RAI Responses 9-12

RESPONSE TO RAI COMMENT 9
ROADMAP TO REFERENCES

REFERENCED DOCUMENT	*EXCERPT LOCATION	REMARK
Draft Section 3116 Determination Salt Waste Disposal SRS.	Excerpt enclosed following response.	

*Excerpt Locations:
1. Excerpt included in response: The excerpt is included within the text of the response or is appended to the response.
2. Excerpt enclosed following response: The excerpt is enclosed on a separate sheet or sheets following the response.
3. Representative excerpt(s) enclosed following response: Representative excerpts from a document that is wholly or largely applicable are enclosed following the response.
4. Other

APPROVED for Release for
Unlimited (Release to Public)

7/14/2005

DOE-WD-2005-001
February 28, 2005

Draft Section 3116 Determination

Salt Waste Disposal

Savannah River Site

4.0 THE WASTE DOES NOT REQUIRE PERMANENT ISOLATION IN A DEEP GEOLOGIC REPOSITORY FOR SPENT FUEL OR HIGH-LEVEL RADIOACTIVE WASTE

Section 3116 (a) of the NDAA provides in pertinent part:

> [T]he term "high-level radioactive waste" does not include waste from reprocessing of spent nuclear fuel that the Secretary of Energy......, in consultation with the Nuclear Regulatory Commission...determines –
>
> > (1) does not require permanent isolation in a deep geologic repository for spent fuel or high-level radioactive waste[.]

The purpose of Section 3116 is to clarify the Secretary's authority, in consultation with the NRC, to determine that certain waste from reprocessing that meets the criteria set out in the section should not be classified as "high-level radioactive waste." It sets out two specific criteria for making this determination in clauses (2) and (3). Clause (2) requires DOE to remove highly radioactive radionuclides to the maximum extent practical. Clause (3) generally mirrors the criteria that the NRC has established for determining whether waste qualifies for land disposal as low level waste(See 10 CFR. 61.55 and 61.58). This includes waste that falls within one of the classes set out in Section 61.55, as well as waste that will be disposed of so as to meet the performance objectives of subpart C of Part 61.

Clause (1) is a broader criterion that requires the Secretary, in consultation with the NRC to consider whether, notwithstanding that waste from reprocessing meets the other two criteria, there are other considerations that, in the Secretary's judgment, require its disposal in a deep geologic repository. Generally speaking, this would be an unusual case. This is because waste that meets the third criterion would be waste that the Secretary, in consultation with the NRC, has already determined either falls within one of the classes set out in section 61.55 that the NRC has specified are considered "generally acceptable for near-surface disposal" or that the Secretary, in consultation with the NRC, has already determined will be disposed of in a manner that meets the Part 61 subpart C performance objectives. These latter objectives, in turn, as the NRC recently explained, "set forth the ultimate standards and radiation limits for: (1) protection of the general population from releases of radioactivity; (2) protection of individuals from inadvertent intrusion; (3) protection of individuals during operations; and (4) stability of the disposal site after closure." In the Matter of Louisiana Energy Services (National Enrichment Services), CLI-05-05, slip op. at 11 (Jan. 18, 2005). It follows that if disposal of a waste stream in a facility that is not a deep geologic repository will meet these objectives, in the ordinary case that waste stream does not "require disposal in a deep geologic repository" because non-repository disposal will be protective of the public health and safety.

That said, it is possible that in rare circumstances, a waste stream that meets the third criterion might have some other unique radiological characteristic or may raise unique policy

RESPONSE TO RAI COMMENT 10
ROADMAP TO REFERENCES

REFERENCED DOCUMENT	*EXCERPT LOCATION	REMARK
Boyles et al 2001	RPP-7702 Section 4.8.3.2 enclosed following response.	Section 4.8.3.2 provides the Hanford scoping cost estimate for building one tank, assuming four tanks are built. (For evaluation purposes, the Hanford costs are assumed comparable within uncertainties, to SRS cost.)
	RPP-7702 Section 4.8.3.3 enclosed following response.	Section 4.8.3.3 provides the Hanford schedule for building one tank, assuming four tanks are built. (For evaluation purposes, the Hanford schedule is assumed comparable to those items applicable to SRS.)
d'Entremont et al. 2005	CBU-PIT-2005-00150 enclosed following response.	This document provides the details concerning the cost and benefit evaluation performed for evaluating the three salt waste treatment cases. It can be used to provide additional details in response to the RAI#10.
d'Entremont et al. 2005	CBU-PIT-2005-00150, Section 3, 4 & 5 enclosed following response.	Section 3.3 and 3.4 of this document quantify worker and population dose. Section 4 addresses cost. Section 3.1 and 3.2 discuss activity reduction and tank years. Section 5 addresses additional qualitative evaluation factors.

7/14/2005

RESPONSE TO RAI COMMENT 10
ROADMAP TO REFERENCES

DRAFT Section 3116 Determination (not specifically called out as a reference in RAI but can be inferred)	DOE-WD-2005-001, Page 7 enclosed following response.	DOE-WD-2005-001, Page 7, para 3 provides a brief general description of the current salt waste disposition strategy.
NCRP 1987	NCRP Report No. 93, Page 14, Table 2.3 enclosed following response.	NCRP Report No. 93, Page 14, Table 2.3 provides the radiation exposure from natural background.
WSRC 1993	WSRC-OS-94-42 Section E enclosed following response.	Section E contains the requirement for DOE to submit to the EPA and SCDHEC schedules for tanks that do not meet secondary containment requirements.

Tank Space Options Report

V. C. Boyles et al
CH2M HILL Hanford Group, Inc., Richland, WA 99352
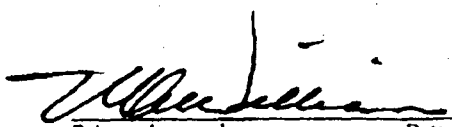Office of River Protection Contract DE-AC27-99RL14047

EDT/ECN: EDT- 632177   UC: 2070
Org Code: 7M100   Contract No.: 7899-32
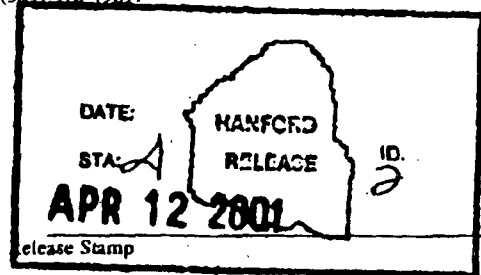B&R Code: EW 3120074   Total Pages: *156*

Key Words: Tank Space, SST Retrieval, DST Capacity

Abstract: N/A

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DATE: HANFORD
STA: RELEASE ID.
APR 12 2001

Release Approval   Date   Release Stamp

Approved for Public Release

4.8 CONSTRUCT NEW DOUBLE-SHELL TANKS

4.8.1 Description

This option would create additional waste storage capacity by construction of DSTs similar to the existing design. These new tanks will meet state and federal regulations and will have a 50-year design life. Each additional DST would increase available capacity by 1200 kgal.

4.8.2 Background

This option is consistent with current program strategy of moving SST wastes to compliant DST storage. The tanks will be designed using lessons learned from the Waste Tank Safety Program, the Multi-Function Waste Tank Facility design (WHC 1995), and the AQ and AT Tank Farm designs. Additional waste storage capacity could be available for use in the 2007 - 2011 time frame. The funding process is expected to take approximately two years; design and construction are expected to take five years. It is expected that savings would be obtained if a tank farm were built instead of individual tanks. The costs were based on a design that contained four tanks.

4.8.3 Evaluation of Option

4.8.3.1 Additional Capacity (Gallons)

Construction of each new DST will create 1200 kgal of additional capacity using an existing design.

4.8.3.2 Cost

The cost estimate for this option is presented in Table 4-25.

Table 4-25. Cost Estimate for Construct New Double-Shell Tanks Option
(Based on Each One of at Least Four Tanks)

Activity Description	Cost ($K)
Obtain Permitting and Regulatory Approval	1000
Design	7000
Procurement and Construction	66000
Start-Up and Testing	1000
OPTION TOTAL	**$ 75,000**

The cost estimate is based on the following assumptions:

- adjustments were made to the $360M Advanced Conceptual Design cost estimate of Project W-236A, "Multi-Function Waste Tank Facility," a four-tank facility (WHC 1995);

- the Project W-236A estimate is used as a basis without detailed evaluation;

- an escalation factor of 3% per year was used to calculate past cost to present cost;

- weather enclosure for the tank farm was deleted from the Project W-236A estimate;

- mixer pumps are not used and were deleted from the Project W-236A estimate;

- tanks are constructed of carbon steel instead of 304L stainless steel;

- modification of the Environmental Impact Statement is required; and

- a tank farm containing four tanks, the same as Project W-236A was used as the basis for the cost of one tank. Additions of less than four tanks are likely to cost more per unit of volume.

An annual estimate of expenditures per tank in groups of four is as follows:

Year	Annual Total
1	$250K
2	$250K
3	$6,100K
4	$18,100K
5	$17,100K
6	$16,100K
7	$17,100K

This preliminary schedule of expenditures is not budget grade, but is for scoping purposes only.

4.8.3.3 Schedule

The schedule for this option is presented in Figure 4-8.

Figure 4-8. Schedule for Construct New Double-Shell Tanks Option

Activity ID	Activity Description	Orig Dur	Years 1	2	3	4	5	6	7	8
OPTION 4.8										
P200100	ACQUIRE FUNDING	519								
P200200	OBTAIN PERMITTING & REGULATORY APPROVAL	1,173								
P200300	DESIGN	300								
P200400	PROCUREMENT & CONSTRUCTION	918								
P200500	STARTUP & TESTING	1,127								

Run Date	09APR01	▭ Early Bar ▬ Progress Bar	OPTN		Sheet 8 of 8
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4-55
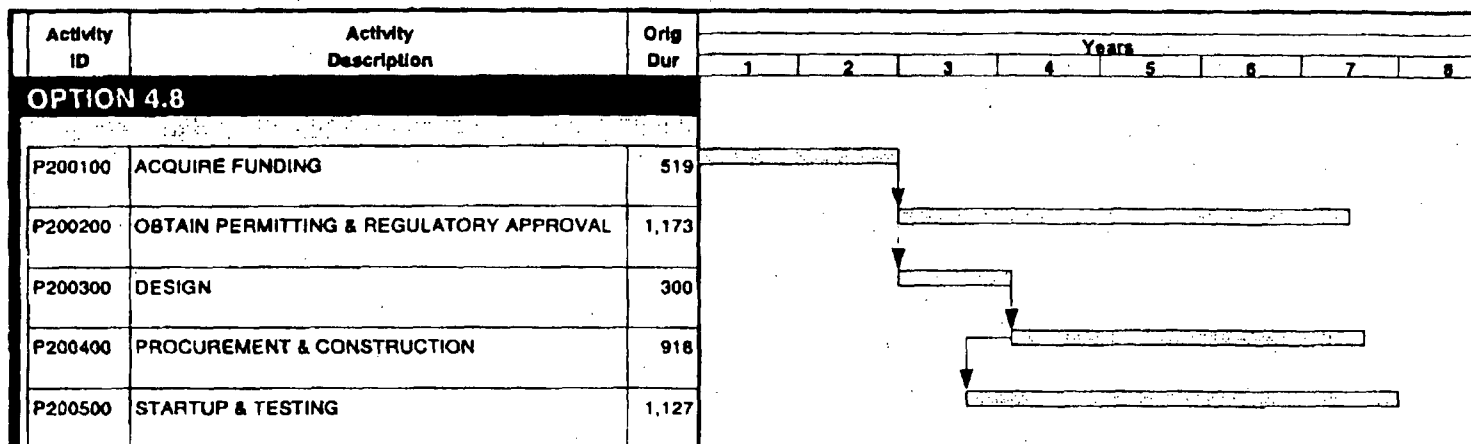
4.8.3.4 Feasibility Considerations

Additional considerations for the feasibility of constructing new DSTs are discussed below. Table 4-26 provides the summary assessment of these considerations.

Table 4-26. Feasibility Considerations for Construct New Double-Shell Tanks Option

Criteria	High Feasibility	Medium Feasibility	Low Feasibility
Technical Feasibility	X		
Environmental Risk	X		
Health and Safety Risk		X	
Maintainability and Operability		X	
Environmental Regulatory Impacts			X
Authorization Basis Impacts		X	

- **Technical Feasibility**

    Construction of new DSTs is technically feasible based upon previous engineering, construction, and operations experience. Detailed analysis of previous conceptual design estimates for DST construction is warranted and has the potential to reduce the cost per gallon derived from cost estimates for the Multi-Function Waste Tank Facility. For example, the River Protection Project plans to build 1500 kgal of low-activity waste feed storage capacity at the WTP for $43M.

- **Environmental Risk**

    New DSTs would be constructed per the requirements of Washington Administrative Code 173-303-640 (4)(c)(iii) that double walled tanks be designed as an integral structure (i.e., an inner tank completely enveloped within an outer shell) and be provided with a built-in continuous leak detection system. Operation of new DSTs would incur minimal incremental environmental risk; however, additional DSTs will result in an increase in closure costs for the Hanford Site.

- **Health and Safety Risk**

    The construction of new DSTs is a major construction activity that presents industrial safety hazards commensurate with other projects of this magnitude. No additional health risk to the public is expected from construction of new DSTs.

- **Maintainability/Operability**

  Additional maintenance or operational expense is incurred for the routine operation of additional DSTs. The maintenance and operational activities are similar current activities, although the newer DSTs are likely to be subject to fewer equipment failures.

- **Environmental Regulatory Impacts**

  Construction of new DSTs would require a change to the RCRA Part B permit to incorporate the new facilities. In addition, supplemental *National Environmental Policy Act* (NEPA) documentation would be needed. Significant public comment would be expected.

- **Authorization Basis Impacts**

  Construction of new DSTs would require an amendment to the tank farm Authorization Basis (CHG 2000a, CHG 2000b) to incorporate the new facilities. However, this would not involve new processes.

Cost and Benefit Evaluation for

Three Salt Waste Treatment Cases

at SRS

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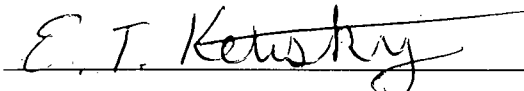July 7, 2005

Westinghouse Savannah River Company
Closure Business Unit
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Cost and Benefit Evaluation for

Three Salt Waste Treatment Cases

at SRS

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Approved by:

S. J. Robertson   _7/7/05_

S. J. Robertson, Technical Integration and Process Development   Date

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SUMMARY OF REVISIONS

6/2005   Revision 0   Initial issue

7/2005   Revision 1   Exposure and cost modified to reflect
                                                    revised assumptions

ACRONYMS

ARP	Actinide Removal Process
B	Used to denote billions of dollars ($B)
CSSX	Caustic Side Solvent Extraction
D&D	Decontamination and Decommissioning
DDA	Dissolution, Deliquification, and Adjustment
DOE	U.S. Department of Energy
DWPF	Defense Waste Processing Facility
EPA	Environmental Protection Agency
ETP	Effluent Treatment Process
FFA	Federal Facilities Agreement
HLW	High Level Waste
IW	Inhibited Water
LCC	Life Cycle Cost
LCS	Low Curie Salt. The use of this term has been replaced with DDA
LIP	Limited Interim Processing
M	Million. Used to denote millions of gallons (Mgal), millions of Curies, and millions of dollars ($M)
MCU	Modular CSSX Unit
NC	Noncomplaint·
NIP	No Interim Processing
NRC	U.S. Nuclear Regulatory Commission
RAI	Request for Additional Information
SCDHEC	South Carolina Department of Health and Environmental Control
SDF	Saltstone Disposal Facility
SPF	Saltstone Production Facility
SRS	Savannah River Site
SWPF	Salt Waste Processing Facility
WD	Waste Determination contained as Reference 1

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GLOSSARY

Baseline Case - The Interim Salt Treatment Strategy is described in the Waste Determination (Ref. 1) (WD) and involves initial salt waste disposition using Deliquification, Dissolution, and Adjustment (DDA) followed by Actinide Removal Process (ARP) and Modular CSSX (Caustic Side Solvent Extraction) Unit (MCU) in combination with DDA until the Salt Waste Processing Facility (SWPF) comes on line. Once the SWPF facility starts up, all salt waste streams will be treated using the SWPF.

Curie Years - The number of years a tank is in service multiplied by the activity (Curies) in the tank.

Life Cycle - The term applied to the entire period from facility conceptualization to completion of Decontamination and Decommissioning (D&D).

Life Cycle Cost (LCC) - The cost of a good or service over its entire life cycle.

Limited Interim Processing (LIP) Case – This case waits to begin initial salt waste disposition until the ARP and MCU facilities are ready to begin operation in 2007. No salt waste is processed using the DDA process. Upon start-up of SWPF, ARP/MCU operations cease and all salt waste is processed using the SWPF.

Noncompliant Tanks – HLW Tanks that do not have full secondary containment.

No Interim Processing (NIP) Case – This case waits to begin initial salt waste disposition until the SWPF is ready to begin operation in 2009. No salt waste is processed using the DDA process or with the ARP/MCU facilities. Using this case, all salt waste is processed using the SWPF.

Old-Style Tanks – Tank Types I, II, and IV.

Sunk Cost – Funds previously expended that cannot be recovered regardless of future events.

(Old Style) Tank Years – A term used to describe material at risk in terms of noncompliant tanks. One "Old Style (noncompliant) Tank Year" is equal to one noncompliant tank in HLW service for one year.

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EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the Draft Salt Waste Disposal Section 3116 Determination (WD) (Ref. 1) and provided a request for additional information (RAI) (Ref. 2) in the form of comments on the WD. This document addresses the comment numbered 10 in the RAI. That comment states, "Additional information is needed to support the conclusion that use of interim treatment measures before the completion of the Salt Waste processing Facility (SWPF) is consistent with removal of highly radioactive radionuclides to the maximum extent practical." The comment further requests that the U.S. Department of Energy (DOE) "Provide a detailed cost/benefit analysis supporting a comparison of the proposed alternative with alternative treatment plans. The response should address the quantitative and qualitative costs and benefits of treating waste with the SWPF alone as well as the costs and benefits of treating waste with both the Actinide Removal Process (ARP)/Modular Caustic Side Solvent Extraction (MCU) and the Salt Waste Processing Facility (SWPF)."

This document evaluates and compares financial and non-financial costs and benefits associated with three different cases and demonstrates that the salt waste disposition strategy described in the Salt Waste Disposal Waste Determination (WD) is the most cost effective case and the case that will provide the lowest overall risk to the site worker and to the general public.

The following three cases were evaluated and compared in this document:

Case #1: Baseline Case - The Interim Salt Treatment Strategy is described in the Salt Waste Disposal Waste Determination and involves initial salt waste disposition using Deliquification, Dissolution, and Adjustment (DDA) followed by Actinide Removal Process (ARP) and Modular CSSX (Caustic Side Solvent Extraction) Unit (MCU) in combination with DDA until the Salt Waste Processing Facility (SWPF) becomes operational. Once the SWPF facility becomes operational, all salt waste streams will be treated using the SWPF.

Case #2: Limited Interim Processing (LIP) Case – This case does not begin salt waste disposition until the ARP/MCU facilities begin operation in 2007. No salt waste is processed using the DDA process. Upon start-up of SWPF, ARP/MCU operations cease and all salt waste is processed using the SWPF.

Case #3: No Interim Processing (NIP) Case – This case does not begin salt waste disposition until the SWPF is ready to begin operation in 2009. No salt waste is processed using the DDA process nor with the ARP/MCU facilities. Using this case, all salt waste is processed using the SWPF.

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Note that the Baseline Case is the case that was described in the WD and that the other two cases were requested for evaluation in the RAI. These other two cases evaluated herein, namely LIP and NIP, represent scenarios for evaluation only and do not represent detailed plans that have been accepted by either the DOE or by the facilities involved.

When these three cases were evaluated, the following fundamental differences were noted between the cases:

Completion of Operations: The Baseline Case completed High Level Waste (HLW) system operations in 2019. The salt waste dispositioned to SDF by DDA and ARP/MCU created compliant tank space in the HLW Tank Farm that permitted Defense Waste Processing Facility (DWPF) operations to continue without interruption and permitted SWPF operations to commence processing at forecast production rates. The Limited Interim Processing Case required 3+ years longer to complete HLW system operations than the Baseline Case. The delay in mission completion resulted from the reduced processing rates through SWPF and DWPF caused by the limited compliant tank space available to prepare the salt and sludge waste streams for processing during initial years of SWPF operation. DWPF production rates are impacted because of the limited compliant tank space prevents sludge washing which is required prior to processing sludge waste at DWPF. The No Interim Processing (NIP) Case required approximately 5+ years longer to complete HLW system operations than the Baseline Case. The delay in mission completion resulted from the reduced processing rates through SWPF and DWPF caused by the limited compliant tank space available to prepare the salt and sludge waste streams for processing during initial years of SWPF operation.

Risk: The doses (exposures) associated with each of the three cases were compared as well as the material/facilities at risk. Dose was further broken down in terms of dose to the facility worker, dose to the public from both ongoing operations and from material dispositioned to the SDF, and dose to the inadvertent intruder from the SDF. In order to appropriately characterize the risks from ongoing operations, the differences between the cases in terms of old style tank closure years and Tank Farm waste disposition rates expressed in Curie Years were evaluated. The evaluations are summarized in the following table.

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Table 1: Summary of Dose, Tank Years and Curie Year Impacts

Case Evaluated	Worker Dose [1] (rem)	Current Public Dose [2] (mrem/yr)	SDF Intruder Dose [3] (rem)	SDF All Pathways Dose (mrem/yr)	Old Style Tank Years [4]	Tank Farm Curie Years [5]
Baseline	890	0.19	36.9	2.3	240	3.7E+09
LIP Case	1100	0.19	36	2.3	300	4.7E+09
(change from baseline)	(+24 %)	(0 %)	(- 2.5 %)	(0 %)	(+25%)	(+25%)
NIP Case	1200	0.19	36	· 2.3	340	5.3E+09
(change from baseline)	(+35 %)	(0 %)	(- 2.5 %)	(0 %)	(+42%)	(+42%)

(1)   Integrated dose to HLW workers

(2)   Dose to public from current SRS operations

(3)   100 year integrated dose (2105 – 2205); includes 360 mrem/yr contribution from background radiation

(4)   Total number of years old style tanks are in service, where 20 tanks in service for 2 years = 40 Tank Years

(5)   Total number of years a curie is in the Tank Farm, where 30 MCi in the Tank Farm for three years = 90M Curie Years

It can be seen from Table 1 that the Baseline Case results in significantly lower worker dose and significantly shorter time that radioactive material remains in the old style tanks. Intruder doses are higher for this case, but the difference is not significant when compared to exposure from background sources of radiation (360 mrem/year) (Ref. 11). The LIP and NIP Cases show significant increases from the Baseline Case for worker exposure and time that radioactive material remains in the Tank Farm. Thus, the Baseline Case provides the lowest risk for worker exposure, as well as, from a time of radiological material at risk perspective, especially in old style tanks.

Financial Cost: The Baseline Case is the most cost effective case. The primary reason that the Baseline Case is the most cost effective is the difference in lifecycle costs associated with extending the HLW system (Tank Farms, DWPF, SWPF, Saltstone Production Facility (SPF), Saltstone Disposal Facility (SDF), etc.) operations by 3+ years for the LIP Case and 5+ years for the NIP Case. This results in an additional cost for operation of approximately $1B ($1.6B escalated) and $1.5B ($2.6B escalated), respectively. Since the sunk costs for ARP/MCU construction are high relative to the total project cost, and since the life cycle costs for the HLW system are much higher than the project construction and D&D (decontamination and decommissioning) costs, life cycle costs dominate the cost comparison. As a result of the relative costs, the case that results in the shortest life cycle will have the lowest financial cost.

Other aspects of the facility operations that were reviewed as a part of this evaluation included consideration of slowing down DWPF rather than shutting down DWPF due to feed streams (sludge batches) to DWPF being unavailable. The slowdown avoids a shutdown of DWPF and subsequent restart. The evaluation shows that slowing down DWPF is preferred over shutdown

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from a cost perspective. Cost comparisons utilized this basis when DWPF operation was evaluated.

Construction of new HLW storage tanks are evaluated against the baseline case. Since the cost of new tank construction was significantly more than the lifecycle cost for ARP/MCU facility (less sunk costs), this was not considered to be cost effective. In addition, the feasibility of designing, constructing and starting up new tanks within the timeframe required to keep from impacting SWPF start-up assumptions is unlikely.

Qualitative Discussion: The primary influence on cost and risk associated with these cases is the duration of facility operation. The evaluation assumes that the dates projected for facility start-up will be achieved and that throughput rates will be as forecasted. Delays in facility start-up and reductions in throughput rates would extend the duration of facility operation with associated increases in cost and a decrease in benefits. It should be noted that this extension in facility operation is likely greater than a day for day match with a delay in facility start-up. Delays in facility start-up will result in less tank space available for salt batch and sludge batch preparation. It would take years of operation at reduced rates to recover the "lost" tank space. In the cases analyzed, it took 4+ years after SWPF start-up for the LIP Case and 7+ years for the NIP Case for SWPF to achieve forecast processing rates. Attaining these forecast processing rates was limited by the availability of compliant tank space to prepare salt batches to feed SWPF at a rate of seven million gallons of salt waste solution per year.

Taken as a whole, the above fundamental differences in the cases evaluated demonstrate that the Baseline Case is the most cost effective option and provides the lowest worker dose. Public doses (including inadvertent intruder doses) are marginally highest with the Baseline Case, but this dose is not significant when compared to exposure from natural sources of radiation. The Baseline Case also reduces radioactive material at risk the most quickly because it facilitates stabilization of radioactive material in the Tank Farm more quickly than in the other cases, as well as permitting closure of old style tanks per the Federal Facility Agreement (FFA) schedule. For these reasons, the Baseline Case provides the greatest overall benefit at the lowest cost.

1 INTRODUCTION

The U.S. NRC staff reviewed the Draft WD (Ref. 1) and provided a RAI (Ref. 2) in the form of comments on the WD. This document responds to the comment numbered 10 in the RAI. That comment states:

Comment: Additional information is needed to support the conclusion that use of interim treatment measures before the completion of the SWPF is consistent with removal of highly radioactive radionuclides to the maximum extent practical.

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Basis: The NRC agrees with the conclusion in Reference 4 that the determination of whether highly radioactive radionuclides have been removed to the maximum extent practical can include a wide variety of considerations. However, it is expected that any factors included in the determination will be supported by a technical basis and, when possible, quantitative comparisons.

For example, although it is stated that risk to the public is reduced by continuing sludge processing at the Defense Waste Processing Facility (DWPF) [4], no information is presented to support the amount of risk reduction achieved by continuing waste processing prior to completion of construction of the SWPF. Furthermore, insufficient information is presented to enable a comparison between the increased risks associated with disposing of Deliquification, Dissolution and Adjustment (DDA) and Actinide Removal Process ARP/MCU waste in saltstone with the risks associated with postponing treatment until all of the waste can be treated at the SWPF.

Similarly, although it is stated that it is necessary to treat waste with interim procedures prior to the completion of the SWPF because shutdown of the DWPF due to tank space limitations will be economically impractical, a comparison between the costs of shutting down and restarting the DWPF with the costs of implementing the proposed interim treatment procedures and disposing of higher activity waste in the SDF has not been provided. Although it was estimated that it would cost $1 billion to halt and restart waste processing with the DWPF [4], no basis for that estimate was given.

Path Forward: Provide a detailed cost/benefit analysis supporting a comparison of the proposed alternative with alternative treatment plans. The response should address the quantitative and qualitative costs and benefits of treating waste with the SWPF alone as well as the costs and benefits of treating waste with both the ARP/MCU and the SWPF. The response should include:

1) A comparison between the dose to the general public, workers, and inadvertent intruders associated with the proposed treatment plan and the two alternatives (e.g., treating waste with the SWPF alone or treating waste with the ARP/MCU and SWPF). The response should also include an estimate of the dose that the tanks currently pose to the public as well as the number of Tank Years of waste storage in old style that would be avoided by treating waste with DDA and ARP/MCU instead of waiting to treat waste with the SWPF (e.g., percent reduction). Consideration should be given to the fact that the wastes that have been proposed to be removed are the lowest activity wastes [4].

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2) A comparison of the financial costs associated with at least three alternatives (i.e., the proposed alternative, treating waste at the SWPF alone, and treating waste with the ARP/MCU and SWPF). The response should address the costs associated with construction and operation of interim procedures and the costs associated with disposing of a higher activity waste on site, as well as the costs of ceasing and restarting sludge processing. Additional alternatives, such as slowing down the throughput of the DWPF or creating new interim tank storage, should be considered. The comparison should also consider factors other than economic cost (e.g., schedule) and the factors should be converted into a comparable metric (e.g., cost and risk) to the extent practical.

The analysis should reflect uncertainties in the timing of when sludge processing would need to cease due to lack of tank space and the uncertainty in the availability of the ARP/MCU, and SWPF treatment facilities.

1.1 Purpose

The purpose of this document is to respond to the NRC RAI comment 10, Ref. 2. To put the response in context, the following background information will be provided. A fundament element of the comment is a request that DOE demonstrate that a technical basis using quantitative comparisons supports the recommended case. The salt disposition cases that the NRC requests this response evaluate are:

Case #1: Baseline Case - The Interim Salt Treatment Strategy is described in the WD and involves initial salt waste disposition using DDA followed by ARP/MCU in combination with DDA until the SWPF comes on line. Once the SWPF facility starts up, all salt waste streams will be treated using the SWPF.

Case #2: Limited Interim Processing (LIP) Case – This case waits to begin initial salt waste disposition until the ARP/MCU facilities are ready to begin operation in 2007. No salt waste is processed using the DDA process. Upon start-up of SWPF, ARP/MCU operations cease and all salt waste is processed using the SWPF. It should be noted that this case is for evaluation only and does not represent detailed plans that have been accepted by either the DOE or by the facilities involved.

Case #3: No Interim Processing (NIP) Case – This case waits to begin initial salt waste disposition until the SWPF is ready to begin operation in 2009. No salt waste is processed using the DDA process or with the ARP/MCU facilities. Using this case, all salt waste is processed using the SWPF. It should be noted that this case is for evaluation only and does not represent detailed plans that have been accepted by either the DOE or by the facilities involved.

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This document provides an evaluation of the costs and benefits associated with the three cases described above and documents the basis for the conclusion that the Baseline Case provides the most cost effective treatment for the salt waste at the greatest benefits.

2 DISCUSSION

This section of the report provides an initial comparison of the three cases so that the reader can understand the differences in terms of salt waste dispositioned by the different processes for each case in terms of curies and gallons processed by year. The Baseline Case is as described in the WD and the other two cases, LIP and NIP, are presented to represent alternate scenarios for evaluation only. These two other cases do not represent detailed plans that have been accepted by either the DOE or by the facilities involved. The information provided in that comparison will then be used to evaluate the impact that these differences have on dose and on cost for each of the alternatives. Risk will be expressed in terms of exposure to the public, to the facility worker and to an inadvertent intruder who is located at the Saltstone Disposal Facility (SDF) after institutional controls restricting site access have ended. Risk will further be quantified in terms of material at risk, both in terms of Old Style Tank Years and Curie Years. These terms will be described in detail in the appropriate sections.

Next, the financial cost associated with each of the cases will be compared to determine the most cost effective case. In addition to the three cases described above, additional discussion will be provided comparing costs for DWPF shutdown vs. slowdown and for construction of additional compliant tank space as compared to the Baseline Case.

Following these quantitative evaluations, qualitative discussion will be provided that describes pertinent information that is difficult to quantify but important to understand to appropriately evaluate the cases described. The last section in the report is a summary of the conclusions that can be drawn from the evaluation of cost and risk for the cases described in the discussion.

2.1 Flowsheets & Curie Balance

Baseline Case (Case #1): DOE is separating the low activity fraction of salt waste at SRS using a two-phase, three part process. The first phase will involve two parts to treat the lower activity salt waste: a) beginning in 2005, processing of a minimal amount of the lowest activity salt waste through a process involving DDA of the waste; and b) beginning in approximately 2007, processing of a minimal amount of additional salt waste with slightly higher activity levels using ARP/MCU, along with deliquification and dissolution of saltcake. The second, and longer term phase, beginning in approximately 2009, involves the separation and processing of the remaining (and by far the majority) of the salt waste using SWPF. This second phase will begin as soon as SWPF is constructed, permitted by the State of South Carolina, and operational.

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Per Ref. 5 after both liquid removal by processing through the Tank Farm evaporator systems and later additions of liquids for saltcake dissolution and chemistry adjustments required for processing, approximately 84 Mgal (5.9 Mgal existing salt waste through the DDA process, 1.0 Mgal future salt waste through the DDA process, 2.1 Mgal through ARP/MCU, 69.1 Mgal existing salt waste through SWPF, and 5.9 Mgal future salt waste through SWPF) of salt solution will be processed by Interim Salt Processing and SWPF resulting in approximately 168 Mgal of grout output from SPF to be disposed in SDF. Refer to Figure 1.
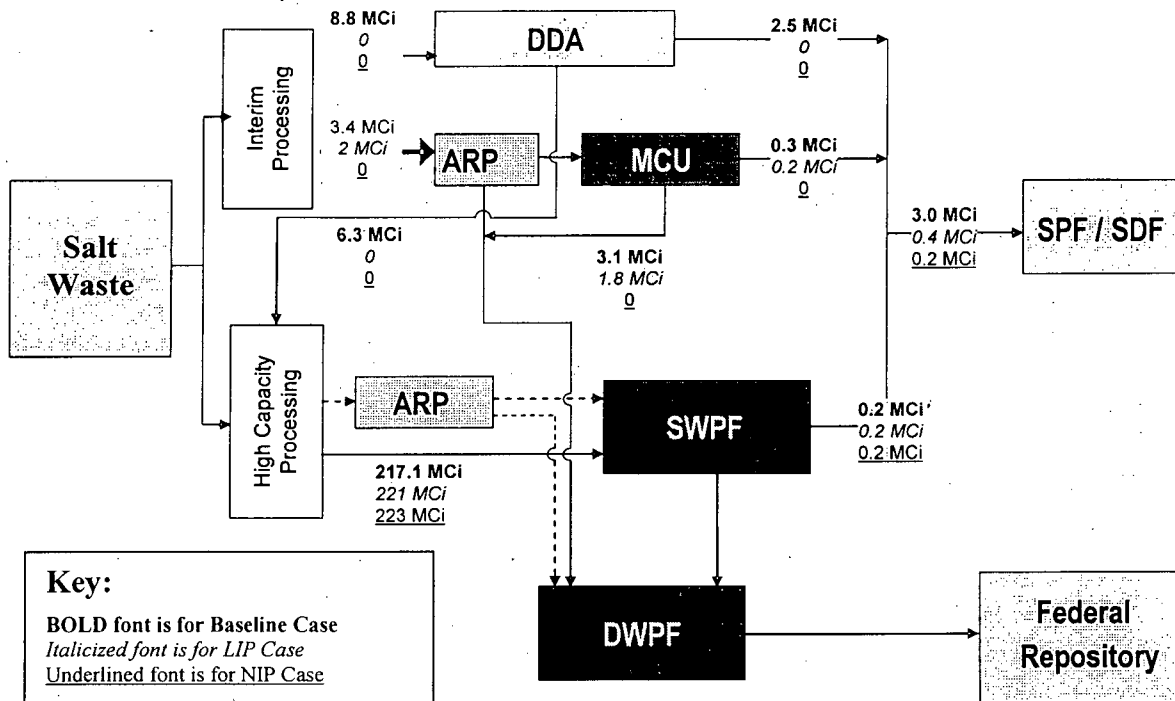

Figure 1: Salt Processing Pathways

Baseline Case (Case #1): This figure shows that for the Baseline Case approximately 2.5 MCi of decontaminated salt waste is planned to be sent to SPF/SDF following processing by DDA and approximately 0.3 MCi of decontaminated salt waste sent to SPF/SDF from following processing by ARP/MCU. This salt waste will be sent in a total of nine batches. Seven of the nine batches are DDA batches. The other two are ARP/MCU batches. Following the completion of interim salt processing, approximately 0.2 MCi of decontaminated salt waste will be sent to SPF/SDF following processing in SWPF.

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LIP Case (Case #2): This case involves limited interim processing of salt waste as compared to the Baseline Case. Phase one salt waste processing in the LIP Case begins in approximately 2007. The initial batches processed in the LIP Case use ARP/MCU to decontaminate the salt waste. The second, and longer term phase, beginning in approximately 2009, involves the processing of the remaining (and by far the majority) of the salt waste using SWPF. This second phase will begin as soon as SWPF is constructed, permitted by the State of South Carolina, and operational.

After both liquid removal by processing through the Tank Farm evaporator systems and later additions of liquids for saltcake dissolution and chemistry adjustments required for processing, approximately 84 Mgal (2.1 Mgal through ARP/MCU, 75 Mgal existing salt waste through SWPF, and 6.9 Mgal future salt waste through SWPF) of salt solution will be processed by Interim Salt Processing and SWPF resulting in approximately 168 Mgal of grout output from SPF to be disposed in SDF.

This figure (Figure 1) shows that for the LIP Case, approximately 0.2 MCi of decontaminated salt waste sent to SPF/SDF following processing by ARP/MCU. This salt waste will be sent in a total of two ARP/MCU batches. Following the completion of interim salt processing, approximately 0.2 MCi of decontaminated salt waste will be sent to SPF/SDF following processing in SWPF.

Note that the fundamental difference between the Baseline Case and the LIP Case are that no salt waste is processed in the LIP Case using the DDA process. Initial processing begins two years later than in the Baseline Case because ARP/MCU does not start-up until approximately 2007. As a result of the LIP, a significant reduction in volume of salt solution is processed prior to SWPF start-up (6.9 Mgal), and insufficient compliant Type III Tank space is available to support SWPF start-up at a rate of 7 Mgal/year throughput. It is estimated that only two compliant Type III Tanks are available to prepare salt batches in 2009, and that it will take approximately 4+ years of SWPF operation to recover sufficient additional compliant Type III Tanks to support a SWPF processing rate of 7 Mgal/year. This initial reduction in the SWPF throughput rates is projected to have an overall impact of a 3+ year extension to the HLW system lifecycle. Note that for financial cost analysis and worker dose, the years of lifecycle increase for the LIP Case over the Baseline Case are conservatively rounded to three years. When using this data to calculate Curie Years, it is estimated that this quantity of radioactive material is at risk for 3+ years.

NIP (Case #3): This case involves "No Interim Processing" of salt waste as compared to the Baseline Case. Salt waste processing in the NIP Case begins in approximately 2009. All batches processed in the NIP Case use SWPF to decontaminate the salt waste. Salt waste processing will begin as soon as SWPF is constructed, permitted by the State of South Carolina, and operational.

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After both liquid removals by processing through the Tank Farm evaporator systems and later additions of liquids for Saltcake dissolution and chemistry adjustments required for processing, approximately 84 Mgal of salt solution will be processed by SWPF resulting in approximately 168 Mgal of grout output from SPF to be disposed in SDF. As Figure 1 shows, for the NIP Case, approximately 0.2 MCi of decontaminated salt waste will be sent to Saltstone following processing in SWPF.

Note that the fundamental difference between the Baseline Case and the NIP Case are that no salt waste is processed in the NIP Case using the DDA, and ARP/MCU process. Initial processing begins four years later than in the Baseline Case because SWPF start-up is not until approximately 2009. As a result of NIP, no salt solution is processed prior to SWPF start-up (vs. approx. 9 Mgal in the Baseline Case), and insufficient compliant Type III Tank space is available to support SWPF start-up at a rate of 7 Mgal/year throughput. It is estimated that only one compliant Type III Tank is available to prepare salt batches in 2009, and that it will take 7+ years of SWPF operation to recover sufficient additional compliant Type III Tanks to support a SWPF processing rate of 7 Mgal/year. This initial reduction in the SWPF throughput rates is projected to have an overall impact of a 5+ year extension to the HLW system lifecycle. Note that for financial cost analysis and worker doses, the years of lifecycle increase for the NIP Case over the Baseline Case are conservatively rounded to five years. When using this data to calculate Curie Years, it is estimated that this quantity of radioactive material is at risk for 5+ years.

2.2 Processing Rates

This section of the report uses the cases described above and shows the quantity of radioactive material that is processed via DDA, ARP/MCU, SWPF, and DWPF each year for each of the cases. The evaluation will be used to show the quantity of material in the Tank Farms at the end of each year in order to provide a comparison of the unstabilized radioactive material at risk through time for each of the cases evaluated. Significant risk reduction is achieved by permanently stabilizing salt and sludge wastes in a solid form (grout and glass, respectively). When the wastes are solidified, they are no longer available to leak from the HLW storage tanks in the Tank Farm. Therefore, it is appropriate to characterize the quantity of radioactive material remaining in the Tank Farms each year when evaluating the differences between the cases.

The following assumptions are used for all of the cases evaluated. Where individual case assumptions are different, they are presented with the review for each case.

- As stated in Reference 6 there are 432 Million Curies (MCi) of supernate, suspended solids, interstitial liquid, and sludge, salt currently in the Tank Farms. Of the total inventory, 199 MCi will be processed through sludge processing and the remainder (233 MCi) will be processed via salt processing. The inventories used in this evaluation are based on most current data (Ref. 3) and differ from the information presented in Figure 1 and in the WD.

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- Annual DWPF processing rates were assumed to equal to the total number of sludge curies to be processed/number of years remaining of DWPF operation unless slowed in early years (2005 through SWPF processing at full production rate) to match SWPF lifecycle.
- Start-up dates for facility operation match those in case description.
- Annual SWPF disposition rates (Curies) are scaled to throughput in terms of gallons.
- Decay is not considered for the purpose of this evaluation.

2.2.1 Baseline Case Assumptions

The assumptions are as follows.

- Five compliant Type III Tanks available for SWPF feed preparation in 2009, so full production rate of 7 Mgal/year is achieved quickly.
- SWPF will complete salt waste processing in 2019.

Table 2: Case #1 = Baseline Case Disposition Rate

Year/ Batch		DDA (MCi)	ARP/MCU (MCi)	SWPF (MCi)	DWPF (MCi)	Total* Processed (MCi)
2005	B0	0	0	0	13	14.
	B1	0.46				
	B2	0.31				
2006	B3	0.52	0	0	13	15
	B4	0.44				
	B5	0.43				
2007	B6	0	0.18	0	13	13
	B7	0.04	0			
2008	B8	0	0.11	0	13	14
	B9	0.26	0			
2009		0	0	10	13	23
2010		0	0	22	13	35
2011		0	0	22	13	35
2012		0	0	22	13	35
2013		0	0	22	13	35
2014		0	0	22	13	35
2015		0	0	22	13	35
2016		0	0	22	13	35
2017		0	0	22	13	35
2018		0	0	22	13	35
2019		0	0	22	13	35
Sum*		2.5	0.29	230	199	432

*Note: Numbers may not sum to actual totals because of rounding

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2.2.2 LIP Case Assumptions

The assumptions are as follows:

- Two compliant Type III Tanks available for SWPF feed preparation in 2009, so full production rate of 7 Mgal/year is not achieved for first four full years of SWPF operation.
- DWPF processing is slowed in years 2006 through 2010 to avoid shutdown costs. Production rates ramped up to match SWPF throughput following slowdown.
- SWPF will complete salt waste processing in year 2023 (vice year 2019 for Baseline Case).

Table 3: Case #2 = LIP Case Disposition Rate

Year/ Batch	DDA (MCi)	ARP/MCU (MCi)	SWPF (MCi)	DWPF (MCi)	Total Processed* (MCi)
2005	0	0	0	13	13
2006	0	0	0	8.0	8
2007	0	1	0	8.0	9
2008	0	1	0	8.0	9
2009	0	0	4.2	8.0	12
2010	0	0	8.5	8.0	16
2011	0	0	9.9	12	22
2012	0	0	13	12	25
2013	0	0	17	12	29
2014	0	0	20	12	32
2015	0	0	20	12	32
2016	0	0	20	12	32
2017	0	0	20	12	32
2018	0	0	20	12	32
2019	0	0	20	12	32
2020	0	0	20	12	32
2021	0	0	20	12	32
2022	0	0	20	12	32
2023	0	0	0.8	0	0.8
Sum*	0	2	231	199	432

*Note: Numbers may not sum to actual totals because of rounding

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2.2.3 NIP Case Assumptions

The assumptions are as follows:

- One compliant Type III Tank is available for SWPF feed preparation in 2009, so full production rate of 7 Mgal/year is not achieved for first seven full years of SWPF operation.
- DWPF processing is slowed in years 2006 through 2010 to avoid shutdown costs. Production rates ramped up to match SWPF throughput following slowdown.
- SWPF will complete salt waste processing in 2025 (vice 2019 for Baseline Case).

Table 4: Case #3 = NIP Case Disposition Rate

Year	DDA (MCi)	ARP/MCU (MCi)	SWPF (MCi)	DWPF (MCi)	Total* Processed (MCi)
2005	0	0	0	13	13
2006	0	0	0	8	8
2007	0	0	0	8	8
2008	0	0	0	8	8
2009	0	0	2.8	8	11
2010	0	0	5.5	8	14
2011	0	0	6.9	10	17
2012	0	0	8.3	10	19
2013	0	0	8.3	10	19
2014	0	0	9.6	10	20
2015	0	0	12	10	23
2016	0	0	17	10	27
2017	0	0	19	10	30
2018	0	0	19	10	30
2019	0	0	19	10	30
2020	0	0	19	10	30
2021	0	0	19	10	30
2022	0	0	19	10	30
2023	0	0	19	10	30
2024	0	0	19	10	30
2025	0	0	8.5	0	8.5
Sum*	0	0	233	199	432

*Note: Numbers may not sum to actual totals because of rounding

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An evaluation of the data in the three tables above makes the following significant points.

- Failure to begin salt waste processing in 2005 will result in extension in facility lifecycle
- Failure to begin salt waste processing in 2005 will result in a reduction in the rate of radioactive material stabilization in the HLW system.

3 MATERIAL AT RISK AND DOSE COMPARISONS

3.1 Activity Reduction Over Time

Figure 2 illustrates the total curies in the Tank Farms as a function of time for the three cases described above in Tables 2, 3, and 4. The area under each curve (Curie Years) would be indicative of the material at risk associated with continued storage of wastes in the Tank Farm. The area under each curve is 3.7E+09 Curie Years, 4.7E+09 Curie Years, and 5.3E+09 Curie Years for the Baseline Case, the LIP Case, and the NIP Case, respectively. The NIP Case increases Curie Years in the Tank Farms by 42% over the Baseline Case. The LIP Case increases the Curie Years in the Tank Farms by 25% over the Baseline Case.

**Curies in the Tank Farm
As a Function of Time**
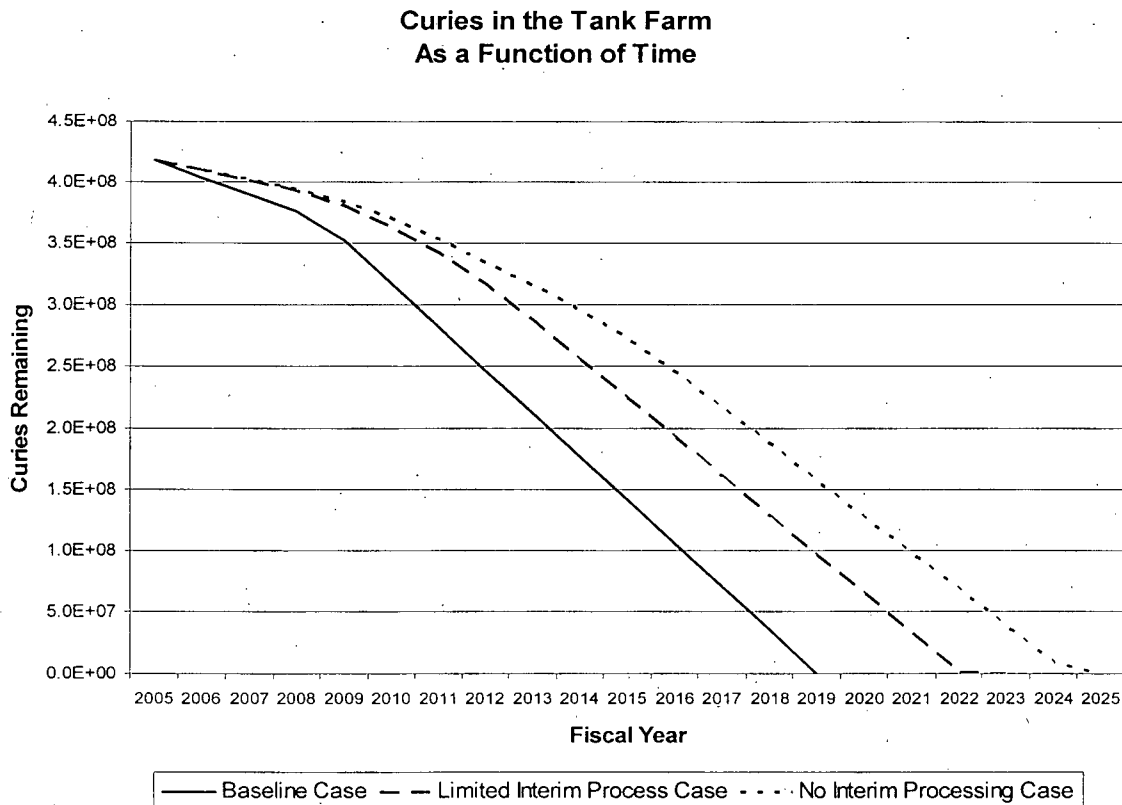
Figure 2: Curies in the Tank Farm as a Function of Time

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Figure 2 shows that the Baseline Case reduces the radioactive material at risk in the Tank Farm at the greatest rate as compared to the other cases.

3.2 Tank Years

The waste tanks in the Tank Farm are of essentially four different types. Tank Types I, II, and IV are considered single shell tanks, while Type III is considered to have secondary containment. The non-Type III Tanks are considered to be noncompliant with current standards for HLW storage tank construction, and therefore, are called noncompliant tanks. In support of the Federal Facilities Agreement (FFA) (Ref. 9), a schedule for the closure of the noncompliant waste tanks has been developed and is summarized below. This closure schedule is a tool used by the regulators to ensure that the highest risk tanks (the noncompliant tanks) are closed expeditiously to minimize the risk associated with their continued use. Failure to meet the closure schedule may result in fines imposed on the DOE as specified in the FFA.

Prior to tank closure, the HLW solutions stored in the tanks must be removed. During waste removal, water that has been chemically treated to prevent corrosion of the carbon steel waste tanks, inhibited water (IW), is added to the waste tanks and agitated by slurry pumps. If the tank contains salt, this IW, and agitation if required, dilutes the concentrated salt or re-dissolves the saltcake. If the tank contains sludge, this water, and agitation, suspends the insoluble sludge solids. In either case, the resulting liquid slurry, which now contains the dissolved salt or suspended sludge, can be pumped out of the tanks and transferred to waste treatment tanks.

Waste removal is a multi-year process. First, each waste tank must be retrofitted with slurry and transfer pumps, infrastructure to support the pumps, and various service upgrades (power, water, air, and/or steam). These retrofits can take between two and four years to complete. Then, the pumps are operated to slurry the waste. Initially, the pumps operate near the top of the liquid and are lowered sequentially to the proper depths as waste is slurried and transferred out of the tanks. Waste removal activities remove the bulk of the waste to prepare the tank for closure.

Following the bulk waste removal process described above, heel removal activities are initiated. The heel removal process includes processes targeted at the "hard to remove" materials remaining in the tank following bulk waste removal. These processes can involve chemical heel removal techniques such as addition of acid, mechanical removal techniques such as targeted high pressure jets, and combinations of mechanical and chemical methods, etc. Similar to the bulk waste removal process, these processes increase the volume of the radioactive waste material that must be stored and processed through the addition of IW and chemical cleaning solutions. Therefore, bulk waste removal and heel removal processes to prepare for tank closure require available waste storage capacity in the Tank Farms.

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If salt waste processing is not initiated to recover compliant Type III Tank space, tank closure operations will be impacted due to the lack of tank space to process the waste generated during the bulk waste and heel removal processes. Therefore, extensions in HLW System lifecycle will delay tank closure by an equivalent number of years. Failure to delay tank closure activities would result in waste generation associated with bulk waste and heel removal processes for the tanks being prepared for closure. The tank space that this waste would consume would result in further lifecycle extensions for the HLW System with associated cost and risk. Therefore, for the purposes of this evaluation, each year of lifecycle extension associated with the cases evaluated will result in a corresponding delay in the continuation of noncompliant tank closures (note that two noncompliant tanks are currently closed).

To assess the differences between the cases in risk associated with using noncompliant tanks, the number of years that the old style, noncompliant tanks will be in service is calculated for each case. This risk will be quantified in terms of old style Tank Years where one old style Tank Year is equal to one old style tank in service for one year. Likewise, twenty old style Tank Years are equal to five old style tanks in service for four years. The difference in the number of years that the old style tanks will be compared between each of the cases evaluated.

Table 5 presents, for each noncompliant (old style) tank, the number of years from 10/2005 to its closure date – the years at risk, and the number of tanks not closed at that closure date for the Baseline Case. The number of at risk tanks and the years at risk can be graphically represented as a step-wise curve (see Figure 3). The area under the step-wise curve would represent the number of Tank Years of risk for the continued storage of wastes in these noncompliant tanks. The area under the step-wise curve is 237 Tank Years. Note that for this evaluation the closure of Tanks 18 and 19 will commence regardless of the salt processing case chosen. Therefore for this analysis, only 20 noncompliant tanks will be addressed.

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Table 5: Closure Dates for Noncompliant Tanks with Years at Risk (as of 10/2005)

Tank #	Closure Date	Years at Risk (years)	Number of Tanks
14	9/30/2010	5	20
11	9/30/2010	5	20
12	9/30/2011	6	18
21	9/30/2012	7	17
22	9/30/2012	7	17
15	9/30/2013	8	15
23	9/30/2014	9	14
24	9/30/2014	9	14
13	9/30/2015	10	12
16	9/30/2015	10	12
2	9/30/2019	14	10
1	9/30/2020	15	9
9	9/30/2020	15	9
10	9/30/2020	15	9
3	9/30/2022	17	6
8	9/30/2022	17	6
4	9/30/2022	17	6
5	9/30/2022	17	6
6	9/30/2022	17	6
7	9/30/2022	17	6

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Number of Noncompliant Tanks after 10/2005 (Excluding Tanks 18 and 19)
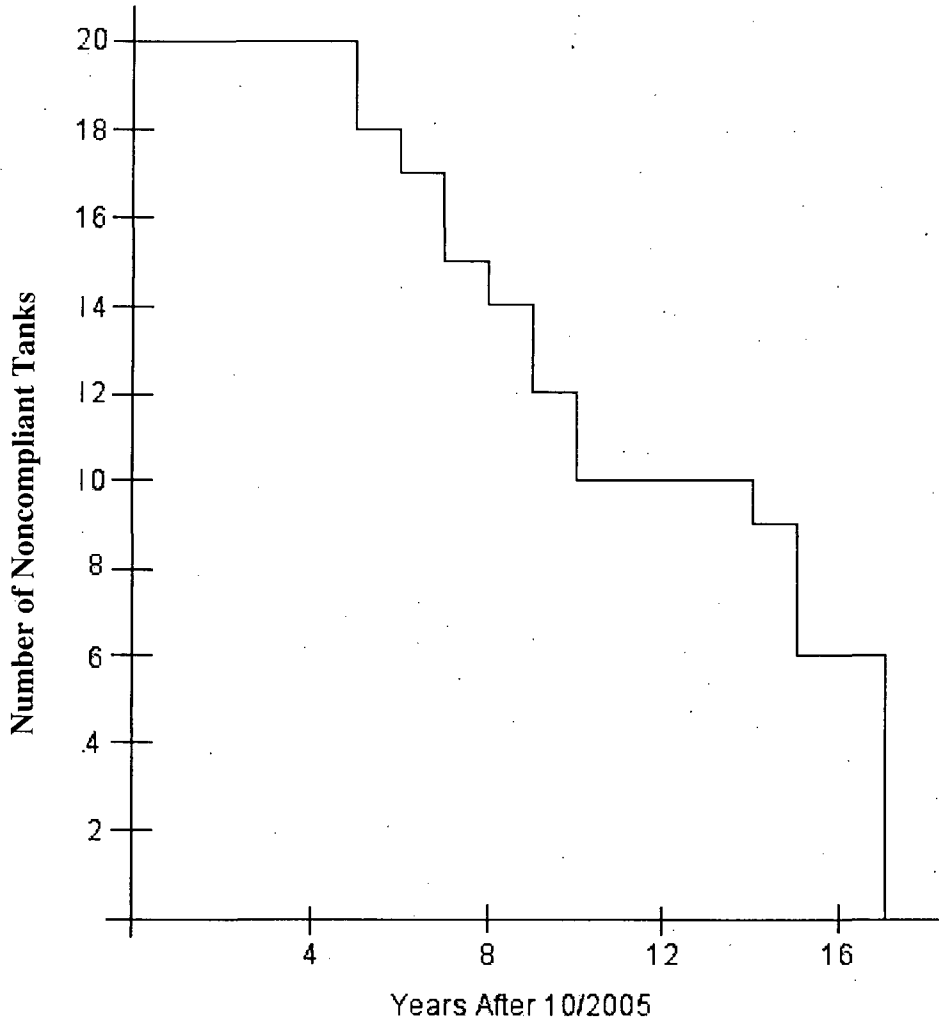
Figure 3: Number of Noncompliant Tanks (after 10/2005)

For each year of life cycle extension caused by limited Type III tank space availability, the step-wise curve would move one year to the right with no decrease in the number of noncompliant tanks. Thus, each year of life cycle extension represents an additional 20 Tank Years.

For the LIP Case, the area under the curve would increase by 60 Tank Years (= 20 Tank Years / year x 3 years). This increase in area represents a 25% increase in the number of Tank Years from the Baseline Case.

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The NIP Case would increase the area under the curve by 100 Tank Years (= 20 Tank Years / year x 5 years). This increase in area represents a 42% increase in the number of Tank Years from the Baseline Case.

The result of the comparison of old style Tank Years between the cases clearly shows that the lowest risk case is the Baseline Case. Both the LIP and the NIP Case result in significant increases in the number of years that the old style tanks remain in service which represents an increase in risk of leaks associated with tank failure. Therefore, the Baseline Case provides the lowest risk in terms of old style Tank Years.

3.3 Worker Dose

Table 6 presents the 2005 annual worker doses for F and H Tank Farms, SDF, DDA, and DWPF based on SRS ALARA reports. These doses will be held constant for the purposes of this analysis. After initiation of SPF/SDF operations to dispose of DDA material, the estimated total worker exposure at SPF will be 1.5 rem/yr (Ref. 6). Since this dose is a result of the relatively high Cs-137 concentration in the DDA waste, for the purposes of this comparison, the total SPF/SDF worker dose during ARP/MCU and SWPF processing periods will be assumed to be 0 rem/year. The additional worker dose resulting from the DDA process is estimated to be 5 rem/year and results from additional maintenance and operations exposure in the Tank Farms during salt removal equipment installation and maintenance. The ARP/MCU dose without DDA is estimated to be 10 rem/year. This includes 5 rem/year for installation and maintenance of salt removal equipment and 5 rem/year for operation and maintenance of the MCU.

Table 6: Current SRS Worker Exposures

	Current SRS Facility				
	Tank Farm		DWPF (rem/yr)	SPF/SDF* (rem/yr)	DDA (rem/yr)
	F (rem/yr)	H (rem/yr)			
Worker Dose	15	22	6.5	0.035	5

Note*: Current dose for SPF/SDF is 0.035, during interim processing value will increase to 1.5 rem/year. After interim processing, value is assumed to be zero (Ref. 6).

Since SWPF operations will be similar to DWPF operations, the dose to workers at SWPF is estimated to be double the 6.5 rem/year to account for the elevated inventory of Cs-137. Lastly, the additional exposure to workers in the Tank Farm resulting from SWPF operation will be the same as the additional exposure for DDA (5 rem/year) since operation of SWPF will require installation and maintenance of salt removal equipment. Therefore, the total projected worker exposure resulting from SWPF operations is 18 rem/year ($\approx$ [6.5 rem/year x 2] + 5 rem/year from Tank Farms).

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Tables 7, 8, and 9 display the worker doses for each of the three cases. The total worker dose is increased in Cases #2 and #3 due to the extension of operations for the facilities by at least three years and at least five years respectively. The total worker exposures are summarized in Table 10.

Table 7: Case #1 = Worker Dose for Baseline Case

Year	DDA (rem)	ARP (rem)	SWPF (rem)	DWPF (rem)	HTF (rem)	FTF (rem)	SPF/SDF (rem)	Total (rem)
2005	5	0	0	6.5	22	15	1.5	50
2006	5	0	0	6.5	22	15	1.5	50
2007	0	10	0	6.5	22	15	1.5	55
2008	0	10	0	6.5	22	15	1.5	55
2009	0	0	18	6.5	22	15	0	62
2010	0	0	18	6.5	22	15	0	62
2011	0	0	18	6.5	22	15	0	62
2012	0	0	18	6.5	22	15	0	62
2013	0	0	18	6.5	22	15	0	62
2014	0	0	18	6.5	22	15	0	62
2015	0	0	18	6.5	22	15	0	62
2016	0	0	18	6.5	22	15	0	62
2017	0	0	18	6.5	22	15	0	62
2018	0	0	18	6.5	22	15	0	62
2019	0	0	18	6.5	22	15	0	62
Total	10	20	200	98	330	230	6	890

Note: Numbers may not sum to actual totals because of rounding.

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Table 8: Case #2 = Worker Dose for LIP Case

Year	DDA (rem)	ARP (rem)	SWPF (rem)	DWPF (rem)	HTF (rem)	FTF (rem)	SPF/SDF (rem)	Total (rem)
2005	0	0	0	6.5	22	15	0	44
2006	0	0	0	6.5	22	15	0	44
2007	0	10	0	6.5	22	15	1.5	55
2008	0	10	0	6.5	22	15	1.5	55
2009	0	0	18	6.5	22	15	0	62
2010	0	0	18	6.5	22	15	0	62
2011	0	0	18	6.5	22	15	0	62
2012	0	0	18	6.5	22	15	0	62
2013	0	0	18	6.5	22	15	0	62
2014	0	0	18	6.5	22	15	0	62
2015	0	0	18	6.5	22	15	0	62
2016	0	0	18	6.5	22	15	0	62
2017	0	0	18	6.5	22	15	0	62
2018	0	0	18	6.5	22	15	0	62
2019	0	0	18	6.5	22	15	0	62
2020	0	0	18	6.5	22	15	0	62
2021	0	0	18	6.5	22	15	0	62
2022	0	0	18	6.5	22	15	0	62
Total	0	20	250	120	400	270	3	1100

Note: Numbers may not sum to actual totals because of rounding

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Table 9: Case #3 = Worker Dose for NIP Case

Year	DDA (rem)	ARP (rem)	SWPF (rem)	DWPF (rem)	HTF (rem)	FTF (rem)	SPF/SDF (rem)	Total (rem)
2005	0	0	0	6.5	22	15	0	44
2006	0	0	0	6.5	22	15	0	44
2007	0	0	0	6.5	22	15	0	44
2008	0	0	0	6.5	22	15	0	44
2009	0	0	18	6.5	22	15	0	62
2010	0	0	18	6.5	22	15	0	62
2011	0	0	18	6.5	22	15	0	62
2012	0	0	18	6.5	22	15	0	62
2013	0	0	18	6.5	22	15	0	62
2014	0	0	18	6.5	22	15	0	62
2015	0	0	18	6.5	22	15	0	62
2016	0	0	18	6.5	22	15	0	62
2017	0	0	18	6.5	22	15	0	62
2018	0	0	18	6.5	22	15	0	62
2019	0	0	18	6.5	22	15	0	62
2020	0	0	18	6.5	22	15	0	62
2021	0	0	18	6.5	22	15	0	62
2022	0	0	18	6.5	22	15	0	62
2023	0	0	18	6.5	22	15	0	62
2024	0	0	18	6.5	22	15	0	62
Total	0	0	290	130	440	300	0	1200

Note: Numbers may not sum to actual totals because of rounding

Table 10: Worker Exposure

Case	Cumulative Worker Exposure (rem)	Percent Difference From Baseline Case	Difference From Baseline Case (rem)
Baseline Case	890	N/A	N/A
LIP Case	1100	24%	210
NIP Case	1200	35%	310

Note: Numbers may not sum to actual totals because of rounding.

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3.4 Population Dose

Three sources of dose to the population were considered for this evaluation. The first is the projected dose that a member of the public receives from on-going operations at the Savannah River Site. The second dose is the projected future dose that a hypothetical intruder could receive from the SDF under certain specified conditions. Third, the All Pathways dose that a member of the public could receive in the future under certain specified scenarios.

3.4.1 Public Dose

Historically, the dose to a member of the public resulting from all operations at SRS has been estimated to be 0.19 mrem/year (Ref. 6). This is the projected dose for the maximally exposed member of the public considering waterborne and airborne pathways of exposure. The operation of DDA, ARP/MCU, and SWPF is not expected to significantly impact this dose. As a result, any case that extends the life-cycle of SRS, increases the cumulative dose from operations. Since this dose (0.19 mrem) is small in comparison to the other doses discussed, the difference between cases in dose to the public from this source is negligible and will not be considered further.

3.4.2 Intruder Dose

For the Baseline Case, Intruder dose from material disposal at the SDF is at its maximum at 100 years (Ref. 6). The projected SDF Cs-137 inventory of 1.35 million curies contributes greater than 99% of the total intruder dose. Starting at the end of institutional control (calendar year 2105), the dose for an intruder living directly on top of Vault 4 would be 22.5 mrem/year (Ref. 6). Table 11 provides the projected intruder dose in 20 year increments for the period of 100 to 300 years after institutional control ends. It can be seen that by the year 2305, the individual intruder dose is projected to be only 0.2 mrem/yr. Also shown in Table 11, is the cumulative dose to an individual intruder living directly on Vault 4. It can be seen that for an intruder living on Vault 4 for the first 100 years after the end of institutional control, the cumulative dose is projected to be ~890 mrem (0.89 rem). During the next 100 year period (2205 – 2305), the cumulative projected dose to an individual intruder drops to ~90 mrem. For convenience, Table 11 also shows the yearly and cumulative doses for a family of four intruders living on Vault 4 for the 200 year period in 20 year increments. For the purposes of this comparison, the Intruder doses for Case #2 and Case #3 are assumed to be 0 mrem/year as a result of the removal of Cs-137 through the MCU and SWPF processes. The individual intruder annual dose and the total cumulative dose to 4 intruders from Table 11 is displayed graphically in Figures 4 and 5. As shown in Figure 4 and in Table 11, the total cumulative projected dose to 4 intruders living on Vault 4 is less than 4 rem (Ref. 6).

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Table 12 below provides a comparison of the 100 year cumulative individual intruder doses for all three cases. The cumulative dose to an individual for the 100 year period also includes the ~ 360 mrem per year an individual would receive from natural and man-made background radiation sources. It can be seen that the total projected 100 year dose to an individual varies by only 2.5% between the cases.

Table 11: Baseline Case Cumulative Dose to 1 and 4 Intruders
(Starting calendar year 2105)

| Calendar Year | Individual | | Four-Person | |
	Dose (mrem/yr)	Cumulative Dose (mrem)	Dose (mrem/yr)	Cumulative Dose (mrem)
2105	22.5	22.5	90.0	90.0
2125	14.2	379	56.8	1520
2145	9.0	604	35.8	2420
2165	5.7	746	22.6	2980
2185	3.6	835	14.3	3340
2205	2.2	892	9.0	3570
2225	1.4	927	5.7	3710
2245	0.9	950	3.6	3800
2265	0.6	964	2.3	3860
2285	0.4	973	1.4	3890
2305	0.2	979	0.9	3910

Table 12: Comparison of Individual Dose Projected at 100 Years

Case	Cumulative Dose from Case[2] (mrem)	Cumulative Dose From Background[3] (mrem)	Total Dose (Case + Background) (rem)
Baseline Case	890	36000	36.9
LIP Case	0	36000	36
NIP Case	0	36000	36

(1) Numbers may not sum to actual totals because of rounding.
(2) For an individual living on Vault 4 for 100 years (2105 – 2205)
(3) Assumes 360 mrem/year for 100 years

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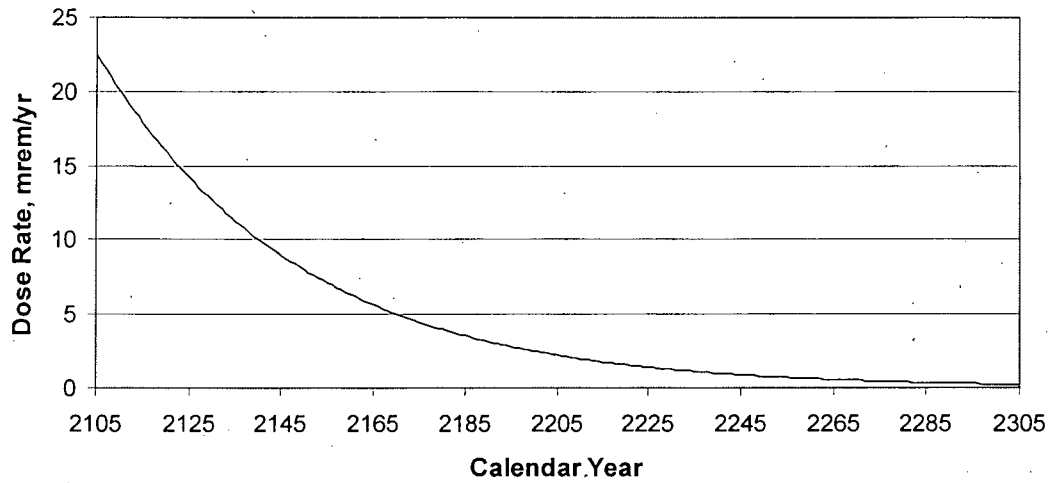
**Figure 4: Annual Projected Individual Intruder Dose
Baseline Case (2105 – 2305)**
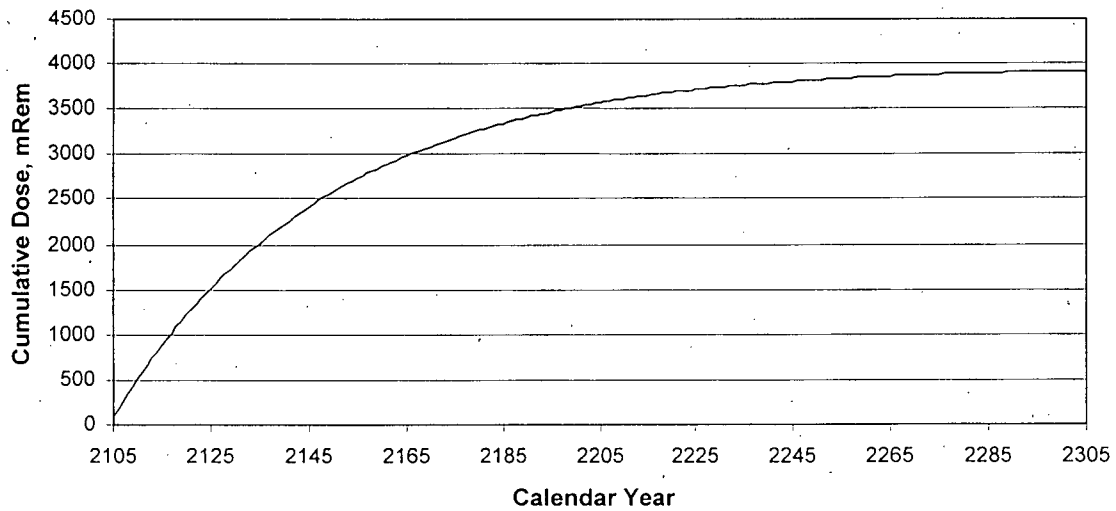
**Figure 5: Cumulative Dose to Four Intruders
Baseline Case (2105 – 2305)**

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3.4.3 All Pathways Dose

The projected all pathways dose from material disposal at the SDF of 2.3 mrem/year is driven by Se-79 and I-129 (Ref. 6). All other nuclides contribute less than 0.01 % of the all pathways dose. The only difference in SDF inventory for Se-79 and I-129 is due to the lower filtration efficiency of the DDA process relative to the ARP/MCU and SWPF processes. For Se-79 and I-129, less than 0.1% of the total SDF inventory is from insoluble solids carry over (based on Ref. 4). Therefore, the difference in Se-79 and I-129 inventories for all three cases is insignificant, and as a result, the differences in all pathways doses for all three cases are negligible. The All Pathways dose of each of the three Cases is shown in Table 13.

Table 13: All Pathways Dose Comparison

Case	All Pathways Dose (mrem/yr)
Baseline Case	2.3
LIP Case	2.3
NIP Case	2.3

It can be seen in Tables 12 and 13 that there is an inconsequential difference in dose to the public between Cases #1, #2, and #3 and therefore, dose to the public should not be used as a criterion for measuring the relative risk between the Baseline Case, the LIP Case, and the NIP Case.

4 FINANCIAL COST COMPARISON

4.1 Case Comparison

The financial cost associated with each of the cases will be compared to determine the most cost effective case. In addition to the three cases described above, additional discussion will be provided comparing costs for construction of additional compliant tank space as compared to the Baseline Case.

There are approximately 85 Mgal of material to be dispositioned via DDA, ARP/MCU, and SWPF at the disposition rates presented in Tables 2, 3, and 4 for the three cases being considered. The assumptions used in the cost comparison are as follows:

For Case #1, DDA and ARP/MCU have made five tanks available for SWPF feed preparation at startup, so full rate is achieved quickly. For Case #2, ARP/MCU has cleared an additional tank of salt, so two tanks are available for SWPF feed preparation at startup. For Case #3, only one tank is available to feed SWPF at startup. Another tank must be cleared of salt and available for feed preparation before the SWPF processing rate increases.

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Because available tank space is scarce in the Tank Farms, the emptied tank space from DDA & ARP/MCU operations is critical to prepare adequate feed for SWPF startup.

Without DDA & ARP/MCU, SWPF has to empty its own tank space to prepare feed. Due to a lack of available feed capacity, SWPF must ramp up slowly to reach its maximum processing rate of 7 Mgal per year.

For Cases #2 and #3, it was assumed that the DWPF would be slowed down, not shutdown. An analysis of the cost of shutting down DWPF showed that a shutdown and restart, which would require replacing the melter and a number of other activities, would be prohibitively expensive. Slowing down the DWPF is the most cost effective strategy (References 7 and 10) for the time period addressed by the three cases.

The annual costs of operating each of the facilities are estimated in Table 14.

Table 14: Approximate Annual Financial Costs
(Non-escalated 2004$)

Facility	Annual Operating Cost ($ M)	Bases
H Tank Farm	125	Actual 2004 cost
F Tank Farm	54	Actual 2004 cost
ETP	15	Actual 2004 cost
DWPF	110	Actual 2004 cost
ARP/MCU	16	Non-escalated. 2007 first year of operating
SWPF	19 (2009) 42 (2010-based on first year full operation)	Non-escalated. 2009 first year of operating. Cost approximated based on processing ratio. For each full year of operation, 100% thereafter.

Note: Summed numbers will not reflect approximate annual total because of variation simplifications.

Estimated capital and D&D Costs for ARP/MCU after 2006 are shown in the table below. Not included capital are costs for 2004 & 2005 which are assumed to be sunk costs. No costs are shown for DDA. Because DDA uses mostly Tank Farm equipment, the costs are difficult to extract from the Tank Farm budget. Also, most of the costs are captured as part of the Tank Farm costs. For example, the transfers required for DDA are already part of the Tank Farm operation. Also, the tanks and equipment must undergo D&D whether DDA operates or not. Therefore, the costs for DDA are small and would not affect the budget comparisons shown in this analysis. Refer to Table 15.

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Table 15: Additional Capital and D&D Costs for ARP/MCU

Fiscal Year	Capital MCU ($M)	Capital ARP ($M)	D&D MCU ($M)	D&D ARP ($M)
2006	38	16		
2007	13			
2008				
2009				
2010			4	1
Totals	51	16	4	1
Grand Total	72			

Based on the above information, a comparison of costs of the three cases is shown below in Table 16.

Table 16: Approximate Life Cycle Costs of SRS Liquid Waste Facilities (Non-escalated, 2004$)

Operating Costs (except as noted)	Case #1 ($M)	Case #2 ($M)	Case #3 ($M)
H Tank Farm	1,900	2,300	2,500
F Tank Farm	550	710	820
ETP	230	270	300
DWPF	1,700	2,000	2,200
Saltstone –Assumed Negligible for all cases	NA[1]	NA[1]	NA[1]
H Canyon	NA[2]	NA[2]	NA[2]
DDA	NA[3]	NA[3]	NA[3]
ARP/MCU	32	32	0
ARP/MCU (Capitol Construction & D&D)	72	72	0
SWPF	440	500	500
Totals	4,900	5,800	6,300
Additional Cost from Baseline	Baseline	Δ 1,000	Δ 1,500
Escalated Additional Cost from Baseline	Baseline	Δ 1,600	Δ 2,600

Note: Numbers may not sum to actual totals because of rounding.

(1) Saltstone operating costs are assumed to be low enough to be neglected. The most significant cost at Saltstone is constructing new vaults to receive the grout. Vault construction will not affect the comparison because the same amount of salt solution is processed for all cases, so the number of vaults is the same for each case.

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(2) Interim Processing and Tank Farm space assumed not to directly impact H Canyon Costs.

(3) DDA costs are assumed to be with F and H Tank Farm Costs.

Additional Notes:

(1) Operating costs for Tank Farms, ETP, and DWPF are based on actual $2004 dollars starting in 2005 and continuing through the end of their useful life.

(2) It is assumed that H Tank Farm, ETP, and DWPF must continue to operate approximately as long as SWPF. An actual sequence for shutting down these facilities has not been determined, but the sequence would be the same for all cases and will not affect the comparison.

(3) No D&D cost is considered or Tank Farms, ETP, DWPF, or SWPF. Again, the non-escalated amount should be the same for all cases and will not affect the comparison.

(4) No Capital cost is considered for all facilities other than ARP/MCU. Again, the non-escalated amount should be approximately the same for all cases and will not affect the comparison.

The following conclusions can be drawn from a review of the cost comparison data provided above:

Case #1 (Baseline) is the most cost effective case evaluated. This is because it has the shortest lifecycle and relatively small capital costs associated with ARP/MCU. Since the annual operating costs dominate the cost evaluation, this is the least expensive case.

Case #2 (LIP) is more expensive than the Baseline Case because it extends the life of SWPF by 3+ years, which requires similar extensions to the lives of DWPF, the Tank Farms, ETP, and Saltstone. The extra expense of operating these facilities for 3 additional years is approximately $1B (2004$) or $1.6B escalated.

Case #3 (NIP) is the most expensive of the three cases because it extends the life of SWPF and the associated facilities by 5+ years. The extra expense of operating these facilities for an additional 5 years (minus the cost saved from not completing construction and subsequent operation of ARP/MCU) is approximately $1.5B (2004$) or $2.6B escalated.

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4.2 Cost Of Building New Tanks

The possibility of building additional tanks was also considered but found to be not economically attractive. A recent study at Hanford indicated that building new waste tanks of 1.2 Mgal capacity would cost approximately $75M (2001) per tank if the tanks were built in sets of four (Ref. 8). Four tanks would be the correct capacity needed to increase the capability to feed SWPF comparable to the same degree as operating DDA and ARP/MCU. Based on the Hanford study, building four tanks would cost about $300M (2001), compared to an estimated additional $72M for finishing the building and eventually decommissioning ARP/MCU. Also, note that new tanks would require additional operating cost and decommissioning cost, which would increase the difference between the two possibilities. A second consideration is that it might not be possible to have new tanks ready to use by 2009. The new tanks would need to be funded, designed, constructed, permitted, and put into service by 2009. Usually, just the funding phase of a project of this size requires several years.

Construction of new HLW storage tanks: In 2001, the cost of new tank construction at Hanford was estimated to be $75M assuming that at least 4 tanks were built. The breakdown of the costs supporting this total is shown in Table 17. In order to support SWPF start-up at full capacity, 4 new tanks would need to be constructed for staging dissolved salt solution. Therefore, a total of $300M (2001) would be required to construct adequate tank space. Since the cost of new tank construction is significantly more than twice the lifecycle cost for ARP/MCU facility, this was not considered to be cost effective. Hanford also estimated an overall schedule of approximately 7 years, the details of which are shown in Figure 6. This schedule is not within the timeframe required to support SWPF start-up assumptions. One further note: The construction of new tank space does not support DOE's and the State of South Carolina's overall objective of risk reduction.

Table 17: Cost Estimate for Constructing New Tanks
(Based on Each One of at Least Four Tanks, Ref. 8)
(2001)

Activity Description	Cost ($K)
Obtain Permitting &Regulatory Approval	1,000
Design	7,000
Procurement and Construction	66,000
Start-up and Testing	1,000
Total	**$75,000**

Note: Numbers may not sum to actual totals because of rounding.

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ACTIVITY	YEARS							
	1	2	3	4	5	6	7	8
Acquire Funding	░░░							
Permit & Regulatory Approval			░░░░░░░░░░░░░░░░░░░░░░░░					
Design			░░░░░					
Procurement & Construction				░░░░░░░░░░░░░░░░				
Startup & Testing			░░░░░░░░░░░░░░░░░░░░░░░░░░					

Figure 6: Schedule for Constructing of New Tanks

5 QUALITATIVE EVALUATION OF ADDITIONAL FACTORS

There are several other factors that are discussed due to their potential for impact on the risk and cost associated with the cases being evaluated.

5.1 Additional Transfers

The first factor that is discussed is that if the Baseline Case is not pursued, tank space available for preparing salt batches and sludge batches in 2009 when SWPF is scheduled to begin processing salt waste will be significantly reduced. The limited tank space available will result in a significant increase in the number of individual transfers required to prepare a salt batch or sludge batch for processing. These additional transfers are required because numerous small volume transfers would replace single large volume transfers due to restrictions in tank space available. These additional transfers incur increased operating risk due to the probability of leaks and inadvertent transfers associated with each transfer evolution. This risk is not quantified in terms of dose or other comparable metric, but is magnified when the compounding considerations of the aging infrastructure used for the transfers and the delay in operation completion for the HLW system are taken into account. The potential consequences of this increase in operating risk include increased leak incidence, operational delays due to compromise of salt batch or sludge batch characterization from receipt of an inadvertent transfer, increased maintenance and worker exposure directly from transfers as well as from maintenance of transfer infrastructure, etc. Directly related to the increase in risk associated with the increased number of transfers is the increase in the cost associated making the transfers (procedures, personnel monitoring the transfer evolution, planning, etc.) and the indirect cost of maintaining the transfer infrastructure (pumps, leak detection systems, electrical distribution, ventilation, etc.).

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5.2 Impact of Delays

Another area that is discussed qualitatively in terms of comparison between the cases is the impact of delays in facility start-up on risk and cost. The cases described above demonstrate that delays in beginning processes that are capable of dispositioning salt waste so that compliant tank space is available to support salt batching and sludge batching result in: 1) the extension of facility lifecycle with the associated increase in exposure to the public and to the worker, 2) the increase in risk by the failure to permanently disposition waste and leaving the waste in the aging HLW system including old style, non-compliant waste tanks for longer periods of time and 3) incurring the additional cost of maintaining the HLW system for the longer durations, etc. Any delays in facility start-up will adversely impact the HLW system operation completion dates. Costs and risks have been presented earlier in this document in terms of annual operating costs and risk in terms of annual exposure and material at risk. While this discussion does not attempt to quantify delays in start-up in terms of additional cost or risk, it has been clearly demonstrated earlier in this document that the cost and risk for the HLW system is minimized by earliest mission completion. It should be noted that this extension in facility operation is likely greater than a day for day match with a delay in facility start-up. Delays in facility start-up will result in less tank space available for salt batch and sludge batch preparation. It would take years of operation at reduced rates to recover the "lost" tank space. In the cases analyzed, it took 4+ years after SWPF start-up for the LIP Case and 7+ years for the NIP Case for SWPF to achieve forecast processing rates. Attaining these forecast processing rates was limited by the availability of compliant tank space to prepare salt batches to feed SWPF at the baseline rate of seven million gallons of salt waste solution per year.

5.3 Aging Infrastructure

A critical element to the discussion on material at risk (expressed in terms of Tank Years and Curie Years) is the consequence of materials leaked from the aging noncompliant tanks and related infrastructure (pipes, valves, secondary containment structures, etc.). While the sections of the document that address worker dose and life cycle costs clearly show the expected increases to exposure and cost associated with lifecycle extensions, they cannot accurately quantify the risk associated with the continued use of the aging tank farms during the period of lifecycle extension.

Clearly the risk of leaks increases proportionally with the increase in facility lifecycle associated directly with the increase in years of operation, e.g. a ten percent increase in lifecycle is a ten percent increase in risk of an incident. The probability of leaks also increases as a result of the fact that tanks and transfer infrastructure continue to degrade due to the corrosive environment and radiation associated with the storage and processing of HLW. No attempt is made to quantify the probability of failure of the degrading infrastructure, but the increased probability is clear. Likewise, no attempt is made to quantify the impact of the contamination to the environment or to quantify the worker/public dose associated with such a leak.

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The quantity and type of material, the location of the leak, duration of the leak, proximity of workers, proximity of transport media, environmental conditions, etc. all effect the impact of such an occurrence. While SRS has robust systems for preventing and/or mitigating such an occurrence through tank inspections, corrosion control programs, solution chemistry management, secondary containment, leak detection systems, etc., the probability of occurrence of a leak increases with facility lifecycle extensions. The quantification of Tank Years and Curie Years is directly related to this increase in risk and demonstrates the exigencies associated with implementation of salt waste stabilization utilizing the Interim Salt Processing Strategy described in the Salt WD.

6 CONCLUSION

When the Baseline, the LIP and the NIP Cases were evaluated, the following fundamental differences were noted between the cases:

Completion of Operations: The Baseline Case completed HLW system operations in 2019. The salt waste dispositioned to SPF/SDF by DDA and ARP/MCU created compliant tank space in the HLW Tank Farms that permitted DWPF operations to continue without interruption and permitted SWPF operations to commence processing at forecast production rates. The Limited Interim Processing Case required 3+ years longer to complete HLW system operations than the Baseline Case. The delay in mission completion resulted from the reduced processing rates through SWPF and DWPF caused by the limited compliant tank space available to prepare the salt and sludge waste streams for processing during initial years of SWPF operation. DWPF production rates are impacted because of the limited compliant tank space prevents sludge washing which is required prior to processing sludge waste at DWPF. The NIP Case required approximately 5+ years longer to complete HLW system operations than the Baseline Case. The delay in mission completion resulted from the reduced SWPF and DWPF processing rates caused by the limited compliant tank space available to prepare the salt and sludge waste feed streams for processing during initial years of SWPF operation.

Risk: The dose (exposure) associated with each of the three cases were compared as well as the material/facilities at risk. Dose was further broken down in terms of dose to the facility worker, dose to the public from both ongoing operations and from material dispositioned to the SPF/SDF, and dose to the inadvertent intruder from the SPF/SDF. In order to appropriately characterize the material at risk from ongoing operations, the differences between the cases in terms of old style tank closure dates and Tank Farm waste disposition rates were also evaluated. The evaluation is shown in Table 18:

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Table 18: Summary of Dose, Tank Years and Curie Year Impacts

Case Evaluated	Worker Dose [1] (rem)	Current Public Dose [2] (mrem/yr)	SDF Intruder Dose [3] (rem)	SDF All Pathways Dose (mrem/yr)	Old Style Tank Years [4]	Tank Farm Curie Years [5]
Baseline	890	0.19	36.9	2.3	240	3.7E+09
LIP Case (change from baseline)	1100 (+24 %)	0.19 (0 %)	36 (- 2.5 %)	2.3 (0 %)	300 (+25%)	4.7E+09 (+25%)
NIP Case (change from baseline)	1200 (+35 %)	0.19 (0 %)	36 (- 2.5 %)	2.3 (0 %)	340 (+42%)	5.3E+09 (+42%)

(1) Integrated dose to HLW workers
(2) Dose to public from current SRS operations
(3) 100 year integrated dose (2105 – 2205); includes 360 mrem/yr contribution from background radiation
(4) Total number of years old style tanks are in service, where 20 tanks in service for 2 years = 40 Tank Years
(5) Total number of years a curie is in the Tank Farm, where 30 MCi in the Tank Farm for three years = 90M Curie Years

It can be seen from Table 18 that the Baseline Case results in significantly lower worker dose and significantly shorter time that curies remain in the old style tanks. Intruder doses are higher for this case, but the difference is not significant when compared to exposure from background sources of radiation. The difference in public doses from sources associated with salt processing are negligible between the cases. The LIP and NIP cases show significant increases from the baseline case for worker exposure and time that radioactive material remains in the Tank Farm. Thus, using the Interim Salt Processing plan provides a significant decrease in perceived risk from a worker exposure perspective as well as from potential tank failures and subsequent potential releases to the environment.

Financial Cost: The Baseline Case was financially the most cost effective. The primary reason that the Baseline Case was the most cost effective was the difference in lifecycle costs associated with extending the HLW system (Tank Farms, DWPF, SWPF, SPF/SDF, etc.) operations by 3+ years for the LIP Case and 5+ years for the NIP Case. This resulted in an additional escalated cost for operation of approximately $1B and $1.5B (2004$) for Case #2 and Case #3, respectively (or $1.6B and $2.6B escalated). Since the sunk costs for ARP/MCU construction are high relative to the total project cost and since the life cycle costs for the HLW system are much higher than the project construction, and D&D (decontamination and decommissioning) costs, the annual operating costs dominate the cost comparison and demonstrate that the case that results in the shortest life cycle will have the lowest financial cost.

Sensitivity to facility start-up delays: Since the primary influence on cost and risk associated with these cases is life cycle, delays in facility start-up will have a significant impact on both risk and cost. The evaluation assumes that the dates projected for facility start-up will be achieved and that throughput rates will be as forecasted. Delays in facility start-up and reductions in

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throughput rates would extend the duration of facility operation with associated increases in cost and a decrease in benefits (risk reductions).

DWPF slowdown versus DWPF shutdown: Other aspects of the facility operations that were reviewed as a part of this evaluation include consideration from a financial perspective of slowing down DWPF rather than shutting down to accommodate processing when feed streams (sludge batches) to DWPF are unavailable. The evaluation shows that slowing down DWPF operations is preferred over shutdown from a cost perspective and cost comparisons utilized this basis when DWPF operation was evaluated.

Construction of new HLW storage tanks: Since the cost of new tank construction was significantly more than the lifecycle cost for ARP/MCU facility, this was not considered to be cost effective. In addition, the feasibility of designing, constructing and starting up new tanks within the timeframe required to keep from impacting SWPF start-up assumptions is unlikely.

Taken as a whole, the above fundamental differences in the cases evaluated demonstrate that the Baseline Case is the most cost effective option and provides the lowest worker dose. Inadvertent intruder doses are marginally higher with the Baseline Case, but this dose is not significant when compared to exposure from natural sources of radiation. The Baseline Case also reduces radioactive material at risk the most quickly because it facilitates stabilization of radioactive material in the Tank Farm more quickly than in the other cases, as well as permitting closure of old style tanks per the Federal Facility Agreement (FFA) schedule. For these reasons, the Baseline Case provides the greatest overall benefit at the lowest cost.

Cost & Benefit Evaluation for
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7 REFERENCES

1) Draft Section 3116 Determination, Salt Waste Disposal Savannah River Site, DOE-WD-2005-001, February 28, 2005.

2) NRC Letter, Request for Additional Information on the Draft Section 3116 Determination for Salt Waste Disposal at the Savannah River Site, Scott C. Flanders to Mark A. Gilbertson, May 26, 2005.

3) Tank Radionuclide Inventories, CBU-PIT-2005-00138, Rev. 0, June 14, 2005.

4) Radionuclide Concentration in Saltstone, CBU-PIT-2005-00013, Rev. 3, June, 2005.

5) P.D. D'Entremont, M.J. Mahoney, "Interim Salt Processing Strategy Planning Baseline," CBU-PED-2004-00027, Rev. 0.

6) R. E. Sheppard, E.T. Ketusky, "Dose/Cost Evaluation for 3 Saltstone Cases," X-ESR-Z-0001 6/21/05.

7) F. Wise to S. Robertson, "NRC Request", CBU-PIT-2005-00159, Rev. 0, June 27, 2005.

8) V. C. Boyles, et al. "Tank Space Options, RPP-7702," CH2M Hill Hanford Group, Rev. 0., April 4, 2001.

9) Federal Facility Agreement for the Savannah River Site, Administrative Document Number 89-05-FF, WSRC-OS-94-42, Effective Date: August 16, 1993.

10) J. E. Owen, "Impacts of Potential Shutdown/Slowdown of DWPF," CBU-PIT-2005-00154, Rev. 0, June 23, 2005.

11) NCRP Report No. 93, Ionizing Radiation Exposure of the Population of the United States National Council on Radiation Protection and Measurements, 1987.

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DOE-WD-2005-001
February 28, 2005

Draft Section 3116 Determination

Salt Waste Disposal

Savannah River Site

certain waste from reprocessing is not HLW and that it may instead be disposed of as low-level waste (LLW) if it meets the criteria set forth in that Section: that it does not require disposal in a deep geologic repository, that it has had highly radioactive radionuclides removed to the maximum extent practical, that it meets concentration limits and/or dose-based performance objectives for near-surface disposal of radioactive waste, and that it will be disposed pursuant to a State-issued permit or State-approved closure plan. In this document, the Secretary proposes to determine that the treated, solidified low-activity salt waste from the tanks will meet all of these criteria. Accordingly, this material may be determined not to be HLW and may be disposed of instead as LLW in SDF.

Although this draft 3116 Determination addresses only the disposal of solidified low-activity salt waste streams, it is important to understand that successful and timely salt waste removal and disposal is integral to DOE's efforts to proceed with all aspects of tank cleanup and closure, extending well beyond disposal of the solidified low-activity salt waste streams themselves. This is not only for the obvious reason that the salt waste must be removed and treated before the tanks may be closed. Less obviously, but more importantly, the disposal pursuant to this draft 3116 Determination will enable DOE to continue without interruption to remove and stabilize the high-activity fraction of the waste. This is because DOE uses the tanks to prepare the high-activity waste so that it may be fed into DWPF. The issue is that the salt waste is filling up tank space needed to allow this preparation activity to continue. Thus, removal and disposal of low-activity salt waste is critical in order to relieve this tank space shortage and assure that vitrification of the high-activity fraction will be able to continue uninterrupted.

The Department of Energy is tailoring the waste treatment for the salt waste based on the radiological properties of those wastes as shown by tank waste characterization data. DOE is separating the salt waste to segregate the low-activity fraction at SRS using a two-phase, three-part process. The first phase will involve two parts to treat the lower activity salt waste: a) beginning in 2005, processing of a minimal amount of the lowest activity salt waste through a process involving deliquification, dissolution, and adjustment (DDA) of the waste; and b) beginning in approximately 2007, processing of a minimal amount of additional salt waste with slightly higher activity levels using an Actinide Removal Process (ARP) and a Modular Caustic Side Solvent Extraction (CSSX) Unit (MCU), along with deliquification and dissolution of saltcake. The second, and longer term phase, beginning in approximately 2009, involves the separation and processing of the remaining (and by far the majority) of the salt waste using a Salt Waste Processing Facility (SWPF) (augmented as necessary by ARP). This second phase will begin as soon as SWPF is constructed, permitted by the State of South Carolina, and operational [3,4].

This document refers to the first phase as Interim Salt Processing because it will be used to initiate treatment and disposal of solidified low-activity salt waste streams using the technologies available during the interim period between 2005 and approximately 2009 when DOE anticipates that SWPF will become operational. As explained in more detail later in this draft 3116 Determination, DOE believes it should proceed with this interim approach because doing so will enable DOE to continue uninterrupted use of DWPF as well as to use SWPF at higher capacity as soon as it comes on line. This will allow DOE to complete cleanup and closure of the tanks years

IONIZING RADIATION EXPOSURE OF THE POPULATION OF THE UNITED STATES

|N|C|R|P|

TABLE 2.3—*Estimated average annual dose equivalents to various tissues for a member of the population in the United States from various sources of natural background radiation (μSv)*[a]

Source	Bronchial epithelium	Other soft tissues	Bone surfaces	Bone marrow
Cosmic	270	270	270	270
Cosmogenic	10	10	10	30
Terrestrial	280	280	280	280
Inhaled[b]	24,000	—	—	—
In the body[c]	360	360	1,100	500
Rounded total	25,000	900	1,700	1,100

[a] 1 μSv = 0.1 mrem.

[b] Doses to other tissues from inhaled radionuclides are included under "In the Body."

[c] This includes all radionuclides in the body (see Table 2.2) excluding the cosmogenic component shown separately in this Table.

of radon. The differences in the dose equivalent rate reported here and those in the earlier NCRP Report (NCRP, 1975) are quite marked. The major change is in the annual dose estimate to the bronchial epithelium from inhaled radon decay products, which increased from 4.5 to 24 mSv (450 to 2,400 mrem). The increases in the estimated dose equivalent from internal emitters were due to the higher quality factor for α radiation, to data showing higher concentrations of the ^{210}Pb-^{210}Po pair in bone, to a higher estimate for the tissue dose from radon decaying in the body, and to higher radon levels indoors as compared to outdoors. The dose equivalent values for cosmic radiation, cosmogenic radionuclides and terrestrial gamma radiation are very little changed from the previous estimates.

The annual dose equivalents have been converted to effective dose equivalent using the weighting factors (w_T) of the ICRP (ICRP, 1977, 1981). The individual contributions are shown (in μSv) in Table 2.4, and their sum is a total of H_E of 3.0 mSv (300 mrem). This estimate of average effective dose equivalent is considered to apply to both sexes and all ages.

In the case of irradiation of the entire public (such as by natural background), the GSD is equal to the gonadal dose equivalent. The value of the gonadal dose equivalent is the same as that for other soft tissues shown in Table 2.3, viz 0.9 mSv/y (90 mrem/y).

We assume that the average effective dose equivalents given in Table 2.4 apply to all members of the U.S. population (230,000,000) and therefore the collective effective dose equivalent from natural sources is 69 × 10^4 person-Sv (69 million person-rem).

FEDERAL FACILITY AGREEMENT

FOR THE

SAVANNAH RIVER SITE

Administrative Document Number 89-05-FF

Effective Date: August 16, 1993

include installed leak detection equipment and procedures, photographic or visual inspections that show liquid or accumulating dried waste, or sampling and analysis.

     E.   Removal of Waste Tank System(s) From Service

     1.   The DOE shall submit to EPA and SCDHEC for review and approval, a plan(s) and schedule(s) for the removal from service of waste tank system(s)/component(s) that do not meet the secondary containment standards of Subsection C herein, or that leak or have leaked. This plan(s) and schedule(s) shall be submitted and/or amended by DOE as follows: (a) within ninety (90) days of the effective date of this Agreement for tank system(s) or secondary containment system(s) known to leak or have leaked prior to the effective date of this Agreement that will not be repaired in accordance with Subsection D herein; (b) within ninety (90) days of discovery of leaks from tank system(s) or secondary containment system(s) not known before the effective date of this Agreement that will not be repaired in accordance with Subsection D herein; and (c) within ninety (90) days of receipt of written notification from SCDHEC of the determination that any waste tank system(s)/component(s) do not meet the secondary containment standards of Subsection C herein. After approval of the schedule by EPA and SCDHEC, DOE shall remove the tanks from service according to the approved plan(s) and schedule(s). In lieu of the requirements of E.1(c) above, DOE may submit a plan that meets the requirements of Subsection C herein to provide secondary containment for substandard tank system(s)/component(s). Subject to the applicable requirements of Appendix B, the DOE may continue to use tanks systems/components scheduled to be removed from service for receipt and storage of wastes according to the approved schedule and prior to approval of the schedule, unless SCDHEC notifies DOE in writing that specific tanks are unsuitable for continued service. Waste tanks deemed unsuitable by SCDHEC shall not receive additional waste prior to schedule approval for such receipt and only if waste receipt is approved as a part of the plan associated with such schedule.

     2.   The DOE's waste tank system(s) removal plan(s) shall provide for the removal or decontamination of all residues, contaminated containment system components (liners, etc.), contaminated soils and structures and equipment contaminated with hazardous and/or radioactive substances. If the DOE demonstrates that it cannot practicably remove or decontaminate soils or structures and equipment, then the DOE shall conduct all necessary response actions under Section XI through XVI of this Agreement for those waste tank system(s). The SCDHEC is the designated oversight agency for review and approval of all response action documents leading up to the Proposed Plan. The DOE must obtain written concurrence from both EPA and SCDHEC prior to publication of Proposed Plans and RODs. SCDHEC will be the designated oversight agency for review/approval of RD/CM and CA/RA documents for the

RESPONSE TO RAI COMMENT 11
ROADMAP TO REFERENCES

REFERENCED DOCUMENT	*EXCERPT LOCATION	REMARK
10 CFR 61	NRC 10 CFR 61 subpart index and sections 61.55, 61.41, 61.42, and 61.43 are enclosed following the response.	The subpart index identifies the breakdown of the performance objectives. Section 61.55 describes waste classification and identifies radionuclide classification limits, by concentration. Section 61.41 addresses protection of the public and gives the 25 mrem/yr public whole body dose limit. Section 61.42 addresses protection of the inadvertent intruder, but does not specify the intruder dose limit. The applicable intruder limit, 500 mrem/yr, is identified in Section 5.2 of NUREG-0945, Vol. 1 (Final Environmental Impact Statement on 10 CFR Part 61 "Licensing Requirements for Land Disposal of Radioactive Waste"). Note that Section 5.2 of NUREG-09, Vol. 1 is enclosed following the response. 10 CFR 61.43 addresses protection of workers, but does not specify the worker dose limit. The applicable worker limit, 5.0 rem/yr, is identified in 10 CFR 20.1201. Note that 10 CFR 20.1201 is enclosed following the response.

RESPONSE TO RAI COMMENT 11
ROADMAP TO REFERENCES

Cook et. al. 2005	WSRC-TR-2005-00074, Section 2.1 is enclosed following the response.	This section provides the methodology for identifying vault limits based on the groundwater transport pathway and identifies that degradation of the closure system, waste form, and vault were taken into account when deriving the vault limits. The groundwater transport pathway contributes to the public dose addressed in the "all pathways" scenario.
Draft Section 3116 Determination	DOE-WD-2005-001, pp. 29 and 30 are enclosed following the response.	The statement regarding maximum practical removal of highly radioactive radionuclides, as taken from Section 3116 (a)(2) of the NDAA, is given at the top of page 29, immediately following the title of Section 5.0. The text addressing radionuclides considered in detail and Footnote 10 are given on pages 29 and 30 of Subsection 5.1.
NRC 2003 (NUREG 1757)	Excerpt enclosed following response	

RESPONSE TO RAI COMMENT 11
ROADMAP TO REFERENCES

Reboul 2005	CBU-PIT-2005-00141 is enclosed following the response.	This document provides the technical bases for a) identifying highly radioactive radionuclides in SRS salt waste and b) quantifying removal efficiencies of planned treatment processes. All inputs, assumptions, and computation methods for determining highly radioactive radionuclides are given in the Methodology section, pp. 7-11. All computational results related to identifying highly radioactive radionuclides are given in the Results and Discussion section, pp. 12-20. All inputs, assumptions, computational methods, and results related to the planned treatment processes and projected removal efficiencies are given in the Methodology section, p. 11, and the Results and Discussion section, pp. 21-26.

*Excerpt Locations:
1. Excerpt included in response: The excerpt is included within the text of the response or is appended to the response.
2. Excerpt enclosed following response: The excerpt is enclosed on a separate sheet or sheets following the response.
3. Representative excerpt(s) enclosed following response: Representative excerpts from a document that is wholly or largely applicable are enclosed following the response.
4. Other

DOE-WD-2005-001
February 28, 2005

Draft Section 3116 Determination

Salt Waste Disposal

Savannah River Site

APPROVED for Release for
Unlimited (Release to Public)
3/24/2005

5.0 THE WASTE HAS HAD HIGHLY RADIOACTIVE RADIONUCLIDES REMOVED TO THE MAXIMUM EXTENT PRACTICAL

Section 3116 (a)(2) of the NDAA provides in pertinent part:

> [T]he term "high level waste" does not include radioactive waste resulting from the reprocessing of spent nuclear fuel that the Secretary of Energy..., in consultation with the Nuclear Regulatory Commission..., determines –

> ***

> (2) has had highly radioactive radionuclides removed to the maximum extent practical[.]

5.1 RADIONUCLIDES CONSIDERED

For the purpose of this draft 3116 Determination, the radionuclides considered in detail are: Cs-137 (including its daughter, Ba-137m), Sr-90 (including its daughter, yttrium (Y)-90), the actinides (isotopes of U, Pu, Am, Np, and Cm), selenium (Se)-79, technetium (Tc)-99, iodine (I)-129, and tin (Sn)-126[10].

The short-lived fission products Cs-137 and Sr-90 and their equilibrium daughters, Ba-137m and Y-90[11], are by far the predominant sources of radioactivity present in the SRS salt waste. Based on process and sampling knowledge as reflected in the current WCS database, more than 99% [2] of the current radioactivity in the SRS tank salt waste is associated with these two radionuclides and their daughters. Indeed, Cs-137 and its daughter Ba-137m alone account for over 95% of the

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[10] DOE has reviewed the inventory of 41 radionuclides in the salt waste in the SRS waste tanks, as reflected in the current Waste Characterization System (WCS) database. WCS is discussed in further detail in Appendix 2. The radionuclides considered in detail for removal in this draft 3116 Determination take into account scientific expertise, knowledge, and health physics principles as applied to the SRS salt waste, and include those radionuclides in Tables 1 and 2 in 10 CFR 61.55 that are in the SRS salt waste and are in quantities such that they may be important to meeting the performance objectives in 10 CFR 61, Subpart C. All radionuclides in Tables 1 and 2 are considered in Section 6.0 and , where relevant, Section 7 of this draft 3116 Determination. However, radionuclides with less than five-year half-lives, H-3, C-14, Co-60, and Ni-63 are present in concentrations so low (well below Class A concentration limits) that they are not discussed for the purposes of removal in Section 5.0. The radionuclides considered in detail for the purposes of removal also include other radionuclides that are not in the above referenced tables that may be important to meeting the performance objectives in 10 CFR 61, Subpart C. Some of the radionuclides considered in detail in this draft 3116 Determination may not be considered in detail in other 3116 Determinations if the circumstances or waste forms do not warrant such consideration.

[11] Cs-137, and its daughter Ba-137m, are typically considered as a single radionuclide for human health protection purposes because the half-life of Ba-137m is so short that it only exists when Cs-137 is present. The same is true for Sr-90 and its daughter Y-90. Accordingly, the discussions that follow in this draft 3116 Determination focus on Cs-137 or Sr-90 since approaches that are effective in removing Cs-137 and Sr-90 also remove Ba-137m and Y-90, respectively.

salt waste's radioactivity [2]. Moreover, Cs-137, Sr-90, and their daughters are present in sufficient concentrations in the salt waste so that without shielding and controls they produce radiation emissions that would present risk to humans simply due to their proximity, without direct inhalation or ingestion. Accordingly, they are of potential acute hazard to occupational workers, the public, and the environment.

Although the actinides in the concentrations present in the salt waste (e.g., U, Pu, Am, Np, and Cm) do not require distance and shielding to protect workers and the public, these radionuclides contain the majority of the long-lived isotopes that may pose the greatest risk in the future to human health because of their long life and because they present human health risk if directly inhaled or ingested. Other radionuclides considered in detail are those that persist in the waste form well into the future due to their long lives, are relatively mobile in the environment, and may present a health risk if ingested. These radionuclides are Se-79, Tc-99, I-129, and Sn-126. These two groups of radionuclides (i.e., the actinides, Se-79, Tc-99, I-129, and Sn-126) together account for less than 1% [2] of the radionuclide inventory in the tank salt waste.

As explained above, the radionuclides considered in detail in this draft 3116 Determination are based on, and take into account, scientific and health physics principles, knowledge, and expertise. In addition, the radionuclides considered in detail account for over 99.9% [2] of the current radioactivity in the SRS tank salt waste, based on process and sampling knowledge.

5.2 REMOVAL TO THE MAXIMUM EXTENT PRACTICAL

Section 3116 (a) of the NDAA provides that certain waste resulting from reprocessing is not high-level waste if the Secretary, in consultation with the NRC, determines, among other things, that the waste has had highly radioactive radionuclides removed "to the maximum extent practical." This section discusses the basis on which the Secretary may conclude that DOE's two-phase, three-part strategy for removing the radionuclides considered above meets this criterion.

Removal to the extent "practical" is not removal to the extent "practicable" or theoretically "possible." Rather, a "practical" approach to removal is one that is "adapted to actual conditions,"[12] "adapted or designed for actual use; useful,"[13] a method that is selected "mindful of the results, usefulness, advantages or disadvantages, etc., of [the] action or procedure."[14] The considerations that bear on whether radionuclide removal will be accomplished to the maximum

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[12] *A Dictionary of Modern English Usage* 453, H. Fowler (1930).

[13] *Random House Unabridged Dictionary*, Copyright © 1997, by Random House, Inc., on Infoplease (www.infoplease.com/ipd/A0598638.html).

[14] *Id.* definition 7. See also Cambridge Dictionary of American English, Cambridge University Press 2004 (http://dictionary.cambridge.org/define.asp?key=practical*2+0&dict=A): "practical (EFFECTIVE): fitting the needs of a particular situation in a helpful way; helping to solve a problem or difficulty; effective or suitable".

It is notable that while prior NRC and Department criteria for waste incidental to reprocessing called for removal "to the maximum extent technically and economically practical," the statute omits the adverbs, thereby suggesting that a broad range of considerations, including but not limited to technical and economic practicalities, may appropriately be taken into account in determining the extent of removal that is practical.

Final Environmental Impact Statement
on 10 CFR Part 61 "Licensing Requirements for Land Disposal of Radioactive Waste"

Summary and Main Report

U.S. Nuclear Regulatory Commission

Office of Nuclear Material Safety and Safeguards

November 1982

Based on the analysis in Chapter 2 of the draft EIS (§ 2.2), the preferred approach selected and followed by NRC in the preparation of the proposed Part 61 was to develop both performance objectives and prescriptive require- ments. Overall performance objectives were developed to define the level of safety that should be achieved in the land disposal of LLW. Minimum technical performance requirements were also developed for each of the major components of an LLW disposal system that should be considered in all cases in the dis- posal of LLW to help ensure that the overall performance objectives for land disposal would be met. Finally, prescriptive requirements were established where they were deemed necessary and where sufficient technical information and rationale were available to support them.

Based on public comments on the Part 61 rule, draft EIS, and NRC's analysis of these comments, NRC has made no change to this approach. It has been followed in the development of the final Part 61 rule.

5.2 DEVELOPMENT OF PERFORMANCE OBJECTIVES

As part of the analysis performed in the draft EIS, NRC analyzed a range of alternative performance objectives for low-level radioactive waste disposal. This analysis involved an extensive series of case studies plus an extensive examination of the case study results. From the analysis NRC staff identified four such overall performance objectives:

1. Protect public health and safety (and the environment) over the long term;

2. Protect the inadvertent intruder;

3. Protect workers and the public during the short-term operational phase; and

4. Long-term stability to eliminate the need for active long-term maintenance after operations cease;

There were few comments from the public on the overall numerical analysis per- formed in the draft EIS to arrive at the preferred performance objectives. There were, however, some comments on the specific details of the analysis such as assumptions on environmental monitoring costs. Based upon the comments, NRC made a number of revisions to the numerical inputs to the impact analysis methodology including an improved method of cost analysis, a more extensive analysis of the impacts of waste classification and analysis of a new pathway (trench overflow and leachate treatment). The effect of the revisions to the analysis methodology had no effect on the overall conclusions but, rather, con- firmed NRC's original conclusions. To provide greater clarity, an effort was made to reduce the number of cases considered and this resulted in the analysis performed in Chapter 4 of this final EIS. Based on public comments on the pro- posed rule, no new areas were identified which should be addressed in the Part 61 rule as overall performance objectives for land disposal of LLW. Commenters generally supported development of performance objectives in the above four areas.

One rule commenter challenged the performance objectives in Part 61 as being premature in advance of relevant EPA standards and beyond the agency's authority to the extent that they are not already embodied in 10 CFR Part 20 and that they are unduly stringent and unsupported. With respect to this comment, EPA, under its ambient environmental standards setting authority assigned by Reorganization Plan No. 3 of 1970 has the authority to prepare a standard that will set limits for releases of radioactivity to the general environment from disposal facilities. Presently there is no such EPA standard. In the absence of such a standard, the Commission examined a range of limits which bound that expected for the EPA standard and selected a proposed performance objective that establishes a release limit for the site boundary, a regulatory action within the limits of NRC authority. In a rulemaking action, the Commission is not solely limited to existing standards in Part 20 and the Commission does not intend to withdraw any portion of the rule that may be related to the performance objectives.

With regard to the specific performance objective for releases to the environment, the Environmental Protection Agency commented that the establishment of an individual exposure limit at the site boundary for releases as proposed in §61.41 is appropriate. They stated that the 25 mrem/yr limit is in the correct range of values (1 to 25 mrem/yr was analyzed by the Commission) which should encompass any future EPA standard for low-level waste disposal facilities. Based on the analysis, NRC does not anticipate any need to change the technical requirements of Part 61 to meet a future EPA standard. In their comments, EPA stated their opinion that it was inappropriate to apply the EPA drinking water standard as proposed in §61.41. Accordingly, this part of the performance objective has been deleted. However, this does not diminish the Commission's concern over protecting sources of drinking water. The Commission will assess the potential impact on drinking water supplies as part of its licensing review.

Reaction to the proposed performance objective to protect potential inadvertent intruders was mixed. There were some who felt the proposed 500 mrem whole body dose to the intruder was too high, some felt that it was the right value for a standard, and others felt that higher values were in order. Those that felt that the standard should be higher suggested values of 5 rem or 25 rem to correspond to limits for occupational exposure or one-time exposures to workers from potential accidents. A number of commenters, in their comments about considering the probability that intrusion will occur, expressed concern about weighting too heavily the protection against inadvertent intrusion in determining disposal requirements for waste. Based on these comments, the Commission believes that the primary concern of those who feel that the intruder protection objective is too restrictive is the effect that this has on the concentrations of certain nuclides that are acceptable for disposal in a near-surface facility and the need to meet waste form requirements such as stability for some wastes. With this in mind, and in response to other comments, the Commission has reevaluated the calculations that establish the waste classification concentration limits to eliminate unnecessarily conservative assumptions with the result that the analysis is more realistic and the limits for several important isotopes have been raised. With this action, the Commission believes that most of the concerns of those who encouraged higher exposure limits or less emphasis on protection of intruders will have been met.

With respect to those who suggested that lower limits would be appropriate, there were no compelling arguments or technical demonstrations presented that persuaded the Commission to lower the dose limit for intruders.

The EPA commented that it was not appropriate to state the 500 mrem (whole body) dose limit as a regulatory limit in the Part 61 rule, since the licensee would not be able to monitor or demonstrate compliance with a specific dose limit that applies to an event that might occur hundreds of years from now. They did recognize use of the 500 mrem whole body dose limit as the basis for determining the concentration limits in Table 1 of Part 61. Noting that, given ALARA, actual exposures to an inadvertent intruder would be lower than 500 mrem per year, the 500 mrem dose limit has been deleted from the performance objective but has been retained as the basis of the waste classification concentration limits.

EPA asked for a clarification of the intent of the performance objective in §61.43 as it pertains to effluents from the site. This performance objective states that operations at the land disposal facility must be conducted in compliance with the standards for radiation protection set out in Part 20. Part 20 contains standards for concentrations of radioisotopes in air and water released from a licensed facility. Section 61.41 sets forth limits on concentrations of radioisotopes released from a land disposal facility which are lower than those in Part 20. It is the Commission's intent that the provisions of Part 20 will apply to all aspects of radiation protection during operation except for releases of radioactivity from the site which will be governed by the more stringent requirements of §61.41. The rule has been modified to clarify this point.

Commenters pointed out a need to be clearer in the rule on how the principle of maintaining radiation exposures to a level that is as low as reasonably achievable (ALARA) will be handled. The Commission intends that the ALARA principle apply to the performance objectives for long-term environmental release and protection of individuals during site operations. It cannot apply to the intruder performance objective, since Part 61 sets out requirements for intrusion protection which are beyond the disposal facility licensee's control. Appropriate changes have been made in §§66.41 and 61.43 to reflect the ALARA principle.

Based upon the EIS analysis, and comments provided on the proposed Part 61 rule, the following performance objectives were derived for the final Part 61 rule:

5.2.1 Protection of the General Population From Releases of Radioactivity

Concentrations of radioactive material which may be released to the general environment in ground water, surface water, air, soil, plants, or animals must not result in an annual dose exceeding an equivalent of 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public. Reasonable effort should be made to maintain releases of radioactivity in effluents to the general environment to levels as low as is reasonably achievable.

5.2.2 Protection of Individuals from Inadvertent Intrusion

Design, operation, and closure of the land disposal facility must ensure protection of any individual inadvertently intruding into the disposal site

Subpart C--Occupational Dose Limits

Source: 56 FR 23396, May 21, 1991, unless otherwise noted.

§ 20.1201 Occupational dose limits for adults.

(a) The licensee shall control the occupational dose to individual adults, except for planned special exposures ι 20.1206, to the following dose limits.

(1) An annual limit, which is the more limiting of--

(i) The total effective dose equivalent being equal to 5 rems (0.05 Sv); or

(ii) The sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue · the lens of the eye being equal to 50 rems (0.5 Sv).

(2) The annual limits to the lens of the eye, to the skin of the whole body, and to the skin of the extremities, v

(i) A lens dose equivalent of 15 rems (0.15 Sv), and

(ii) A shallow-dose equivalent of 50 rem (0.5 Sv) to the skin of the whole body or to the skin of any extremity.

(b) Doses received in excess of the annual limits, including doses received during accidents, emergencies, and special exposures, must be subtracted from the limits for planned special exposures that the individual may re the current year (see § 20.1206(e)(1)) and during the individual's lifetime (see § 20.1206(e)(2)).

(c) The assigned deep-dose equivalent must be for the part of the body receiving the highest exposure. The a: shallow-dose equivalent must be the dose averaged over the contiguous 10 square centimeters of skin receivir exposure. The deep-dose equivalent, lens-dose equivalent, and shallow-dose equivalent may be assessed fron other radiation measurements for the purpose of demonstrating compliance with the occupational dose limits, individual monitoring device was not in the region of highest potential exposure, or the results of individual mc unavailable.

(d) Derived air concentration (DAC) and annual limit on intake (ALI) values are presented in table 1 of append 20 and may be used to determine the individual's dose (see § 20.2106) and to demonstrate compliance with t occupational dose limits.

(e) In addition to the annual dose limits, the licensee shall limit the soluble uranium intake by an individual to in a week in consideration of chemical toxicity (see footnote 3 of appendix B to part 20).

(f) The licensee shall reduce the dose that an individual may be allowed to receive in the current year by the a occupational dose received while employed by any other person (see § 20.2104(e)).

[56 FR 23396, May 21, 1991, as amended at 60 FR 20185, Apr. 25, 1995; 63 FR 39482, July 23, 1998; 67 FR 5, 2002]

10 CFR 61 REFERENCES

are as follows: §§60.1, 60.2, 60.3, 60.5, 60.6, 60.7, 60.8, 60.15, 60.16, 60.17, 60.18, 60.21, 60.22, 60.23, 60.24, 60.31, 60.32, 60.33, 60.41, 60.42, 60.43, 60.44, 60.45, 60.46, 60.51, 60.52, 60.61, 60.62, 60.63, 60.64, 60.65, 60.101, 60.102, 60.111, 60.112, 60.113, 60.121, 60.122, 60.130, 60.131, 60.132, 60.133, 60.134, 60.135, 60.137, 60.140, 60.141, 60.142, 60.143, 60.150, 60.151, 60.152, 60.162, 60.181, and 60.183.

[57 FR 55076, Nov. 24, 1992]

PART 61—LICENSING REQUIREMENTS FOR LAND DISPOSAL OF RADIOACTIVE WASTE

Subpart A—General Provisions

AUTHORITY: Secs. 53, 57, 62, 63, 65, 81, 161, 182, 183, 68 Stat. 930, 932, 933, 935, 948, 953, 954, as amended (42 U.S.C. 2073, 2077, 2092, 2093, 2095, 2111, 2201, 2232, 2233); secs. 202, 206, 88 Stat. 1244, 1246, (42 U.S.C. 5842, 5846); secs. 10 and 14, Pub. L. 95-601, 92 Stat. 2951 (42 U.S.C. 2021a and 5851) and Pub. L. 102-486, sec. 2902, 106 Stat. 3123, (42 U.S.C. 5851); sec. 1704, 112 Stat. 2750 (44 U.S.C. 3504 note).

SOURCE: 47 FR 57463, Dec. 27, 1982, unless otherwise noted.

§ 61.30 Transfer of license.

(a) Following closure and the period of post-closure observation and maintenance, the licensee may apply for an amendment to transfer the license to the disposal site owner. The license shall be transferred when the Commission finds:

(1) That the closure of the disposal site has been made in conformance with the licensee's disposal site closure plan, as amended and approved as part of the license;

(2) That reasonable assurance has been provided by the licensee that the performance objectives of subpart C of this part are met;

(3) That any funds for care and records required by §61.80 (e) and (f) have been transferred to the disposal site owner;

(4) That the post-closure monitoring program is operational for implementation by the disposal site owner; and

(5) That the Federal or State government agency which will assume responsibility for institutional control of the disposal site is prepared to assume responsibility and ensure that the institutional requirements found necessary under §61.23(g) will be met.

[47 FR 57463, Dec. 27, 1982, as amended at 61 FR 24674, May 16, 1996]

§ 61.31 Termination of license.

(a) Following any period of institutional control needed to meet the requirements found necessary under §61.23, the licensee may apply for an amendment to terminate the license.

(b) This application must be filed, and will be reviewed, in accordance with the provision of §61.20 and of this section.

(c) A license is terminated only when the Commission finds:

(1) That the institutional control requirements found necessary under §61.23(g) have been met; and

(2) That any additional requirements resulting from new information developed during the institutional control period have been met, and that permanent monuments or markers warning against intrusion have been installed.

(3) That the records required by §61.80 (e) and (f) have been sent to the party responsible for institutional con-

trol of the disposal site and a copy has been sent to the Commission immediately prior to license termination.

[47 FR 57463, Dec. 27, 1982, as amended at 61 FR 24674, May 16, 1996]

Subpart C—Performance Objectives

§ 61.40 General requirement.

Land disposal facilities must be sited, designed, operated, closed, and controlled after closure so that reasonable assurance exists that exposures to humans are within the limits established in the performance objectives in §§61.41 through 61.44.

§ 61.41 Protection of the general population from releases of radioactivity.

Concentrations of radioactive material which may be released to the general environment in ground water, surface water, air, soil, plants, or animals must not result in an annual dose exceeding an equivalent of 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public. Reasonable effort should be made to maintain releases of radioactivity in effluents to the general environment as low as is reasonably achievable.

§ 61.42 Protection of individuals from inadvertent intrusion.

Design, operation, and closure of the land disposal facility must ensure protection of any individual inadvertently intruding into the disposal site and occupying the site or contacting the waste at any time after active institutional controls over the disposal site are removed.

§ 61.43 Protection of individuals during operations.

Operations at the land disposal facility must be conducted in compliance with the standards for radiation protection set out in part 20 of this chapter, except for releases of radioactivity in effluents from the land disposal facility, which shall be governed by §61.41 of this part. Every reasonable effort shall be made to maintain radiation exposures as low as is reasonably achievable.

completed closure and stabilization measures.

(11) Only wastes containing or contaminated with radioactive materials shall be disposed of at the disposal site.

(b) Facility operation and disposal site closure for land disposal facilities other than near-surface (reserved).

[47 FR 57463, Dec. 27, 1982, as amended at 56 FR 23474, May 21, 1991; 56 FR 61352, Dec. 3, 1991; 58 FR 67662, Dec. 22, 1993]

§ 61.53 Environmental monitoring.

(a) At the time a license application is submitted, the applicant shall have conducted a preoperational monitoring program to provide basic environmental data on the disposal site characteristics. The applicant shall obtain information about the ecology, meteorology, climate, hydrology, geology, geochemistry, and seismology of the disposal site. For those characteristics that are subject to seasonal variation, data must cover at least a twelve month period.

(b) The licensee must have plans for taking corrective measures if migration of radionuclides would indicate that the performance objectives of subpart C may not be met.

(c) During the land disposal facility site construction and operation, the licensee shall maintain a monitoring program. Measurements and observations must be made and recorded to provide data to evaluate the potential health and environmental impacts during both the construction and the operation of the facility and to enable the evaluation of long-term effects and the need for mitigative measures. The monitoring system must be capable of providing early warning of releases of radionuclides from the disposal site before they leave the site boundary.

(d) After the disposal site is closed, the licensee responsible for post-operational surveillance of the disposal site shall maintain a monitoring system based on the operating history and the closure and stabilization of the disposal site. The monitoring system must be capable of providing early warning of releases of radionuclides from the disposal site before they leave the site boundary.

§ 61.54 Alternative requirements for design and operations.

The Commission may, upon request or on its own initiative, authorize provisions other than those set forth in §§ 61.51 through 61.53 for the segregation and disposal of waste and for the design and operation of a land disposal facility on a specific basis, if it finds reasonable assurance of compliance with the performance objectives of subpart C of this part.

§ 61.55 Waste classification.

(a) Classification of waste for near surface disposal. (1) Considerations. Determination of the classification of radioactive waste involves two considerations. First, consideration must be given to the concentration of long-lived radionuclides (and their shorter-lived precursors) whose potential hazard will persist long after such precautions as institutional controls, improved waste form, and deeper disposal have ceased to be effective. These precautions delay the time when long-lived radionuclides could cause exposures. In addition, the magnitude of the potential dose is limited by the concentration and availability of the radionuclide at the time of exposure. Second, consideration must be given to the concentration of shorter-lived radionuclides for which requirements on institutional controls, waste form, and disposal methods are effective.

(2) Classes of waste. (i) Class A waste is waste that is usually segregated from other waste classes at the disposal site. The physical form and characteristics of Class A waste must meet the minimum requirements set forth in § 61.56(a). If Class A waste also meets the stability requirements set forth in § 61.56(b), it is not necessary to segregate the waste for disposal.

(ii) Class B waste is waste that must meet more rigorous requirements on waste form to ensure stability after disposal. The physical form and characteristics of Class B waste must meet both the minimum and stability requirements set forth in § 61.56.

(iii) Class C waste is waste that not only must meet more rigorous requirements on waste form to ensure stability but also requires additional measures at the disposal facility to

protect against inadvertent intrusion. The physical form and characteristics of Class C waste must meet both the minimum and stability requirements set forth in §61.56.

(iv) Waste that is not generally acceptable for near-surface disposal is waste for which form and disposal methods must be different, and in general more stringent, than those specified for Class C waste. In the absence of specific requirements in this part, such waste must be disposed of in a geologic repository as defined in part 60 or 63 of this chapter unless proposals for disposal of such waste in a disposal site licensed pursuant to this part are approved by the Commission.

(3) Classification determined by long-lived radionuclides. If radioactive waste contains only radionuclides listed in Table 1, classification shall be determined as follows:

(i) If the concentration does not exceed 0.1 times the value in Table 1, the waste is Class A.

(ii) If the concentration exceeds 0.1 times the value in Table 1 but does not exceed the value in Table 1, the waste is Class C.

(iii) If the concentration exceeds the value in Table 1, the waste is not generally acceptable for near-surface disposal.

(iv) For wastes containing mixtures of radionuclides listed in Table 1, the total concentration shall be determined by the sum of fractions rule described in paragraph (a)(7) of this section.

TABLE 1

Radionuclide	Concentration curies per cubic meter
C-14	8
C-14 in activated metal	80
Ni-59 in activated metal	220
Nb-94 in activated metal	0.2
Tc-99	3
I-129	0.08
Alpha emitting transuranic nuclides with half-life greater than 5 years	[1]100
Pu-241	[1]3,500
Cm-242	[1]20,000

[1] Units are nanocuries per gram.

(4) Classification determined by short-lived radionuclides. If radioactive waste does not contain any of the radionuclides listed in Table 1,

classification shall be determined based on the concentrations shown in Table 2. However, as specified in paragraph (a)(6) of this section, if radioactive waste does not contain any nuclides listed in either Table 1 or 2, it is Class A.

(i) If the concentration does not exceed the value in Column 1, the waste is Class A.

(ii) If the concentration exceeds the value in Column 1, but does not exceed the value in Column 2, the waste is Class B.

(iii) If the concentration exceeds the value in Column 2, but does not exceed the value in Column 3, the waste is Class C.

(iv) If the concentration exceeds the value in Column 3, the waste is not generally acceptable for near-surface disposal.

(v) For wastes containing mixtures of the nuclides listed in Table 2, the total concentration shall be determined by the sum of fractions rule described in paragraph (a)(7) of this section.

TABLE 2

Radionuclide	Concentration, curies per cubic meter		
	Col. 1	Col. 2	Col. 3
Total of all nuclides with less than 5 year half-life	700	[1]	[1]
H-3	40	[1]	[1]
Co-60	700	[1]	[1]
Ni-63	3.5	70	700
Ni-63 in activated metal	35	700	7000
Sr-90	0.04	150	7000
Cs-137	1	44	4600

[1] There are no limits established for these radionuclides in Class B or C wastes. Practical considerations such as the effects of external radiation and internal heat generation on transportation, handling, and disposal will limit the concentrations for these wastes. These wastes shall be Class B unless the concentrations of other nuclides in Table 2 determine the waste to be Class C independent of these nuclides.

(5) Classification determined by both long- and short-lived radionuclides. If radioactive waste contains a mixture of radionuclides, some of which are listed in Table 1, and some of which are listed in Table 2, classification shall be determined as follows:

(i) If the concentration of a nuclide listed in Table 1 does not exceed 0.1 times the value listed in Table 1, the class shall be that determined by the concentration of nuclides listed in Table 2.

Removal of Highly Radioactive Nuclides from SRS Salt Waste

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Summary of Revisions

6/2005   Revision 0, Initial Issue

Removal of Highly Radioactive Nuclides from SRS Salt Waste

Executive Summary

Removal of highly radioactive nuclides from salt waste is required to assure that the waste can be dispositioned in accordance with all applicable performance objectives of 10 CFR 61 Subpart C, as well as all applicable state and federal environmental regulations. Highly radioactive nuclides in SRS salt waste were identified through assessment of risks to the public, SRS workers, and environment. Specifically, inventories of radionuclides in untreated salt waste (without salt-based nuclide removal treatments) were compared against inventory limits based on Nuclear Regulatory Commission (NRC) Class A and 10 CFR 61 Subpart C criteria. Using this approach, Sr-90, Cs-137, and the alpha-emitting transuranic (TRU) nuclides were determined to be highly radioactive, based on contributing most significantly to the radiological risks to the workers, the public, and the environment and therefore requiring treatment for removal to the maximum extent practical as required by Section 3116 (a)(2). The following four observations were instrumental in identifying these highly radioactive nuclides. First, Sr-90, Cs-137, and the alpha-emitting TRU nuclides are the only radionuclides in the untreated salt waste having total inventories exceeding the Saltstone Disposal Facility (SDF) inventory limits based on NRC Class A criteria (10 CFR 61.55). Second, no radionuclides in untreated salt waste have average inventories in an SDF vault exceeding 10% of the vault inventory limits based on a maximum annual public dose rate of 25 mrem (10 CFR 61.41). Third, Cs-137 is the only radionuclide in untreated salt waste with an average untreated inventory in an SDF vault exceeding 10% of the vault inventory limit based on a maximum annual intruder dose of 500 mrem (10 CFR 61.42). Fourth, Cs-137 is the only radionuclide with a total inventory in untreated salt waste exceeding 10% of the SDF inventory limit based on a maximum annual worker gamma dose of 5.0 rem (CFR 61.43). Additionally, Sr-90, Cs-137, and the alpha-emitting TRU nuclides are the radionuclides driving worker inhalation dose.

Removal of the highly radioactive nuclides will be accomplished using a combination of the following five treatment processes: 1) deliquification, dissolution, and adjustment (DDA); 2) actinide removal process (ARP) without monosodium titanate (MST) sorption; 3) ARP with MST sorption; 4) modular caustic side solvent extraction unit (MCU); and 5) Salt Waste Processing Facility (SWPF) treatments. Removal efficiencies for each of the planned treatment processes are identified in the table below.

Treatment Process	Removal Efficiency, %								
	Sr-90			Cs-137			α-emitting TRU		
	Nom	Low	High	Nom	Low	High	Nom	Low	High
DDA	66	46	86	50	30	70	63	43	83
ARP w/o MST	99.6	98.0	99.9	~0	~0	~0	78	50	93
ARP w/ MST	99.997	99.4	99.999	~0	~0	~0	98	90	99.9
MCU	0	0	0	91	90	92	0	0	0.
SWPF	99.98	99.4	99.995	99.998	99.99	99.998	96	90	99.5

For strontium removal, the relative treatment efficacies are:
ARP w/ MST ≈ SWPF > ARP w/o MST > DDA >> MCU

For cesium removal, the relative treatment efficacies are:
SWPF > MCU > DDA >> ARP

For alpha-emitting TRU removal, the relative treatment efficacies are:
ARP w/ MST ≈ SWPF > ARP w/o MST > DDA >> MCU

Introduction

Removal of highly radioactive nuclides from SRS salt waste is required prior to disposition, to ensure the final waste meets the Nuclear Regulatory Commission (NRC) Class C limits (10 CFR 61.55) and all performance objectives of 10 CFR 61 Subpart C. These performance objectives include compliance with public safety and environmental dose limits as defined in 10 CFR 61.41, intruder protection as defined in 10 CFR 61.42, and worker protection as defined in 10 CFR 61.43. Compliance with the NRC Class C limits and all performance objective limits will be achieved by processing the waste through a series of treatments tailored to removal of the highly radioactive nuclides. This document provides the technical basis for:

a) identifying which nuclides in untreated salt waste are highly radioactive; and
b) quantifying treatment removal efficiencies for the highly radioactive nuclides.

The treatment processes addressed in this document include:

- Deliquification, Dissolution, and Adjustment (DDA)
- Actinide Removal Process (ARP) w/o Monosodium Titanate (MST) Sorption
- ARP w/ MST Sorption
- Modular Caustic Side Solvent Extraction Unit (MCU)
- Salt Waste Processing Facility (SWPF)

Specific Objectives

1) Identify approach for determining highly radioactive nuclides;

2) Identify soluble and insoluble radionuclide inventories in untreated salt waste;

3) Identify nuclides in untreated salt waste that are highly radioactive;

4) Identify bases and assumptions of planned radionuclide treatment processes; and

5) Quantify treatment removal efficiencies for the highly radioactive nuclides.

Methodology

Background

Thirty-six million gallons of radioactive waste are currently stored in 49 underground tanks at the Savannah River Site (SRS). Most of this material was generated through defense-related reprocessing of spent nuclear fuel. By volume, approximately 93% of the material is "salt waste" consisting of supernatant salt solution and solid precipitated salts. Disposition of the salt waste will be accomplished through a three-step approach. First, the salt waste will be treated to remove highly radioactive nuclides to the maximum extent practical. Second, the treated salt waste will be stabilized in a grout matrix. Lastly, the stabilized waste will be disposed in vaults at the SRS Saltstone Disposal Facility (SDF).

This document provides the technical basis for identifying the highly radioactive nuclides in SRS salt waste and for quantifying the removal efficiencies of the planned treatment processes. Note that in this document the term "treatment" applies solely to the processes performed for the purpose of removing highly radioactive nuclides from salt waste. This includes DDA, ARP, MCU, and SWPF treatments. In this document, "treatment" does not refer to processes performed: a) before waste was received into the underground storage tanks (i.e., segregation of waste and pH neutralization); b) for the purpose of maximizing tank space (i.e., evaporation); or c) for the purpose of stabilization at SDF (i.e., grouting). Consistent with this approach, the term "untreated waste" refers solely to waste that has not undergone treatments tailored to removing highly radioactive nuclides. Specifically, this means the waste has not been treated by DDA, ARP, MCU, or SWPF.

Approach for Determining Highly Radioactive Nuclides

From a technical perspective, the approach for identifying highly radioactive nuclides should be based on quantification of the potential risks impacting public, worker, and environmental health. For radionuclides, this includes consideration of the potential internal and external radiation doses associated with near proximity exposure to the waste, as well as remote exposures associated with inhalation and/or

ingestion of material transported via environmental media. For final disposition of the treated waste, NRC Class C limits (10 CFR 61.55) and the performance objectives of 10 CFR 61 Subpart C provide clear bases for gauging acceptable radiological risks. However, for the purpose of determining highly radioactive nuclides in the untreated waste, application of these bases is less clear.

For identifying highly radioactive nuclides, an approach was taken that recognizes the disposition requirements of 10 CFR 61 (sections 55, 41, 42, and 43) and is sufficiently conservative to take variabilities and uncertainties into account. The approach provides a basis for identifying the radionuclide removal needs. As such, the approach is based on comparisons of quantities of radionuclides in untreated waste versus applicable radionuclide limits. An observation that the quantity of a nuclide exceeds a significant fraction of any one disposal limit is consistent with a conclusion that the given nuclide is highly radioactive. Correspondingly, an observation that the quantity of a nuclide is an insignificant fraction of all applicable disposal limits is consistent with a conclusion that the given radionuclide is not highly radioactive.

For these comparisons, the first measure of radiological risk addresses the need for the final waste form to meet NRC Class C limits (10 CFR 61.55). In this case, the basis for comparison is set at the NRC Class A limits, which are significantly lower than the Class C limits. In the case of the long-lived nuclides, the Class A limits are 10% of the Class C limits. In the case of most of the short-lived nuclides, the Class A limits are vanishingly small fractions of the Class C limits (from 6E-4% to 5E-1% of the Class C limits). The rationale for using Class A limits rather than a predefined fraction of the Class C limits is based on the belief that quantities below Class A limits have minimal disposal requirements and pose an insignificant risk to the public and to workers.

The second, third, and fourth measures of radiological risk address the need for the final waste form to meet the performance objectives pertaining to public, environmental, intruder, and worker safety (10 CFR 61.41, 61.42, and 61.43). In these cases, the bases for comparisons are set at 10% of the performance objectives limits, namely 10% of the public and environmental annual dose limit of 25 mrem; 10% of the intruder annual dose limit of 500 mrem; and 10% of the worker annual dose limit of 5.0 rem.

In this analysis, comparisons were made between the total radionuclide inventories in untreated salt waste (or some fraction of the total inventories, when applicable) and the inventory limits derived for each radiological measure identified above. Radionuclides with inventories exceeding the limits based on any of the four radiological measures (Class A, 10% of the 25 mrem public annual dose, 10% of the 500 mrem intruder annual dose, or 10% of the worker annual dose) were identified as highly radioactive and assumed to require removal to the maximum extent practical, prior to disposition.

It is recognized that using the total radionuclide inventories as bases for the risk-based decisions produces conclusions reflective of average waste compositions, rather than the full range of waste compositions. Although this approach may seem non-conservative, it is assumed appropriate for this analysis for the following reasons: 1) the uncertainties of the total radionuclide inventories are significantly lower than the uncertainties of the inventories in individual waste tanks; 2) the total radionuclide inventories provide a good indication of the relative impacts of the various radionuclides; and 3) the risk measures are sufficiently conservative to accommodate most concentration fluctuations.

The methods for deriving the radionuclide inventory limits for the various measures of radiological risk are provided in the section entitled Calculation of Inventory Limits.

Radionuclide Inventories in Untreated Salt Waste

Soluble and insoluble radionuclide inventories in untreated salt waste were identified based on the characterization data reported by Tran (2005). Soluble inventories of C-14, Na-22, and Al-26 were determined by summing the inventories reported for the precipitated salt phase (referred to by Tran as the insoluble salt phase) and the total free supernatant phase. Soluble inventories of all other radionuclides were those reported for the total free supernatant phase.

Insoluble inventories of all radionuclides were calculated based on the assumption that entrained sludge exists in salt feed at a concentration of 600 mg/L (600 mg/L is the design basis for the various salt treatment processes). Based on a total projected salt feed volume of 197 Mgal (DOE, 2005) and a total estimated dry sludge mass of 2.9E+6 kg (WCS1.5, 2005), the entrained sludge represents 8.4% of the total sludge. Consequently, the insoluble radionuclide inventories in salt were determined by multiplying 0.084 by the total dry sludge radionuclide inventories.

Soluble and insoluble inventories of alpha-emitting transuranic (TRU) nuclides were calculated by summing the respective soluble and insoluble phase inventories of Np-237, Pu-238, Pu-239, Pu-240, Pu-242, Am-241, Am-243, Cm-242, Cm-243, Cm-244, and Cm-245.

Total inventories of radionuclides were computed by the summing the soluble and insoluble inventories.

Note that the radionuclides quantified in this analysis were those radionuclides important from the perspective of NRC classification and the performance objectives identified in 10 CFR 61 Subpart C (public and environmental risk, intruder risk, and worker risk).

Calculation of Inventory Limits and Comparisons Against Radionuclide Inventories

For the NRC Class A radiological risk measure, comparisons were made between the total radionuclide inventories in untreated salt waste and the total curie limits for the SDF under the assumption that NRC Class A concentration limits applied and radionuclide inventories were evenly distributed across the total volume of grout at the SDF. Specifically, the total curie limits for the SDF were determined by multiplying the Class A concentration limits by: a) the total projected grout volume (6.4E+5 cubic meters (DOE, 2005)) when the Class A concentration limit was given in units of curies per cubic meter; and b) the product of the total projected grout mass (1.1E+12 g, which assumes a grout density of 1.7 g/mL) and the activity conversion factor (1E-9 Ci/nCi), when the Class A concentration limit was given in units of nCi/g. Ratios of "the total curies in untreated salt waste" to "the Class A total curie limit" were then calculated and reported. Ratios exceeding one are indicative of nuclides considered to be highly radioactive.

For the public, environmental, and intruder risk measures, comparisons were made between the average untreated radionuclide inventories in 9.6 Mgal of grout (Vault 4 capacity) and the applicable inventory limits for Vault 4. This was done under the assumption that the Vault 4 limits would be representative of the limits of other vaults of the same size and thus would provide a reasonable measure of which nuclides require treatment to meet public and intruder dose limits. The average untreated inventories in 9.6 Mgal of grout were determined by multiplying the total radionuclide inventories by a factor of 0.057. This factor represents the proportion of the waste in 9.6 Mgal grout and is computed by dividing the Vault 4 volume (9.6 Mgal) by the total projected grout volume (168 Mgal, as reported by DOE in 2005). For the public and environmental risk measures, the inventory limits for a 9.6 Mgal vault were those values reported by Cook et. al. (2005) under the "all pathways" scenario, which assumes a public dose rate of 25 mrem/yr. Ratios of the "average radionuclide inventories in 9.6 Mgal grout" to the "Vault 4 public dose inventory limits" were then calculated and reported. Ratios exceeding 0.10 (10%) are indicative of nuclides considered to be highly radioactive.

For the intruder risk measure, comparisons were made between the average untreated inventories in 9.6 Mgal of grout and the inventory limits for Vault 4 assuming a 500 mrem annual intruder dose. In the intruder case, the inventory limits for Vault 4 were computed by multiplying by five the inventory limits reported by Cook et. al. (2005) under the "intruder" scenario, which assumes an intruder dose rate of 100 mrem/yr. Ratios of the "average radionuclide inventories in 9.6 Mgal grout" to the "Vault 4 intruder dose inventory limits" were then calculated and reported. Ratios exceeding 0.10 (10%) are indicative of nuclides considered to be highly radioactive.

For the worker dose risk measure, two scenarios were considered, one focusing on worker gamma dose and the other focusing on radionuclides driving potential inhalation dose. For the worker gamma dose risk measure, comparisons were made between the total radionuclide inventories in untreated salt waste (normalized for Cs-137/Ba-137m dose) and the SDF Cs-137/Ba-137m inventory limit based on a 5.0 rem annual worker gamma dose. Note that the SDF worker gamma limit is defined as 0.2 curies Cs-137 per gallon of salt solution (Howell, 2005), based on a maximum annual worker gamma dose of 1.0 rem/yr. To convert this to an inventory limit representing a 5.0 rem annual worker gamma dose, the facility concentration limit was multiplied by the total projected volume of salt feed (107 Mgal) and a factor of five (5.0 rem/1.0 rem). This results in a total facility limit of 1.1E+8 curies Cs-137. For comparisons against this limit, the inventories of photon-emitting radionuclides in the untreated salt waste were normalized for Cs-137/Ba-137m dose, by multiplying each radionuclide inventory by the ratio of the radionuclide's photon dose constant to the Cs-137/Ba-137m photon dose constant (this ratio is referred to as the dose normalization factor). Photon dose constants used in these calculations were those reported by Unger and Trubey (1982). In cases where the nuclide had short-lived decay products (cases such as Ru-106, Sn-126, Sb-125, Ce-144, Np-237), the photon dose contributions of the decay products were taken into account. For example, in the case of Sn-126, the overall photon dose constant was the sum of the Sn-126 constant, the Sb-126m dose constant, and 14% of the Sb-126 constant (this takes into account that 100% of the Sn-126 disintegrations produce Sb-126m and 14% of the Sb-126m disintegrations produce Sb-126). Ratios of the "normalized radionuclide inventories" to the "worker gamma dose inventory limit" were then calculated and reported. Ratios exceeding 0.10 (10%) are indicative of nuclides considered to be highly radioactive.

For the worker inhalation dose measure, the primary radionuclides contributing to the total potential inhalation dose were identified. This was accomplished by: a) multiplying the radionuclide inventories in untreated salt waste by the nuclide-specific inhalation dose factors to identify the total potential inhalation dose associated with each nuclide; b) summing the contributions of all the total potential inhalation doses; and c) computing the percentage of the cumulative potential inhalation dose associated with each nuclide. Radionuclides contributing one percent or more of the cumulative potential inhalation dose were considered to be the primary drivers of the potential inhalation dose. The inhalation dose factors used in these calculations were those reported by EPA (1988), converted from units of Sv/Bq to units of rem/Ci by multiplying the Sv/Bq values by factors of 3.7E+12 rem-Bq/Sv-Ci. Note that this risk measure does not address the significance of worker inhalation dose, however, it serves two important functions – it provides a basis for comparison of the nuclides deemed highly radioactive by the other risk measures and assures that the primary worker inhalation risk drivers are identified.

Calculation of Radionuclide Removal Efficiencies

Radionuclide removal efficiencies for the planned treatment processes were calculated for all nuclides deemed highly radioactive. Assumptions regarding removal efficacies of the various processes are given in the Results and Discussion section, along with flow schemes and process descriptions for each treatment scheme. In every case, removal efficiencies were calculated by applying the appropriate treatment assumptions to the expected soluble and insoluble constituent phases. The bases for the soluble-insoluble distributions are the soluble and insoluble inventories identified for the untreated salt waste (Table 1).

Nominal removal efficiencies were calculated based on the exact soluble-insoluble distributions given in Table 1 and the nominal decontamination assumptions. Lower and upper bounds of the removal efficiencies were calculated based on conditions where the soluble-insoluble distributions increased and decreased by a factor of four (a total factor of sixteen), and lower and upper bounds of the decontamination assumptions were applied, when available. In cases where bounding decontamination assumptions were not available, lower and upper bounding removal efficiencies were calculated based on nominal decontamination assumptions applied to lower and upper bounding soluble-insoluble distributions. Note that in several cases, the nominal decontamination factors were chosen conservatively and actually represent lower bounding decontamination factors.

Removal efficiencies for alpha-emitting TRU were identified based on weighted averages of the individual removal efficiencies of Pu-238, Am-241, Cm-244, and Pu-239. These four nuclides contribute approximately 99% of the TRU alpha activity and therefore are considered the primary nuclides impacting TRU alpha removal.

Results and Discussion

Radionuclide Inventories in Untreated Salt Waste

Soluble, insoluble, and total radionuclide inventories in untreated salt waste are given in Table 1. Soluble radionuclides are those radionuclides present as dissolved ions in supernatant solution or as precipitated salts that will dissolve upon addition of water. Insoluble radionuclides are those radionuclides present in the form of entrained sludge. Note that for a given radionuclide, the soluble-insoluble distribution is an important factor determining treatment requirements for effective removal. Soluble constituents typically require chemically-specific removal technologies tailored to the particular constituent. Insoluble constituents are typically removed through physical methods such as filtration or centrifugation.

Based on the information in Table 1, it can be seen that the soluble-insoluble distribution varies considerably as a function of radionuclide, as would be expected based upon the chemical differences of the various constituents. About 20% of the radionuclides exist primarily as soluble constituents, 40% exist primarily as insoluble constituents, and 40% exist with soluble and insoluble contributions of the same order of magnitude. This suggests about 20% of the isotopes would be relatively unaffected by a physical separation process (such as filtration), 40% would be highly affected by physical separation, and 40% would be moderately affected by physical separation.

Based on the total radionuclide inventories, Cs-137 is clearly the predominant radionuclide from an activity perspective. (It should be noted that Ba-137m, the short-lived decay product of Cs-137, is also a predominant radionuclide. Ba-137m is not included in Table 1, but is taken into account when calculating inventory limits for Cs-137). As seen in the table, activities of all other radionuclides (except for Ba-137m) are one or more orders of magnitude below that of Cs-137.

Also of note are the primary radionuclides contributing to alpha-emitting TRU activity. Approximately 82% of the alpha-emitting TRU inventory is Pu-238, 11% is Am-241, 4% is Cm-244, and 3% is Pu-239. All other alpha-emitting TRU nuclides contribute minimally to the TRU inventory.

Comparison with Inventory Limits Based on the Class A Risk Measure

Table 2 gives: a) the radionuclide inventories in untreated salt waste, b) the NRC Class A inventory limits for the total projected grout volume, and c) the ratios of the inventories to the limits. Based on the ratios, it is clear that the inventories of Sr-90, Cs-137, and alpha-emitting TRU in untreated salt waste are significantly greater than the inventory limits. Specifically, the inventory of Sr-90 is 290 times the limit, the inventory of Cs-137 is 170 times the limit, and the inventory of alpha-emitting TRU is 25 times the limit. Consequently, Sr-90, Cs-137, and alpha-emitting TRU are considered to be highly radioactive nuclides requiring treatment prior to disposition. Because 99% of the TRU alpha activity is attributed to Pu-238, Am-241, Cm-244, and Pu-239, it is these TRU nuclides that are considered to be highly radioactive and require treatment prior to disposition. Most important is the removal of Pu-238, since it is responsible for 82 % of the TRU alpha activity.

For all other nuclides compared against the Class A risk measure, the ratios of the inventories to the limits are clearly less than one (by a factor of three or more), suggesting these nuclides are not highly radioactive from a Class A perspective.

Comparison with Inventory Limits Based on the Public Dose Risk Measure

Table 3 gives: a) the average radionuclide inventories projected for a 9.6 Mgal volume of grout (the Vault 4 grout volume); b) the Vault 4 inventory limits based on a 25 mrem/yr public dose limit, and c) the ratios of the average inventories to the inventory limits. For all nuclides, the ratios of the average inventories to the inventory limits are significantly less 0.10 (by one or more orders of magnitude), suggesting no nuclides are highly radioactive from the public dose perspective.

Comparison with Inventory Limits Based on the Intruder Dose Risk Measure

Table 4 gives: a) the average radionuclide inventories projected for a 9.6 Mgal volume of grout (the Vault 4 grout volume); b) the Vault 4 inventory limits based on a 500 mrem/yr intruder dose limit, and c) the ratios of the average inventories to the inventory limits. Based on the results, it is clear that Cs-137 is the only nuclide with a ratio greater than 0.10. Specifically, the ratio for Cs-137 is 0.21, approximately twice the intruder dose risk measure. Although already deemed a highly radioactive nuclide based on the Class A risk measure, the intruder dose measure confirms the importance of providing Cs-137 treatment prior to disposition. All other nuclides have ratios significantly less than 0.10 (by two or more orders of magnitude), suggesting these other nuclides are not highly radioactive from the intruder dose perspective.

Comparison with Inventory Limits Based on the Worker Dose Risk Measure

Results of the comparisons based on the gamma exposure and inhalation dose risk measures are given in Table 5A and 5B, respectively.

Table 5A gives: a) the radionuclide inventories in untreated salt waste, b) the Cs-137/Ba-137m dose normalization factors, and c) the ratios of the normalized inventories to the Cs-137/Ba-137m inventory limit, based on a worker gamma dose rate of 5.0 rem/yr. From the results, it is clear that Cs-137 is the only nuclide with a ratio greater than 0.10. Specifically, the ratio for Cs-137 is 1.0, tens times the worker gamma dose risk measure. Although already deemed a highly radioactive nuclide based on the Class A and intruder dose risk measures, the worker gamma dose risk measure confirms the importance of providing Cs-137 treatment prior to disposition. All other nuclides have ratios significantly less than 0.10 (by one or more orders of magnitude), suggesting these other radionuclides are not highly radioactive from the worker gamma dose perspective.

Table 5B gives: a) the radionuclide inventories in untreated salt waste, b) the inhalation dose factors, and c) the percentages of the total potential inhalation dose due to each nuclide. From the results, it is clear that Sr-90, Cs-137, and the four primary alpha-emitting TRU nuclides (Pu-238, Pu-239, Am-241, and Cm-244) contribute the majority of the potential inhalation dose. Specifically, the percentages of the

inhalation doses contributed by these nuclides are approximately 7, 3, 73, 3, 10, and 2, respectively. Collectively, these nuclides contribute approximately 98% of the total potential inhalation dose. Each other nuclide contributes less than one percent of the total potential inhalation dose, with contributions from most of the nuclides being two or more orders of magnitudes lower than one percent. Clearly, Pu-238 is the largest potential contributor of inhalation dose (73% of the total potential inhalation dose), with Am-241 and Sr-90 being the second and third largest contributors (10 and approximately 7%, respectively), and Cs-137, Pu-239, and Cm-244 being minor contributors (approximately 3, 3, and 2%, respectively). Based on the results, it is clear that the nuclides most important from a worker inhalation dose perspective are the same nuclides identified by the other radiological risk measures.

Summary of Highly Radioactive Nuclides

A summary of the nuclides considered to be highly radioactive and the technical bases for their risk-based determination is given in Table 6. Sr-90 and alpha-emitting TRU nuclides are considered highly radioactive based on exceeding the NRC Class A limits and being important worker inhalation dose drivers. Cs-137 is considered highly radioactive based on exceeding the NRC Class A limit, 10% of the intruder limit, 10% of the worker gamma limit, and being an important worker inhalation dose driver. Based on the risk measures assessed in this analysis, no other nuclides are considered to be highly radioactive.

Table 1
Soluble, Insoluble, and Total Inventories in Untreated Salt Waste

Radionuclide	Soluble Curies in Untreated Salt Waste	Insoluble Curies in Untreated Salt Waste	Total Curies in Untreated Salt Waste
H-3	9.4E+3	Negligible	9.4E+3
C-14	5.2E+2	2.8E-1	5.2E+2
Na-22	5.1E+3	9.3E-1	5.1E+3
Al-26	2.4E+1	4.9E-1	2.4E+1
Co-60	8.6E+1	2.9E+4	2.9E+4
Ni-59	2.4E0	2.2E+2	2.2E+2
Ni-63	2.1E+2	1.9E+4	1.9E+4
Se-79	8.9E+1	1.3E+2	2.2E+2
Sr-90	2.8E+4	7.3E+6	7.3E+6
Nb-94	7.0E-4	5.9E-2	6.0E-2
Tc-99	3.3E+4	2.2E+3	3.5E+4
Ru-106	2.3E+3	5.7E+2	2.9E+3
Sn-126	4.5E+2	1.7E+2	6.2E+2
Sb-125	9.2E+3	1.5E+4	2.4E+4
I-129	1.8E+1	9.0E-3	1.8E+1
Cs-134	2.3E+5	8.8E+2	2.3E+5
Cs-135	3.9E+2	1.5E0	3.9E+2
Cs-137	1.1E+8	4.4E+5	1.1E+8
Ce-144	5.9E0	5.0E+2	5.1E+2
Pm-147	3.8E+3	3.2E+5	3.2E+5
Sm-151	4.3E+3	3.6E+5	3.6E+5
Eu-152	2.1E+1	1.7E+3	1.7E+3
Eu-154	9.1E+2	7.6E+4	7.7E+4
Eu-155	2.4E+2	2.0E+4	2.0E+4
Th-232	1.0E-1	2.4E-1	3.4E-1
U-232	2.9E-2	4.7E-2	7.6E-2
U-233	2.7E0	8.3E0	1.1E+1
U-234	4.2E0	3.2E0	7.4E0
U-235	8.4E-2	1.3E-1	2.1E-1
U-236	3.6E-1	5.2E-1	8.8E-1
U-238	6.8E0	5.5E0	1.2E+1
Np-237	4.2E0	8.3E0	1.3E+1
Pu-238	5.7E+4	1.7E+5	2.3E+5
Pu-239	3.4E+3	4.0E+3	7.4E+3
Pu-240	9.1E+2	1.7E+3	2.6E+3
Pu-241	3.8E+4	1.0E+5	1.4E+5
Pu-242	9.4E-1	2.8E0	3.7E0
Am-241	3.6E+2	3.0E+4	3.0E+4
Am-242m	2.1E-1	1.8E+1	1.8E+1
Am-243	7.0E-2	5.8E0	5.9E0
Cm-242	1.7E-1	1.5E+1	1.5E+1
Cm-243	4.1E-2	3.4E0	3.4E0
Cm-244	1.5E+2	1.2E+4	1.2E+4
Cm-245	1.4E-2	1.2E0	1.2E0
α-emitting TRU	6.2E+4	2.2E+5	2.8E+5

Table 2
Comparison of Inventories in Untreated Salt Waste with NRC Class A Limits

Radionuclide	Total Curies in Untreated Salt Waste	Class A Limit, Curies Per Total Projected Grout Volume	Ratio of "Total Curies in Untreated Salt Waste" to "Class A Limit"
H-3	9.4E+3	2.5E+7	3.8E-4
C-14	5.2E+2	5.1E+5	1.0E-3
Co-60	2.9E+4	4.5E+8	6.4E-5
Ni-63	1.9E+4	2.2E+6	8.6E-3
Sr-90	7.3E+6	2.5E+4	2.9E+2
Tc-99	3.5E+4	1.9E+5	1.8E-1
I-129	1.8E+1	5.1E+3	3.3E-3
Cs-137	1.1E+8	6.4E+5	1.7E+2
Pu-241	1.4E+5	3.8E+5	3.7E-1
Cm-242	1.5E+1	2.2E+6	6.8E-6
α-emitting TRU	2.8E+5	1.1E+4	2.5E+1

Table 3
Comparison of Inventories in Untreated Salt Waste and Inventory Limits
Based on a Public Dose Rate Limit of 25 mrem/yr

Radionuclide	Average Untreated Curies in 9.6 Mgal of Grout (Vault 4 Volume)	Vault 4 Curie Limit Based on a Maximum Public Dose Rate of 25 mrem/yr	Ratio of "Average Untreated Curies in 9.6 Mgal Grout" to "Vault 4 Curie Limit Based on a 25 mrem/yr Public Dose Rate"
H-3	5.4E+2	1.3E+12	4.2E-10
C-14	3.0E+1	1.1E+8	2.7E-7
Al-26	1.4E0	2.3E+10	6.1E-11
Co-60	1.7E+3	> 1.0E+20	< 1.7E-17
Ni-59	1.3E+1	1.6E+19	8.1E-19
Se-79	1.3E+1	1.0E+3	1.3E-2
Sr-90	4.2E+5	1.4E+17	3.0E-12
Nb-94	3.4E-3	7.0E+17	4.9E-21
Tc-99	2.0E+3	1.1E+17	1.8E-14
Sn-126	3.5E+1	2.9E+19	1.2E-18
Sb-125	1.4E+3	> 1.0E+20	< 1.4E-17
I-129	1.0E0	4.0E+3	2.5E-4
Cs-134	1.3E+4	> 1.0E+20	< 1.3E-16
Cs-135	2.2E+1	> 1.0E+20	< 2.2E-19
Cs-137	6.3E+6	> 1.0E+20	< 6.3E-14
Eu-152	9.7E+1	> 1.0E+20	< 9.7E-19
Eu-154	4.4E+3	> 1.0E+20	< 4.4E-17
Eu-155	1.1E+3	> 1.0E+20	< 1.1E-17
Th-232	1.9E-2	> 1.0E+20	< 1.9E-22
U-232	4.3E-3	> 1.0E+20	< 4.3E-23
U-233	6.3E-1	> 1.0E+20	< 6.3E-21
U-234	4.2E-1	> 1.0E+20	< 4.2E-21
U-235	1.2E-2	> 1.0E+20	< 1.2E-22
U-236	5.0E-2	> 1.0E+20	< 5.0E-22
U-238	6.9E-1	> 1.0E+20	< 6.9E-21
Np-237	7.4E-1	8.9E+18	8.3E-20
Pu-238	1.3E+4	> 1.0E+20	< 1.3E-16
Pu-239	4.2E+2	> 1.0E+20	< 4.2E-18
Pu-240	1.5E+2	> 1.0E+20	< 1.5E-18
Pu-241	8.0E+3	> 1.0E+20	< 8.0E-17
Pu-242	2.1E-1	> 1.0E+20	< 2.1E-21
Am-241	1.7E+3	> 1.0E+20	< 1.7E-17
Am-242m	1.0E0	> 1.0E+20	< 1.0E-20
Am-243	3.4E-1	> 1.0E+20	< 3.4E-21
Cm-243	1.9E-1	> 1.0E+20	< 1.9E-21
Cm-244	6.9E+2	> 1.0E+20	< 6.9E-18
Cm-245	6.9E-2	> 1.0E+20	< 6.9E-22

Table 4
Comparison of Inventories in Untreated Salt Waste and Inventory Limits
Based on a Resident Intruder Dose Rate Limit of 500 mrem/yr

Radionuclide	Average Untreated Curies in 9.6 Mgal of Grout (Vault 4 Volume)	Vault 4 Curie Limit Based on an Intruder Dose Rate of 500 mrem/yr	Ratio of "Average Untreated Curies in 9.6 Mgal Grout" to "Vault 4 Curie Limit Based on a 500 mrem/yr Intruder Dose Rate"
H-3	5.4E+2	> 5.0E+20	< 1.1E-18
C-14	3.0E+1	> 5.0E+20	< 6.0E-20
Al-26	1.4E0	8.0E+2	1.8E-3
Co-60	1.7E+3	2.9E+10	5.9E-8
Ni-59	1.3E+1	> 5.0E+20	< 2.6E-20
Se-79	1.3E+1	> 5.0E+20	< 2.6E-20
Sr-90	4.2E+5	> 5.0E+20	< 8.4E-16
Nb-94	3.4E-3	5.0E+3	6.8E-7
Tc-99	2.0E+3	1.9E+14	1.1E-11
Sn-126	3.5E+1	6.0E+3	5.8E-3
Sb-125	1.4E+3	7.0E+17	2.0E-15
I-129	1.0E0	> 5.0E+20	< 2.0E-21
Cs-134	1.3E+4	2.1E+20	6.2E-17
Cs-135	2.2E+1	> 5.0E+20	< 4.4E-20
Cs-137	6.3E+6	3.0E+7	**2.1E-1**
Eu-152	9.7E+1	3.2E+7	3.0E-6
Eu-154	4.4E+3	6.0E+8	7.3E-6
Eu-155	1.1E+3	5.5E+19	2.0E-17
Th-232	1.9E-2	8.0E+2	2.4E-5
U-232	4.3E-3	4.5E+4	9.6E-8
U-233	6.3E-1	7.0E+4	9.0E-6
U-234	4.2E-1	2.3E+4	1.8E-5
U-235	1.2E-2	5.0E+5	2.4E-8
U-236	5.0E-2	1.6E+9	3.1E-11
U-238	6.9E-1	3.3E+5	2.1E-6
Np-237	7.4E-1	3.4E+5	2.2E-6
Pu-238	1.3E+4	6.5E+7	2.0E-4
Pu-239	4.2E+2	7.0E+10	6.0E-9
Pu-240	1.5E+2	1.5E+13	1.0E-11
Pu-241	8.0E+3	5.0E+10	1.6E-7
Pu-242	2.1E-1	2.5E+11	8.4E-13
Am-241	1.7E+3	1.7E+9	1.0E-6
Am-242m	1.0E0	4.9E+7	2.0E-8
Am-243	3.4E-1	1.5E+6	2.3E-7
Cm-243	1.9E-1	3.5E+10	5.4E-12
Cm-244	6.9E+2	5.5E+15	1.3E-13
Cm-245	6.9E-2	4.2E+7	1.6E-9

Comparison of Inventories in Untreated Salt Waste and Facility Curie Limit
Based on a Worker Gamma Dose Rate of 5.0 rem/yr

Radionuclide	Total Curies in Untreated Salt Waste	Cs-137/Ba-137m Dose Normalization Factor	Ratio of "Normalized Total Curies in Untreated Salt Waste" to "Cs-137 Curie Limit Based on a 5.0 rem/yr Worker Gamma Dose Rate"
Al-26	2.4E+1	3.9E0	8.7E-7
Co-60	2.9E+4	3.6E0	9.8E-4
Nb-94	6.0E-2	2.6E0	1.5E-9
Tc-99	3.5E+4	1.2E-6	3.9E-10
Ru-106	2.9E+3	3.6E-1	9.8E-6
Sn-126	6.2E+2	3.3E-1	1.9E-6
Sb-125	2.4E+4	9.9E-1	2.2E-4
I-129	1.8E+1	3.3E-1	5.6E-8
Cs-134	2.3E+5	2.6E0	5.6E-3
Cs-137	1.1E+8	1.0E0	**1.0E0**
Ce-144	5.1E+2	6.1E-2	2.9E-7
Pm-147	3.2E+5	7.0E-6	2.1E-8
Sm-151	3.6E+5	2.4E-4	8.1E-7
Eu-154	7.7E+4	2.0E0	1.4E-3
Eu-155	2.0E+4	1.7E-1	3.2E-5
Th-232	3.4E-1	1.8E-1	5.7E-10
U-232	7.6E-2	2.3E-1	1.6E-10
U-233	1.1E+1	7.6E-2	7.8E-9
U-234	7.4E0	2.0E-1	1.4E-8
U-235	2.1E-1	8.9E-1	1.7E-9
U-236	8.8E-1	1.9E-1	1.6E-9
U-238	1.2E+1	1.7E-1	1.9E-8
Np-237	1.3E+1	1.2E0	1.5E-7
Pu-238	2.3E+5	2.1E-1	4.5E-4
Pu-239	7.4E+3	7.9E-2	5.5E-6
Pu-240	2.6E+3	2.0E-1	4.9E-6
Pu-242	3.7E0	1.6E-1	5.5E-9
Am-241	3.0E+4	8.2E-1	2.3E-4
Am-242m	1.8E+1	4.8E-1	8.1E-8
Am-243	5.9E0	8.2E-1	4.5E-8
Cm-242	1.5E+1	2.4E-3	3.4E-10
Cm-244	1.2E+4	1.7E-1	1.9E-5
Cm-245	1.2E0	1.2E0	1.3E-8

Table 5B
Inhalation Dose Contributions in Untreated Salt Waste

Radionuclide	Total Curies in Untreated Salt Waste	Inhalation Dose Factor, rem/Ci	% of Total Potential Inhalation Dose in Untreated Salt Waste
H-3	9.4E+3	6.3E+1	4.6E-7
C-14	5.2E+2	2.4E+1	9.3E-9
Na-22	5.1E+3	7.8E+3	3.1E-5
Al-26	2.4E+1	8.1E+4	1.5E-6
Co-60	2.9E+4	2.2E+5	5.0E-3
Ni-59	2.2E+2	2.7E+3	4.6E-7
Ni-63	1.9E+4	6.3E+3	9.3E-5
Se-79	2.2E+2	1.0E+4	1.7E-6
Sr-90	7.3E+6	1.3E+6	**7.4E0**
Nb-94	6.0E-2	4.1E+5	1.9E-8
Tc-99	3.5E+4	8.5E+3	2.3E-4
Ru-106	2.9E+3	4.8E+5	1.1E-3
Sn-126	6.2E+2	1.0E+5	4.8E-5
Sb-125	2.4E+4	1.2E+4	2.2E-4
I-129	1.8E+1	1.7E+5	2.4E-6
Cs-134	2.3E+5	4.8E+4	8.5E-3
Cs-135	3.9E+2	4.4E+3	1.3E-6
Cs-137	1.1E+8	3.2E+4	**2.7E0**
Ce-144	5.1E+2	3.7E+5	1.5E-4
Pm-147	3.2E+5	4.1E+4	1.0E-2
Sm-151	3.6E+5	3.0E+4	8.5E-3
Eu-152	1.7E+3	2.2E+5	2.9E-4
Eu-154	7.7E+4	2.8E+5	1.7E-2
Eu-155	2.0E+4	4.1E+4	6.4E-4
Th-232	3.4E-1	1.6E+9	4.2E-4
U-232	7.6E-2	6.7E+8	4.0E-5
U-233	1.1E+1	1.4E+8	1.2E-3
U-234	7.4E0	1.3E+8	7.4E-4
U-235	2.1E-1	1.2E+8	1.9E-5
U-236	8.8E-1	1.2E+8	8.5E-5
U-238	1.2E+1	1.2E+8	1.1E-3
Np-237	1.3E+1	5.6E+8	5.7E-3
Pu-238	2.3E+5	4.1E+8	**7.3E+1**
Pu-239	7.4E+3	4.4E+8	**2.6E0**
Pu-240	2.6E+3	4.4E+8	8.5E-1
Pu-241	1.4E+5	8.1E+6	8.5E-1
Pu-242	3.7E0	4.1E+8	1.2E-3
Am-241	3.0E+4	4.4E+8	**1.0E+1**
Am-242m	1.8E+1	4.4E+8	6.1E-3
Am-243	5.9E0	4.4E+8	2.0E-3
Cm-242	1.5E+1	1.7E+7	2.0E-4
Cm-243	3.4E0	3.1E+8	8.5E-4
Cm-244	1.2E+4	2.5E+8	**2.3E0**
Cm-245	1.2E0	4.4E+8	4.1E-4

Table 6
Highly Radioactive Nuclides in Untreated Salt Waste

Highly Radioactive Nuclide	Technical Basis			
	Exceeds Class A	> 10% Intruder Limit	> 10% Worker Gamma Limit	Drives Worker Inhalation Dose
Sr-90	X			X
Cs-137	X	X	X	X
Alpha-emitting TRU	X			X

Planned Treatment Processes and Projected Removal Efficiencies

<u>Deliquification, Dissolution, and Adjustment (DDA):</u>

```
                Step 3:
                Add dissolution water
                │
                │
                ▼
┌────────────────┐    Dissolved        ┌──────────────┐   Clarified
│   Saltcake     │    Salt Solution    │              │   Salt Solution (CSS)
│     and        │──────────────────▶  │   Settling   │──────────────────▶
│  Supernatant   │                     │              │
└────────────────┘                     └──────────────┘
        │                                      │
        │  Step 1:                             │      Sludge
        │  Remove free supernatant             │
        │                                      │
        │  Step 2:                             │
        │  Remove interstitial                 │
        │  supernatant                         ▼
        ▼                                   DWPF
  Removed supernatant will be sent to an
  alternate HLW tank and held for future
  treatment at SWPF
```
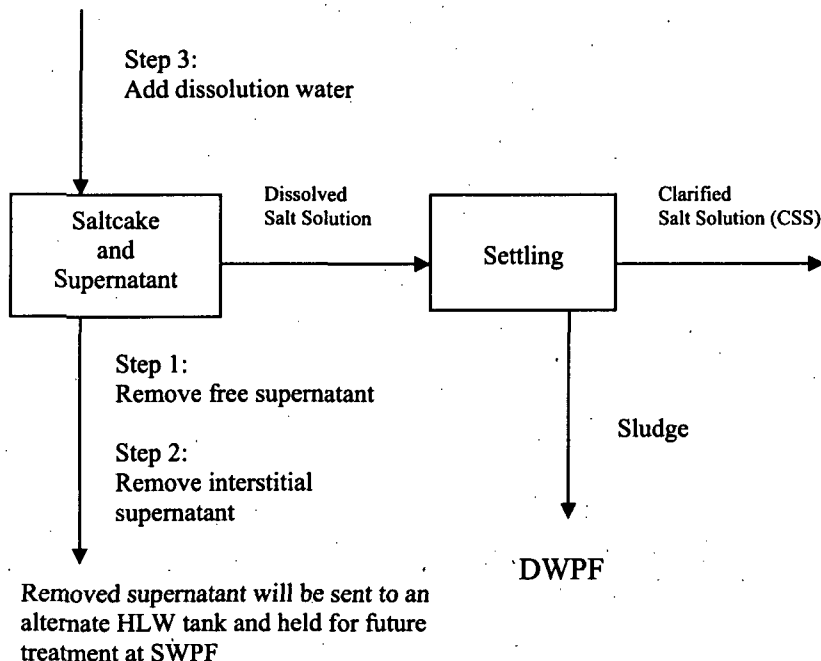
The DDA process relies on two removal mechanisms, removal of supernatant fluid through pumping/draining and removal of suspended solids (sludge) through gravity settling/clarification. As shown in Step 1 of the flow diagram, the DDA process is initiated when free supernatant solution (supernatant above saltcake) is pumped from the tank. During Step 2, interstitial supernatant fluid is drained/removed from the saltcake after a well is generated through the saltcake. All fluid removed is sent to an alternate tank for future treatment at the SWPF. In Step 3, the saltcake is dissolved and transferred to a settling tank. Following a settling period, the clarified salt solution (CSS) is decanted out of the tank and dispositioned to SPF. In the future, the settled solids will ultimately be removed from the tank and dispositioned to DWPF. Note that early batches of CSS containing elevated Cs-137 concentrations will undergo further treatment via ARP/MCU before being dispositioned to SPF.

In determining the overall removal efficiencies of DDA, the following individual removal efficiencies are assumed. Deliquification typically removes 50% of the supernatant solution (Shah and Hopkins, 2004), with a lower bound of 30% and an upper bound of 70% (these assumptions incorporate the lessons learned from Tank 41 salt). For a thirty day period, gravity settling typically removes two-thirds of the suspended solids (Gillam, 2005), with a lower bound of 50% and an upper bound of 80%. Given the magnitude of these variabilities, the uncertainty of the overall removal efficiency is typically ± 20%, regardless of the soluble/insoluble distribution.

Removal efficiencies for the highly radioactive nuclides are given in Table 7. Nominal removal efficiencies range from 50% to 66%, depending on the solubility of the constituent (50% for highly soluble constituents; 66% for highly insoluble constituents). For Sr-90, the nominal removal efficiency is 66%, with a lower bound of 46% and an upper bound of 86%. For Cs-137, the nominal removal efficiency is 50%, with a lower bound of 30% and an upper bound of 70%. For TRU, the nominal removal efficiency is 63%, with a lower bound of 43% and an upper bound of 83%.

Table 7
Removal of Highly Radioactive Nuclides Using DDA

Radionuclide	Projected Removal Efficiency, %		
	Nominal	Lower Bound	Upper Bound
Sr-90	66	46	86
Cs-137	50	30	70
Pu-238	63	43	83
Am-241	66	46	86
Cm-244	66	46	86
Pu-239	59	39	79
α-emitting TRU	63	43	83

Actinide Removal Process (ARP) w/o Monosodium Titanate (MST):
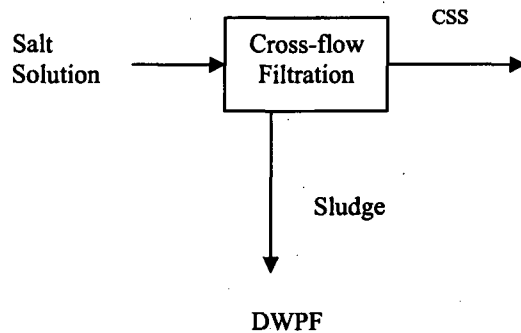
The ARP process (w/o MST sorption) relies on one removal mechanism, removal of suspended solids (sludge) by cross-flow filtration. Removed solids are dispositioned to DWPF. CSS is dispositioned to SPF.

Cross-flow filtration is assumed to nominally remove 100% of the suspended solids, although it is recognized that actual removal will be slightly lower. A lower bound of 99.5% removal is assumed, based on industrial filtration experience.

Removal efficiencies for the highly radioactive nuclides are given in Table 8. For Sr-90, the removal efficiencies are high (98.0 – 99.9%), due to the low solubility which makes particulate removal significant. In contrast, for Cs-137, the removal efficiencies are negligible, due to the high solubility which makes particulate removal insignificant. For TRU, the range of removal efficiencies is relatively broad (51 – 93%), reflecting the expectation that appreciable quantities of both soluble and insoluble phases will be present.

Table 8
Removal of Highly Radioactive Nuclides Using ARP w/o MST

Radionuclide	Projected Removal Efficiency, %		
	Nominal	Lower Bound	Upper Bound
Sr-90	99.6	98.0	99.9
Cs-137	~ 0	~ 0	~ 0
Pu-238	75	43	92
Am-241	98.8	94.9	99.7
Cm-244	98.8	94.8	99.7
Pu-239	54	23	82
α-emitting TRU	78	50	93

ARP w/ MST:
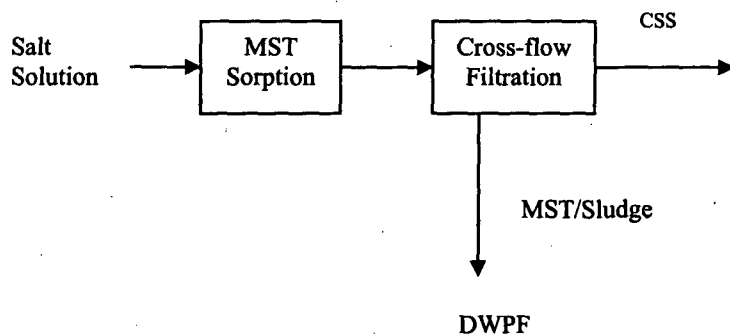
The ARP process (w/ MST sorption) relies on two removal mechanisms, removal of soluble constituents by MST sorption and removal of suspended solids (MST and sludge) by cross-flow filtration. Removed solids are dispositioned to DWPF. CSS is dispositioned to SPF.

Duration of the MST strike is assumed to be 24 hours. Assumed MST decontamination factors (DFs) are given in the table below. Nominal DFs are those reported by d'Entremont (2005) for a twenty four hour duration strike. Lower and upper bounding DFs are those reported by Le (2005) under conditions of four to twenty four hour duration strikes. Assumptions regarding efficiency of the cross-flow filter are the same as in the previous case (ARP w/o MST).

Constituent	ARP MST Soluble Phase Decontamination Factor		
	Nominal	Lower Bound	Upper Bound
Strontium	130	20	130
Cesium	0	0	0
Plutonium	13	5.5	13
Americium	1.7	1.0	4.6
Curium	1.7	1.0	1.7

Removal efficiencies for the highly radioactive nuclides are given in Table 9. For Sr-90, the removal efficiencies are extremely high (99.4 – 99.999%), due to a) the very low solubility of strontium that makes particulate removal significant and b) the very high removal efficiency of MST for soluble phase strontium. For Cs-137, the removal efficiencies are negligible due to a) the high solubility of cesium that makes particulate removal insignificant and b) the inability of MST to sorb soluble phase cesium. For TRU, the removal efficiencies are relatively high (96 – 99%), due to the combination of low solubility and reasonably high soluble phase removal. Clearly, the combination of MST and cross-flow filtration is an effective treatment for Sr-90 and TRU nuclides.

Table 9
Removal of Highly Radioactive Nuclides Using ARP w/ MST

Radionuclide	ARP w/ MST Projected Removal Efficiency, %		
	Nominal	Lower Bound	Upper Bound
Sr-90	99.997	99.4	99.999
Cs-137	~ 0	~ 0	~ 0
Pu-238	98.1	89.4	99.9
Am-241	99.3	94.9	99.9
Cm-244	99.3	94.8	99.8
Pu-239	96.4	85.8	98.6
α-emitting TRU	98.1	90.1	99.9

Modular Caustic Side Solvent Extraction Unit (MCU):
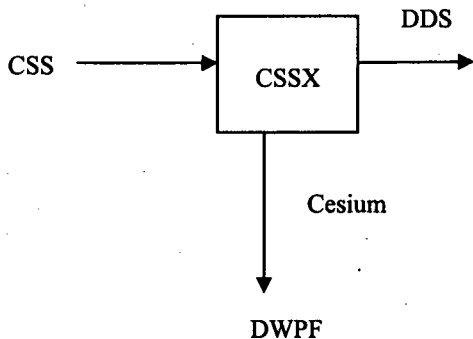
The MCU process relies on one removal mechanism, removal of soluble phase cesium by liquid-liquid extraction utilizing the Caustic Side Solvent Extraction (CSSX) technology. In this process, CSS is the feed stream and the effluents include a concentrated cesium stream that is dispositioned to DWPF and a decontaminated salt solution (DSS) that is dispositioned to SPF.

For MCU, a DF of 12 is assumed for soluble phase cesium (d'Entremont, 2005). For Sr-90 and TRU nuclides, the MCU removal efficiency is assumed to be zero.

The nominal removal efficiency for Cs-137 is 91%, with a lower bound of 90% and an upper bound of 92%.

Salt Waste Processing Facility (SWPF) Treatment:
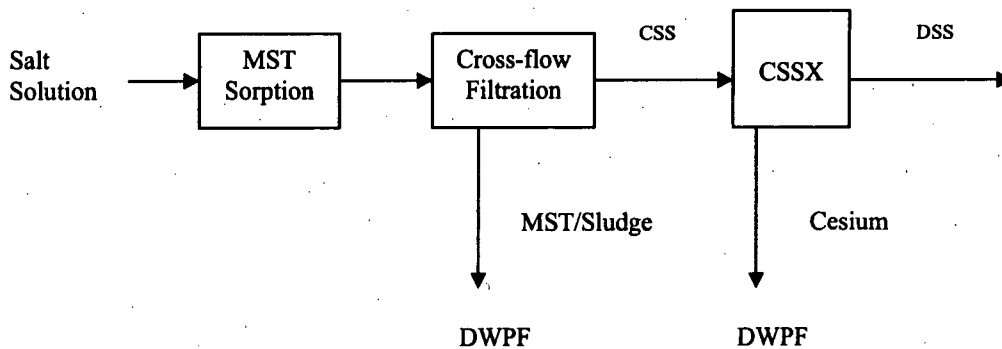
The SWPF treatment process relies on three removal mechanisms: 1) removal of soluble constituents by MST sorption; 2) removal of suspended solids by cross-flow filtration; and 3) removal of cesium by liquid-liquid extraction utilizing CSSX. In this process, salt solution is first treated with MST and then

filtered to produce a CSS that is subsequently treated with CSSX. The removed solids and the concentrated cesium streams are dispositioned to DWPF, and the DSS stream is dispositioned to SPF.

Duration of the MST strike is assumed to be 12 hours (Parsons, 2004). Assumed DFs for the MST treatment are given in the table below. Nominal MST DFs are those reported by d'Entremont (2005) for a twelve hour duration strike. Lower and upper bounding MST DFs are those reported by Le (2005) under conditions of four to twenty four hour duration strikes. Assumptions regarding efficiency of the cross-flow filter are the same as in the previous ARP cases. For SWPF, the CSSX DF for soluble phase cesium is assumed to be 40,000 (d'Entremont, 2005).

Constituent	SWPF MST Soluble Phase Decontamination Factor		
	Nominal	Lower Bound	Upper Bound
Strontium	20	20	130
Cesium	0	0	0
Plutonium	5.5	5.5	13
Americium	4.6	1.0	4.6
Curium	1.0	1.0	1.7

Removal efficiencies for the highly radioactive nuclides are given in Table 10. For Sr-90, the removal efficiencies are very high (99.4 to 99.999%), due to the combination of effective particulate removal and high soluble phase decontamination. For Cs-137, the removal efficiencies are extremely high (99.990 to 99.998%, respectively), due to the extremely high removal efficiency of CSSX for soluble phase cesium. For TRU, the removal efficiencies are high (91 to 99%), although lower than those of Sr-90 and Cs-137. Clearly, the SWPF treatments offer an effective means of removing Sr-90, Cs-137, and TRU nuclides.

Table 10
Removal of Highly Radioactive Nuclides Using SWPF

Radionuclide	SWPF Projected Removal Efficiency, %		
	Nominal	Lower Bound	Upper Bound
Sr-90	99.98	99.4	99.999
Cs-137	99.998	99.990	99.998
Pu-238	95.5	89.4	99.4
Am-241	99.7	94.9	99.94
Cm-244	98.8	94.8	99.8
Pu-239	91.6	85.8	98.6
α-emitting TRU	96	90	99.5

Conclusions

1) Cs-137 in untreated salt waste is considered highly radioactive from the perspective of exceeding NRC Class A limits and 10% of the public, environmental, and worker dose limits.

2) Sr-90 and alpha-emitting TRU nuclides in untreated salt waste are considered highly radioactive from the perspective of exceeding NRC Class A limits.

3) Other than Sr-90, Cs-137 and alpha-emitting TRU, no nuclides in untreated salt waste are considered highly radioactive from the perspective of exceeding NRC Class A limits or public, environmental, and worker dose limits.

4) For strontium removal, the relative efficacies of the planned treatments are:

ARP w/ MST ≈ SWPF > ARP w/o MST > DDA >> MCU

5) For cesium removal, the relative efficacies of the planned treatments are:

SWPF > MCU > DDA >> ARP

6) For alpha-emitting TRU removal, the relative efficacies of the planned treatments are:

ARP w/ MST ≈ SWPF > ARP w/o MST > DDA >> MCU

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WSRC-TR-2005-00074
Revision 0

KEY WORDS: Performance Assessment
Low-level Radioactive Waste Disposal

SPECIAL ANALYSIS:
REVISION OF SALTSTONE VAULT 4 DISPOSAL LIMITS (U)

PREPARED BY:
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MAY 26, 2005
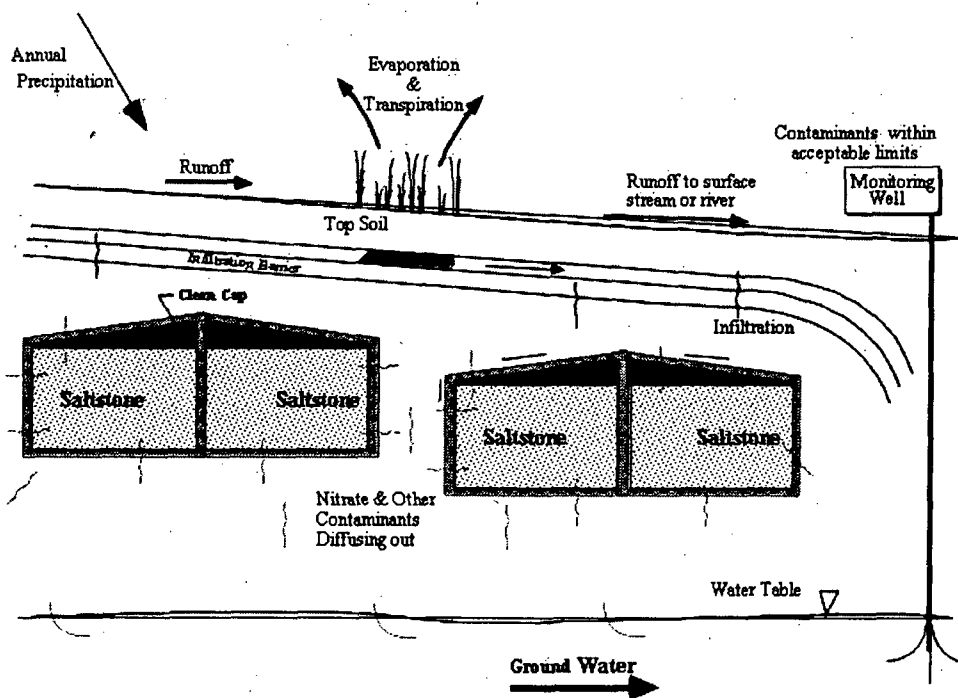
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**Prepared for the U.S. Department of Energy Under
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SRNL

2.0 GROUNDWATER ANALYSIS

2.1 Methodology

The groundwater pathway analysis for each radionuclide involves two steps. First a vadose zone flow and transport simulation is done to estimate flux to the water table for a disposed radionuclide parent and any subsequent progeny. Then saturated zone flow and transport modeling is used to estimate the groundwater concentration(s) at a hypothetical well placed 100 meters down-gradient from the disposal unit.

The vadose zone flow model was developed to reflect the current Z-Area closure concept (Phifer and Nelson 2003), which calls for a geosynthetic cover system instead of a kaolin cap as assumed in the 1992 PA. After completion of the institutional control period, infiltration is predicted to gradually increase over time as the closure system degrades due to phenomena such as intrusion of deep-rooted plants (e.g., trees) and silting of drainage layers (Phifer 2004). While it is assumed that tree root penetration will contribute to closure system degradation, tree roots should not penetrate into the Saltstone, itself, and uptake radionuclides for the following reasons:

- Several layers of the multi-layered cover system above the vault roof are frequently at or near saturation. Since tree roots are opportunistic and seek sources of water, the roots will concentrate in these layers above the vault roof, which contain significant water.

- While roots might penetrate to the vault roof, the concrete roof presents a hardened surface over which roots are more likely to extend along rather than penetrate.

- The pore fluid within Saltstone is essentially a salt solution (brackish water) which the trees could not utilize.

- It is unlikely that roots would be able to extract water from Saltstone due to the matrix potential within Saltstone.

The purpose of the deeper roots of pine trees is to seek sources of water. The multi-layered cover system will produce local zones of saturated water in the drainage layers overlying the barrier layers. The pine tree roots will tend to follow these layers rather than attempt to penetrate to deeper levels since it is much easier for the roots to extract water from saturated soil than unsaturated soil. Therefore, pine tree roots are not expected to penetrate the vault roof.

A potential PA concern is the effects of cracks developing in the Saltstone monolith over time. A structural analysis (Peregoy 2003) predicts that cracks will develop and their aperture will increase with increasing time. However, the analysis shows that the cracks will open either at the top or at the bottom and will be pinched closed at the opposite end. Therefore, no through-wall cracks will develop. A separate modeling study (Yu and Cook 2004) concluded that cracks of this nature have very little effect on contaminant transport rate. Based on this finding cracks are not considered in this SA.

The conceptual model describes the materials, layout, and dimensions of the SDF. Figure 2-1 depicts the conceptual model used for the Vault No. 4. The Saltstone monolith is approximately 200×600×25 ft. Only half of a vault in the short dimension is modeled, taking advantage of symmetry. The top of the modeling domain is the bottom of the upper GCL layer. Infiltration through this layer as a function of time is calculated by the HELP code (USEPA 1994a, 1994b). The constant infiltration rate is used as a flow boundary condition at the top of the modeling domain. The bottom of the modeling domain is the water table. Capillary pressure at the water table is set to zero to simulate 100% water saturation. The vertical boundary through the center of the vault is modeled as a no-flow boundary due to symmetry. The right boundary is also assumed to be a no-flow boundary because it is sufficiently far away from the vault and the predominant
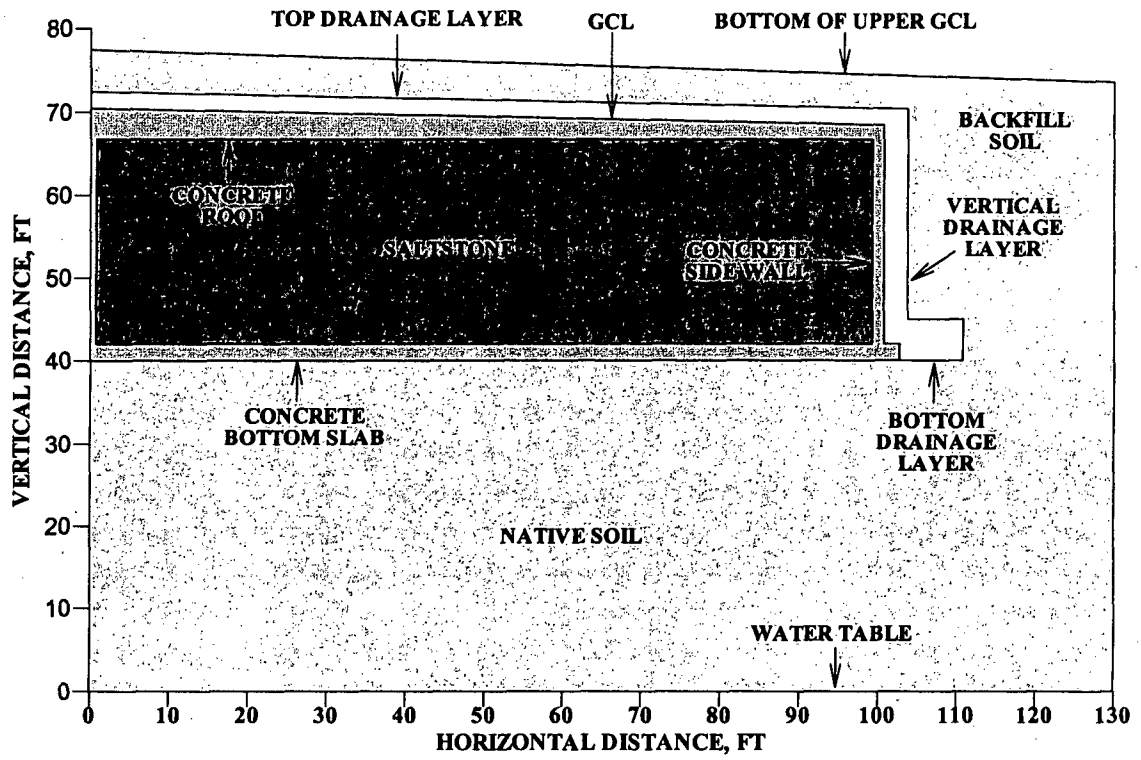
Figure 2-1. Conceptual Model for the Saltstone Vault No. 4
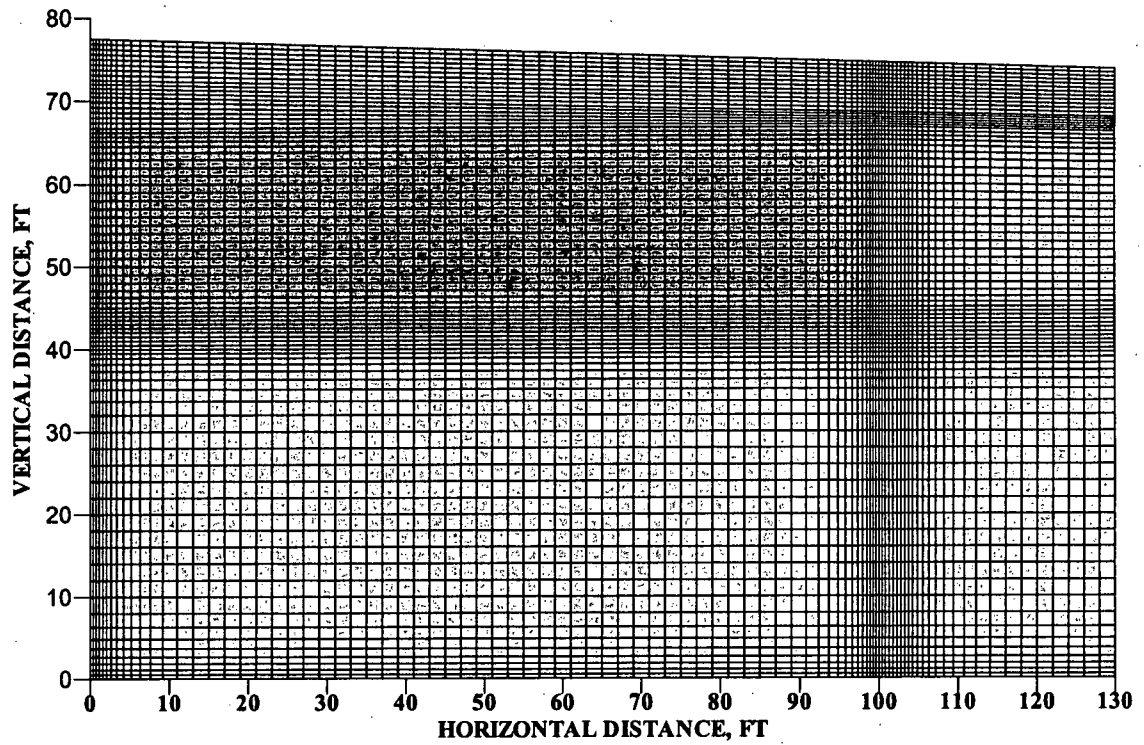
Figure 2-2. Modeling Grid

contaminant transport mechanism is downward convection. Figure 2-2 shows the gridding used in the model.

The vadose zone flow simulation was performed as a sequence of steady-state runs approximating average conditions during a number of time intervals over 10,000 years based on the HELP code results. Time zero is when closure operations are complete. Material properties were varied for each time interval to represent degradation of the closure system, the Saltstone waste form, and the vault. These properties are given in Appendix A.

A total of 45 radionuclides were selected for analysis based on a screening study for the SRS Low Level Waste Facility (Cook and Wilhite 2004). Nitrate was also run in the analysis because it occurs in high concentrations and has a relatively low groundwater limit.

The new plutonium chemistry implemented for the trench disposal units in the E-Area Low-Level Waste Facility (Cook 2002, Kaplan 2004) has been included in the present special analysis. The Pu (III/IV) oxidation state is far more abundant than Pu (V/VI), but the latter is significantly more mobile in sediments: a soil-solute distribution coefficient of $K_d = 370$ mL/g is assumed for Pu (III/IV) versus $K_d = 15$ mL/g for Pu (V/VI). Although present in trace amounts, the relatively high mobility of Pu (V/VI) could potentially lead to a significant contribution to the dose at the 100-meter well. The two pairs of oxidation states are tracked separately in the vadose zone transport simulations to accommodate the difference in mobility.

In addition to the geochemistry modifications described above, some distribution coefficients were updated to reflect current knowledge. Appendix A provides a complete listing of K_d values used in the groundwater analysis and other key input data such as, radionuclides analyzed, half-lives, atomic mass, concentration limits, solubility limits, and assumed decay chains.

The FACT code model of the General Separations Area (GSA) was recently superseded by an equivalent model using the PORFLOW code, in order to consolidate PA subsurface flow and transport modeling to a single software product (Flach 2004). The flow field computed by GSA/PORFLOW is used in the present study. GSA/PORFLOW is a regional scale model with a mesh resolution in the horizontal plane of 200 ft, compared to a width of about 200 ft for Vault 4.

Figure 2-3 illustrates locations of the existing Vaults, 1 and 4, and the aquifer model mesh. Figure 2-3 also shows the extent of the aquifer flow and transport model (blue border) and the mesh resolution in the horizontal plane (light gray dashes). Particle tracking results starting from the four corners of the combined facility indicate the groundwater flow direction. Time markers (red dots) are shown every 10 years of travel. Figure 2-3 indicates a possibility of plume overlap, which is the subject of a sensitivity study presented in Section 7.

2.2 Results

The magnitude and time of maximum concentration, the Maximum Contaminant Level (MCL) (USEPA 2004) and the Vault 4 inventory limit for the key radionuclides for two time periods of interest, 1000 years and 10,000 years, are given in Tables 2-1 and 2-2, respectively. These limits for the groundwater pathway are compared with limits derived for the other pathways and with the projected Vault 4 inventory in Section 7. For the projected Vault 4 inventory, none of the radionuclides produces a significantly large fraction of the groundwater limit.

Plots of fractional flux and concentration for each radionuclide modeled with PORFLOW are presented in Appendix A.

Consolidated NMSS Decommissioning Guidance

Characterization, Survey, and Determination of Radiological Criteria

Final Report

Division of Waste Management
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
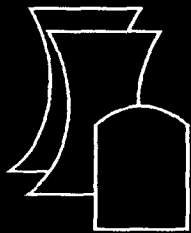
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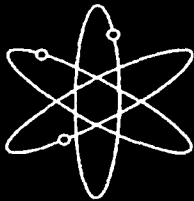
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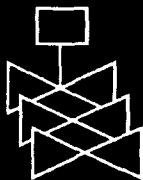
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NRC staff has observed that licensees have had difficulties in developing DQOs, especially during the optimization step, and have not taken full advantage of the DQO process. Experience has shown that the process is often rigidly structured by relying too much on characterization data and not readily open to the possibility of incorporating new information as it becomes available. This rigid approach makes implementing any changes difficult and is an inefficient use of resources, since it imposes time delays (e.g., the additional time required to determine how to implement any changes). Refer to Section O.2 from Appendix O of this volume, for guidance on lessons learned regarding use of the DQO process related to recently submitted decommissioning plans.

3.3 INSIGNIFICANT RADIONUCLIDES AND EXPOSURE PATHWAYS

> **Licensees should note that they are required to comply with the applicable dose criteria; nothing in this discussion should be interpreted to allow licensees to exceed the criteria.**

This section provides guidance on conditions under which radionuclides or exposure pathways may be considered insignificant and may be eliminated from further consideration. The dose criteria in 10 CFR Part 20, Subpart E, apply to the total dose from residual radioactivity. Thus, demonstrations of compliance should generally address the dose from all radionuclides and all exposure pathways. However, NRC staff recognizes that there may be large uncertainties associated with survey data and with dose assessment results. In a risk-informed, performance-based paradigm, NRC staff has determined it is reasonable that radionuclides or pathways that are insignificant contributors to dose may be eliminated from further detailed consideration.

NRC staff considers radionuclides and exposure pathways that contribute no greater than 10 percent of the dose criteria to be insignificant contributors. Because the dose criteria are performance criteria, this 10 percent limit for insignificant contributors is an aggregate limitation only. That is, the sum of the dose contributions from all radionuclides and pathways considered insignificant should be no greater than 10 percent of the dose criteria. No limitation on either single radionuclides or pathways is necessary. In cases of restricted release, where two dose criteria apply (one for the possibility of restrictions failing), the 10 percent limitation should be met for each dose criterion.

Once a licensee has demonstrated that radionuclides or exposure pathways are insignificant, then (a) the dose from the insignificant radionuclides and pathways must be accounted for in demonstrating compliance, but (b) the insignificant radionuclides and pathways may be eliminated from further detailed evaluations. For example, after sufficient site characterization, suppose a licensee shows that the dose from Sr-90 at the facility is 0.02 mSv/y (2 mrem/y), which is less than 10 percent of the dose criterion for unrestricted use. In this case, Sr-90 can be considered insignificant and eliminated from the FSS and from detailed consideration in the dose modeling. However, the dose from Sr-90 has to be considered in demonstrating compliance with the dose criterion.

It is important that the licensee documents the radionuclides and pathways that have been considered insignificant and eliminated from further consideration and that the licensee justifies the decision to consider them insignificant. However, licensees and NRC staff should be aware that remediation techniques (or other activities or processes) may increase concentrations above those previously deemed insignificant. Thus, licensees should also demonstrate that the concentrations deemed insignificant will not increase from other activities. Refer to Section O.1 from Appendix O and Questions 1 and 2, all of this volume, for guidance on which radionuclides can be considered and deselected from further consideration, respectively.

Summary of Determining Insignificant Radionuclides and Exposure Pathways

- Licensees may eliminate insignificant radionuclides and exposure pathways from further detailed consideration. However, the dose from the insignificant radionuclides and pathways must be accounted for in demonstrating compliance with the applicable dose criteria.

- Insignificant means no greater than 10 percent of applicable dose criterion.

- Ten percent is an aggregate limit; total dose contributions of all radionuclides and all exposure pathways considered insignificant should not exceed the 10 percent limitation.

- No additional limit on single radionuclides or pathways.

- Licensees should also address potential for concentrations to increase during remediation activities.

3.4 CONSIDERATIONS FOR OTHER CONSTRAINTS ON ALLOWABLE RESIDUAL RADIOACTIVITY

There can be situations or standards other than the dose criteria and ALARA requirements of Subpart E that may constrain the final dose below 0.25 mSv/y (25 mrem/y). There are two main causes for constraining the Subpart E dose limit: these causes are (1) partial site release and (2) other standards or regulations.

Partial site release is a situation where a licensee releases a portion of its site for unrestricted use prior to terminating the entire license. While the licensee should demonstrate that the residual radioactivity at the time of unrestricted release of the specific area meets the Subpart E dose limit, the residual radioactivity of the area should also be taken into account during final termination to demonstrate that the entire site met the appropriate release criteria. Dose modeling considerations for partial site release are discussed in Appendix K of this volume. In general, the comments below are also applicable to partial site releases.

Demonstrating compliance with the Subpart E dose limit does not eliminate the licensee's requirement for meeting other applicable Federal, State, or local rules and regulations. These regulations from other governmental agencies may conflict with the requirements of Subpart E,

RESPONSE TO RAI COMMENT 12
ROADMAP TO REFERENCES

REFERENCED DOCUMENT	*EXCERPT LOCATION	REMARK
Barnes and Flach 2005	Representative excerpt enclosed following response (WSRC-TR-2005-00085)	Executive Summary enclosed, full document has been supplied
Barton 2005	S-112 Retrieval Progress Presentation enclosed following response	
Churnetski 1981	Representative excerpt enclosed following response (DPST-81-366)	Summary to document enclosed, full document has been supplied
d'Entremont and Drumm 2005	Table A-12, excerpt enclosed following response (CBU-PIT-2005-00013)	
Drumm and Hopkins 2003	Table 13, excerpt enclosed following response (WSRC-TR-2001-00559)	
Drumm and Tran 2004	Table 9, excerpt enclosed following response (CBU-SPT-2004-00157)	See Average salt composition column
Flach 2003	Figure 7, excerpt enclosed following response(WSRC-TR-2003-00080)	
Flach 2004	Figure 3, excerpt enclosed following response (WSRC-TR-2003-00533)	
Fowler 1980	Table II, excerpt enclosed following response (DPST-80-569)	Per Table II, Tank 19 Saltcake Insoluble Residue was 51 g/L (= 51,000 mg/L). Concentration in dissolved salt calculated assuming 3:1 dilution required for salt dissolution (51,000/3 = 17,000 mg/L).
Fowler 1981a	Page 10, excerpt enclosed following response	$(1.52 \text{ g/mL}) \times (1.3 \text{wt\%} / 100) \times (1000)^2 = 19{,}800$ mg/L. Concentration in dissolved salt calculated assuming 3:1 dilution required for salt dissolution (19,800/3 = 6,600 mg/L).

Fowler 1981b	Page 21, excerpt enclosed following response	Insolubles = <0.1 g/L = <100 mg/L
Fowler 1981c	Page 22 left margin, excerpt enclosed following response	Wt. filter unit = wt. unit plus solids. Therefore, no insolubles detected.
Fowler 1982	Table I, excerpt enclosed following response (DPST-82-871)	$(1.30 \text{ g/mL}) \times (2.1\text{wt}\% / 100) \times (1000)^2 = 27{,}300 \text{ mg/L}$
Gilliam 2005 (ref. on pg 54 of response)	Page 10, excerpt enclosed following response (X-CLC-H-00546)	% Solids removed is calculated by comparing initial concentration to concentration at 18 and 30 days.
Gilliam 2005 (ref. on pg 64 of response)	Table page 10 included within response (X-CLC-H-00546)	
Goodlett 1968	Representative excerpt enclosed following response (DP-1135TL)	Introduction and summary to document enclosed, full document has been supplied
Handy 1975	Representative excerpt enclosed following response (ARH-C-6)	Summary to document enclosed, full document has been supplied
Hester 2004	Reference enclosed following response (X-ESR-G-00004)	
Hill 2005	Reference enclosed following response (CBU-PIT-2005-00127)	
Kirk 1980	Representative excerpt enclosed following response (RHO-CD-925)	Introduction and summary to document enclosed, full document has been supplied
Kiser 1979	Reference enclosed following response (DPST-79-269)	
Mahoney and d'Entremont 2004	Excerpt enclosed following response	
McCabe 1995	Representative excerpt enclosed following response (WSRC-TR-95-0337)	Summary to document enclosed, full document has been supplied

RESPONSE TO RAI COMMENT 12
ROADMAP TO REFERENCES

National Research Council 2000	Page 84, excerpt enclosed following response	
Norton et al. 2003	Reference enclosed following response (CBU-SPT-2003-00141)	
Pike 2005	Pages 11-15, excerpt enclosed following response (CBU-PIT-2005-00081)	Lowest residual based on case 5. Residual volume = 150 Kgal. (Saltcake Volume = 1.1 Mgal)
Poirer 2000	Reference enclosed following response (WSRC-TR-2000-00288)	
Porier et al. 2001	Reference enclosed following response (WSRC-TR-2001-00555)	
Reboul 2005	Table 1, excerpt enclosed following response (CBU-PIT-2005-00141)	"Portion of Total Salt Waste Inventory in Supernate" in Table 12-1 of response was calculated by dividing Soluble Curies by Total Curies
Seufert and Norton 2003	Reference enclosed following response (CBU-SPT-2003-00157)	
Simmons 1995	Representative excerpt enclosed following response (PNL-10803/UC-600)	Executive Summary enclosed, full document has been supplied
SRNL 1997	Page 1, excerpt enclosed following response (SRT-EDS-970022)	
Staheli and Peters 1998	Representative excerpt enclosed following response (GL-98-3)	Portion of Introduction to document enclosed, full document has been supplied
Staheli and Peters 1998 (ref. on pg 55 of response)	Page 5 and Table 8, excerpt enclosed following response, (GL-98-3)	Range of residual is calculated as follows: 22% * 0.29 = 6.4% to 22% * 0.52 = 11%

RESPONSE TO RAI COMMENT 12
ROADMAP TO REFERENCES

Tran 2005	Reference enclosed following response (CBU-PIT-2005-00050)	
Van Pelt 2000	Reference enclosed following response (WSRC-TR-2000-00287)	
Walker and Hamm 1983	Page 6, Table I.A, excerpt enclosed following response (DPST-83-695)	Concentration is calculated by multiplying the average Insoluble Solids concentration of the three Tank 24 samples (79 ppm) by the average Density (1.30 g/mL) and converting to mg/L
West 1982	Page 1. Reference enclosed following response	
Wiersma 1996	Representative excerpt enclosed following response (WSRC-TR-96-0160)	Summary to document enclosed, full document has been supplied
WSRC 1998	Page 60, excerpt enclosed following response (WSRC-RP-98-00162)	

*Excerpt Locations:
1. Excerpt included within response: The excerpt is included within the text of the response or is appended to the response.
2. Excerpt enclosed following response: The excerpt is enclosed on a separate sheet or sheets following the response.
3. Representative excerpt(s) enclosed following response: Representative excerpts from a document that is wholly or largely applicable are enclosed following the response.
4. Other

MODEL RESULTS OF THE DRAINING OF TANK 25 INTERSTIAL LIQUID (U)

C. D. Barnes
G. P. Flach

APPROVED for Release for
Unlimited (Release to Public)

REPORT DATE FEBRUARY 14, 2005

SRNL

SAVANNAH RIVER NATIONAL LABORATORY

™

1.0 EXECUTIVE SUMMARY

SRNL was tasked to simulate the draining of interstitial liquid from Tank 25 saltcake which is scheduled to take place in 2005. The salt processing plan baseline[1] identifies a target of 135,000 gallons of interstitial liquid to be removed from Tank 25. Due to the uncertainty of the Tank 25 material properties and conditions, several cases were modeled varying the saltcake and interstitial liquid properties. The cases present a wide range of performance. The nominal baseline, case 1, removed the 135,000 gallons in approximately 1,030 hours of pump operation. The cases with optimal drain characteristics (high intrinsic permeability, high temp.) drain the 135,000 gallons in less time. Those with less favorable drain conditions did not approach the 135,000 gallons in a reasonable amount of time. Common to all cases unable to achieve the target volume was the low temperature at which they were run, 30°C (the lowest modeled), though there were additional contributing factors. A summary of the results are shown in Table 1.

Table 1. Summary of Tank 25 Drain Model Results

Case	Time (hrs.)	Volume Removed (gal.)	Volume Remaining (gal.)	Pump Rate (at given time, gpm)	Time to Drain 135k gals. (hrs)
1 (nominal)	500	126,816	211,626	1.05	1,030
	1030	135,535	202,907	intermittent	
2	500	103,567	234,875	1.16	NA (>1,500)
	1500	120,476	217,967	Intermittent	
3	500	134,391	204,051	0.98	550
	836	139,761	198,681	intermittent	
4	450	74,505	263,937	0.98	NA (»1,530)
	1530	89,564	248,878	intermittent	
5	500	180,963	157,479	1.31	180
6	500	95,273	186,762	1.01	NA (»500)
7	500	144,107	303,675	1.11	385

TECHNICAL DIVISION
SAVANNAH RIVER LABORATORY

DPST-81-366

ACC. NO. 135891

CC: L. M. Papouchado
      R. E. Eibling
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      E. W. Holtzscheiter
      D. W. Jones
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      File TIS (2)

April 7, 1981

TO: J. F. ORTALDO, 773-A

FROM: B. V. CHURNETSKI, 676-1G

BENCH SCALE SIMULATION OF WASTE REMOVAL IN TANKS CONTAINING ALTERNATING LAYERS OF SALT AND SLUDGE

Summary

Two type IV tanks, 17F and 22H, have alternating layers of salt
and sludge. Waste removal from these tanks was simulated in the
Savannah River Laboratory using four liter and twelve liter
beakers. The proposed mode of removal involves the stepwise
insertion of slurry pumps to agitate the salt and sludge layers.
Between agitation steps the sludge would be allowed to settle, then
the salt rich supernate would be decanted.

The four liter beaker simulation predicted removal of 94% of the
salt in three dissolutions with a time requirement of about six
months. The twelve liter beaker simulation predicted removal of
99.5% of the salt in six dissolutions with a time requirement
of about eight and a half months. In both simulations, settling
of the simulated sludge constituted a considerable percentage of
the total time requirement. To date, a correlation between the
settling behavior of actual sludge and that of simulated sludge
does not exist. This precludes an accurate determination of the
amount of time required for settling.

Background

As part of the tank retirement program, both salt and sludge are
to be transferred from 22 older waste tanks to new stress relieved
tanks. The sludge will be suspended using long shafted, centrifugal
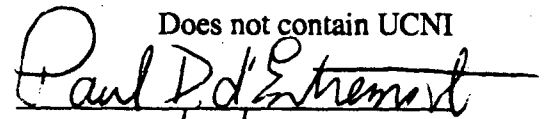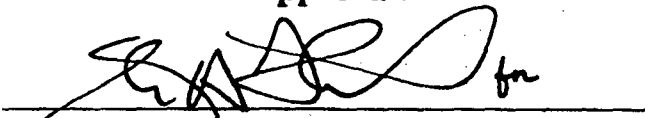
6/21/05   ADC/RO

Radionuclide Concentrations in Saltstone

P. D. d'Entremont, 766-H
M.D. Drumm, 766-H

Approval:

L. B. Romanowski, Manager,
CBU Salt Disposition Planning

APPROVED for Release for
Unlimited (Release to Public)

Ci	DDA	ARP/MCU	SWPF	Total
Sb-126	6.91E-01	8.68E-01	6.15E+01	6.30E+01
Sb-126m	4.93E+00	6.20E+00	4.39E+02	4.50E+02
Sm-151	3.00E+02	5.93E+01	4.19E+03	4.55E+03
Eu-152	1.45E+00	2.87E-01	2.03E+01	2.20E+01
Eu-155	1.70E+01	3.35E+00	2.37E+02	2.57E+02
Ra-226	2.44E-01	2.13E-08	1.27E+01	1.30E+01
Ra-228	6.52E-06	2.80E-07	1.04E-01	1.04E-01
Ac-227	1.40E-06	3.42E-08	1.77E-05	1.91E-05
Th-229	2.80E-03	2.52E-05	4.70E-03	7.53E-03
Th-230	1.49E-03	2.60E-06	3.38E-02	3.53E-02
Pa-231	3.90E-06	9.49E-08	4.92E-05	5.32E-05
Pu-244	1.16E-05	1.21E-07	7.85E-04	7.96E-04
Am-243	4.85E-03	9.58E-04	1.47E-02	2.05E-02
Cm-242	4.46E-03	2.37E-03	9.85E-02	1.05E-01
Cm-243	2.84E-03	5.61E-04	2.33E-02	2.67E-02
Cm-247	5.48E-13	1.08E-13	4.49E-12	5.15E-12
Cm-248	5.71E-13	1.13E-13	4.68E-12	5.36E-12
Bk-249	4.17E-20	8.23E-21	5.81E-19	6.31E-19
Cf-249	3.16E-12	6.24E-13	4.41E-11	4.79E-11
Cf-251	1.08E-13	2.14E-14	1.51E-12	1.64E-12
Cf-252	3.51E-15	6.93E-16	4.90E-14	5.32E-14

Table A- 12: Concentrations Sent to Saltstone

Ci/gal	DDA	ARP/MCU	SWPF	Total
H-3	2.33E-04	2.03E-04	6.91E-05	8.68E-05
C-14	6.94E-06	6.05E-06	4.56E-06	4.80E-06
Co-60	4.96E-06	2.09E-06	6.00E-07	1.01E-06
Ni-59	4.78E-08	1.17E-08	2.44E-08	2.61E-08
Ni-63	1.02E-05	5.06E-06	1.47E-06	2.31E-06
Se-79	1.11E-07	4.32E-07	9.06E-07	8.25E-07
Sr-90	6.14E-04	1.35E-04	1.39E-05	6.88E-05
Y-90	6.14E-04	1.35E-04	1.39E-05	6.88E-05
Nb-94	3.38E-12	3.40E-12	7.12E-12	6.70E-12
Tc-99	3.82E-05	1.60E-04	3.36E-04	3.06E-04
Ru-106	2.71E-06	1.10E-05	2.32E-05	2.11E-05
Rh-106	2.71E-06	1.10E-05	2.32E-05	2.11E-05
Sb-125	1.24E-05	4.48E-05	9.39E-05	8.56E-05
Sn-126	5.30E-07	2.18E-06	4.58E-06	4.17E-06
I-129	2.21E-08	6.62E-08	1.81E-07	1.65E-07
Cs-134	2.63E-04	9.20E-05	5.79E-08	2.51E-05
Cs-135	4.53E-07	1.56E-07	9.84E-11	4.33E-08
Cs-137	1.31E-01	4.58E-02	2.89E-05	1.25E-02
Ba-137m	1.24E-01	4.34E-02	2.73E-05	1.18E-02
Ce-144	4.44E-08	2.88E-08	6.03E-08	5.81E-08

Feed Basis for Processing Relatively Low Radioactivity Waste Tanks

M. D. Drumm
M. D. Hopkins

APPROVED for Release for
Unlimited (Release to Public)

Westinghouse Savannah River Company
Savannah River Site
Aiken, South Carolina 29808

SRS
SAVANNAH RIVER SITE

ENGINEERING DOC. CONTROL-SRS

00722735

Table 13: Comparison of Real Salt Cake Sample Results with Predicted Compositions

	MW	Tank 38 (avg. Sp.G. = 1.725) Ave. Sample Weight %	Sample Conc (M)	Average Sample Conc @6.44 M Na (M)	WCS Projected Feed @6.44 M Na (M)	Tank 41 - 4 Samples (Avg. Sp.G. = 1.53) Average Sample Weight Frac.	Sample Conc (M)	Average Sample Conc @6.44 M Na (M)	WCS Projected Feed @6.44 M Na (M)	Tank 37 (Avg. Sp.G. = 1.524) Average Sample Weight %	Sample Conc (M)	Average Sample Conc @6.44 M Na (M)	WCS Projected Feed @6.44 M Na (M)
Na	22.98977	2.98E+01	2.24E+01	6.44E+00	6.44E+00	2.75E-01	1.83E+01	6.44E+00	6.44E+00	2.87E+01	1.91E+01	6.44E+00	6.44E+00
Al(OH)4	26.98154	< 2.01E-01	1.29E-01	3.70E-02	4.01E-01	9.60E-03	1.55E-01	5.45E-02	3.91E-01	7.40E-01	4.18E-01	1.41E-01	3.93E-01
CO3	60.0092	—	—	—	1.35E-01	7.03E-02	1.79E+00	6.31E-01	1.35E-01	—	—	—	1.30E-01
C2O4	88.0196	—	—	—	1.43E-02	1.00E-03	1.74E-02	6.13E-03	1.39E-02	—	—	—	1.40E-02
NO2	46.0055	—	—	—	7.45E-01	7.33E-03	2.44E-01	8.59E-02	7.39E-01	—	—	—	7.52E-01
NO3	62.0049	—	—	—	2.67E+00	4.41E-01	1.09E+01	3.83E+00	2.65E+00	—	—	—	2.61E+00
OH	17.0073	—	—	—	1.87E+00	1.45E-02	1.30E+00	4.60E-01	1.87E+00	—	—	—	1.91E+00
PO4	94.9714	—	—	—	1.58E-02	7.43E-03	1.20E-01	4.22E-02	1.58E-02	—	—	—	1.54E-02
SO4	96.0576	—	—	—	1.84E-01	5.57E-03	8.87E-02	3.13E-02	1.80E-01	—	—	—	1.78E-01
Ag	107.868	2.08E-03	3.33E-04	9.58E-05	4.38E-08	2.34E-06	3.32E-05	1.17E-05	3.37E-08	8.93E-03	1.26E-03	4.27E-04	3.34E-08
As	74.91	—	—	—	3.56E-06	1.95E-05	3.98E-04	1.40E-04	4.36E-05	—	—	—	6.58E-05
Ba	137.33	≤ 9.41E-04	1.18E-04	3.40E-05	7.47E-09	1.58E-06	1.76E-05	6.21E-06	—	< 3.03E-03	3.37E-04	1.14E-04	—
Ca	40.08	6.08E-02	2.62E-02	7.54E-03	3.24E-05	4.21E-05	1.61E-03	5.67E-04	3.03E-01	< 2.47E-02	9.40E-03	3.18E-03	3.01E-05
Cd	112.41	≤ 1.42E-03	2.18E-04	6.28E-05	7.30E-08	4.29E-07	5.84E-06	2.06E-06	1.06E-06	< 3.12E-03	4.23E-04	1.43E-04	1.93E-06
Cl	35.453	—	—	—	3.86E-02	2.10E-01	9.06E-01	3.19E-01	3.78E-02	—	—	—	3.75E-02
Co	58.9332	< 5.67E-03	1.66E-03	4.78E-04	1.99E-07	9.56E-07	2.48E-05	8.75E-06	2.63E-07	< 1.40E-02	3.62E-03	1.22E-03	3.54E-07
Cr	51.996	7.06E-03	2.34E-03	6.75E-04	2.01E-04	7.18E-04	2.11E-02	7.44E-03	2.20E-04	< 1.70E-02	4.99E-03	1.69E-03	2.00E-04
Cu	63.546	< 1.89E-03	5.13E-04	1.48E-04	2.56E-05	1.97E-06	4.74E-05	1.67E-05	6.31E-05	< 6.64E-03	1.59E-03	5.38E-04	1.14E-04
Fe	55.847	3.17E-01	9.79E-02	2.82E-02	1.95E-06	2.51E-04	6.88E-03	2.42E-03	2.90E-06	5.00E-02	1.36E-02	4.61E-03	5.15E-06
P	18.9984	—	—	—	3.63E-02	—	—	—	3.53E-02	—	—	—	3.52E-02
Hg	200.59	—	—	—	1.02E-06	6.18E-07	4.71E-06	1.66E-06	1.20E-05	—	—	—	1.24E-05
K	39.0983	—	—	—	4.73E-03	2.64E-04	1.03E-02	3.64E-03	5.08E-03	—	—	—	6.03E-03
Mg	24.305	1.60E-02	1.14E-02	3.27E-03	1.38E-07	1.04E-05	6.52E-04	2.30E-04	8.20E-08	3.21E-03	2.01E-03	6.80E-04	1.02E-08
Mn	54.93805	2.82E-02	8.85E-03	2.55E-03	5.98E-08	3.21E-06	8.94E-05	3.15E-05	1.66E-07	< 2.15E-03	5.95E-04	2.01E-04	2.44E-07
Nd	144.27	—	—	—	4.75E-06	—	—	—	1.59E-06	—	—	—	—
Ni	58.69	≤ 9.27E-03	2.72E-03	7.85E-04	4.13E-07	2.04E-06	5.32E-05	1.87E-05	5.87E-07	< 2.03E-02	5.28E-03	1.79E-03	8.30E-07
Pb	207.2	< 6.65E-02	5.54E-03	1.59E-03	2.97E-06	1.25E-05	9.23E-05	3.25E-05	—	< 1.56E-01	1.15E-02	3.88E-03	—
Ru	101.1	—	—	—	3.76E-06	—	—	—	7.38E-06	—	—	—	1.23E-05
Se	78.96	—	—	—	5.85E-05	1.19E-05	2.31E-04	8.13E-05	1.74E-04	—	—	—	1.46E-04
Si	28.0855	8.85E-02	5.44E-02	1.57E-02	1.42E-04	7.40E-05	4.03E-03	1.42E-03	—	5.21E-01	2.83E-01	9.55E-02	—
Sr	87.62	< 7.07E-04	1.39E-04	4.01E-05	3.23E-06	2.48E-07	4.33E-06	1.53E-06	7.95E-07	< 1.72E-03	3.00E-04	1.01E-04	—
Zn	65.39	< 2.88E-03	7.60E-04	2.19E-04	4.80E-05	8.80E-04	2.06E-02	7.26E-03	8.82E-05	< 4.65E-03	1.08E-03	3.66E-04	1.43E-04
Zr	91.224	2.05E-03	3.88E-04	1.12E-04	9.28E-07	1.86E-06	3.12E-05	1.10E-05	2.09E-06	9.57E-03	1.60E-03	5.40E-04	3.68E-06
Water	—	2.21E-01				1.18E-01				2.11E-01			
Insoluble Solids	—	0.795	13,700 mg/l	3,900 mg/l	600 mg/l	0.850	13,000 mg/l	4,580 mg/l	600 mg/l	0.678	25,794 mg/l	8,719 mg/l	600 mg/l

Tank 41 insoluble solids calculation base five sample average of 0.85 wt% insoluble solids at Sp. G. of 1.53 and 18.3M Na corrected to 6.44M Na.

T38   T41   T37

Westinghouse
Savannah River Company
Aiken, SC 29808

CBU-SPT-2004-00157, Rev. 0

Retention: Until dismantlement or disposal of
facility, equipment, system, or process
Disposal Auth.: DOE 14-1.c (1)
Tracking #: ASD 10067

July 16, 2004

Recommended Waste Characterization System (WCS) Chemical and Radionuclide Attribution to SRS Salt Solids

_____   7-22-04
Prepared by: M. D. Drumm, CBU Flowsheet Development & Optimization   Date

_____   7-22-04
Prepared by: H. Q. Tran, Salt Processing Engineering   Date

_____   7/26/04
Prepared by: J. R. Hester, F&H Process Support Engineering   Date

_____   07/22/04
Radionuclide Portion Reviewed by: Q. L. Nguyen, Salt Processing Eng.   Date

_____   7/22/04
Chemical Portion Reviewed by: J.A. Pike, CBU FD&O   Date

_____   8/24/04
Approved by: S. J. Robertson, Manager, CBU FD&O   Date

_____   8/3/04
Approved by: E. J. Freed, Manager, F&H Process Support Engineering   Date

Table 8: Combined Tank 41 Dry Salt Composition

Compounds	wt%
$NaNO_3$	83
$Na_2CO_3.H_2O$	11
$NaNO_2$	0.29
$NaAlO_2 \cdot 2H_2O$	0.72
$Na_2C_2O_4$	0.19
Na_2SO_4	3.7
NaCl	0.0070
NaF	0.26
NaOH	0.76
Na_3PO_4	0.61

All the dry salt data that was not considered anomalous was then averaged to give a dry salt composition to be used for all salt tanks. The current salt compositions[8] in WCS for F and H Area salt wastes are listed for comparison.

Table 9: Comprehensive Dry Salt Compositions

Sample (wt%)	Tank 2	Tank 3	Tank 10	Tank 38	Tank 41	AVG	WCS-F	WCS-H
NaNO3	88	97	94	85	83	86	66	50
Na2CO3·H2O	3.7	2.0	2.4	10	11	5.7	5.7	3.7
NaNO2	---	1.4	---	---	0.29	0.82	1.4	11
NaAlO2·2H2O	7.4	0.10	---	0.80	0.72	2.2	7.8	10
Na2C2O4	0.10	---	---	1.1	0.19	0.45	0.62	0.42
Na2SO4	0.90	---	3.8	3.7	3.7	2.9	9.5	5.8
NaCl	---	---	---	---	0.0070	0.0068	0.51	0.50
NaF	---	---	---	0.10	0.26	0.17	0.62	0.34
NaOH	---	---	---	---	0.76	0.73	7.4	16
Na3PO4	---	---	---	---	0.61	0.59	0.00040	0.0060
Total Na	27	27	28	29	29	28	30	32

RADIONUCLIDE COMPOSITION FOR DRY SALT CAKE

As with the chemical composition, the sample results are analyzed to determine the radionuclide composition of the solid and liquid phases. A list of major radionuclides as seen in Table 10 is chosen from the final reports[1-4, 10] for Tanks 2F, 3F, 10H, 29H, 38H, and 41H. Except for C-14, Sr-90, Cs-137, Pu-238, Pu-239/240, and Am-241 that were analyzed by the more accurate radiochemistry method, the other radionuclei species were obtained from ICP-MS method. Data shown for the radionuclide constituents analyzed by ICP-MS do not add up to exactly 100% due to tolerances in the method used. Tables 10 and 11 present the radionuclide concentrations in the wet saltcake and interstitial liquid/supernate for samples collected from undrained saltcake of Tanks 2F, 3F, 10H, 29H, 38H, and 41H (post-dissolution).

Additionally, the bulk of the Sr-90 and the actinides in these samples were found to be associated with dry salt, but this finding can be explained by the presence of insoluble solids (2.582E-03 weight fraction[8]), which WCS already attributes to salt. Note that the insoluble solid composition does not account for the C-14, U-235, or U-238. Carbon occurs most dominantly in the crystalline salt solids as carbonates and less so as oxalates. C-14 should be added to the salt composition in WCS. Uranium is known to precipitate when saltcake is formed via evaporation, thus, is present in saltcake as sodium diurinate.[13, 14] This component is generally considered part of the insoluble solids in WCS and, therefore, the composition of insoluble solids should be changed to include U-235 and U-238.

Recommended changes to WCS radionuclide composition are shown in Table 14. Data should be developed to support a similar determination for I-129, which is a constituent of interest to the Integrated Flowsheet Model and the Salt Waste Processing Facility design effort.

Table 12: Major Radionuclide Concentrations in Dry Saltcake

Tank	2F	3F		10H	29H	38H	41H	Average	WCS
pCi/mL	T2F-1-1	T3F-1-1	T3F-1-4	HTF-610	T29H-B6-1	HTF-E-03-114	HTF-E-03-146	pCi/mL	
^{99}Tc	1.13E+04	1.02E+05	5.98E+04	5.38E+04	6.75E+04	8.37E+03	5.04E+03	3.72E+04	0
^{135}Cs	5.87E+02	7.76E+02	8.82E+02	1.29E+02	4.62E+02	5.14E+01	2.35E+02	5.06E+02	0
^{230}Th	1.34E+02	2.03E+03	1.31E+03	6.24E+03	6.49E+02	1.80E+03	1.11E+03	1.28E+03	0
^{232}Th	5.62E-02	1.75E-02	1.07E-02	4.53E+00	1.30E-01	4.79E-01	5.05E-03	1.14E-01	0
^{233}U	6.44E+01	9.75E+02	6.26E+02	1.97E+04	3.75E+03	8.64E+02	1.49E+02	5.36E+02	0
^{234}U	4.16E+01	6.29E+02	4.04E+02	4.07E+03	5.00E+03	1.23E+04		3.35E+03	0
^{235}U	4.97E-01	3.20E-01	1.48E-01	2.91E+01	1.40E+01	1.38E+01		**3.70E+00**	0
^{236}U	3.02E+00	6.52E+00	4.18E+00	2.68E+02	1.88E+02	6.17E+01		1.89E+01	0
^{237}Np	4.69E+00	7.10E+01	4.56E+01	1.12E+03	1.39E+02	6.46E+02	2.29E+01	1.58E+02	0
^{238}U	1.42E+01	1.22E+01	6.73E+00	6.16E+01	9.75E+00	2.99E+02		**8.30E+01**	0
^{239}Pu	1.03E+04	6.26E+03	4.02E+03	2.71E+04	1.86E+04	6.13E+04	3.42E+03	1.71E+04	0
^{240}Pu	1.07E+04	2.30E+04	1.47E+04	7.04E+04	7.33E+03	5.04E+04	1.25E+04	2.23E+04	0
^{241}Pu	8.52E+05	1.04E+07	6.66E+06	3.18E+07	3.31E+06	9.20E+06	5.67E+06	6.55E+06	0
^{242}Pu	2.54E+01	3.85E+02	2.47E+02	1.18E+03	1.23E+02	1.16E+03	2.10E+02	4.06E+02	0
^{14}C	3.57E+03	7.09E+03	8.18E+03	4.79E+04	2.12E+03	1.09E+04	4.66E+02	**6.04E+03**	0
^{90}Sr	2.55E+06	3.59E+06	1.64E+06	1.16E+08	2.79E+07			4.48E+06	9.40E+07
^{137}Cs		3.96E+06	3.91E+07	3.10E+06			1.56E+07	2.15E+07	4.20E+06
^{238}Pu	2.20E+04	1.15E+04	6.17E+03	2.76E+06	2.37E+06	7.02E+06	2.66E+04	1.42E+06	1.20E+06
$^{239/240}$Pu	1.31E+04	4.71E+03	4.14E+03		3.64E+04		1.80E+03	2.97E+04	4.00E+04
^{241}Am	7.28E+03	2.71E+03	2.71E+03	5.07E+04	2.26E+04			4.23E+03	0

Porous Medium Analysis of Tank 41 Drain Operations (U)

G. P. Flach

APPROVED for Release for
Unlimited (Release to Public)

SRS

SAVANNAH RIVER SITE
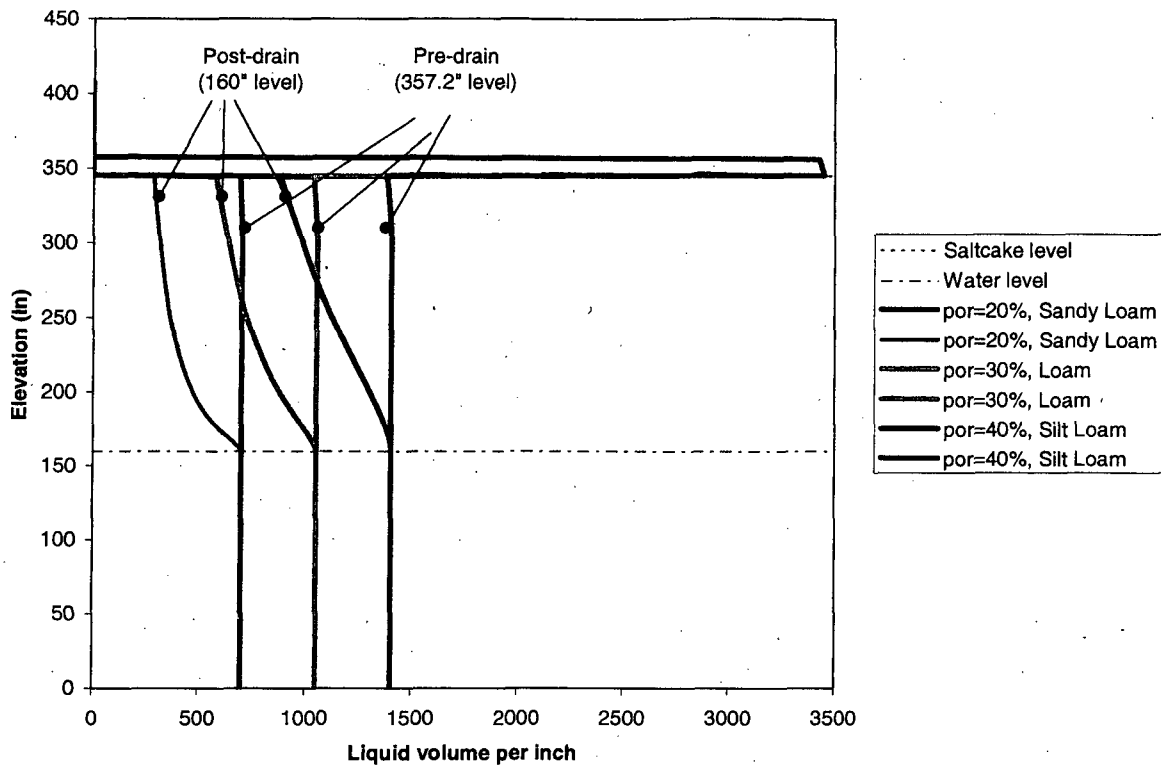
Figure 7   Liquid inventories in Tank 41 before and after pumping for three optimal parameter settings (345" saltcake level).

Porous Medium Analysis of Interstitial Liquid Removal from Tank 41 and Tank 3 (U)

SRS
savannah river site

Westinghouse Savannah River Company
Savannah River Site
Aiken SC 29808

Porous Medium Analysis of Interstitial Liquid Removal from Tank 41 and Tank 3 (U)

G. P. Flach

SRS
savannah river site

Westinghouse Savannah River Company
Savannah River Site
Aiken SC 29808

Interstitial liquid drainage

The prior study included an approximate analytical expression for estimating the time required to remove a specified volume of interstitial fluid, or equivalently to lower the average interstitial level from a specified initial height to a specified final elevation (cf. Flach 2003a, equation (39)). The drain time expression was based on an analytical flow model that had been calibrated to the well recovery event during downtime between 9/22/02 and 10/11/02 (cf. Flach 2003a, equation (32)). Figure 3 compares predicted interstitial liquid volume left to be drained, "gallons-to-go", as a function of pump operating time. The agreement is excellent through nearly 300 hours. After that intermittent pump operation cause poorer agreement, although still good except near the end. The analytical solution assumes pseudo steady-state, slow transient conditions, which was not the case when the sump pump operated intermittently. The comparison indicates that the analytical expression, once calibrated to initial drainage data, can be expected to produce accurate estimates of subsequent drain rates while pump operation is reasonably continuous.
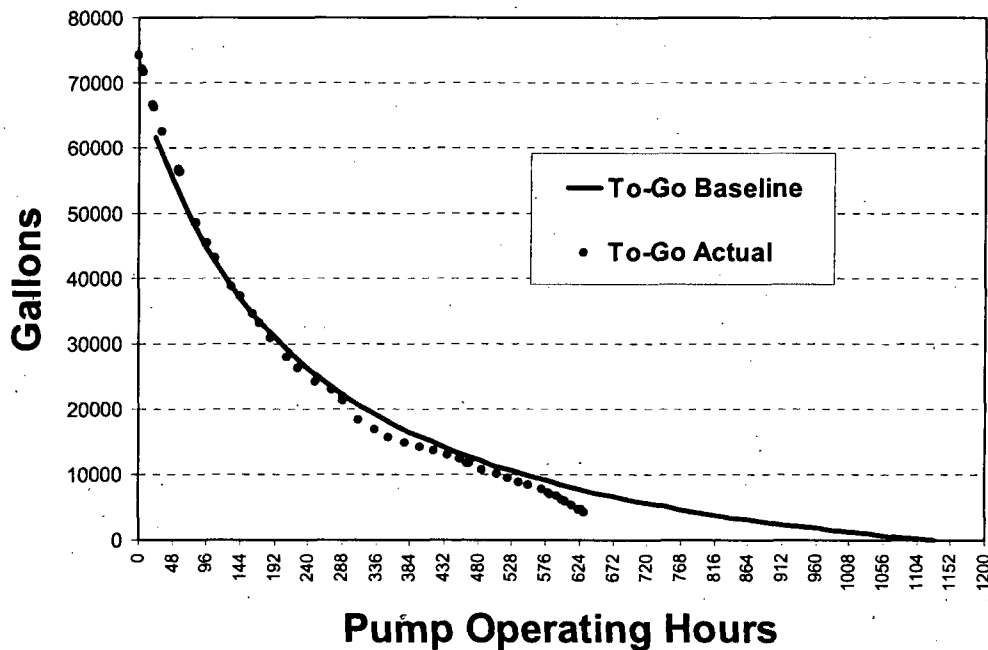
Figure 3   Predicted and actual drainage progress for Tank 41 from March through June 2003.

TECHNICAL DIVISION
SAVANNAH RIVER LABORATORY

DPST-80-569

ACC. NO. 143083

CC: D. L. McIntosh
J. A. Kelley
J. K. Okeson
R. B. Ferguson
E. W. Holtzscheiter
M. H. Tennant
M. D. Boersma
J. F. Ortaldo
J. R. Chandler
C. T. Randall
L. F. Landon
→ TIS File Copy (2)

MEMORANDUM

October 7, 1980

TO: M. J. PLODINEC

FROM: J. R. FOWLER

TIS FILE
RECORD COPY

ANALYSIS OF TANK 19 SALT CAKE

INTRODUCTION AND SUMMARY

Plans to retire waste tanks from service include several tanks that are presently
used to store salt cake. In support of the tank retirement program, a solid
sample of the salt cake in Tank 19F was obtained for physical, chemical and
radiochemical characterization. Analysis of this and other salt samples will
provide a data base for interim and long-term waste management programs.

DISCUSSION

Based on tests in the Shielded Cell Facility, 90 volume % (96 wt %) of the salt
cake was soluble in water using a 1.3:1 water-to-salt cake volume ratio to
approximate the 1.3-1.4 specific gravity projected for salt dissolution. The
specific gravity of the resulting solution was 1.41. Specific concentrations
of critical anions (NO_3^-, NO_2^-, OH^-) are shown in Table I.

The chemical composition of the salt cake is listed in Table II, assuming all
anions are present as sodium salts. Based on the water insoluble residue in
the sample, approximately 177 metric tons (dry basis) of sludge solids are
contained in the salt cake representing approximately 93,600 gallons of wet
sludge. This sludge is in addition to the 12,000 gallons already estimated to

be in Tank 19F. Only Cs-137 and Sr-90 were detected in the reconstituted salt solution (Table III). The insoluble solids in this sample were not analyzed.

Additional salt samples will be obtained from Tank 10H, 24H, 1F, 9H, 2F and 3F during the next six months[1] to provide additional information on waste salt. Solids content and radionuclide content of these samples will provide additional information for the DWPF flowsheet.

JRF:ln
Attachments

---

[1]W. L. West to G. M. Johnson, memo, "Salt Sampling," September 24, 1980.

Table I

SOLUTION PROPERTIES OF DISSOLVED SALT FROM TANK 19F

Specific Gravity,	1.41
NO_3^-,M	4.5
NO_2^-,M	0.096
Free OH^-,M	0.77
Insoluble Solids	10 vol. % of original salt cake
Solution Makeup	1.3 liters H_2O/liter of salt cake

Table II

CHEMICAL ANALYSES OF TANK 19 SALT CAKE

Component	g/ℓ of Salt Cake	Wt %[a]	Metric Tons in Tank[d]
$NaNO_3$	641	50.1	2270
$NaNO_2$	11	0.9	41
$NaAl(OH)_4$	186	14.5	658
Na_2SO_4	154	12.0	545
Na_2CO_3	44	3.4	156
NaOH	52	4.1	186
$Na_2C_2O_4$	1	0.1	4.5
NaF	1	0.1	4.5
NaCl	Matrix Interference[e]	-	-
Na_3PO_4	Trace	-	-
H_2O[b]	90	7.0	3.8
Insoluble Residue[c]	51	4.0	182
Total	1231	96.2	4365

[a] Based on measured bulk density of 1.28 kg/ℓ for the salt sample; sodium salts assumed for all anions identified.

[b] Wt loss on heating air-dried salt sample to 120°C.

[c] Insoluble residue was 10 volume % of original dry sample volume; dissolution of soluble salts will leave ~75,000 gallons of additional sludge.

[d] Assumes 3.54×10^6 liters (9.36×10^5 gallons) of salt cake in Tank 19 of the composition shown in this table.

[e] Presence of organic anion prevented determination of chloride content of sample, attempts to identify organic species were unsuccessful.

Table III

SOLUBLE RADIONUCLIDES IN TANK 19F SALT CAKE

Radionuclide	mCi/Liter of Salt Cake	Ci in Tank
Cs-137	45	1.6×10^5
Sr-90	0.21	7.4×10^2

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ATOMIC ENERGY DIVISION

SAVANNAH RIVER LABORATORY
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DPST-82-871-TL

CC: J. A. Kelley
    W. V. Wright
    J. R. Fowler

Acc. No. 150440

September 22, 1982

J. T. GRANAGHAN
SAVANNAH RIVER PLANT

ATTN: E. B. SNELL
      WASTE MANAGEMENT TECHNOLOGY

ANALYSES OF DISSOLVED SALT SOLUTION FROM TANK 24

In support of the demonstration of the precipitatation process to decontaminate soluble waste, the requested analysis of the feed solution from Tank 24 has been completed. Results are summarized in the attached memorandum.

W. R. Stevens, III
Waste Processing Technology

JRF:pmc
Att
Disc 4

TECHNICAL DIVISION
SAVANNAH RIVER LABORATORY

DPST-82-871

ACC. NO. 150440

CC: W. B. Stevens
     R. B. Ferguson
     L. M. Lee
     J. B. Pickett
     TIS File Copy (2)

September 22, 1982

TO: W. V. WRIGHT

FROM: J. R. FOWLER

ANALYSIS OF TANK 24 DISSOLVED SALT SOLUTION
(Sampled 6/82)

Summary

At the request of Waste Management Technology, chemical and
radiochemical analyses of a sample of Tank 24 dissolved salt
solution were obtained. These analyses were requested to support
the demonstration of soluble waste decontamination by the
precipitation process planned for the first quarter of 1983.
Results are shown in the attached Table I.

Details

The sample from Tank 24 was received June 17, 1982. The
sample was prepared for analyses by H. Edwards in High Level Cell
14 according to the procedure shown on page 62 of DPSTN-3302.
The tests in Cell 14 determined the sample density to be 1.30 g/ml,
and that this sample contained 2.1 wt% insoluble solids. These
insoluble solids were removed by vacuum filtration. The filtrate,
containing soluble waste components, was submitted to Analytical
Development Division (ADD) for analyses by their standard
analytical procedures. ADD quality assurance procedures as
outlined in DPST-81-595 were used to qualify the methods for these
analyses.

JRF:pmc
Att
Disc 4

TABLE I

ANALYSES OF TANK 24 DISSOLVED SALT SOLUTION

Physical Properties

Wt% Insoluble Solids   2.1 ⟵
Density, g/ml   1.30 ⟵

Chemical Analyses

Species	Moles/liter
Na^+	6.6
K^+	6.1E-3
Ca^{+2}	7.0E-5
NO_3^-	3.8
NO_2^-	0.40
Free OH^-	0.89
$CO_3^=$	0.34
$Al(OH)_4^-$	0.45
$SO_4^=$	0.21
Cl^-	0.012
$MO_4^=$	Not Detected
$CrO_4^=$	9.6E-4
$SiO_3^=$	<0.01

Radiochemical Analyses	mCi/liter
Cs-137	220
Ru-106	0.59
Tc-99	0.071
Se-79	0.0024
Sr-90	<background
Pu-238,239	0.075

Disc 4

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DPST-82-871-TL

CC: J. A. Kelley
W. V. Wright
J. R. Fowler

Acc. No. 150440

September 22, 1982

J. T. GRANAGHAN
SAVANNAH RIVER PLANT

ATTN: E. B. SNELL
WASTE MANAGEMENT TECHNOLOGY

ANALYSES OF DISSOLVED SALT SOLUTION FROM TANK 24

In support of the demonstration of the precipitatation process to decontaminate soluble waste, the requested analysis of the feed solution from Tank 24 has been completed. Results are summarized in the attached memorandum.

W. R. Stevens, III
Waste Processing Technology

JRF:pmc
Att
Disc 4

Salt from
Tank 1 #1 experiment

V. If the volume of residual solids is ≥ 1 ml, go to
Residual   step VI. If not add ~10 ml of salt to the acid residue
Solid >1ml and ~20 ml of H_2O, mix well, centrifuge 30 minutes, discard
the solution. Repeat until >1 ml of insoluble solids is in
the cone. Record the total volume of salt required to give ≥ 1 ml
solids. (No. of 10 ml portions of salt = 1 , total volume soln = 10.0).

cone broken
before weighed!

VI. Wash the residual solid with 15 ml H_2O using mix/centrifuge/decant
procedure. Discard wash. Dry solid at 120°C. Reweigh cone + solid.
Transfer solid to sample vial. Reweigh empty cone. (Wt cone + solid = 6.92   Wt cone = 6.85
Sample vial No. 14-316)   ~ 0.1 g

Bulk density: $\frac{13.6 g}{10.0 ml}$ = 1.36 g/ml wet

$\frac{12.7 g}{10.0 ml}$ = 1.27 g/ml dry

An additional salt sample will be carried through this procedure
because cone was broken before it was weighed after drying. Wt of
solid was estimated by weighing sample vial before adding
the solid to it, then reweighing the vial. Wt of solid
is very approximate.

Wt % H_2O: $\frac{13.6 - 12.7}{13.6} \times 100$ = 6.62 (from I)

The six samples from Step VI will be combined
by ACD into a single sample for analysis.
The solid sample will be dissolved by ACD using
the Na_2O_2 Technique for solid sample dissolution, then
analyzed.

Wt % insoluble solid: $\frac{0.1}{13.7} \times 100$ = 0.72 %
(from VI).

Vol % insoluble solid: $\frac{1.5}{10.0}$ = 15 vol % (from step III).

ACD obtained ~0.2 g sample ⟹ 1.4 wt % solid wet   cont'd on p. 6
~ 1.6 wt % dry

EXPERIMENTER [signature]   DATE 6/1/81
WITNESS   DATE

10   Fowler 1981a

DATE

SUBJECT OF EXPT. Tank 1 Salt Sample — Experiment #2   Results
(See p.5,6 this notebook for discussion)

Bulk density: $\dfrac{15.25g}{10ml}$ = 1.52 g/ml (wet salt) ⟵

$\dfrac{11.0}{10}$ = 1.10 g/ml dry

~1.3 wt % insoluble solid based on wet salt ⟵
~1.8 wt %   "   "   "   "   dry salt
% H₂O = $\dfrac{4+2}{15.2}$ = 27.6

ACD Sample No. 5482   Request dated 6/3/81
My sample ID 14-317 through 14-322 salt solution (see p.9 this NB)
obtained by dissolving salt from Tank 1 (10 ml salt) in distilled H₂O.

15.25 g wet salt dried to yield 11.0 g dry salt, dissolved in H₂O

Total solution volume = 40.5 ml. vol of insol. residue = 1.5 ml (15 vol %)

See p.6 for method of calculation. Factors for this sample are 0.0405 liters
and 15.25 g in original sample (wet) or 11.0 g (dry).

<u>Anion Analyses:</u>

Species Measured	Found Mole/l	equiv g Na Salt wet/dry	Wt % as Na Salt Wet/Dry orig. (redried)	Re RTI Avg
Cl⁻	0.019 (IC)	0.059 (SIE) 0.140	/1.27	0.7
OH⁻	0.155	0.155 0.251	/2.28	3.
Al(OH)₄⁻	0.117	0.117 0.539	/5.08	4
CO₃⁼	0.683	0.683 2.93	/26.6	2
NO₃⁻	2.56M (Recheck)	0.30	1.03 (5.81)	9.39 (50.2)	7.
NO₂⁻	≤0.4 M	n.d. (<3.20)	— (<1.12)	
SO₄⁼	0.144	0.0088	0.0506 (0.80)	/0.46 (7.32)	
F⁻	0.004	0.0148	0.025 (.0068)	/8.22 (.004)	
C₂O₄⁼	n.d.	—		
PO₄⁼	n.d.	—		
SiO₃⁼				
I⁻				
Hg O(OH)⁻	<2×10⁻⁸	<6 ×10⁻⁴ g	0.001	
CrO₄⁼	1.44×10⁻³	1.40E-3	0.086	
Ag(OH)₂⁻				
MoO₄⁼	3.44×10⁻⁴	2.87E-3	0.026	

* Bracketed #'s not used — obviously too low

DATE
SUBJECT OF EXPT. Analysis of Tank 18 Supernate – Transferred from Tank 19

LCD No. 51-8175

$\rho = 1.20 \text{ g/ml}$

$$\frac{Final}{} \times \frac{Dilution}{factor} = Original, \text{ moles/l}$$

NO_3^-	$= 1.49 M$	$\times 1.95 =$	2.91
NO_2^-	$= 0.03 M$		$= 0.059$
$SO_4^=$	$= 0.12 M$		$= 0.23$
F^-	$= 0.04 M$		$= 0.078$
I^-	$= < 0.004 M$		
Cl^-	$= 0.03 M$		$= 0.059$
$Al(OH)_4^-$	$= 0.101 M$		$= 0.20$
Free OH$^-$	$= 0.396 M$		0.77
$CO_3^=$	$= 0.032 M$		0.062
$Pu\alpha$	$= 2.27 E3$	$4.04 E3 \text{ d/m/ml}$	$= 0.0018 \text{ mCi/l}$
$Tc-99$	$= 1.4 E4$	$2.7 E4 \text{ d/m/ml}$	$= \cancel{1.2 E-5 \text{ mCi/l}}$ 0.012 mCi/l
$Sr-90$	$= 1.63 E5$	$3.18 E5 \text{ d/m/ml}$	$= 0.14 \text{ mCi/l}$
$Se-79$	$= 1.8 E3$	$3.51 E3$	$= 0.00016 \text{ mCi/l}$ 34 mCi/g
Cu	$= 2.31 E-5 M$	$4.5 E-5$	
Mo	$< 6 \times 10^{-6} M$	$1.2 E-5$	
Cr	$= 1.32 \times 10^{-4} M$	$2.57 E-4$	
Na	$= 2.15$	4.19	
K	$= 6.10 \times 10^{-3} M$	$1.19 E-2$	
Hg	$= 5 \times 10^{-8} M$	$9.8 E-8$	
Ag	$= < 1.6 E-6$	$< 3.1 E-6$	
Zn	$= < 2.7 E-3$	$< 5.27 E-3$	
Insoluble	< 0.1	$< 0.1 g$	

Salt Composition Tank 18		
Assumed Species	g/l	Wt %
$NaNO_3$	247.4	69.94
$NaNO_2$	4.07	1.15
Na_2SO_4	32.66	9.2
NaF	3.28	0.93
$NaCl$	3.45	0.98
$NaAl(OH)_4$	23.60	6.67
$NaOH$	30.80	8.70
Na_2CO_3	6.57	1.85
$CaSO_4$	0.0061	0.00
Na_2MoO_4	0.0025	0.00
Na_2CrO_4	0.0412	0.011
KNO_3	1.20	0.33
$NaHgO(OH)$	2.5 E-5	0.000
$NaAg(OH)_2$	$< 5.1 E-4$	< 0.000
Na_2SiO_3	0.641	0.1
Insoluble	$< 0.1 g$	< 0.0
Total g/l	353.721	

EXPERIMENTER
WITNESS

DATE 10/30/81
DATE

22   Tank No. **20**   Date Received _____ Date Prepared _____

Procedure   Dissolved Salt Solution

① Weigh sample bottle + lid + sample, record wt

   Wt = 425.7g.   ←—

② Transfer entire contents to 250 ml graduated cylinder (preweighed).
   Reweigh graduate + sample. Reweigh empty sample bottle + lid
   Record sample volume in graduate.   density = 1.36

   Wt Graduate = ___   Volume of sample = 93.6 ml
   Wt Graduate + sample = ___   wt sample = 372.5 g

③ Standardize (properly) _____ pH buffer. Check pH of solution
   in graduate   pH =

④ Rinse sample bottle three times with distilled water (20 ml portions).
   Add all three _____ to the sample in the graduate. Reweigh ____
   containing sample plus rinses and record wt. Record total volume
   in graduate. Reweigh sample bottle + lid after rinsing.

   wt graduate + sample + rinses = 453.5 g   wt bottle + lid = 372
   ∴ total volume sample + rinses = 165.5 ml   after rinsing
                                               Dilution factor = 2.02

⑤ Pour contents of graduate into a poly bottle labeled with tank No.
   and date received. Mix well by shaking. Return contents of
   bottle to the graduate.

⑥ Filter entire contents of graduate using a preweighed 0.45μ
   disposable filter unit (will probably ~~require~~ need to do in
   3 or 4 increments). Pour clean filtrate into the poly bottle used
   in step 5. After transferring all of the filtrate to the bottle,
   rinse the solids on the filter with two 10 ml portions
   distilled water. Discard the ~~Dry the~~ Place
   unit in the oven at 50°C to dry ~~overnight~~. Reweigh

wt filter
unit = ___

wt unit
plus solid
= ___

⑦ Take 3 liquid samples from the poly bottle. Transfer
   dried solid from the filter unit to a sample bottle.

Liquid samples: 14-___, 14-___, 14-___. Solid sample

Calculation Cover Sheet

Project N/A	Calculation No. X-CLC-H-00546	Project No. N/A
Title Settling of Insoluble Solids in Supernate from Salt Dissolution	Functional Classification PS	Sheet 1 of 12 /4 _Jul 3/31/05_
	Discipline Chemical Process	

Calc Level ☒ Type 1 ☐ Type 2	Type 1 Calc Status ☐ Preliminary ☒ Confirmed

Computer Program No. Microsoft EXCEL ☐ N/A	Version/Release No. 10.0.6501.0

Purpose and Objective
Provide a method to project settling rates for insoluble solids suspended in supernate resulting from the dissolution of salt. This will initially be applied towards settling of Tank 41 dissolved salt transferred to Tank 49. Projection of settling rates will allow for planning of settling times and decant pump placement.

Summary of Conclusion
The calculation method estimates, based on SRNL settling data from Tank 41 salt, the amount of supernate available at any target composition and any settling period. Inputs to the calculation are the tank level, initial insoluble solids concentration, the desired concentration for decanting, estimated solids density, and an estimated initial settled solids height. The calculation is presented in spreadsheet form. An electronic version of the spreadsheet is available from the author.

	Revisions
Rev No.	**Revision Description**
0	Original Issue

		Sign Off		
Rev No.	**Originator (Print) Sign/Date**	**Verification/ Checking Method**	**Verifier/Checker (Print) Sign/Date**	**Manager (Print) Sign/Date**
0	JEFFREY M. GILLAM _Jeffrey M. Gillam_ 3/31/05	Design Review	Mark Hopkins _Mark Hopkins_ 4/7/05	DC Bridgewater _DC Bridgewater_ 4/7/05

Design Authority — (Print) M. J. AUGAM	Signature	Date 4/20/05
Release to Outside Agency – (Print)	Signature	Date

Security Classification of the Calculation
Unclassified

FIGURE 1: EXAMPLE SETTLING CALCULATION

	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P	Q
4		Settling rates per WSRC-TR-95-0249:						Conservative settling rates:								
6								If fract </=	0.667	in/day	=	-42.00	*	frac. settled	+	37.00
7								If fract >/=	0.667	in/day	=	-25.95	*	frac. settled	+	26.30
9			DF	Fraction of solids	Rate, in./day*			Fraction of		Rate, in./day						
10				removed	spg 1.194	spg 1.399		solids removed		Conservative prediction						
11								0		37.00						
12			2	0.500	16	23		0.500		18.00						
13			3	0.667	9.0	11		0.667		9.00						
14			4	0.750	6.8	8.7		0.750		6.84						
15			5	0.800	5.8	6.1		0.800		5.54						
16			10	0.900	3.6	3.2		0.900		2.95						
17			15.5	0.935	2.9	2.7		0.935		2.03						
18			20	0.950	2.1	1.8		0.950		1.65						
19			28	0.964	1.7	1.3		0.964		1.28						
20			40	0.975	1.3	1.0		0.975		1.00						
21								1		0.35						
22			* bold defines conservative settle rates for both densities													
26		Tank level, in. =			285	Beginning settled heel, in =	2.00	Tank fill factor, gal/in - =				3510				
27		Initial conc I.S., mg/L =			550	Pump/jet buffer distance, in =	18	Minimum pump/jet suction level, in -				26.5				
28		Target conc., mg/L =			180.00	Bounding level for hindered settling	6									
29		Applicable settling rate, in/day =		8.843												

	Volume or Elevation vs. Time for Fixed Target Concentration							Concentration vs Time for a Fixed Elevation					
	Target conc. mg/L	Applicable settling rate, in/day	Days of settling	Interface height, in	Minimum pump/jet suction level, in.	Gal available for transfer		Concentration, mg/L	Applicable settling rate, in/day	Days of settling	Tank Elevation, in.	Minimum pump/jet suction level, in.	
37	180.00	8.843	3.0	258.47	276.47	29934		550.00	37.000	1.0	10.00	28.00	
38	180.00	8.843	6.0	231.94	249.94	123048		550.00	37.000	3.0	10.00	28.00	
39	180.00	8.843	9.0	205.42	223.42	216163		550.00	37.000	6.0	10.00	28.00	
40	180.00	8.843	12.0	178.89	196.89	309277		465.61	30.556	9.0	10.00	28.00	
41	180.00	8.843	15.0	152.36	170.36	402391		365.58	22.917	12.0	10.00	28.00	
42	180.00	8.843	18.0	125.83	143.83	495505		305.56	18.333	15.0	10.00	28.00	
43	180.00	8.843	21.0	99.30	117.30	588619		265.54	15.278	18.0	10.00	28.00	
44	180.00	8.843	24.0	72.77	90.77	681733		238.96	13.095	21.0	10.00	28.00	
45	180.00	8.843	27.0	48.25	64.25	774848		215.53	11.458	24.0	10.00	28.00	
46	180.00	8.843	30.0	19.72	37.72	867962		198.85	10.185	27.0	10.00	28.00	
47	180.00	8.843	31.5	6.45	26.50	907335		185.52	9.167	30.0	10.00	28.00	
48	180.00	8.843	36.0	6.00	26.50	907335		177.61	8.730	31.5	10.00	28.00	
49	180.00	8.843	39.0	6.00	26.50	907335		154.48	7.639	36.0	10.00	28.00	
50	180.00	8.843	42.0	6.00	26.50	907335		142.02	7.051	39.0	10.00	28.00	
51	180.00	8.843	42.0	6.00	26.50	907335		131.35	6.548	42.0	10.00	28.00	

DP-1135TL

663571

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31

AUG 17 1968

J. W. CROACH, DIRECTOR
TECHNICAL DIVISION - AED
EXPLOSIVES DEPARTMENT
WILMINGTON

DP-1135, CONCENTRATION OF RADIOACTIVE WASTES

by C. B. Goodlett

The physical chemical data obtained during our studies of waste concentration appear in this report, collected from our letters and monthly reports. These data include slurry density, boiling point, and solids content during concentration of supernatant liquid; the distribution of ^{137}Cs between solid and liquid phases; the results of heating the solid phase to 700°C; and the rate of redissolution of the crystals.

The mechanical operations involved in waste concentration are discussed in the companion report, DP-1136, *Transfer of Radioactive Slurries*.

D. S. Webster, Research Manager
Separations Engineering Division

INTRODUCTION

Waste solutions from the processing of radioactive fuel elements in two separate areas at the Savannah River Plant (SRP) are stored in carbon steel underground tanks of 3/4 to 1-1/3 million gallons capacity. The solutions are alkaline, with a dissolved solids content of 30-35 wt %. The wastes are of two general types: "high level waste," which contains sufficient radioactive fission products to produce decay heat at 0.5 to 5 Btu/(hr)(gal); and "low level waste," with fission product content 1/1000 to 1/100,000 that of the high level waste, but still too high to discard to a seepage pond or to streams and rivers. The low level waste contains principally sodium aluminate from the caustic dissolution of the aluminum jacket on the irradiated fuel elements. The high level waste, principally sodium nitrate with some sulfate and carbonate, contains nearly all of the radioactive fission products from the processing of irradiated fuel elements. Both wastes separate into a layer of sludge and a layer of relatively clear supernatant liquid.

Costs are reduced in the long-term storage of radioactive waste solutions at SRP by transferring the waste supernate, after sufficient time for decay of highly active fission products, from storage tanks with cooling coils (Fig. 1) to less expensive uncooled tanks (Fig. 2). The tanks with coils are then reused for fresh waste. Further economy is obtained by evaporative reduction of volume of the aged wastes during the transfer to the uncooled tanks.

A tank farm evaporator, shown in Fig. 3, was installed in both waste storage areas at SRP (1960 in F Area and 1963 in H Area). Each evaporator, central to four large uncooled storage tanks, has operated satisfactorily since installation. The present evaporators are designed to concentrate the supernatant solution from tanks (which contain either aged high activity waste or low level waste) from ~35 to ~70 wt % solids (total solute plus solid phase) and to discharge the concentrated waste by steam lift and gravity drain or a pumped recirculating loop to the storage tanks. These radioactive slurries were transported from one location to another in tests at the Savannah River Laboratory.[1]

To keep storage costs at a minimum and to ensure an operable concentration system, it was necessary to determine the characteristics of waste solutions during concentration and to determine the distribution of radioisotopes between solid and liquid phases as the salts crystallize during cooling. The distribution of radioisotopes, particularly ^{137}Cs, could affect the cooling requirements for the evaporated waste, and require additional shielding around the evaporator during the latter concentrations. Also of interest are changes in the solid residues as the temperature increases because of fission product decay heat.
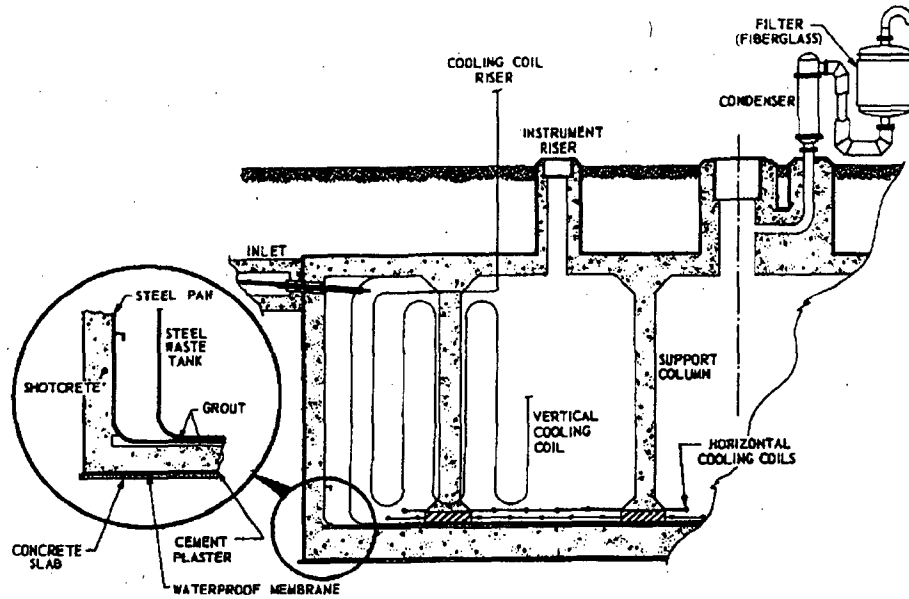
FIG. 1 COOLED STORAGE TANK
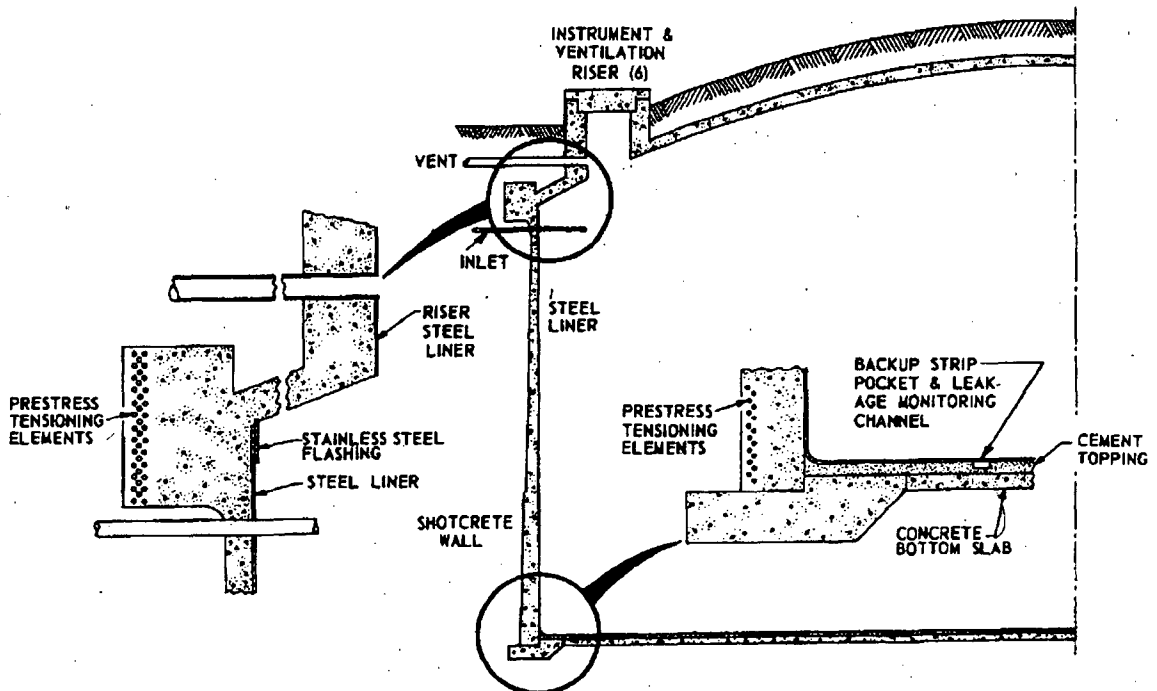
FIG. 2 UNCOOLED STORAGE TANK

- 8 -
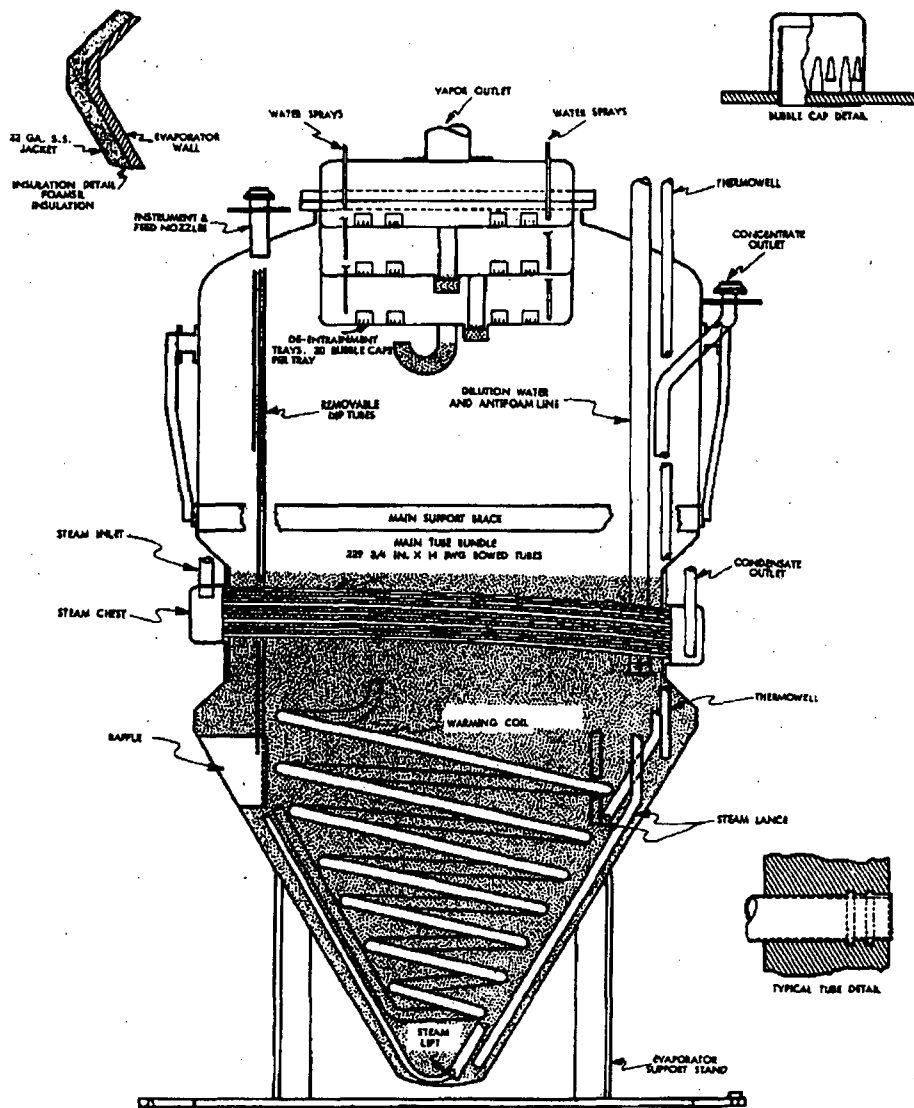
FIG. 3 TANK FARM EVAPORATOR

In some types of waste, large crystalline masses collect on the walls and cooling coils in the waste tanks. Following removal of the liquid phase (predominantly sodium aluminate de-canning solution), crystals 8 feet deep on the bottom of the tank, and large deposits of crystals on the walls, thermocouple wells, and dip tubes have been noted. In October 1964, a crystal forma-tion was observed around the center thermowell in Tank 20 that was estimated to be 5-1/2 to 7 feet in diameter and to weigh 30 to 40 tons; a crystal deposit 8 to 12 feet in diameter was on the instrument pipes in this tank (Fig. 4). Without the high-density liquid phase to help support these large crystals, their weight could pull the cooling coils from the tank roof, or collapse the roof. Consequently, the rate of dissolution of synthetic waste crystals was determined to provide basic data for removal of the crystals from the tanks, should this step be necessary.
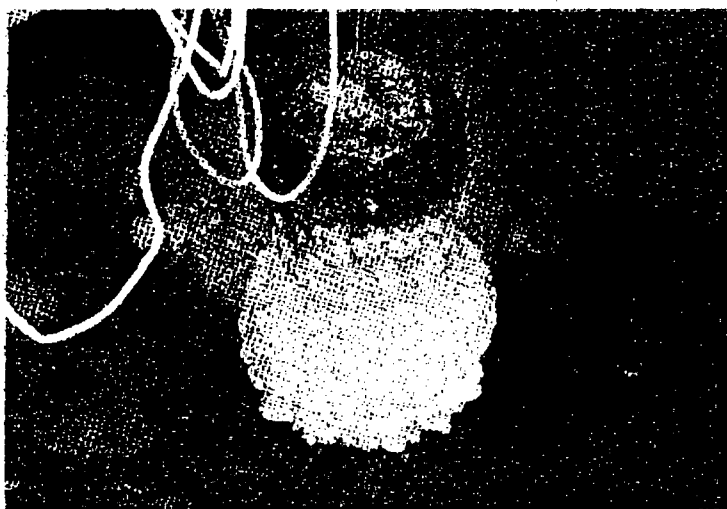
FIG. 4 TWELVE-FOOT-DIAMETER MASS OF CRYSTALS ON 0.8-INCH PIPES

SUMMARY

Tests with both synthetic and actual plant waste showed that all of the radioactive waste stored in the separations areas can be reduced to ~1/3 of its original volume and immobilized by suc-cessive evaporation of the decantable liquid. The concentration of ^{137}Cs in the liquid phase of plant waste evaporations increased as much as 17-fold. No vigorous reactions were observed when the solid residues were heated to 700°C.

The measured rates of dissolution for synthetic crystalline wastes indicate that solidified separations wastes from both natural and enriched uranium processing can be dissolved in water in a reasonable time (rates exceeding 0.1 ft/hr).

Atlantic Richfield Hanford Company

RICHLAND, WASHINGTON

TITLE AND AUTHOR

FLOW PROPERTIES OF SALT CAKE FOR INTERSTITIAL LIQUID REMOVAL/
IMMOBILIZATION DEVELOPMENT PROGRAM

Lyman L. Handy, Consultant

APPROVED for Release for
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J. C. Petrie	234-5	200-W			

ROUTE TO	PAYROLL NO.	LOCATION	FILES ROUTE DATE

FLOW PROPERTIES OF SALT CAKE

FOR

INTERSTITIAL LIQUID REMOVAL/IMMOBILIZATION

DEVELOPMENT PROGRAM

Summary Report for Consultant

Agreement CA-186

Lyman L. Handy

Consultant

September 30, 1975

SUMMARY

1. Permeabilities measured with synthetic salt cake in glass columns were about 20 darcies. These permeabilities are higher than data obtained with Hassler sleeves at Atlantic Richfield Hanford. The difference can best be resolved by in situ measurements.

2. Sludge permeabilities were estimated to be less than 10 md.

3. Steady state and pseudo steady state flow equations have been derived for salt cake in cylindrical tanks. They can be used to predict drainage rates; or, from flow rates and liquid levels, to calculate in situ permeabilities. If the permeability is assumed to be 20 darcies, maximum flow rates will be 200 to 300 gals per hour, depending on liquid levels.

4. Minimum drainage times have been calculated for representative salt cake and storage tank properties. The times are approximately proportional to the difference in reciprocals of the final and initial liquid levels at the periphery of the tank. An example calculation showed that 140 days would be required to drain an initial 20 feet of heel to 2 feet of remaining liquid for a 20 darcy salt cake in a 75 foot tank.

X-ESR-G-00004
Rev. A

August 30, 2004

TO: E. J. Freed, 703-H

FROM: J. R. Hester, 766H   *[signature]* 8/31/04

TECHNICAL RI IEW: J. K. Jeffrey, 766-H   *[signature]* 8/31/04

APPROVED: D. J. Martin, 766-H   *[signature]* 8/31/04

WCS   pernate Radionuclide Concentration Algorithms

INTRODUCTI(

Closure Busines Jnit Planning and Execution (P&E) is developing an Integrated
Material Balance id Flowsheet Model to be used for planning and coordination among
high level was   process facilities.   The model draws data from the Waste
Characterization stem (WCS), some of which is not currently included in the system.
Therefore, WCS being expanded and upgraded to support the modeling effort. The
purpose of this n iorandum is to recommend algorithms (rules of thumb) to estimate the
concentrations o: :y radionuclides in supernate, when sample data is unavailable.

SUMMARY

Recent supernat(id saltcake liquid analyses from thirteen tanks provide a reasonable,
best-estimate b: for expanding and improving WCS algorithms for supernate
radionuclide con trations.

DISCUSSION

Background

P&E identified 1 nty key radionuclides important to their model (Attachment A). Of
these, WCS sup   ate data currently covers all the actinides, except Cm-242, and Cs-
137/Ba-137m. I   es not cover seven fission/activation products listed in Attachment A.

Recent studies o   ae liquid (supernate) from Tanks 13H, 30H, 37H, 39H, 45F, 46F, and
49H [1, 2] and c   iltcake liquids from Tanks 3F and 2F [3], 10H [4], 29H [5], 38H [6],
and 41H [7] pro\ : an opportunity to expand and improve WCS supernate data.

Besides providing temporary data for thirteen tanks, the recent studies provide a basis for constructing algorithms to estimate the concentrations of radionuclides in supernate where sample data is lacking or rendered inapplicable because of changed conditions.

Further, these studies provide a glimpse into the effect of filtering supernate and of the difference between free supernate and interstitial salt liquid. Comparing free supernate with interstitial salt liquid results indicates whether separate algorithms are needed for these two forms. Whereas, comparing filtered versus unfiltered results indicates whether a constituent is present in dissolved form or as suspended solids, which distinction is important to the construction of algorithms for dissolved constituents.

The following section discusses the results and recommended supernate concentration algorithms for the key radionuclides. The algorithms are of three kinds:

- Constant values for low-level constituents, such as C-14
- Ratio to Cs-137 (i.e., gross gamma) for Sr-90 and Tc-99
- Partition factors for actinides, according to their abundance in associated sludge.

Sample Analysis Results and Algorithms for Individual Radionuclides

Attachments B and C consist of extracts of results from the referenced reports for fission/activation products and for actinides, respectively.

Some samples were diluted prior to analysis (Attachment B and C, Dilution Factor column). In these cases the results are multiplied by the appropriate dilution correction to estimate as-received concentrations. This procedure is conservative and may overestimate as-received concentrations of constituents that fall below detectable limits (i.e., "less than" results).

H-3

H-3 is a low-yield activation product created in the reactors. Its presence in the form of tritiated water is expected at low concentration in all the waste tanks. Tritiated water is soluble in all proportions, so no filtering effect is possible. Supernate blending during future processing should tend to average the tritium concentration over time.

H-3 analyses of free supernate samples from seven tanks yielded six results over the detection limit, ranging from 8.45E+03 to 1.14E+05 pCi/mL (Attachment B, H-3 column). A seventh sample was found to contain less than 2.38E+04 pCi/mL. Counting the "less than" result, these values average 4.27E+04 pCi/mL, with a standard deviation of 4.64E+04 pCi/mL.

A constant value of 1.00E+05 pCi/mL is recommended as a reasonable algorithm for supernate H-3 concentrations.

C-14

C-14 is a minor activation product created in the reactors. It also occurs in nature and is deposited in waste tanks as the liquid takes up CO_2 from the environment. Carbon is mildly soluble (~24 percent) in alkaline media, so C-14 replenishment is expected if fresh liquid is added to a tank. This "buffering" effect should tend to stabilize the C-14 concentration in supernate.

C-14 analyses of free supernate or salt interstitial liquid samples from eight tanks yielded two results (2.69E+02 and 1.71E+03 pCi/mL) above the detection limit (Attachment B, C-14 column). The other analyses indicated "less than" results ranging from 3.43E+02 to 7.11E+03 pCi/mL. Combined, the eight samples averaged 1.76E+03 pCi/mL, with a standard deviation of 2.27E+03 pCi/mL.

A constant value of 1.76E+03 pCi/mL is recommended as a reasonable algorithm for supernate C-14 concentrations.

Co-60

Co-60 is a minor activation product created in the reactors. Cobalt is sparingly soluble (~0.02 percent) in alkaline media; therefore, its concentration in supernate should be low.

Co-60 analyses were performed on free supernate samples from seven tanks, yielding two results (1.90E+03 and 2.74E+03 pCi/mL) above the detection limit (Attachment B, Co-60 column). The other six analyses indicated "less than" concentrations from 3.60E+01 to 1.17E+03 pCi/mL. The average was 1.03E+03 with a standard deviation of 1.02E+03 pCi/mL.

A constant value of 1.03E+03 pCi/mL is recommended as a reasonable algorithm for supernate Co-60 concentrations.

Ni-63

Ni-63 is a minor activation product created in the reactors. Nickel is sparingly soluble (~0.02 percent) in alkaline media; therefore, its concentration in supernate is expected to be low.

Ni-63 analyses were performed on free supernate samples from seven tanks with no results above the detection limit (Attachment B, Ni-63 column). The analyses indicated concentrations of less than 1.14E+02 to 5.72E+03 pCi/mL with an average of 2.50E+03 pCi/mL and a standard deviation of 2.40E+03 pCi/mL.

A constant value of 2.50E+03 pCi/mL is recommended as a reasonable algorithm for supernate Ni-63 concentrations.

Sr-90/Y-90

Sr-90 is a major fission product created in the reactors. The radiological impact of Sr-90 is doubled by its daughter, Y-90, which is in secular equilibrium with it. Because the radiological impacts of Sr-90 are more pronounced than those of other, less prominent, radionuclides, greater care is needed in selecting an algorithm for it.

Strontium is sparingly soluble in alkaline media (~0.025 percent); therefore, Sr-90 is expected in low concentration in supernate and salt interstitial liquid. Strontium combines with CO_2 to form $SrCO_3$, which may become suspended in the liquid as finely divided solids. In this case, the fines would contribute to the Sr-90 concentration in the liquid, and filtering should affect the results of analyses.

Sr-90 analyses were performed on supernate samples from twelve tanks. All samples yielded results above the detection limit. Some samples were analyzed unfiltered and after being filtered through 0.45μ, 0.1μ, or 0.02μ filters. There is no apparent filtration trend among the results (Attachment B, Sr-90 filter-size columns). Nor is there an apparent trend between free and interstitial liquid samples.

The sample results for each tank were averaged, yielding an overall average Sr-90 concentration of 5.37E+05 pCi/mL with a standard deviation of 8.72E+05 pCi/mL (Attachment B, Sr-90 Tank Average column). Figure 1 is a histogram of the tank averages, which indicates skewness to the right. The most frequent value, 1.07E+06 pCi/mL, over estimates the overall average considerably, so it is not an ideal algorithm.

As an alternative, the Sr-90 averages for each tank were divided by the corresponding Cs-137 values to obtain Sr-90/Cs-137 ratios (Attachment B, Sr-90/Cs-137 column). These ratios averaged 2.36E-04. Figure 2 is a histogram of the Sr-90/Cs-137 ratios. The most frequent value of 2.75E-04 is reasonably close to the average value. Taking an intermediate value of 2.60E-04 is reasonable.

Multiplying the Cs-137 (gross gamma) concentration by 2.60E-04 is recommended as a reasonable algorithm for deriving Sr-90 concentrations in supernate.

Tc-99

Tc-99 is a long-lived decay product of the minor short-lived fission product Mo-99, which was created in the reactors. It is fairly soluble (~57 percent) in alkaline media; therefore, it is expected to be prevalent in supernate.

Tc-99 analyses were performed on free supernate from seven tanks, yielding results above the detection limit for all. The indicated concentrations ranged from 1.70E+05 to 1.56E+06 pCi/mL with average, 6.64E+05 pCi/mL, and standard deviation, 5.04E+05 pCi/mL (Attachment B, Tc-99 column).

Figure 3 is a histogram of the Tc-99 results, which is decidedly non-symmetric. The most frequent value of 8.66E+05 pCi/mL is a fair approximation only of the average value.

Tc-99/Cs-137 ratios averaged 4.85E-04 with a standard deviation of 7.23E-04 (Attachment B, Tc-99/Cs-137 column). Figure 4 is a histogram of the Tc-99/Cs-137 ratios, which is symmetric, but the most frequent value, 1.12E-03 overestimates the average value by more than 2X.

Neither the straight-value rendering nor the Tc-99/Cs-137 ratio treatment is clearly preferable; however, the symmetry and conservatism of the ratio recommends it as the better of the two. Therefore, multiplying the Cs-137 (gross gamma) concentration by 1.12E-03 is recommended as a reasonable algorithm for deriving Tc-99 concentrations in supernate.

I-129

Iodine is a low-yield fission product created in the reactors; therefore, it is present in low concentration in all tanks. Iodine is completely soluble, and, like tritium, supernate blending during future processing will tend to average out its concentration.

I-129 analyses were performed on free supernate samples from six tanks, yielding two results (3.78E+02 and 5.36E+02 pCi/mL) above the detection limit of (Attachment B, I-129 column). The other five analyses indicated "less than" concentrations from 8.78E+01 to 1.03E+03 pCi/mL. The average was 4.49E+02 with a standard deviation of 3.22E+02 pCi/mL.

A constant value of 4.49E+02 pCi/mL is recommended as a reasonable algorithm for supernate I-129 concentrations.

Cs-137/Ba-137m

Cs-137 is a major fission product created in the reactors. It is in secular equilibrium with its daughter, Ba-137m, which emits the hard gamma normally associated with Cs-137. Cesium is highly soluble and is expected to be the dominant radionuclide in supernate.

Gross gamma (commonly referred to as Cs-137) determinations are performed on virtually all supernate samples submitted for analysis from the tank farms. WCS uses these results to infer Cs-137/Ba-137m concentrations because this pair is the overwhelmingly dominant gamma source in current, aged waste. Therefore, no algorithm is needed for Cs-137/Ba-137m.

Actinides

Most of the actinide inventory in the waste tanks was created from reactor fuel elements, containing U-235 and U-236, and from target elements, containing U-238 or Np-237.

Actinides are sparingly soluble in alkaline media. They are major radiological constituents of sludge, but, barring the presence of suspended sludge particles, actinide concentrations in supernate are expected to be low.

Actinide analyses were performed on thirteen tanks as listed in Attachment C. (Cm-245 was not included in the analyses.) Attachment C contains the results of the analyses and useful derived quantities. When multiple results were available, average values are cited.

No filtration trend for actinides was discernable in the analyses, and the measured values for plutonium and uranium were below their predicted solubility [8]. The absence of a filtration trend and low concentrations imply that the samples contained little or no suspended sludge, which would otherwise mask the dissolved constituent concentrations.

The individual actinide results in each row of Attachment C are in general agreement with expectations (e.g., Pu-238 is dominant). The 7.47E+05 pCi/mL Pu-238 result for Tank 39H is an exception. This value is more than 800 times higher than expected. Analysis of Tank 39H supernate in November 2003 indicated a Pu-238 concentration of 2.00E+06 dpm/L (9.01E+02 pCi/mL) [9]. The high Pu-238 coupled with high Pu-239/240 and Pu-241 values seem to imply sludge in the Tank 39H sample, the lack of a filtration effect notwithstanding.

The current WCS methodology for determining supernate actinide concentrations involves partitioning constant total (gross) alpha values according to tank type and associated sludge [10]. A gross alpha value of 1.88E+06 pCi/mL (7.1E-03 Ci/gal) is used for Type III/IIIA tanks, and a value of 5.55E+04 pCi/mL (2.1E-04 Ci/gal) is used for Type I, II, and IV tanks.

Even with the Tank 39H results included, the gross alpha values (Attachment C, Gross Alpha column) are lower on average than the WCS assumed value for Type III/IIIA tanks (1.09E+05 pCi/mL versus 1.88E+06 pCi./mL). Omitting the Tank 39H values drops the average to less than the WCS lower value (3.19E+04 pCi/mL versus 5.55E+04 pCi/mL).

Based on the information at hand, the current WCS methodology should be retained, but a constant value of 1.09E+05 pCi/mL should be used in lieu of the current values. This approach will lower the dissolved actinide concentrations in Type III/IIIA tank supernate and raise those in other tank types. Average actinide concentrations may be used for tanks with no associated sludge.

CONCLUSIONS

Recent analyses of supernate and saltcake liquids from thirteen waste tanks provide a good basis for expanding and improving WCS algorithms for supernate radionuclide concentrations. Table 1 is a list of recommended algorithms for the key radionuclides listed in Attachment A.

Table 1
Recommended Algorithms

Radionuclide	Value	Application Notes
H-3	1.00E+05 pCi/mL	Constant
C-14	1.76E+03 pCi/mL	Constant
Co-60	1.03E+03 pCi/mL	Constant
Ni-63	2.50E+03 pCi/mL	Constant
Sr-90/Y-90	2.60E-04	Ratio to measured Cs-137 (gross gamma)
Tc-99	1.12E-03	Ratio to measured Cs-137 (gross gamma)
I-129	4.49E+02 pCi/mL	Constant
Cs-137/Ba-137m	N/A	Use current WCS methodology
Actinides	1.09E+05 pCi/mL	Constant used with WCS sludge partitioning

REFERENCES

[1]   WSRC-TR-2004-XXXXX, Characterization of Supernate Samples from High Level Waste Tanks 13H, 30H, 37H, 39H, 45F, 46F, and 49H, M. E. Stallings, et al., DRAFT G, July 2004

[2]   WSRC-NB-2000-00135, SRNL Laboratory Notebook, pp. 85-90

[3]   WSRC-TR-2004-00131, Tanks 3F and 2F Saltcake Core and Supernate Sample Analysis, C. J. Martino, et al., Rev. 0, April 2004

[4]   WSRC-TR-2004-00164, Tank 10H Saltcake Core Sample Analysis, C. J. Martino, et al., Rev. 0, April 2004

[5]   WSRC-TR-2004-00130, Tank 29H Saltcake Core and Supernate Sample Analysis, C. J. Martino, et al., Rev. 0, April 2004

[6]   WSRC-TR-2004-00129, Tank 38H Saltcake Core and Supernate Sample Analysis, C. J. Martino, et al., Rev. 0, April 2004

[7]   WSRC-TR-2004-00165, Tank 41H Post-Dissolution Saltcake Core and Supernate Sample Analysis, C. J. Martino, et al., Rev. 0, April 2004

[8]   WSRC-TR-2004-XXXXX, Characterization of Supernate Samples from High Level Waste Tanks 13H, 30H, 37H, 39H, 45F, 46F, and 49H, M. E. Stallings, et al., DRAFT G, July 2004, Section 1.0, Executive Summary

[9]   WG08, WASTE8, HLW Tank Chemistry, NTANK39, November 2003 entry

[10]   WSRC-TR-1996-00264, High Level Waste Characterization System (WCS), J. R. Hester, Rev. 0, December 1996

Figure 1
Sr-90 Concentration Histogram

Figure 2
Sr-90/Cs-137 Ratio Histogram

Figure 3
Tc-99 Concentration Histogram

Figure 4
Tc-99/Cs-137 Ratio
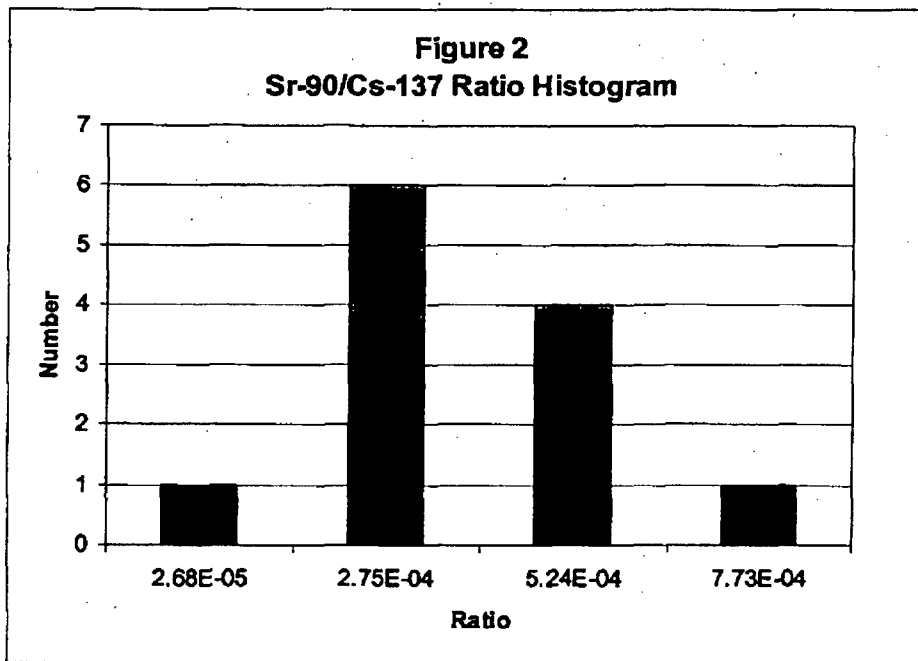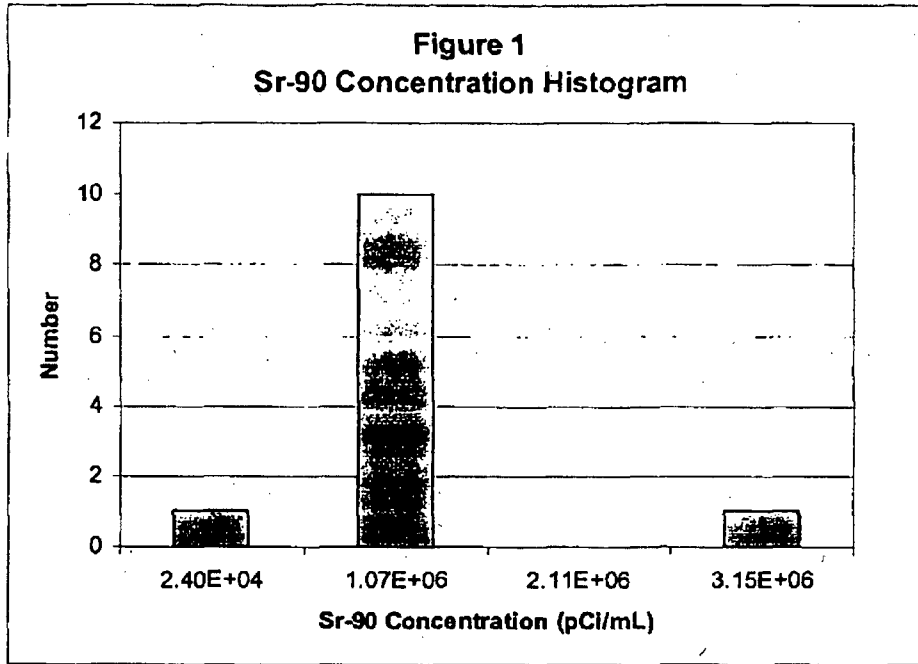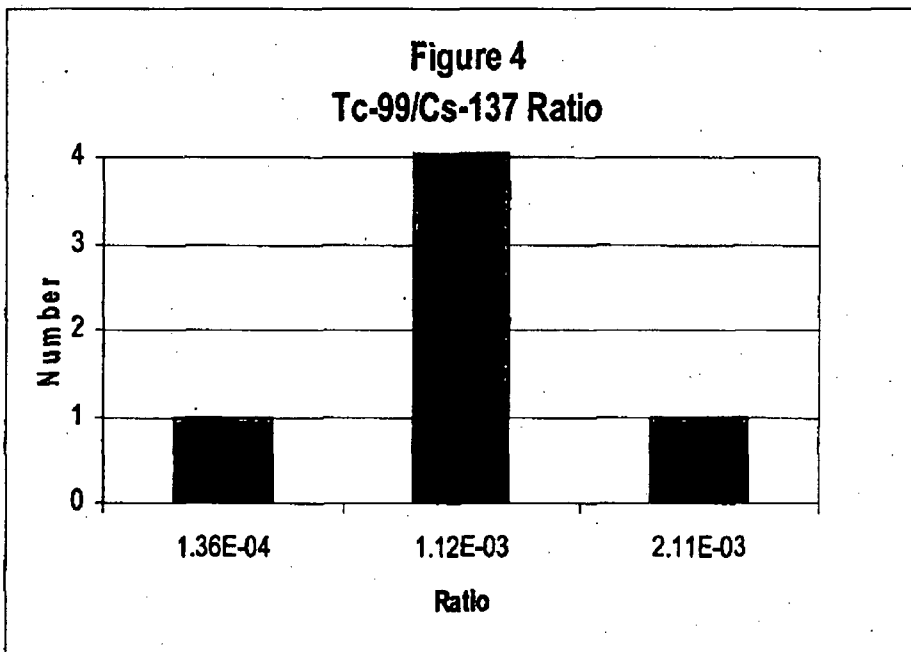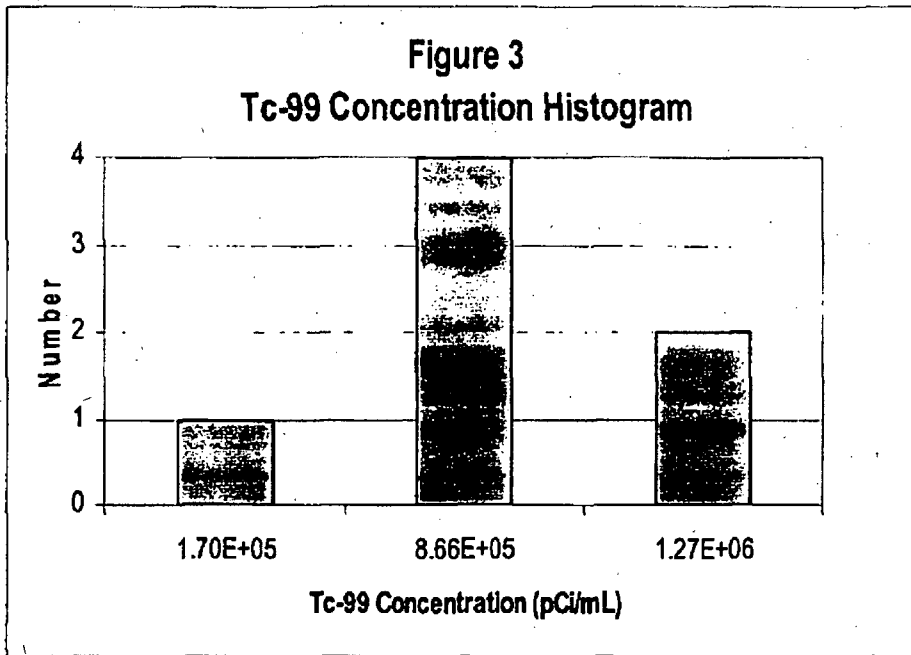
Attachment A

Key Radionuclides for Material Balance and Flowsheet Modeling

Radionuclide	Basis
H-3	NRC Class A
C-14	NRC Class A
Ni-63	NRC Class A
Co-60	NRC Class A
Sr-90	NRC Class A
Tc-99	NRC Class A
I-129	NRC Class A
Cs-137	NRC Class A
U-232	DSA Source Term
U-233	Fissile and likely present in residual sludge
U-235	Fissile and likely present in residual sludge
Np-237	Contributes to total alpha
Pu-238	Contributes to total alpha
Pu-239	Contributes to total alpha
Pu-240	Contributes to total alpha
Pu-241	NRC Class A
Am-241	Contributes to total alpha
Am-242m	DSA Source Term
Cm-242	NRC Class A
Cm-245	DSA Source Term

Attachment B

Sample Analysis Results for Fission/Activation Products

Tank	H-3 pCi/mL	C-14 pCi/mL	Co-60 pCi/mL	Ni-63 pCi/mL	Sr-90 (Unfiltered) pCi/mL	Sr-90 (0.45µ) pCi/mL	Sr-90 (0.1µ) pCi/mL	Sr-90 (0.02µ) pCi/mL	Sr-90 Tank Average pCi/mL	Sr-90/ Cs-137 pCi/mL/ pCi/mL	Tc-99 pCi/mL	Tc-99/ Cs-137 pCi/mL/ pCi/mL	I-129 pCi/mL	Cs-137 pCi/mL	Cs-137 Tank Average pCi/mL
2FL	?	1.22E+03	?	?	5.10E+04	7.40E+03	3.74E+05	?	1.44E+05	8.79E-05	?	?	?	1.64E+09	1.64E+09
2IL	?	?	?	?	?	?	?	?	.	.	?	?	?	?	-
3FL	?	?	?	?	?	5.99E+05	?	?	5.24E+05	2.90E-04	?	?	?	1.74E+09	1.81E+09
3IL	?	2.20E+03	?	?	?	4.50E+05	?	?	.	.	?	?	?	1.88E+09	-
10IL	?	?	?	?	?	2.40E+04	?	?	2.40E+04	2.73E-05	?	?	?	8.79E+08	8.79E+08
13FL	1.06E+05	4.51E+02	4.67E+01	8.83E+02	?	6.99E+05	5.59E+05	4.83E+05	5.80E+05	1.64E-04	6.30E+05	1.78E-04	5.36E+02	3.54E+09	3.54E+09
29FL	?	?	?	?	1.44E+04	1.59E+05	1.21E+05	?	4.94E+05	3.07E-04	?	?	?	7.15E+08	1.61E+09
29IL	?	?	?	?	?	1.68E+06	?	?	.	.	?	?	?	2.50E+09	-
30FL	2.40E+04	7.56E+02	1.90E+03	3.08E+03	?	6.72E+05	1.27E+06	8.85E+05	9.42E+05	4.07E-04	9.83E+05	4.24E-04	3.78E+02	2.32E+09	2.32E+09
37FL	9.53E+03	2.69E+02	2.74E+03	2.30E+02	?	3.48E+06	3.10E+06	2.88E+06	3.15E+06	7.73E-04	8.11E+05	1.99E-04	1.03E+03	4.08E+09	4.08E+09
38FL	?	?	?	?	?	6.20E+04	?	?	3.54E+04	1.93E-04	?	?	?	1.93E+08	1.83E+08
38IL	?	?	?	?	2.95E+04	1.47E+04	?	?	.	.	?	?	?	1.73E+08	-
39FL	1.14E+05	1.71E+03	2.64E+02	5.64E+03	?	1.90E+05	3.20E+05	3.76E+05	2.96E+05	3.99E-04	1.56E+06	2.11E-03	8.78E+01	7.40E+08	7.40E+08
45FL	2.38E+04	?	1.17E+03	4.72E+03	?	8.26E+04	9.32E+04	5.94E+04	7.85E+04	6.28E-05	1.70E+05	1.36E-04	?	1.25E+09	1.25E+09
46FL	1.33E+04	7.11E+03	1.06E+03	1.84E+03	?	4.81E+04	3.57E+04	3.45E+04	3.94E+04	2.68E-05	2.77E+05	1.88E-04	3.94E+02	1.47E+09	1.47E+09
49FL	8.45E+03	3.43E+02	3.60E+01	1.14E+02	?	1.45E+05	1.20E+05	1.25E+05	1.30E+05	9.64E-05	2.18E+05	1.61E-04	2.65E+02	1.35E+09	1.35E+09

Total

	H-3	C-14	Co-60	Ni-63	Sr-90 (Unfiltered)	Sr-90 (0.45µ)	Sr-90 (0.1µ)	Sr-90 (0.02µ)	Sr-90 Tank Average	Sr-90/ Cs-137	Tc-99	Tc-99/ Cs-137	I-129	Cs-137	Cs-137 Tank Average
Avg	4.27E+04	1.76E+03	1.03E+03	2.50E+03	3.16E+04	5.54E+05	6.65E+05	6.92E+05	5.37E+05	2.36E-04	6.64E+05	4.85E-04	4.49E+02	1.54E+09	1.74E+09
SD	4.64E+04	2.27E+03	1.02E+03	2.40E+03	1.84E+04	9.23E+05	9.88E+05	1.01E+06	8.72E+05	2.17E-04	5.04E+05	7.23E-04	3.22E+02	1.14E+09	1.12E+09
Max	1.14E+05	7.11E+03	2.74E+03	5.72E+03	5.10E+04	3.48E+06	3.10E+06	2.88E+06	3.15E+06	7.73E-04	1.56E+06	2.11E-03	1.03E+03	4.08E+09	4.08E+09

Key: ? = no result
nnnE+nn = positive result
[nnnE+nn] = less than value

Attachment C

Sample Analysis Results for Actinides

Tank	U-232 (pCi/mL)	U-233 (pCi/mL)	U-235 (pCi/mL)	Np-237 (pCi/mL)	Pu-238 (pCi/mL)	Pu-239/240 (pCi/mL)	Pu-241 (pCi/mL)	Am-241 (pCi/mL)	Am-242m (pCi/mL)	Cm-242 (pCi/mL)	Gross Alpha (pCi/mL)	Density (g/ml)	Dilution Factor
2	?						?	?	?	?	?	1.500	1
3	?				3.19E+03	2.13E+03		?	?	?	?	1.495	1
10	?		4.09E-01		1.92E+04	3.10E+03		?	?	?	?	1.437	1
13			6.15E-01	2.14E+01	4.14E+04	8.46E+02					2.60E+04	?	2.3
29	?		2.35E-01		1.13E+04	3.15E+03		?	?	?	?	1.480	1
30		?	6.78E-01	3.03E+01	3.41E+04	1.13E+04				9.44E+00	4.82E+04	?	2.8
37			5.66E-01	2.70E+01	1.15E+04	1.18E+04					5.83E+04	?	2.4
38	?		3.78E-01		1.44E+04	1.27E+03		?	?	?	?	1.440	1
39		?	4.39E+00	1.36E+02		4.51E+04	7.33E+05	2.73E+02		3.71E+00	6.47E+05	?	1.1
41	?	?	?	?	9.48E+03	1.48E+03	?	?	?	?	1.10E+04	1.427	1
45		?	1.40E-01	BDL	2.58E+03	1.99E+03		1.24E+03				?	3.6
46		?	6.87E-02	BDL	5.42E+03	1.35E+03		9.72E+04	?	1.12E+01		?	2.9
49		8.98E+01	3.18E-01	2.24E+02	1.06E+04	3.13E+03					1.38E+04	?	1.7
Avg	8.29E+02	7.55E+02	6.86E-01	7.59E+01	7.01E+04	4.89E+03	3.48E+06	1.55E+04	3.48E+04	1.79E+02	1.09E+05		
SD	1.02E+03	4.95E+02	1.18E+00	6.39E+01	2.04E+05	1.21E+04	6.70E+06	3.60E+04	7.21E+04	4.04E+02	2.18E+05		
Max	3.08E+03	1.46E+03	4.39E+00	2.24E+02	7.47E+05	4.51E+04	2.05E+07	9.72E+04	1.82E+05	1.00E+03	6.47E+05		

Key: ? = no result
nnnE+nn = positive result
= less than value
= questionable result
BDL = below detection limit

TO: A. S. JENNINGS

FROM: D. L. KISER

        DLK.

BENCH-SCALE INVESTIGATION OF DIFFERENT CONCEPTS FOR WASTE TANK SALT DISSOLUTION

INTRODUCTION AND SUMMARY

For the Phase I waste removal program, salt will be removed from waste tanks
to be retired by batching in fresh water as a layer above the salt cake.
Steam agitation jets are to be mined into the salt cake and will recirculate
the supernate to dissolve the salt. Steam, used to power the agitation jets,
will add condensate to the supernate and heat the contents of the tank. After
four volume turnovers of the supernate and a cool-down period, the supernate will
be jetted to an evaporator feed tank. The process will be repeated until most
of the salt is dissolved. The major disadvantage with this method is the long
cool down periods (totaling up to 10 months/tank for Type I and 24 months/tank
for Type IV) required for liquid transfer due to elevated temperature in the
tank.

Two alternate salt dissolution concepts have been investigated on a bench-scale.
The density driven concept appears to have about the same dissolution rate as a
steam agitation jet without the elevated temperature condition. Mechanical
agitation is superior to the density driven concept in both dissolution rate and
fresh water requirements. Both the density driven concept and mechanical agitation
are proposed to be tested in the Phase I program, to confirm the bench scale test
results with radioactive waste salt.

INFORMATION ONLY

Soluble Phase Selenium-79, Technetium-99, and Tin-126 Inventories

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Soluble Phase Selenium-79, Technetium-99, and Tin-126 Inventories

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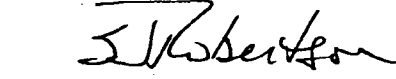
Paul D. d'Entremont   5/25/05
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Table of Contents

Summary of Revisions

5/2005   Revision 0; Initial Issue

1 Summary

Previously, soluble Se-79, Tc-99, and Sn-126 inventories were not tracked in WCS. In 2004, an algorithm for modeling Tc-99 was developed for the Integrated Flowsheet model[6]. This algorithm conservatively overestimates the Tc-99 concentrations. However, a revised estimation method based on analytical data and process history that would project more reliable supernatant inventories is required to support salt solution disposition planning efforts. The purpose of this memorandum is to provide the basis to identify the soluble curie inventory for Se-79, Tc-99, and Sn-126. Using these bases, the projected curie inventories in the supernatant phase for Se-79, Tc-99, and Sn-126 are 8.9E+1 curies, 3.3 E+4 curies, and 4.5E+2 curies respectively.

2 Selenium-79

In 1995, Georgeton and Hester[1] estimated that only 5% of the Se-79 received into the tank farms would be soluble. At the time, only the sludge phase constituents were of concern, and therefore, the soluble portion was not incorporated into WCS. The insoluble portion of the Se-79 (95%) was incorporated into WCS for tracking. As of 5/1/05, the Se-79 sludge phase inventory was 1.6E+3 curies, and the total Se-79 transferred in sludge to DWPF has been 1.1E+2 curies[2]. This is a total WCS tracked sludge inventory of 1.7E+3 curies. Multiplying this by the soluble to insoluble ratio of 0.05 (5% / 95%) yields a supernatant inventory of 8.9E+1 curies. A review of recent supernatant sample results contained in WSRC-TR-2004-00386 indicates that all Se-79 sample analyses reported less than minimum detectable concentrations of Se-79. The lowest detection limit for each tank is shown in Table 1. The average of the lowest detection limit from each tank was 2.3E+3 pCi/ml. If this average concentration is applied to the tank farm inventory of 24M gallons of supernatant (free and interstitial), a total supernatant Se-79 inventory of <2.1E+2 curies is obtained. Since this is in reasonable agreement with the theoretical fission yield total of 8.9E+1 curies, the value of 8.9E+1 curies is recommended for use during salt solution disposal planning.

Table 1 Se-79 Minimum Detection Limits in WSRC-TR-2004-00386, R1

Tank	Lowest Se-79 Detection Limit[3] (pCi/ml)	Dilution Factor[6]	Actual Se-79 (pCi/ml)
13	2.0E+2	2.3	<4.6E+2
30	9.1E+2	2.8	<2.5E+3
37	2.1E+3	2.4	<5.0E+3
39	4.4E+3	1.1	<4.8E+3
45	2.5E+2	3.6	<9.0E+2
46	3.4E+2	2.9	<9.9E+2
49	8.5E+2	1.7	<1.4E+3

3 Technetium-99

Two methods will be used to calculate the soluble Tc-99 inventory. The first will use theoretical fission yield relationships and solubility expectations to determine a predicted inventory. The second will review available sample results to derive a calculated inventory. In 1995, Georgeton and Hester[1] calculated predicted canyon waste stream compositions based on theoretical fission yield relationships, and these compositions were used as the basis for constituent inventories tracked in the Waste Characterization System (WCS). The report also determined that 53% of the Tc-99 received by the Tank Farms was soluble. At the time, only the sludge phase constituents were of concern, and therefore, the soluble portion was not incorporated into WCS. The insoluble portion of the Tc-99 (47%) was incorporated into WCS for tracking. As of 5/1/05, the Tc-99 sludge phase inventory was 2.7E+4 curies, and the total Tc-99 transferred in sludge to DWPF has been 1.8E+3 curies[2]. This is a total WCS tracked sludge inventory of 2.9E+4 curies. Multiplying this by the soluble to insoluble ratio (53% / 47%) yields a soluble inventory of 3.3E+4 curies. This inventory includes Tc-99 that has precipitated into saltcake.

A search of available sample data has identified 22 supernatant sample results for Tc-99. These results are shown along with the Cs-137 concentrations in Table 2. A statistical evaluation was performed to determine if a correlation exists between Tc-99 concentration and Cs-137 concentration in supernatant. Cs-137 was chosen for comparison because the Cs-137 supernatant concentrations have been well characterized by sampling. Figure 1 shows the regression plot and equation for this analysis.

Table 2. Tc-99 Supernatant Sample Results

Date	Tank	Cs-137 (pCi/ml)	Tc-99 (pCi/ml)	Ratio (Tc-99/Cs-137)	Source
7/2/2004	13	3.5E+9	6.3E+ 5	1.8E-4	X-ESR-G-00004[6]
7/2/2004	30	2.3E+9	9.8E+5	4.2E-4	
7/2/2004	37	4.1E+9	8.1E+5	2.0E-4	
7/2/2004	45	1.3E+9	1.7E+5	1.4E-4	
7/2/2004	49	1.4E+9	2.2E+5	1.6E-4	
7/2/2004	46	1.5E+9	2.8E+5	1.9E-4	
7/2/2004	39	7.4E+8	1.6E+6	2.1E-3	
05/01/87	20	2.7E+7	8.1E+3	3.0E-4	Ntank files
10/09/86	21	2.4E+7	3.9E+3	1.6E-4	
09/24/86	21	1.1E+8	1.6E+4	1.5E-4	
09/22/86	21	1.9E+8	3.2E+4	1.7E-4	
05/21/86	22	3.0E+7	3.8E+3	1.3E-4	
08/14/92	26	1.4E+9	1.8E+5	1.3E-4	
07/21/92	27	9.9E+8	1.4E+5	1.4E-4	
07/21/92	28	1.2E+9	1.8E+5	1.5E-4	
11/28/92	29	2.9E+9	5.2E+5	1.8E-4	
11/29/92	30	2.2E+9	2.6E+5	1.2E-4	
04/10/92	30	2.5E+9	3.4E+5	1.4E-4	
11/29/92	32	9.9E+8	1.9E+5	2.0E-4	
04/30/92	34	5.0E+8	1.4E+5	2.8E-4	
11/24/92	38	9.9E+8	2.0E+5	2.0E-4	
11/24/92	43	6.8E+8	1.4E+5	2.1E-4	

Figure 1. Tc-99 vs Cs-137 Regression Plot

Regression Plot

Log(10) Tc = -3.71004 + 1.00104 Log(10) Cs

S = 0.271245   R-Sq = 86.9 %   R-Sq(adj) = 86.3 %

Regression
95% CI
95% PI
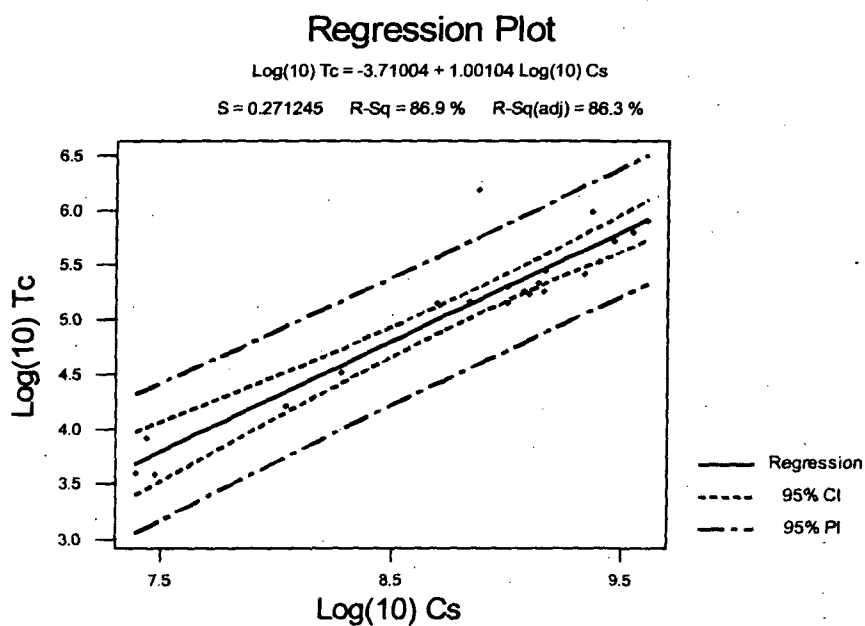
The R-Sq (adj) value is a measure of what percentage of the change in Tc-99 concentration can be attributed to the change in Cs-137 concentration. The R-Sq (adj) value of 86.3% shows that there is a very strong correlation between the supernatant Tc-99 concentration and the Cs-137 concentration, and that this correlation can be described by the equation at the top of the plot. Converting from Log_{10}, this equation becomes Tc-99 = (1.95E-4) x (Cs-137). It should be noted that, in X-ESR-G-00004, Hester calculated an average Tc-99 to Cs-137 ratio of 4.85E-4, but his ratio was based on more limited data (only 7 sample results). With the inclusion of the additional Ntank data (15 sample results), the number of sample results is tripled, and therefore, the value derived in this report is believed to be more representative. As of 5/1/05, the total supernatant Cs-137 inventory in the Tank Farms was 1.13E+8 curies[2]. Inserting this into the equation calculates a Tc-99 supernatant inventory of 2.2E+4 curies.

The theoretical method and the analytical method for estimating total supernatant inventory are in reasonable agreement. For the purposes of salt solution disposal planning, the more conservative inventory of 3.3E+4 curies should be used.

4   Tin-126

In 1995, Georgeton and Hester[1] estimated that only 5% of the Sn-126 received into the tank farms would be soluble. At the time, only the sludge phase constituents were of concern, and therefore, the soluble portion was not incorporated into WCS. The insoluble portion of the Sn-126 (95%) was incorporated into WCS for tracking. As of 5/1/05, the Sn-126 sludge phase inventory was 2.0E+3 curies, and the total Sn-126 transferred in sludge to DWPF has been 1.4E+2 curies[2]. This is a total WCS tracked sludge inventory of 2.1E+3 curies. Dividing this by the insoluble ratio of 0.95 (95%) yields a total Sn-126 inventory received into the tank farm of 2.2E+3 curies (decayed from receipt date to 5/1/05). Multiplying this by the 0.05 (5%) yields a soluble inventory of 1.1E+2 curies. This inventory includes Sn-126 that has precipitated into saltcake.

Since Georgeton and Hester estimated the solubility fraction in 1995, a number of supernatant and saltcake samples have been analyzed for Sn-126. The supernatant results are displayed in Table 3. When multiple results were listed, the value that was above detection limits was used. The average concentration for the listed sample results is 2.6E+3 pCi/ml. If this concentration is applied to the total tank farm supernatant inventory of 24M gallons (free and interstitial), a total supernatant inventory of 2.4E+2 curies is obtained.

A value of 2.4 nCi/g for Sn-126 in dry saltcake is reported in WSRC-TR-94-057[4]. Unfortunately, the recent salt samples for tanks 2, 3, 10, 29, 38 and 41 were not analyzed for Sn-126. Multiplying the single sample point value of 2.4 nCi/g times the tank farm inventory of 8.9E+7 kg of salt yields a salt inventory of 2.1E+2 curies. Combining this value with the calculated supernatant inventory of 2.4E+2 curies results in a total soluble Sn-126 inventory of 4.5E+2 curies. Since this number is higher than the value obtained from the theoretical fission yield method, the conservative value of 4.5E+2 curies should

be used for Saltstone disposal planning. It is also noted that the Sn-126 solubility fraction is closer to 20% rather than the estimated 5%.

Table 3 Sn-126 Supernatant Sample Results

Tank	Measured Sn-126 (pCi/ml)	Dilution Factor[6]	Sn-126 Actual (pCi/ml)	Source
13	1.5E+3	2.3	3.4E+3	
30	2.0E+3	2.8	5.5E+3	
37	2.2E+3	2.4	5.2E+3	WSRC-TR-2004 00386 R1[3]
45	5.3E+2	3.6	1.9E+3	
46	9.0E+2	2.9	2.6E+3	
49	6.2E+2	1.7	1.0E+3	
25	2.6E+3	NA	2.6E+3	
26	2.7E+3	NA	2.7E+3	
27	2.2E+3	NA	2.2E+3	
28	2.8E+3	NA	2.8E+3	
29	4.5E+3	NA	4.5E+3	WSRC-RP-93-1009[5]
30	2.0E+3	NA	2.0E+3	
32	1.8E+3	NA	1.8E+3	
38	7.7E+2	NA	7.7E+2	
43	7.2E+2	NA	7.2E+2	

5   Conclusion and Recommendation

The supernatant phase Se-79, Tc-99, and Sn-126 inventories, not currently tracked in WCS, are established in this document. Theoretical fission yield and solubility expectations were used to calculate these inventories, and sample results were used to confirm the calculations when available. For Se-79, the fission yield method and the sample data were in reasonable agreement, and, since the sample data were all less than detectable, the theoretical value of 8.9E+1 should be used. For Tc-99, the theoretical fission yield and solubility method yielded the more conservative value of 3.3E+4 curies. For Sn-126, available sample data indicate the solubility fraction of 5% is probably low by a factor of 4, and therefore, the soluble inventory for Sn-126 is calculated to be 4.5E+2 curies.

6   *References*

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4. Hobbs, D.T. and C. J. Coleman, Final Report: analysis of tank 41H saltcake sample #2, WSRC-TR-94-057, January 1994.

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6. Hester, J. R., WCS Supernate Radionuclide Concentration Algorithms, X-ESR-G-00004, August 2004.

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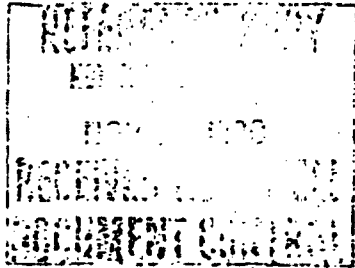
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PERMEABILITY, POROSITY AND CAPILLARITY OF HANFORD
WASTE MATERIAL AND ITS LIMITS OF PUMPABILITY

J. J. Kirk

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PERMEABILITY, POROSITY AND CAPILLARITY OF
HANFORD WASTE MATERIAL AND ITS LIMITS OF PUMPABILITY

J. J. Kirk

Waste Processing Technology Unit
Process Technology Group
Systems Engineering Department
Research and Engineering Function

August 1980

ABSTRACT

Parameters which determine the pumpability and drainability of the
interstitial liquid (IL) contained in the 500,000 to 1,000,000 gallon
single shell waste tanks located at Hanford are presented. The comple-
tion point of jet pumping is defined and the total quantity of drainable
interstitial liquid in single shell tanks is calculated.

INTRODUCTION

The 242-T evaporator and the In-Tank Solidification (ITS) Unit 1 and
ITS 2 facilities produced a salt cake that was deposited in single shell
tanks in the 1960's. The 242-S and 242-A evaporator-crystallizers produced
more salt cake material in the 1970's. A study was begun to determine the
flow properties of the liquid contained in the intersticies of this salt
cake in 1975. Mathematical formulae for modeling the inflow process of
interstitial liquid (IL) to a centralized salt well were developed . A low
flowrate jet pump prototype was then developed that could remove the IL
at the flow rates predicted by the equations.

The original flow studies relied on salt cake property data based on
laboratory work with synthetic material and engineering judgement. Actual
pumping data have now been obtained from the two jet pump prototypes
241-BY-107 and 241-S-111. Data have also been obtained from the 241-S-104,
241-S-106, 241-S0110 and 241-S-112 jet pump production units. These data
have been used to determine the salt cake properties that control the
inflow process to the salt well.

Material that existed before the evaporation-crystalization processes
has been historically identified as sludge. From an IL flow standpoint
this material would be undrainable because of capillary forces within the
material. In actuality some of these sludges will behave similar to
salt cake.

All drainable liquid is not necessarily pumpable. The liquid may drain
at such a low rate no pumping system available will remove the liquid.
Therefore, a limit has to be defined to differentiate between pumpable
and drainable liquid. The limit of pumpability has been defined as a
pumpout rate from the low flowrate jet pump system. The pumpout rate is
based on the equipment available and the IL flow characteristics from the
waste material.

SUMMARY

Current estimates of salt cake interstitial liquid (IL) flow properties
have changed somewhat from original predictions. Permeability values of
10-20 Darcies were initially estimated for salt cake. Calculations have
shown permeability in Tank 241-BY-107 varied from 10-14 Darcies. Calcula-
tions based on other tank data suggest that a range of 5-20 Darcies will
probably be encountered in pumping the waste tanks. Porosity estimates
of 30%-35% were used initially. Jet pumping data indicate that the range
of 30%-35% is too low and should be 40%-48%. Capillary heights of salt
cake have been predicted to vary from 0.5 to 2.0 feet. A capillary
height of one foot should be used for waste volume projections. Use of
one foot capillary height will result in a bulk volume of 33,000 gallons
of salt cake per tank being undrainable.

Sludge properties have been harder to obtain since most pumping experience
has been in pumping salt cake. A true sludge material would be unpumpable.
Pumping of tanks containing sludge has shown that some material defined
as sludge will have IL flow properties similar to salt cake.

The only method now available to determine whether a material classified as sludge is pumpable is to install a jet pump and attempt pumping. In an effort to quantify the amount of sludge that will behave similar to salt cake, a value of about 25 percent of the sludge volume was used.

The 0.05 gpm value of inflow rate to a salt well should continue to be used to define the completion of jet pumping. There will be 1 to 2 feet of drainable interstitial liquid remaining in a salt containing tank at the completion of jet pumping. This is equivalent to 15,000 to 30,000 gallons of drainable interstitial liquid or an average of 22,500 gallons per tank.

PERMEABILITY, POROSITY AND CAPILLARITY

Data collected from the jet pump prototypes and jet pump production units will be analyzed for permeability, porosity, and capillarity in this section. Each of these properties will be covered for both salt cake and sludge. Analysis of the production jet pump data has shown that more reliable data need to be taken: data from 241-S-105, 241-S-108, and 241-S-109 were not useable and some of the data from the other pumps had to be discarded because of inaccuracy. It is a relatively simple matter to correct this problem since most of the data collection is adequate except for "before and after pumping" interstitial liquid (IL) level readings from the liquid level control well. The liquid level in the salt well should be accurately taken before starting jet pumping and then again at shutdown when the liquid level in the well has equilibrated with the tank liquid level. Better determinations of porosities and permeabilities will result. Better estimates for liquid addition through dip tubes will need to be made in the future At the present pumping rates, steam condensate additions are small percentages of the total liquid pumped; however, as the liquid inflow rate drops, they will become a greater percentage. Liquid additions will have to be factored into permeability and porosity calculations.

Savannah River Site

Closure Business Unit
Salt Decontamination Project Development Team

Tank 41 Dissolved Salt Filtration Options

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August 16, 2003

PURPOSE

The purpose of this document is to propose filtration methods to treat dissolved Tank 41 salt solution.

BACKGROUND

Tank 41 was identified as a Low Curie tank. Interstitial supernate was drained and pumped away to Tank 49 and 39. What remained in Tank 41 was a salt cake with some percentage of residual supernate and sludge. The intent of the process is to pump the dissolved salt cake to Tank 50 where it would be sent on to Saltstone for final processing into grout.

Analysis performed on dip samples taken from Tank 41 following the salt cake dissolution indicate that Pu^{238} concentrations are above the Saltstone WAC limits[1]. On the expectation that further analysis of the samples will show that a significant portion of the Pu^{238} is filterable as an insoluble (in caustic solution) solid, a request has been made for filtration methods that would bring the Pu^{238} concentration within the Saltstone WAC limit.

Key Basis:

- Pu^{238} concentration must be reduced to meet Saltstone WAC limits (1.88E+5 pCi/mL)
- Pu^{238} is an insoluble solid in the dissolved salt supernate.
- The target filtrate rate is 20 gpm.
- The maximum case volume of supernate to be processed is 400,000 gallons.
- Short lead-in period is required to begin operation (approximately one month).

REVISION SUMMARY

Revision 0 included three proposals for filtration methods, key basis, and identified risks.

REFERENCES:

1. SRT-LWP-2003-00061, Rev 0, "Initial Results of Tank 41H Saltstone WAC Analyses"
2. USC-FRED-PSP-RPT-09-B-015, "Final Report on the Crossflow Filter Optimization with 5.6 M Sodium Salt Solution"

ATTACHMENTS:

1. Option Diagrams

Design Authority: _____ Date: _____

Concurrence: _____ Date: _____
Design Authority Engineer Manager: Michael R. Norton

Options #1
ITP Demonstration Filter:

OVERVIEW:
A proof of concept filter test was performed for ITP circa 1982-83. That test involved a crossflow filter unit (four filters in series) mounted in a Tank 48 riser. The filter unit's performance was supported by various equipment in other Tank 48 risers that included pumps, filtrate hold vessels, chemical cleaning and backpulsing capacity.

SCOPE
- Build and install a filter assembly based on the design of the ITP demonstration equipment:
    - Four crossflow filters in series
    - Pump, diaphragm or centrifugal
    - Backpulse reservoir
- Ancillary equipment:
    - Minimum three pressure transmitters
    - High pressure air source for backpulse (~ 150 psig)
    - Two flowmeters.
    - Three control valves
        - Filter back-pressure valve
        - Filtrate flow control valve
        - Air pressure to backpulse reservoir control valve

KEY ASSUMPTIONS:
- Filtration of dissolved Tank 41 salt through a nominal 0.5 micron filter will be sufficient treatment for it to meet the Saltstone WAC.
- The flux rate for a crossflow sintered metal filter of nominally 0.5 micron pore size is .05 gpm/ft^2 when processing supernate w/entrained sludge and MST.[2] The flux rate will be similar for a feed stream of dissolved salt with entrained sludge without MST.
- The filter assembly used for the demonstration has been disposed of.

RISKS:
- The demonstration filter design ran filtrate outside of the tank. A portion of filtrate was contained in a backpulse vessel. That demonstration included MST and STPB treatment of the waste which produced a low activity decontaminated filtrate. Filtrate produced from Tank 41 dissolved salt will not have gone through that same treatment.
- Significant design and construction time.
- This is a temporary application. The duration is limited by that of Tank 41 Low Curie operations. Significant costs may be put into the development of this filter assembly that does not see an equivalent return of product to justify the effort. The design and hardware developed for this may not be transferable to other tanks that may require similar treatment.
- The test in which this design demonstrated a filtrate rate meeting the 20 gpm target of this proposal was performed with a MST and STPB treated feed. Feeds without STPB but with MST and/or entrained sludge typically have much lower flux rates.

COSTS:
- The filter assembly is a unique piece of equipment requiring it to be fabricated.

- Though some vessels like the filtrate hold tanks and backpulse reservoir may be supplied from surplus tanks, significant reworking would be required to use them in this design.
- Structural supports for the vessels and any required additional shielding.
- Control software and electronics would be required.
- Total design and construction rough estimate: $900,000

SCHEDULE:
- Six to eight months due to the significant levels of design and construction required.

Options #2
Cleanable Sintered Metal Tube Filter:

OVERVIEW:
The concept of this proposal is to install a vendor built filtration unit to be mounted in a Tank 41 riser. The filter would incorporate sintered metal tubes similar to those in the 512-S filter unit with the tubes bundled and mounted in an outer shell.

The flow of waste will be into the tubes of the filter. Filtrate is forced through the capped sintered metal tubes out to the shell. The outlet of the shell side will connect to the existing transfer line from Tank 41 to 50 allowing the filtrate to be collected in Tank 50. Concentrated solids are collected inside tubes. At some level of accumulation the solids would be back-flushed out and back into Tank 41.

The filter unit will be mounted vertically within the C3 riser of Tank 41. Flush water would be used to discharge the accumulated solids back into Tank 41.

To accommodate the filter, the existing pump assembly will be removed and rebuilt. Otherwise the filter assembly may be designed for another riser. That would require two above ground transfer routes. One to connect the pump's riser to the filter and a second to return the filtrate to the appropriate nozzle located in C3.

SCOPE:
- Riser modification:
  - Mounting for the filter assembly
  - Inlet/outlet ports
- Addition of a flush line.
- Five manually operated (reach rod operated) valves will be built.
- Rebuild the Tank 41 pump assembly

KEY ASSUMPTIONS:
- The period between filter flushes will be long relative to the periods of filtrate production.
- The means of recovering filter operation, lowering the pressure delta across the media due to fouling, are limited to backflushing and chemical cleaning.
- The flux rate for a crossflow sintered metal filter of nominally 0.5 micron pore size is .05 gpm/ft^2 when processing supernate w/entrained sludge and MST.[2] The flux rate will be similar for a feed stream of dissolved salt with entrained sludge without MST.
- The filter will be of fine enough porosity to produce filtrate that meets the Saltstone WAC requirements.
- The filter will be capable of processing 400,000 gallons of filtrate before requiring replacement.
- The pumping mechanism can supply sufficient pressure to overcome the filter's pressure differential and push the filtrate on to Tank 50.

RISKS:
- Space constraints may limit the surface area of the filter such that the target of 20 gpm would not be maintainable.
- Final disposal of the filter will require special decontamination and or storage.

COST ESTIMATES:
- Engineering: $50, 000
- Material and construction: $100,000
  - Vendor supplied filter assembly: $25,000
  - Pump assembly, valve modifications, piping modifications: $75,000

SCHEDULE:
Estimated at six to eight weeks for design and construction.

Options #3
Disposable Cartridge Filters:

OVERVIEW:
The concept of this proposal is to install a vendor built filtration unit. The unit would consist of a bank of disposable cartridge type filters arranged in parallel in a filter housing. The filter housing with filters will be replaceable. Dissolved Tank 41 salt solution would flow into the bank and through the filters. Entrained solids will be captured on and in the filter media while filtrate passes through to Tank 50. To accommodate the filter, the existing pump assembly will be removed and rebuilt. Otherwise the filter assembly may be designed for another riser. That would require two above ground transfer routes. One to connect the pump's riser to the filter and a second to return the filtrate to the appropriate nozzle located in C3.

SCOPE:
- Riser modification:
  - Mounting for the filter assembly
  - Inlet/outlet ports
- Addition of a flush line.
- Five manually operated (reach rod operated) valves will be built
- Rebuild the Tank 41 pump assembly

KEY ASSUMPTIONS:
- The pumping mechanism can supply sufficient pressure to overcome the filter's pressure differential and push the filtrate on to Tank 50.
- The filter will be of fine enough porosity to produce filtrate that meets the Saltstone WAC requirements.
- The frequency of filter change-out will be low relative to the duration of the filters' operational duration.

RISKS:
- In the absence of a particle size distribution for Tank 41 dissolved salt, the duration of the cartridge filters' operation before full loading is unknown.
- Backflushing and/or chemical cleaning as means of filter recovery are not compatible with this design. Once the pressure delta limit is reached, the filters will require removal and replacement.
- The filter media of the cartridges is non-metallic. This will present special disposal challenges following filter removal.

COSTS:
- Engineering: $50, 000
- Material of construction: $95,000
  - Vendor supplied filter assembly (Two, one initial and one spare): $20,000 ($10,000 each)
  - Pump assembly, valve modifications, piping modifications: $75,000

SCHEDULE:
Estimated at six to eight weeks for design and construction.

OPTION COMPARISON:

Schedule:
Option #1 is expected to be the longest of the three due to the extensive engineering and construction of unique equipment required. It falls far outside the one-month target. Options #2 and #3 have similar schedule requirements.

Risk:
Relative to the other two Option #1 has the least technical risk. It incorporates technology that has been demonstrated to perform the filtration required. It also has the best flux recovery options since it would have a built in backpulse option and have a chemical cleaning capacity.

Option #2 uses similar technology, sintered metal tubes, but in a different way. Where Option #1 would cycle the concentrate stream from the tank through the filter and back to the tank where solids are collected, Option #2 is a single pass filter that collects the solids inside the filter itself. Option #2 would not have the same flux recovery options as #1. It can be backflushed but not backpulsed.

There are two primary technical risks for Option #3. First, the filter media is untested in this application. How it will hold up under a radiation field and in a caustic solution is unknown. Additionally, it is unknown how long the filters will take to reach their maximum solids load. Second there is no flux recovery option for this design. Once the filters reach their maximum solids load the filter housing must be replaced.

Cost:
The estimated cost associated with Option #1 far exceeds that of Options #2 and #3. Options #2 and #3 will have the same initial cost. Option #3 has the potential to be more expensive over the life of the project should the filters and housing require frequent replacement.

RECOMMENDATION:
Comparing the three options based on the criteria of schedule, risk, and cost leads the conclusion that Option #2 is preferable over the others. It has only marginally more technical risk than Option #1 and much less risk than Option #3. Option #2 falls much closer to the schedule target than #1. Option #2 has significantly less cost than #1 and based on the capacity for flux recovery may cost less than #3.

Option #1: ITP Demonstration Filter

A   filter shutoff   SS-2-12315
B   Backpluse air relief   SS-2-12396
C   backpluse air   13E-8-H-SD-15
G   air from compressor
F   filtrate drain to tank 48
K   process return to
    tank 48(0-100 A/O
    valve controlled
    by regulator on
    panalboard

Detector
(signal to
panalboard)

Level sensor detector a)
(signal to panalboard)

pressure c) switch

valve C

pressure transmitter

Valve A

Flowmeter c)
(0-80GPM)

valve G

150 PSIG AIR TO RESERVOIR

Air compressor
(see Dpsol 138-H-SD-10)

Backpulse
Reservoir

valve B

Valve F

Filtrate flow
to hold tanks 1,2
(see Dpsol 138-B-
H-SD-2,2A)

Filtrate return to
tank 48

Leaking
Valve

Filtrate flow
(Deconned salt)
to backpulse
reservoir

G
RISER
TANK 48

pressure transmitter c)

Valve K

Pressure Trans-c)
mitter

Flowmeter c)
(0-250 gpm)

Process flow from
Diaphragm or transfer pump

1) The backpressure reservoir has two liquid level detectors
One(LLL)which senses low level during step 9 of the
backpulse sequence and the second(HLL)which indicates the
backpulse reservoir has refilled with liquid during step
3 of the backpulse sequence.

1) Detector installed within the backpulse air exhaust pipe
in order to measure when filtrate has displaced the
backpulse air from the reservoir during step 3 of
backpulse sequence. Air exhaust through valve D.

Process return to
tank 48

Mott Filter

c)Readout on panalboard
inside tank 48 valvehouse

MOTT FILTER (manual valves not shown,see Dpsol 138-B-H-SD-3

Option #1:
ITP Demonstration Filter
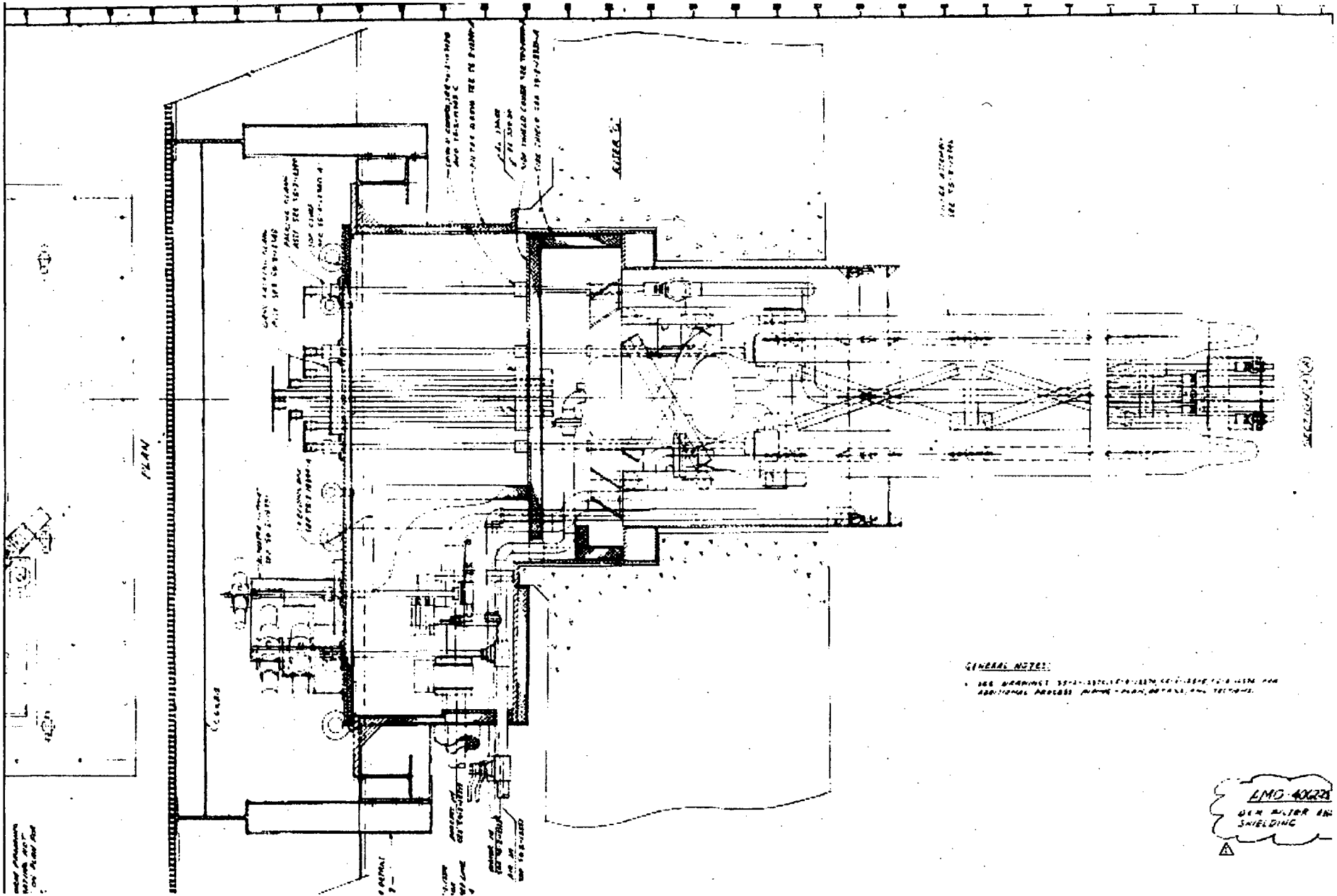
Option #2: Cleanable Sintered Metal Tube Filter

Water for Backflushing (Cleaning) Filter

Sintered Metal Tube Filter

3-Way Valves

C-3 Riser

Support Frame

Waste Flow "inside" Capped Sintered Metal Tubes

Flexible Hose

Tank 50 to Tank 41 Backflushing Path

Existing Gravity Drain Line to 2H Evaporator (Flow Path to Tank 50)

20 GPM Diaphragm Pump

Existing Caisson

Waste Flow Path

Backflushing Path

Backflushing Path Tank 50 to Tank 41
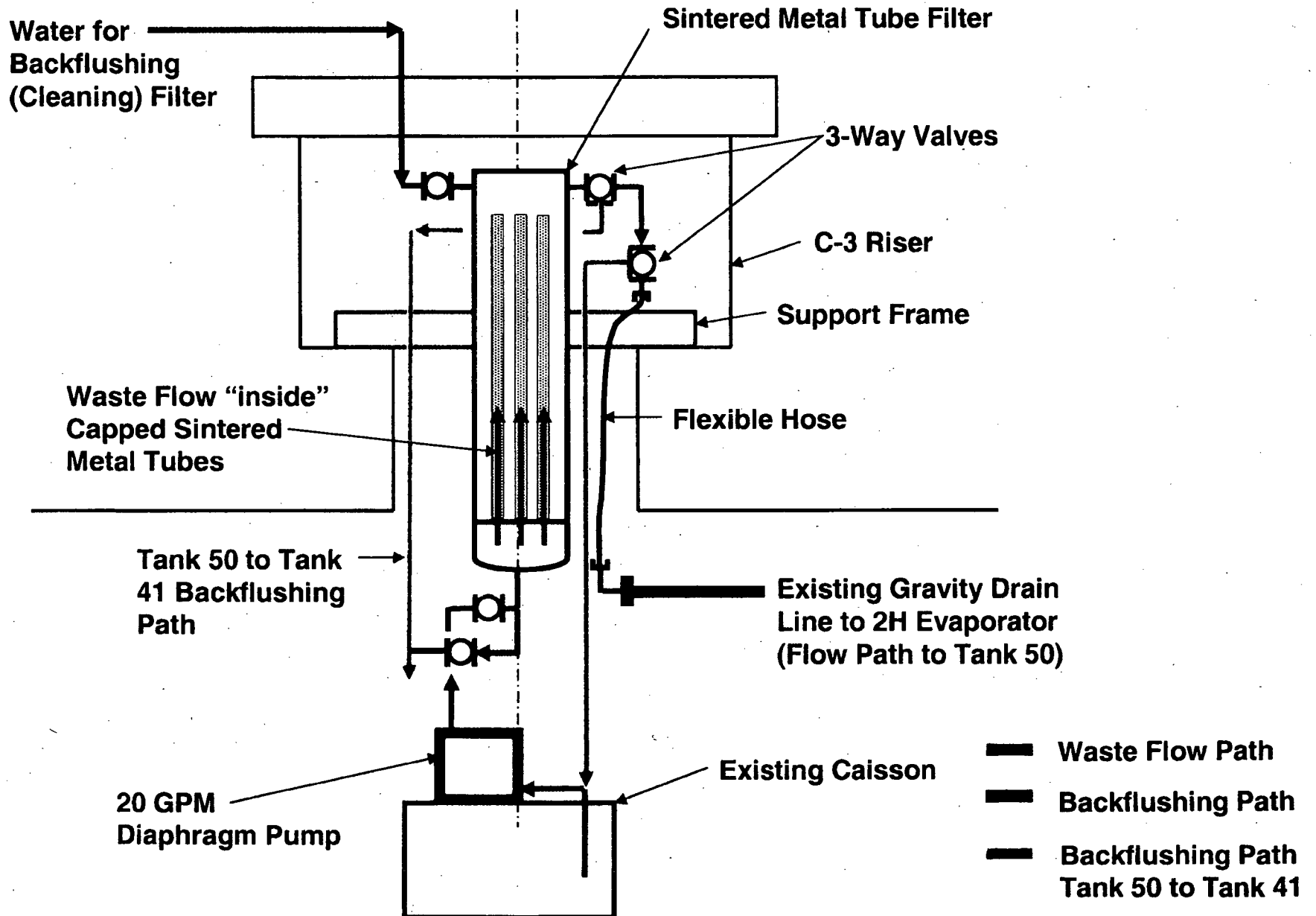
Option #3: Disposable Cartridge Filters

Disposable Cartridge Filters arranged In Filter Housing. (Note: This Filter Housing will be designed to be replaceable as a unit.)
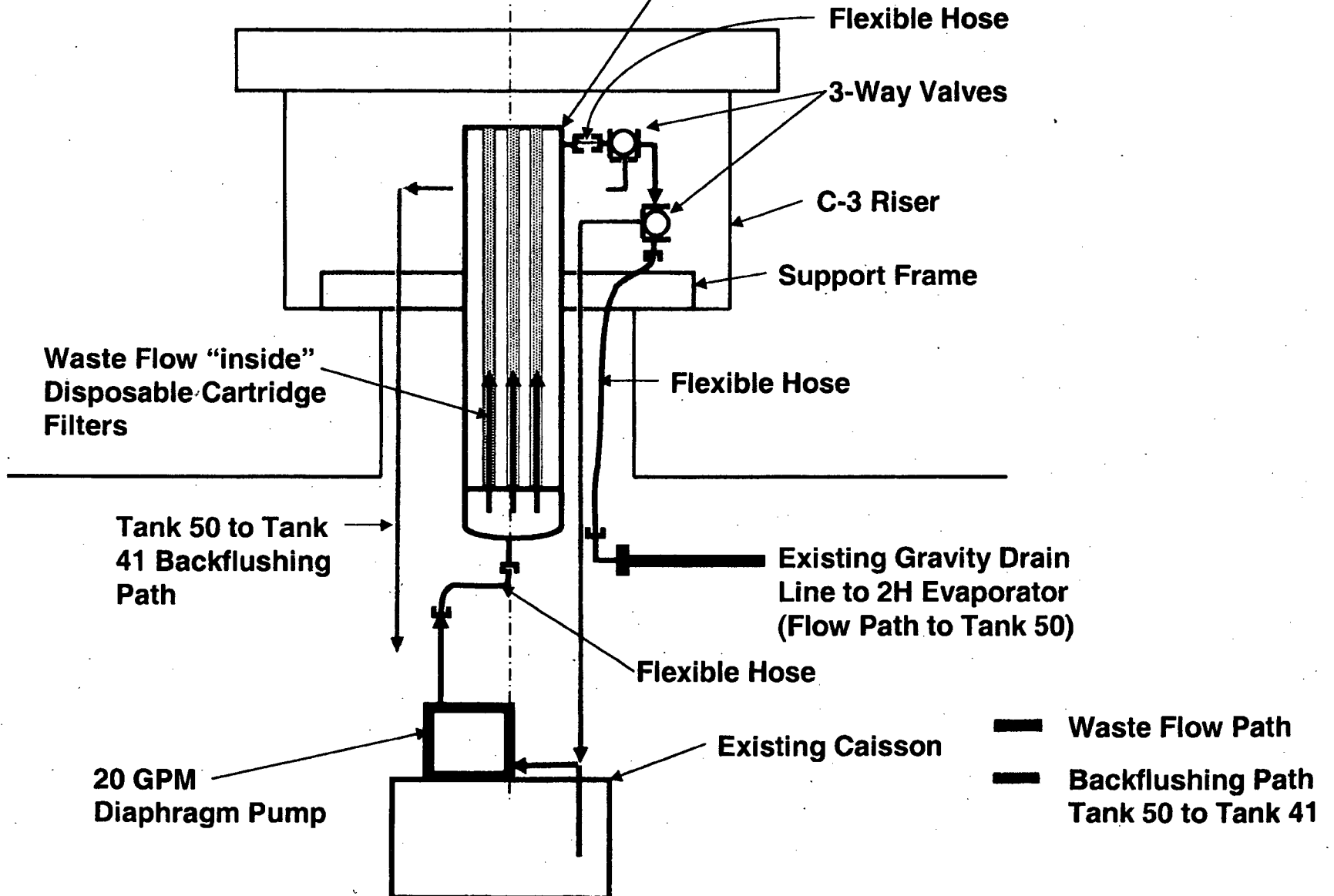
Flexible Hose

3-Way Valves

C-3 Riser

Support Frame

Waste Flow "inside" Disposable Cartridge Filters

Flexible Hose

Tank 50 to Tank 41 Backflushing Path

Existing Gravity Drain Line to 2H Evaporator (Flow Path to Tank 50)

Flexible Hose

20 GPM Diaphragm Pump

Existing Caisson

Waste Flow Path

Backflushing Path Tank 50 to Tank 41

ADC/RO 8/27/04

Interim Salt Processing Strategy Planning Baseline

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1. Summary

The Interim Salt Processing Strategy Planning Baseline contained in this document describes the planning basis for processing salt solutions through the Liquid Waste and Waste Solidification System until the start-up of the Salt Waste Processing Facility (SWPF) in April 2009. The purpose of the document is to provide a basis for planning salt processing activities during this period. The Planning Baseline is based on the Salt Processing Strategy, which has the following objectives:

*   Maintain sufficient space in the Tank Farms to allow continued Defense Waste Processing Facility (DWPF) Operations at a rate of 250 canisters per year
*   Support Sludge Batch preparation for DWPF
*   Provide tank space to support staging of salt solution adequate to feed 5 Mgal of salt solution to SWPF during the initial year of operation starting in April 2009
*   Ensure that the curies to Saltstone during the Interim Salt program are acceptably low (less than 5 MCi total).
*   Meet DOE Contract Minimum Gate of 500 kgal of salt solution dispositioned at Saltstone

Preliminary modeling showed that meeting all these objectives requires dispositioning the tetraphenylborate waste in Tank 48 so that the 1.3 Mgal of space in this tank can be used for Tank Farm service in staging salt solution for processing. Thus, Tank 48 must be dispositioned in some manner for the strategy to be successful. The two possible methods are 1) aggregate Tank 48 waste with other wastes going to Saltstone in such a manner that all Waste Acceptance Criteria (WAC) limits are met, or 2) chemically and thermally degrade the tetraphenylborate so that the Tank 48 waste can be processed with other Tank Farm wastes.

Four cases were examined with different methods and degrees of difficulty in dispositioning the material in Tank 48:

1. Aggregate Tank 48 waste with negligible issues
2. Aggregate Tank 48 with modest modifications and technical issues (recommended planning case)
3. Aggregate Tank 48 waste with more extensive modifications and technical issues
4. Thermally and chemically degrade the tetraphenylborate so that this waste can be sent to the Tank Farm.

Case 2 is the recommended case for the Planning Baseline. This case allows a reasonable amount of time to begin dispositioning the waste in Tank 48, considering the risks and technical uncertainties involved.

This case has the following attributes:

*   Processes salt solutions originating from Tanks 25, 28, 38, 41 and 48
*   Assumes that Tank 48 can be dispositioned through aggregation to Saltstone
*   Processes about 12 Mgal of salt solution to Saltstone containing less than 5 MCi (total of all radionuclides)

- Requires the start-up of Modular Caustic Side Solvent Extraction (CSSX) Unit (MCU) and Actinide Removal Process (ARP) by August 2007
- Includes disposition to Saltstone of 400 kgal of Low-Level Waste from processing unirradiated Highly Enriched Uranium (HEU) fuel, which reduces the amount of waste that must be managed in the Tank Farm
- Assumes regulatory issues and permits are completed per the schedule described in this document
- Accommodates current sludge batching plans, SWPF startup schedule, and Tank Farm minimum space requirements
- Feeds salt solution to Saltstone at a planned maximum activity of 0.2 Ci/gal of Cs-137.
- Assumes the use of Tank 24 to store concentrated recycle that is within limits for Type IV tank storage.

Case 4 is the backup case for the Planning Baseline. The main difference between this case and Case 2 are that Case 4 assumes the Tank 48 tetraphenylborate waste is thermally and chemically degraded, then sent to the Tank Farm, whereas Case 2 assumes the Tank waste is aggregated with the wastes going to Saltstone. Case 4 has a high technical risk— at this time, conditions required to adequately degrade the tetraphenylborate so that it can be managed as normal Tank Farm waste have not been identified.

These cases, their bases (including technical bases and assumptions), associated risks, and opportunities for improvement are described in the sections that follow. The opportunities for improvement identified in this document include continuing to pursue Tank 48 disposition by thermal and chemical degradation (the backup case) to minimize the amount of material requiring aggregation, acceleration of MCU operational start-up, increasing the Saltstone processing rate, and Slurry Mix Evaporator Condensate Tank (SMECT) stream processing. The Planning Baseline described in this document will be used in the Contract Execution development for all of the associated facilities.

This document will be revised when significant changes occur in the planning bases that impact successful implementation of the Planning Baseline. The document and any subsequent revisions will be incorporated into future HLW System Plan revisions. It will be maintained as a controlled document.

Removal of Highly Radioactive Nuclides from SRS Salt Waste

S. H. Reboul

APPROVED for Release for
Unlimited (Release to Public)

Table 1
Soluble, Insoluble, and Total Inventories in Untreated Salt Waste

Radionuclide	Soluble Curies in Untreated Salt Waste	Insoluble Curies in Untreated Salt Waste	Total Curies in Untreated Salt Waste
H-3	9.4E+3	Negligible	9.4E+3
C-14	5.2E+2	2.8E-1	5.2E+2
Na-22	5.1E+3	9.3E-1	5.1E+3
Al-26	2.4E+1	4.9E-1	2.4E+1
Co-60	8.6E+1	2.9E+4	2.9E+4
Ni-59	2.4E0	2.2E+2	2.2E+2
Ni-63	2.1E+2	1.9E+4	1.9E+4
Se-79	8.9E+1	1.3E+2	2.2E+2
Sr-90	2.8E+4	7.3E+6	7.3E+6
Nb-94	7.0E-4	5.9E-2	6.0E-2
Tc-99	3.3E+4	2.2E+3	3.5E+4
Ru-106	2.3E+3	5.7E+2	2.9E+3
Sn-126	4.5E+2	1.7E+2	6.2E+2
Sb-125	9.2E+3	1.5E+4	2.4E+4
I-129	1.8E+1	9.0E-3	1.8E+1
Cs-134	2.3E+5	8.8E+2	2.3E+5
Cs-135	3.9E+2	1.5E0	3.9E+2
Cs-137	1.1E+8	4.4E+5	1.1E+8
Ce-144	5.9E0	5.0E+2	5.1E+2
Pm-147	3.8E+3	3.2E+5	3.2E+5
Sm-151	4.3E+3	3.6E+5	3.6E+5
Eu-152	2.1E+1	1.7E+3	1.7E+3
Eu-154	9.1E+2	7.6E+4	7.7E+4
Eu-155	2.4E+2	2.0E+4	2.0E+4
Th-232	1.0E-1	2.4E-1	3.4E-1
U-232	2.9E-2	4.7E-2	7.6E-2
U-233	2.7E0	8.3E0	1.1E+1
U-234	4.2E0	3.2E0	7.4E0
U-235	8.4E-2	1.3E-1	2.1E-1
U-236	3.6E-1	5.2E-1	8.8E-1
U-238	6.8E0	5.5E0	1.2E+1
Np-237	4.2E0	8.3E0	1.3E+1
Pu-238	5.7E+4	1.7E+5	2.3E+5
Pu-239	3.4E+3	4.0E+3	7.4E+3
Pu-240	9.1E+2	1.7E+3	2.6E+3
Pu-241	3.8E+4	1.0E+5	1.4E+5
Pu-242	9.4E-1	2.8E0	3.7E0
Am-241	3.6E+2	3.0E+4	3.0E+4
Am-242m	2.1E-1	1.8E+1	1.8E+1
Am-243	7.0E-2	5.8E0	5.9E0
Cm-242	1.7E-1	1.5E+1	1.5E+1
Cm-243	4.1E-2	3.4E0	3.4E0
Cm-244	1.5E+2	1.2E+4	1.2E+4
Cm-245	1.4E-2	1.2E0	1.2E0
α-emitting TRU	6.2E+4	2.2E+5	2.8E+5

WESTINGHOUSE SAVANNAH RIVER COMPANY
INTEROFFICE MEMORANDUM

APPROVED for Release for
Unlimited (Release to Public)

CBU-SPT-2003-00157
Rev. 0
Retention: Until dismantlement or disposal
of facility, equipment, system, or process
Disposal Auth.: DOE 14-1.c (1)
Tracking #: ASD 100067

September 24, 2003

To: Renee H. Spires, Manager Salt Program Engineering, 766-H

From: E. Edward Seufert, Salt Program Design Authority Engineer, 766-H
        Michael R. Norton, Salt Program Design Authority Manager, 766-H

Evaluation of Tank 41 In-Riser Filter for Tank 41 to Tank 50 Transfer Path

This memorandum documents the evaluation of a proposal to install a filter in a Tank 41 Riser. The filter would be used to filter out insoluble actinides contained in a salt slurry being pumped from Tank 41 to Tank 50. The goal of this filtration process is to reduce the actinide content of the filtrate to a level that meets the Saltstone WAC (Waste Acceptance Criteria).

Summary and Conclusion:

Although it is technically possible to filter out the insoluble actinides from the transfer stream of salt slurry using a "dead end" type filter, the necessity to frequently backwash the filter with flush water to restore filtering capability makes the operation of the filter unacceptable. Depending on the solids loading (assumed to range from 300 ppm to 3000 ppm) the filter will have to be backwashed on a frequency ranging from 2 times/hour to 2 times/24 hour day. Each backwash cycle will introduce 200 to 300 gallons of flush water into Tank 41. In order to position the valving for backwashing, an operator will have to "dress out" in protective clothing and enter an enclosed hut. The necessity to backwash the filter frequently is operationally unacceptable.

Background:

The following background information was partially gleaned from **Reference 1**:

Tank 41 was identified as a Low Curie tank. Interstitial supernate was drained and pumped to Tank 49 and 39. What remained in Tank 41 was a salt cake with some percentage of residual supernate and sludge. The original intent of the Low Curie Process was to pump the dissolved salt cake to Tank 50 where it would be sent to Saltstone for final processing into grout.

An Analysis performed on a dip sample taken from Tank 41 following the salt cake dissolution indicated that Pu^{238} concentrations are above the Saltstone WAC limits. (Note: Pu^{238} is the primary actinide in Tank 41.) The sample contained insoluble solids at a concentration of approximately 3000 ppm.

A portion of the sample taken of the salt slurry in Tank 41 was run through a 0.45 micron absolute filter. An analysis showed that 83% of the Pu^{238} was filtered out **(See Page 11 of Reference 2)**. This is enough to meet the Saltstone WAC (Waste Acceptance Criteria). Based on this result it was proposed to install a "dead end" filter in a riser in Tank 41 to filter out Pu^{238}. (Note: A "dead end" filter passes the entire stream through the filter, retaining the solids on the filter elements and passing the liquids.)

Proposed Filter Specifications:

Based on the Tank 41 sample analysis and studies done of actual SRS Sludge **(See Reference 3)**, a filter specification was developed for filtering a 20 gpm salt slurry containing up to 3000 ppm solids (See Reference 4) and sent to Pall Filter and Mott Filter **(See Attachment 1)**.

Note: Actual SRS Sludge samples show that sludge is present in the hard salt at levels of from 300 to 600 ppm **(Reference 3)**.

Vendor Response:

Mott Filter did not respond to our specification.

Pall Filter responded with a Proposal for a "Backwashable Filter" **(See Attachment 2)**.

The proposed Pall Filter consists of a vertically mounted, nominal 12" OD by 85" long Filter Assembly consisting of 76 nominal ¾" sintered metal filter tubes. One end of the tubes would be welded into a tube sheet at the top end of the vessel. The other end of the tubes would be closed or "dead-ended".

The Filter Operation is as follows:
Salt Slurry flows around the outside of the tubes and is filtered through the tubes. Retained solids build up as cake on the outside of the tubes. Filtered liquid flows inside the tubes to the tube sheet and exits the vessel. A solid cake builds up on the outside of the tubes until the filter pressure drop is around 30 psig. At that time, Flush Water is used to "backwash" the filter by reversing flow through the filter tubes. A backwash flow rate of at least 100 gpm for 1 to 3 minutes (total flow of 100 to 300 gallons) is needed to clean the filter. This has the effect of knocking the filter cake off of the outside of the tubes. The dislodged filter cake falls to the bottom of the vessel and is drained out of the vessel through the normal "Inlet" nozzle. The total filter area is approximately 70 ft². A 20 gpm feedrate to the filter produces a filter flux rate of 0.29 gpm/ft².

The budget cost of this filter is $25,000 with a carbon steel outer shell and 316 stainless steel tubes. If SRS proceeds with this proposal, the filter will be rebid with a 304 or 316 stainless steel shell. The cost would be less than $5,000 more than the carbon steel vessel. It is assumed that SRS will purchase at least 2 filters. This will provide an option to replace the entire filter assembly if frequent backwashing results in a plugged filter unit.

Other Filter Designs Considered:
Pall Filter considered other filter designs and rejected them as follows:
(See Attachment 2, Sheet 2 of 3)

1. "Disposable filtration system is not recommended due to the high media viscosity and no way to backwash the system."

2. "Crossflow filtration system is not recommended due to much lower flux rate and the requirement for a larger recirculation pump. The overall system would be much more complex."

Proposed Design:

The proposed Filter System Design is illustrated in **Attachment 3** as follows:

1. The existing "Six Pumps on a Stick" Assembly originally designed to be installed in Tank 41 Riser C-3 will be installed in Tank 41 Riser C-1 after the existing "Interstitial Pump Assembly" is removed.
2. The proposed "Backwashable Filter" will be installed in Tank 41 Riser C-3. An above ground transfer line (hose in a hose) will be used to connect the discharge line from the "Pumps on a Stick" to the Inlet Nozzle on the Filter. The filtrate will be routed to the existing GDL (Gravity Drain Line) connection in Riser C-3. There is an existing defined Transfer Path to Tank 50 via the GDL in Tank 41 to the 2H Evaporator, through a new Jumper between Evaporator Nozzles 9 and 6, and then down the GDL to Tank 50.
3. The Filter will be backwashed using the existing Flush Water at Tank 41. The backwashed solids and the flush water will be drained into Tank 41.

Evaluation:

The major limitation of the proposed Filter System is the necessity to backwash the Filter to restore filter effectiveness after a period of filtering. As the Filter is used, solids will build up as a cake on the outside of the filter tubes. Assuming 20 gpm of 1.5 S.G. salt slurry at a insoluble solids loading of 3000 ppm (or 0.3 weight percent), after one hour the filter will have removed 40.5 lbs of insoluble solids assuming a 90% filtration rate:

Solids = .90 x 20 gal/min x ft^3/7.48 gal x 62.4 lbs/gal x 1.5 (S.G.) x 60 min/hr x 0.003 = 40.5 lbs.

Pall Filter was asked to estimate the backwash frequency assuming filter cake thicknesses (before backwashing) ranging from 0.094 inch thick to 0.160 inches thick at a solids loading of 3000 ppm. Their estimates were based on determining the time it would take to build up a specific filter cake thickness on the filter tubes. The results (**See Attachment 4**) show that the Filter would have to be backwashed from once every 47 minutes to once every 79 minutes depending on the assumed filter cake thickness. This is approximately once/hour. In a 24 hour period (assuming 5 minutes per backwash cycle including 2 minutes of actual backwash flow at 100 gpm) 26,400 gallons of waste would be filtered (and removed from Tank 41), but 4800 gallons of flush water would be added to Tank 41.

Even assuming an order of magnitude lower solids concentration of around 300 ppm (0.03 weight percent), the filter will have to be backwashed 2 to 3 times per 24 hour day.

The manual backwashing system will require an operator to "dress out" and enter a hut on the Tank Top to operate the valves. This is unacceptable at a frequency of 2 to 3 times/24 hour for a lower solids (300 ppm) concentration period. It is even more unacceptable at a solids concentration of 3000 ppm which will require backwashing at a frequency of once/hour or 24 times in a 24 hour period.

References:

Reference 1: CBU-SPT-2003-00141, Rev 0, "Tank 41 Dissolved Salt Filtration Options", August 16, 2003.

Reference 2: WSRC-TR-2003-00380, Rev 0, "Tank 41H Dissolved Saltcake Sample (HTF-E-03-91 – 92) Saltstone Waste Acceptance Criteria Analysis, August 28, 2003.

Reference 3: WSRC-TR-2003-00221, Particle Size of Simulated SRS Sludge, Actual SRS Sludge, and Monosodium Titanate, May 20, 2003.

Attachments:

Attachment 1: Filter Specification

Attachment 2: Pall Filter Budget Quotation GD091503 dated 9/23/03

Attachment 3: Pumps and Filter in Risers for Tank 41 to Tank 50 Transfer Path via 2H Evaporator

Attachment 4: Pall Filter Calculations by J. Larkin dated 9/17/03.

cc:
Electronic Distribution to:
Earl Brass
Dennis Conrad
Ginger Dickert
Brent Gifford
Lynn Harkey
Glen Johnson
Nick Kennedy
Bill Kerley
George Matis
Terry Ortner
Eloy Saldivar
SP Design Authorities, 766-H
File, 766-H

Attachment 1: LIQUID FILTER APPLICATION DATA SHEET

Completed By:E. Edward Seufert_____ Date:_8-28-03_____

Customer: Savannah River Site	Contact Name: E. Edward Seufert
Address:	Title: Design Authority Engineer
WSRC 766-H; Room 2032	Phone: 803-208-0310
Aiken, SC 29808	Fax: 803-208-8234
	E-mail: e.seufert@srs.gov

PROCESS DESCRIPTION: Filter Radioactive Salt Slurry with entrained sludge particles

REQUIRED FILTRATION PERFORMANCE: REMOVE 90% OF SLUDGE PARTICLES

LIQUID PROPERTIES	SOLIDS PROPERTIES
Composition: Radioactive Salt Slurry	Composition: Insoluble sludge containing radioactive actinides
Flow Rate: 20 gpm	Content (ppm or wt%): **Sample max at 3000 ppm (average 300 to 600 ppm)**
Specific Gravity or Density: 1.4 to 1.5	**Particle Size: 90% > 1.0 micron absolute; Would like to use a 0.5 micron absolute as the filter opening size.**
Viscosity: 6.1 cP @_____ °F or _25°C	Preferred Disposition of Solids:
Operating Temperature: °F or 29 °C	O Recovery X Disposal
Operating Pressure: <100 psig	**MATERIALS OF CONSTRUCTION:**
Maximum ΔP: 30 to 40 psig	Filter Porous Media: 304 or 316 stainless steel
DESIGN REQUIREMENTS	Filter Element Hardware: 304 or 316 stainless steel
Design Temperature (Min): 29 deg C	Vessel (Wetted): 304 or 316 stainless
Design Temperature (Max): 35 deg C	Vessel (Non-wetted): 304 or 316 stainless
Design Pressure (Min): Atmospheric	Housing Gasket: radiation resistant
Design Pressure (Max): Less than 150 psig	Element Seal Type: radiation resistant
	Element Seal Material: radiation resistant

**FLUID
FLOW**

PRODUCTS, INC.
MEASUREMENT/CONTROL
FILTRATION

QUOTATION

Page: 11
Date: 9/23/0309/15/03
Quote No.: GD091503

Please address order to:

FLUID FLOW OF THE CAROLINAS, INC.

Westinghouse Savannah River Co.
Attn.: Sam Newton & Ed Scuferi
WSRC
Aiken, SC 29808
Ref.: TANK RISER FILTERS

Item	Qty	Description	Unit Price	Extension
1	1	**Pall Backwash Filtration System**	$25,000.00	

Housing:
All welded Carbon Steel housing to ASME code. Design
drawing will not be submitted for customer review and
approval, due to delivery constraints. Please review attach
drawing.

Filter elements:
76 AccuSep 316 L Stainless Steel elements rated at 2 um
Absolute liquid particle removal rating at 99.9%
efficiency. Total filter area = 70 ft2, at 20 gpm the 70 ft2
system will be at 0.29 gpm / ft2. This flux rate is within
the range of other experiences with backwash applications.

5990-C Unity Drive
Norcross, GA 30071
770/446-5747
Fax 770/446-0386

**Fluid Flow of
Georgia, Inc.**
800-849-5947

2108 Crown View Drive
Charlotte, NC 28227
704/847-4464
Fax 704/847-2377

**Fluid Flow of
Carolinas, Inc.**
800-222-2229

6701 Baum Dr. Suite 245
Knoxville, TN 37919
865/588-1012
Fax 865/588-1094

**Fluid Flow of
Tennessee, Inc.**
800-275-0140
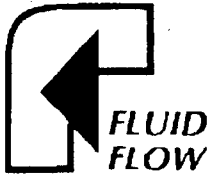
FLUID FLOW

PRODUCTS, INC.
MEASUREMENT/CONTROL
FILTRATION

QUOTATION

Page: 11
Date: 9/23/0309/15/03
Quote No.: GD091503

Please address order to:

FLUID FLOW OF THE CAROLINAS, INC.

Westinghouse Savannah River Co.
Attn.: Sam Newton & Ed Seufert
WSRC
Aiken, SC 29808
Ref.: TANK RISER FILTERS

NOTES:

1- Disposable filtration system is not recommended due high media viscosity and no way to backwash the system

2- Crossflow filtration system is not recommended due to much lower flux rate and require a larger recirculation pump. Overall system would be more complex.

3- Backwash with water will have to be optimized during testing and start with 100 gpm flush for 1-2 minutes to clear all solids from the vessel.

Terms & Conditions Attached
Quote valid for 30 days
Terms: Net 30 days
F.O.B.: DESTINATION
Delivery: 8 WEEKS ARO
NO penalties for late delivery.

3390 C Unity Drive
Norcross, GA 30071
770/446-5747
Fax 770/446-0385

2108 Crown View Drive
Charlotte, NC 28227
704/847-4464
Fax 704/847-2377

GREG DOUGLAS@843-349-5450
8701 Baum Dr. Suite 245
Knoxville, TN 37919
865/588-1012
Fax 865/588-1094

**Fluid Flow of
Georgia, Inc.**
800-849-5947

**Fluid Flow of
Carolinas, Inc.**
800-222-2229

**Fluid Flow of
Tennessee, Inc.**
800-275-0140

UPPER VENT
1 INCH 3000 LB CPLG

PROCESS OUTLET
1 1/2 INCH 3000 LB CPLG

BACKWASH INLET
1 1/2 INCH 3000 LB CPLG

TUBESHEET VENT
1 INCH 3000 LB CPLG

ELEMENT LAYOUT
(76) ELEMENTS LOCTED ON A 1 1/8 INCH PTCH
SCALE: NONE

PROCESS INLET AND PROCESS DRAIN
2 INCH 3000 LB CPLG

Attachment 3: Proposed Filter System Design
(Pumps and Filter in Risers for Tank 41 to Tank 50 Transfer Path via 2H Evaporator.)

1-1/2" Gates "Renegade" Internal Hose

3" Gates "Renegade" External Hose

Air Vent

Flush Water

Flow meter

Flush Water for Backwashing Filter

1 ½" 3-Way Ball Valve;

1-1/2" Gates "Renegade" Hose

Air Vent

1 ½" Ball Valve

C-1 Riser

1-1/2" Gates "Renegade" Hose

C-3 Riser

Drain Line

Tank 41

Gravity Drain Line to 2H Evaporator; Route to Tank 50

Discharge Lines from "Pumps on a Stick"

Sintered Metal Back-Washable Filter

Filter Operation; Flow is From Outside the tubes to Inside the Tubes; 76 Nominal ¾" Tubes; Filter Area = 70 sq ft.; Flux Rate = 0.29 gpm/sq ft at 20 gpm flowrate.

"Pump on a Stick"; Typical 6 Places at Various tank elevations.

Backwash Drain

Existing Caissons

9-24-03
Revised: Z Robinson

PALL CORPORATION
PTM CORTLAND, NY 13045 · USA
CALCULATION SHEET
PROJECT: WSRC – TANK RISER

$Q = 20\ gal/min$ @ $\rho = 93.6\ lbm/ft^3$

$\dot{m} = 20\ \frac{gal}{min}\left(\frac{ft^3}{7.48\ gal}\right)\left(\frac{93.6\ lbm}{ft^3}\right) = \boxed{250.3\ lbm/min}$

SOLIDS @ 300 TO 3,000 ppmW

$\dot{m}_{sol} = 300\ ppmW\ \therefore\ 0.0003(250.3\ lbm/min) = \boxed{0.0751\ lbm/min}$

$\dot{m}_{sol} = 3000\ ppmW\ \therefore\ 0.003(250.3\ lbm/min) = \boxed{0.751\ lbm/min}$

$\dot{V}_{sol} = 0.0751\ \frac{lbm}{min}\left(\frac{ft^3}{50\ lbm}\right) = \boxed{0.0015\ ft^3/min}$ ASSUMED SOLIDS ρ

$\dot{V}_{sol} = 0.751\ \frac{lbm}{min}\left(\frac{ft^3}{50\ lbm}\right) = \boxed{0.0150\ ft^3/min}$

<u>CASE 1</u>: ⅛" CAKE THICK ($\therefore\ 0.886\ ft^3/VSL$)

@ 300 ppmW: $\frac{min}{0.0015\ ft^3}\left(\frac{0.886\ ft^3}{VSL}\right) \sim \boxed{590\ mins/BW\ CYCLE}$

@ 3,000 ppmW: $\frac{min}{0.015\ ft^3}\left(\frac{0.886\ ft^3}{VSL}\right) \sim \boxed{59\ mins/BW\ CYCLE}$

✉ **CONFIDENTIAL**

DESIGN ORIGINATOR | DATE | CHECKED BY | DATE 9/17/03

PALL CORPORATION
PTM CORTLAND, NY 13045 — USA
CALCULATION SHEET
PROJECT:- WRSC (Cont) -

PROJECT No.
PAGE ② OF ②
ORIGINATOR J. LIPKIN
DATE 9/17/03

CASE 2: 0.160" CAKE THICK ($\therefore$ 1.186 ft³/VSL)

@ 300 ppmW: $\dfrac{\text{min}}{0.0015 \text{ ft}^3}\left(\dfrac{1.186 \text{ ft}^3}{\text{VSL}}\right) \sim$ | 790 mins / BW CYCLE |

@ 3000 ppmW: $\dfrac{\text{min}}{0.015 \text{ ft}^3}\left(\dfrac{1.186 \text{ ft}^3}{\text{VSL}}\right) \sim$ | 79 mins / BW CYCLE |

CASE 3: 3/32" CAKE THICK ($\therefore$ 0.640 ft³/VSL)

@ 300 ppmW: $\dfrac{\text{min}}{0.0015 \text{ ft}^3}\left(\dfrac{0.640 \text{ ft}^3}{\text{VSL}}\right) \sim$ | 427 mins / BW CYCLE |

@ 3000 ppmW: $\dfrac{\text{min}}{0.015 \text{ ft}^3}\left(\dfrac{0.640 \text{ ft}^3}{\text{VSL}}\right) \sim$ | 17 mins / BW CYCLE |

✉ CONFIDENTIAL

DESIGN ORIGINATOR | DATE | CHECKED BY | DATE 9/17/03

Letter Report for Westinghouse Hanford Company

A Simplified Model of Saltcake Moisture Distribution

C. S. Simmons

September 1995

⊛Battelle

PNL-10803

MASTER

Executive Summary

This letter report describes the formulation of a simplified model for finding the moisture distribution in a saltcake waste profile that has been stabilized by pumping out the drainable interstitial liquid. The model is based on assuming that capillarity mainly governs the distribution of moisture in the porous saltcake waste. A steady upward flow of moisture driven by evaporation from the waste surface is conceptualized to occur for isothermal conditions. To obtain hydraulic parameters for unsaturated conditions, the model is calibrated or matched to the relative saturation distribution as measured by neutron probe scans. The model is demonstrated on Tanks 104-BY and 105-TX as examples. A value of the model is that it identifies the key physical parameters that control the surface moisture content in a waste profile. Moreover, the model can be used to estimate the brine application rate at the waste surface that would raise the moisture content there to a safe level. Thus, the model can be applied to help design a strategy for correcting the moisture conditions in a saltcake waste tank.

US-CE-C Property of the
United States Government

US Army Corps
of Engineers
Waterways Experiment
Station

Interstitial Fluid Displacement for Preferential Recovery of Cesium from Saltcake

by Kimberlie Staheli, John F. Peters

1 Introduction

Background

The Savannah River Site (SRS) currently has 51 tanks that contain high- level radioactive waste created from fuel reprocessing activities. These tanks hold approximately 30 million gal[1] of waste that must be converted into more stable waste forms for long term storage. Each tank contains varying volumes of sludge, saltcake, and salt solution. In tanks with predominantly saltcake, the radioneuclide present in the highest concentration is Cesium-137. Because Cesium's daughter product, Barium-137m, is a strong emitter of gamma radiation, pipelines containing solutions of dissolved saltcake emit high radiation rates. Such pipelines require heavy shielding, usually several inches of lead or several feet of soil or concrete, to reduce the radiation rates. Due to the porous nature of the saltcake, it has been suggested the Cesium-137 could be removed from the saltcake by displacing the interstitial fluid with uncontaminated water. If the preferential removal of Cesium with fluid displacement methods is feasible, the material remaining in the tank after Cesium-137 removal could be transported through pipelines with much less shielding, making its removal, processing, and handling much less expensive.

Objective

The purpose of this study was to determine the feasibility of using interstitial fluid displacement (IFD) methods for preferential Cesium-137 recovery. This was determined by performing numerical analyses using a three-dimensional finite element computer model for simulating flow and transport, FEMWATER (Lin and Richards 1997).[2] The finite element mesh that was developed to model the system was created on the Department of Defense (DoD) Groundwater

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[1] A table of factors for converting non-SI units of measurement to SI units is found on page vi.

[2] Lin, Hsin-Chi J., and Richards, David R. (1997). "FEMWATER: A three-dimensional finite element computer model for simulating density dependent flow and transport," Technical Report CHL-97-12, U.S. Army Engineer Waterways Experiment Station, Vicksburg, MS.

Modeling System (GMS). Details on the GMS system can be found in the DoD GMS reference manual (1996).[1]

Scope and Approach

The goal of this study was to determine the feasibility of reducing the amount of Cesium-contaminated interstitial pore fluid in the saltcake media. It was hoped that by reducing the mass of Cesium-137 in the tank by pumping or draining the pore fluid, the gamma radiation emitted by the saltcake media could be lowered to allow transportation of the saltcake in unshielded transfer lines. Experience at the SRS tank farm shows that pipelines transporting waste with Cesium contamination levels less than 0.05 Ci/gal generally do not require shielding. For the purpose of this study, a contamination level of 0.05 Ci/gal was used as the pipeline shielding limit.

Due to the high levels of contamination in the tanks, substantial constraints exist in the process by which the removal of Cesium can take place. These constraints include limited access to the tanks and a restricted pumping location not central to the saltcake media. These restrictions hinder the recovery process. Additionally, each tank contains a series of cooling coils, which will interrupt the flow of fluid toward the pumping location. Further, the physical properties of the saltcake media are unknown due to the complications associated with collecting and testing samples from the tank. Therefore, it was necessary to make assumptions pertaining to the material properties and to run a series of simulations establishing reasonable limits on the results.

The salt forms around the cooling coils more quickly than in the areas between coils leading to a nonuniform salt structure. In addition, large volumes of waste are added to the tanks periodically, causing dissolving of the existing saltcake and crystallization of new saltcake. From viewing photographs of the inside of the tanks, it was evident that the saltcake around the cooling coils had different material properties than the material between the coils. It was assumed, therefore, that the saltcake in the tanks possessed a high level of heterogeneity which is known to lead to more localized flow paths than those expected in homogeneous media. Because the success of the Cesium removal depends on uniform movement of interstitial fluid, variability in saltcake permeability was considered as important as its mean value. Accordingly, two types of saltcake media were examined in this study: homogeneous media and heterogeneous media.

Two different homogeneous media simulations were performed to establish the sensitivity of the Cesium removal to the permeability of the media. The first was assigned the best estimate of the media average of permeability, based on a comparison in size and porosity to sand. The second simulation was assigned a media average with an optimistically low permeability. Although the permeability of the saltcake material in the tanks was presumed to be lower than the

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[1] Department of Defense. (1996). *Groundwater Modeling System Reference Manual* Vol 2. Brigham Young University.

permeability assigned in this simulation, the results form a basis for determining the sensitivity of the results to variation in permeabilities.

Two heterogeneous media simulations were performed to establish the sensitivity of the system to the degree of heterogeneity. A preliminary simulation was performed on a simple mesh to determine the effects of the heterogeneous salt forming process by assigning lower permeability properties around simulated cooling coils. This simulation resulted in discontinuities in the mesh resulting in numerical difficulties. These results of the preliminary simulation were found to be very similar to a simulation where material properties were randomly distributed through the saltcake matrix. In the random distribution, preferential flow paths were established that were similar to the simulation with the cooling coils; however, the mathematical discontinuities were not observed. Based on the results of these preliminary simulations, the heterogeneous media was developed by randomly assigning material properties to the saltcake media.

In the first simulation, the heterogeneous media contained nine materials with varying permeabilities and that ranged over four orders of magnitude (from 10^{-3} to 10^{-7} cm/sec). These materials were distributed randomly in the model. The second heterogeneous simulation also had nine materials with varying permeabilities; however, the assigned permeabilities were varied over a smaller range (from 10^{-2} to 10^{-4} cm/sec).

The simulations with homogeneous media were designed to determine the "best case" scenario with uniform flow and drainage paths. In addition, these simulations would establish a baseline for comparison of results of simulations where heterogeneity was introduced.

The purpose of the heterogeneous simulations was to evaluate the sensitivity of the Cesium recovery to heterogeneity that is undoubtably caused by the cooling coils. In both types of simulations, a uniform Cesium concentration was assumed throughout the tank. Details on the material properties are provided later in this report.

Summary of Results

In the drained method, Cesium removal was limited by the residual saturation of the media. In both the drained and pumping methods, removal was limited by the nonuniform flow patterns that caused a significant amount of the saltcake media to be by-passed. Both methods were found to be highly susceptible to heterogeneity because of the increased tendency for flow to become localized, thus by-passing an even greater volume of media. For both methods, continued pumping contributed little additional Cesium recovery but produced considerably more contaminated water.

The draining method removed a larger percentage of Cesium from the system compared with continuous pumping and recharge. In addition, the volume of waste produced during the draining removal process was substantially lower than the volume generated in the pump and recharge case. While the Cesium level in the tanks was reduced significantly by both methods of removal, concentrations

were not reduced to the low levels required for transportation without shielding. This was largely due to zones of little or no flow through the salt media.

The problem of moving the water from dead zones is one of changing the flow patterns. The pump location cannot be altered without incurring large operational costs; however, it is possible to insert recharge wells with reasonable operational costs. Simulations showed that the recharge wells were very efficient at moving water from the dead zones.

Due to the complexity of the intermittent removal, it is recommended that a Cesium recovery method be designed based on draining the tank and displacing Cesium left in dead zones with a controlled flood through recharge wells. Finally, the tank could be cleaned with a controlled, top-down dissolution and selective removal of water. This combined method has the advantage of being least sensitive to media properties and provides the greatest control of the Cesium recovery.

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Interstitial Fluid Displacement for Preferential Recovery of Cesium from Saltcake

by Kimberlie Staheli, John F. Peters

2   Development of the Model

Finite Element Mesh

A finite element mesh was designed to represent the saltcake contained within a Type IIIA tank. The mesh consisted of 3,021 nodes and 2,030 elements. Figure 1 displays the finite element mesh used in this analysis. The volume of a Type IIIA tank is 1.3 million gal (4,925 m³). The volume of the saltcake media was assumed to be 4,099 m³ with an interstitial fluid volume of 902 m³ (22 percent). No-flow boundary conditions were established on the walls and bottom of the mesh, representing the tank walls. For the drain simulations, a fixed-pressure boundary condition was set at seven nodes located at the bottom of the mesh, representing an area of zero pressure head. For the pumping simulations, one node was assigned a source/sink boundary condition with a constant flow rate and concentration. For all simulations, the initial conditions consisted of a constant total head, which represented hydrostatic conditions with the fluid level at the top surface of the saltcake media.

Material Properties

The first series of model studies was performed assuming the tank was filled with a completely homogeneous media representing the salt matrix. The porosity of the material was assumed to be 22 percent. Based on this assumption, the permeability of the material was assumed to be 5×10^{-4} cm/sec (0.018 m/hr) and was equal in the x-, y-, and z-directions. The bulk density of the material was assigned a value of 95 lb/ft³ (1524 kg/m³).

The relationship between moisture content and pressure head in the saltcake was established using the correlations developed by van Genuchten (1980)[1] for a typical sand with a residual saturation of 17 percent. The moisture content curve is shown in Figure 2. This curve defines the relationship between moisture content and pressure head in the unsaturated zones created in the media as the water level is lowered. For the saltcake media, the value of moisture content ranged between 0.22 for the saturated moisture content and 0.0374 for the residual moisture content.

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[1] van Genuchten, M. T. (1980). "A closed-form equation for predicting the hydraulic conductivity of unsaturated soil," *Soil Sciences Society of America Journal* 44, 892-898.

After the first drain cycle, 29 percent of the Cesium was retained in the tank, compared with 52 percent in the heterogeneous media with a broader range of permeabilities. The amount of Cesium remaining in the tank with the lower range of permeabilities is similar to the homogeneous material with a narrow range of permeabilities except there is no reduction in the Cesium after the second drain. The difference in the amount of Cesium retained in the two heterogeneous simulations is not substantial considering the material properties were randomly distributed in the media. However, the development of preferential flow paths is evident in this simulation as no additional Cesium is removed during the third or fourth drain cycle. This simulation clearly shows that heterogeneity will have an enormous impact on the amount of Cesium that can be recovered from the system.

Overview of Drain Simulations

Table 8 shows the results of the drain simulations in the four different salt-cake media. As expected, after four complete drains, generating 2,450 m^3 (646,680 gal) of high-level waste, the lowest contamination levels were seen in the tanks with homogeneous media. When heterogeneity was introduced to the system, the development of preferential flow paths was evident as very small amounts of Cesium were removed after the second drain cycle.

Table 8
Cesium Retained in Tank (Percent Original Mass), Overview of All Drain Simulations

Number of Drain Cycles	Homogeneous Permeability, 5×10^{-4} cm/sec	Homogeneous Permeability, 1×10^{-2} cm/sec	Heterogeneous Permeability, 10^{-3} to 10^{-7} cm/sec	Heterogeneous Permeability, 10^{-2} to 10^{-4} cm/sec
1	32	29	52	29
2	17	20	39	18
3	13.4	18	36	18
4	12.8	17	35	18

During the drain simulations, the highest level of Cesium recovery was found to be 87.2 percent, after four complete drains of the completely homogeneous media system. The mass of the Cesium remaining in the tank was 12.8 percent of the original mass. It is hoped that after the completion of the four drain cycles, the saltcake can be dissolved and removed from the tanks by mixing the media at a 2:1 ratio of water to saltcake and the contamination levels will be below the pipeline shielding limit (0.05 Ci/gal). The volume of the saltcake media is 4099 m^3; therefore, the dissolved saltcake and water has the volume of 12,297 m^3. The maximum allowable concentration of Cesium is 0.00015 kg/m^3. To remain below the pipeline shielding limit, the maximum mass of Cesium contained in the tank after the fourth drain is 1.845 kg. Since the volume in the tank represents 12.8 percent of the original mass of Cesium, the maximum original mass of Cesium per tank is 14.41 kg, equivalent to a concentration of

Supernatant Phase Iodine-129 Inventory

H. Q. Tran

APPROVED for Release for
Unlimited (Release to Public)
6/23/2005

Westinghouse Savannah River Company
Closure Business Unit
Planning Integration & Technology Department
Aiken, SC 29808

Supernatant Phase Iodine-129 Inventory

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Table of Contents

Summary of Revisions

2/2005   Revision 0; Initial Issue

1   Summary

A previous supernatant I-129 inventory was estimated based on a constant I-129 concentration in all waste tanks.[1] However, a revised estimation method based on analytical data and process history would project a more reliable supernatant I-129 inventory. The purpose of this memorandum is to provide the basis to determine the soluble Curie inventory for I-129. Using this basis, the projected I-129 curie inventory in the supernatant phase is 17.4 Ci.

2   Introduction

Projections of the total inventory of I-129 in the SRS Tank Farm were reported in CBU-PIT-2005-00033, Rev. 0.[1] These projections were based on theoretical fission yield data and on the assumption that the I-129 supernatant phase concentration was constant in all waste tanks. Although there were clear uncertainties associated with these bases, the projection approach offered a relatively simple means of estimating total I-129 based on very limited data.

Since the CBU-PIT-2005-00033 document was issued, additional I-129 data were identified and a request for honing in on the supernatant phase I-129 inventory was received. Available analytical supernatant I-129 data are utilized to determine an average I-129 /Cs-137 activity ratio for the supernatant phase. I-129 supernatant inventories are estimated for each waste tank based on the activity ratio and knowledge of the supernatant Cs-137 inventories. Because supernatant Cs-137 concentrations in waste tanks are measured on a regular basis, this projection method is very well suited for estimating I-129.

3   Inputs and Assumptions

Reference date for WCS information: 2/1/2005
Interstitial supernatant volumes accounted for in total supernatant volume, Ref. 2
Volumes of Tanks 26 and 39, the currently H & F Area-Canyon receipt tanks, were updated through 11/2004, Ref. 3
I-129 Half life = 1.57E+07 yr, Ref. 4
I-129 Specific Activity = 1.77E-04 Ci/g, Ref. 4
The interstitial liquid fraction is assumed to be 0.3
Tank 12H is adjusted to be a wet tank

4   Computational Methods

Analytical data listed in Table 1 is utilized to estimate the I-129 inventory in the supernatant phase. The supernatant phase includes the free supernatant liquid, interstitial liquid in saltcake, and interstitial liquid in sludge.

Table 1: Iodine-129 and Cesium-137 Analytical Concentrations

pCi/mL	I-129	Cs-137
13H*	5.36E+02	3.54E+09
30H*	3.78E+02	2.32E+09
Average	4.57E+02	2.93E+09

   *   Supernatant sample analysis, Ref. 5

4.1   Estimates based on I-129 reported concentration

Reported I-129 concentrations of 7.66 pCi/mL (Ref. 6, 100" sample) and 2.15 pCi/mL (Ref. 7, average of sample 1 and sample 2 results) are utilized for the Tank 24H and 50H supernatant phase inventory estimates, respectively. The Tank 23H reported I-129 concentration of <0.658 pCi/mL (Ref. 6, 100" sample) is applied to this tank and other Type IV tanks (21H, 22H). The I-129 inventories for these tanks are estimated by multiplying the analytical concentrations by the appropriate total supernatant volumes, and data can be found in Table 2.

Table 2: I-129 Inventory in the Supernatant Phase

Tank	Total Supernatant Volume (gal)	I-129 (pCi/mL)	I-129 (Ci)
21H	696,261	<6.58E-01	<1.73E-03
22H	819,701	<6.58E-01	<2.04E-03
23H	1,294,029	<6.58E-01	<3.22E-03
24H	1,217,760	7.66E+00	3.53E-02
50H	347,841	2.15E+00	2.83E-03
Total	N/A	N/A	<4.51E-02

1.2   Estimates based on I-129 to Cs-137 ratio

The I-129 to Cs-137 activity ratio is established based on reported concentrations for Tanks 13H and 30H as listed in Table 1. The average analytical activity ratio is 1.57E-07 and is utilized to estimate the I-129 inventory for each waste tank not already quantified in section 4.1. A previous study established a bounding ratio of 3.23E-07, based on data of reactor assemblies that H-canyon processes to generate the HM waste[8]. The analytical ratio (1.57E-07) is about 50% of the bounding ratio (3.23E-07), which seems reasonable when uncertainties surrounding iodine processing losses are taken into account.

Cesium-137 inventory in the supernatant phase is reported in WCS.[2] Projected I-129 inventory is then established for each waste tank by multiplying the I-129/Cs-137 activity ratio by the appropriate Cs-137 inventory. Results of the calculations are given in Table 3. Based on the results, the total I-129 inventory in the supernatant phase is estimated to be 17.4 curies.

Table 3: Supernatant Phase Cs-137 and I-129

Tank	Reported Cs-137 Inventory(Ci)	Projected I-129 Inventory (Ci)
1	3.33E+06	5.23E-01
2	1.21E+06	1.89E-01
3	1.22E+06	1.92E-01
4	3.52E+06	5.54E-01
5	4.93E+04	7.75E-03
6	9.98E+03	1.57E-03
7	2.40E+05	3.78E-02
8	7.06E+03	1.11E-03
9	1.22E+06	1.92E-01
10	8.22E+04	1.29E-02
11	2.61E+03	4.10E-04
12	4.16E+05	6.54E-02
13	1.18E+07	1.85E+00
14	1.25E+06	1.97E-01
15	NS	NS
16	NS	NS
17	CLOSED	CLOSED
18	1.38E+02	2.16E-05
19	2.12E+02	3.33E-05
20	CLOSED	CLOSED
25	1.74E+06	2.74E-01
26	5.59E+06	8.78E-01
27	5.10E+06	8.02E-01
28	2.22E+06	3.48E-01
29	4.73E+05	7.43E-02
30	1.06E+07	1.67E+00
31	5.23E+06	8.22E-01
32	5.26E+06	8.27E-01
33	2.91E+06	4.58E-01
34	1.42E+06	2.22E-01
35	4.15E+06	6.53E-01
36	1.11E+07	1.74E+00
37	4.60E+06	7.22E-01
38	1.89E+05	2.97E-02
39	8.35E+05	1.31E-01
40	5.19E+04	8.16E-03
41	3.51E+05	5.51E-02
42	1.23E+07	1.93E+00
43	1.47E+05	2.30E-02
44	2.99E+06	4.70E-01
45	2.55E+06	4.01E-01
46	5.02E+06	7.89E-01
47	6.72E+05	1.06E-01
48	1.72E+04	2.70E-03
49	5.39E+05	8.48E-02
51	2.17E+05	3.42E-02
Total from Tanks 21-24 and Tank 50	N/A	<4.51E-02
Tank Farm Total	1.11E+08	1.74E+01
NS: No Supernate		

5   Conclusion and Recommendation

The supernatant phase I-129 inventory, not currently tracked in WCS, is established in this document. Analytical data were utilized to establish an I-129 to Cs-137 activity ratio, and this ratio was multiplied by the reported Cs-137 inventories to yield the estimated I-129 inventories. Using this approach, the total projected I-129 inventory for the SRS Tank Farm is 17.4 Ci. It is recommended that the results of this analysis be incorporated into the Waste Characterization System.

To reduce the uncertainty of I-129 projections in the future, it is recommended that additional supernatant I-129 analytical data be obtained.

6 References

[1] CBU-PIT-2005-00033, Rev. 0, "Total Iodine-129 Curies Inventory Report", H. Q. Tran, February 14, 2005.

[2] //WG17/WCS1.5PROD/WCS 1.5, February 1, 2005

[3] //WG17/WCS1.5PROD/SLUDGE 1.5, January 31, 2005

[4] WSRC-TR-99-00344, Rev. 0, "Compilation of Recent Radionuclide Data for Specific Activity, Specific Heat and Fission Product Yields", A. Gibbs, October, 1999.

[5] X-ESR-G-00004, Rev. A, "WCS Supernate Radionuclide Concentration Algorithms", J. R. Hester, August, 31, 2004.

[6] WSRC-TR-2003-00112, Rev. 0, "Results of Analyses of Tank 23H and 24H Saltstone WAC Samples HTK-521-HTK-528"; R. F. Swingle, March 12, 2003.

[7] WSRC-TR-2004-00625, Rev. 1, "Characterization of Tank 50H Material Per Saltstone Waste Acceptance Criteria Analysis Requirements-2004", L. N. Oji, D. J. McCabe,

[8] WSRC-TR-2001-00564, Rev. 0, "Impact of Iodine-129 on Low Level Waste from Separation Facility Operations", M. Chandler, November 20, 2001.

DPST-83-695

TECHNICAL DIVISION
SAVANNAH RIVER LABORATORY

MEMORANDUM

ACC. NO. 105552

July 21, 1983

TO: M. A. EBRA

FROM: D. D. WALKER, B. A. HAMM *DDW*
BAH

SAMPLE ANALYSES FROM THE FULL SCALE IN-TANK
DEMONSTRATION OF THE PRECIPITATION PROCESS

SUMMARY

A full scale demonstration of in-tank salt processing was completed
by the Savannah River Plant (SRP) in April 1983. In this test,
427,000 gallons of radioactive salt solution were decontaminated by
a combination of precipitation, adsorption, and filtration. At
each stage of the demonstration, samples of the salt solution and
slurry were processed in the High Level Cells (HLC) and then ana-
lyzed by Analytical Development Division (ADD) of the Savannah
River Laboratory (SRL). The results of these analyses are reported
in this document. It is intended to be a complete listing of ana-
lytical information for future interpretive reports.

INTRODUCTION

Between February and April of 1983, a major achievement in high
level waste processing was accomplished at the Savannah River
Plant. A full scale demonstration of in-tank processing of salt
solution was successfully completed. More than 99.9% of the radio-
activity in 427,000 gallons of high level waste salt solution was
removed by precipitation and adsorption. Salt solution from Tank
24H was transferred to Tank 48H (a new, unused Type IV waste stor-
age tank) for processing. Radioactive cesium was removed by pre-

TABLE I.A

PHYSICAL PROPERTIES OF THE INITIAL SALT SOLUTION

Property	1AP[1]	1AD[2]	Sample 1BP[3]	1BD[4]	1CV[5]
Density (g/ml)	1.28	1.30	1.31	1.26±.01	1.26±.01
Insoluble Solids (ppm)	59	111	67	--[6]	--[6]
Soluble Solids (wt%)	--	--	--	--	33±1

---

[1] Tank 24H supernate pump sample.

[2] Tank 24H dip sample.

[3] Tank 24H peristaltic pump sample taken during transfer.

[4] Tank 48 dip sample after transfer (cold chemical heel solution included).

[5] Tank 48 dip sample after transfer completed.

[6] Insoluble solids not measured due to presence of cold chemical heel solution.

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WSRC-TR-96-0160 (U)

1189549

AN INVESTIGATION OF DENSITY DRIVEN SALT DISSOLUTION TECHNIQUES (U)

B. J. Wiersma

Savannah River Technology Center
Applied Science and Engineering Technology Department
Materials Technology Section

APPROVED for Release for
Unlimited (Release to Public)

Publication Date: August, 1996

Westinghouse Savannah River Company
Savannah River Site
Aiken, SC 29808

An Investigation of Density Driven Salt Dissolution Techniques (U)

1.0 Summary

Laboratory experiments were performed to support the salt dissolution demonstration in Tank 41H. This demonstration is sponsored by the Office of Science and Technology. The tests were designed to investigate three techniques of density gradient type dissolution: (1) Drain-Add-Sit-Remove, (2) Modified Density Gradient, and (3) Continuous Salt Mining. Removal of saltcake left attached to equipment after the solution was removed (i.e., perched saltcake) with a water jet was also simulated. The desire was to improve upon past salt dissolution operations and develop a safe, efficient and cost effective means for future operations. The experiments were performed on simulated Tank 41H saltcake that was formed in a three foot long rectangular trough.

The primary safety issue is whether or not the dissolved salt solutions that are produced will cause significant corrosion degradation of the tank walls or cooling coils. The parameter investigated during the tests to answer this question was the concentration of corrosion inhibitors in the dissolution water added to the simulated saltcake. It was observed that dissolution with 1 M sodium hydroxide solution resulted in salt solutions that were within the current corrosion technical standards for fresh waste storage. This result was independent of the density gradient technique. However, if inhibited water (0.01 M sodium hydroxide and 0.011 M sodium nitrite) was utilized, the salt solutions were frequently outside the technical standards. In the early stages of dissolution, the interstitial liquid which has high concentrations of corrosion inhibitors maintained the salt solution within corrosion standards. The Drain-Add-Sit-Remove technique was the most effective means for maintaining the interstitial liquid. Interstitial liquid entrapped by the surface tension between crystals, mixes with the salt solution as it flows through the saltcake matrix. The other two techniques, modified density gradient and continuous salt mining tend to displace the interstitial liquid. However, even with the Drain-Add-Sit-Remove technique as more salt was dissolved, the chemistry of the resulting salt solutions became outside the corrosion technical standards.

The corrosion technical standards are being evaluated for their application to waste removal. The current standards apply primarily to storage of fresh waste. The temperatures during waste removal are expected to be much lower than the maximum temperatures allowed during waste storage. The saturated salt solutions which form during waste removal have very high nitrate concentrations (> 4 M) and usually have a very high pH (12-13). Stress corrosion cracking may be a concern at these chemistries and at high temperatures (> 75 °C). Pitting is usually a concern in dilute solutions (nitrate concentrations < 1M) and at lower pH (9.5-10).

The lower temperatures may require less corrosion inhibitors to prevent stress corrosion cracking. Corrosion tests are in progress to investigate the potential for stress corrosion cracking in salt solutions similar to those removed during tests with inhibited water as the dissolution water. Therefore these solutions will have lower levels of corrosion inhibitors than are specified in the corrosion technical standards. If these tests show that carbon steel is not susceptible to corrosion damage, recommendations for a corrosion technical standard applicable to waste removal will be made.

A comparison between the three dissolution techniques and several key observations of the dissolution process that impact the efficiency and cost of operations are summarized

in Table 1. The primary parameters investigated during the tests to address these issues were the dissolution water addition rates, the salt solution removal rates, and the depth at which the outlet line was located. Salt dissolution was observed to be a very rapid process as salt solutions with densities between 1.38-1.4 were frequently removed. Given the fast dissolution rate, the techniques needed to be performed in a manner that ensured good contact between the unsaturated salt solution and the saltcake (i.e., no channeling or short-circuiting). Slower addition and removal rates and locating the outlet line at deeper levels below the top of the saltcake provided better contact between the dissolution water and the saltcake. Near the bottom of the trough, salt solutions with lower densities were removed (1.15-1.25). It is likely that the removal rate was too fast, given the shallow depth of the outlet line, to provide adequate contact time between the dissolution water and the saltcake. Slower removal rates than those tested are recommended in order to achieve a higher salt solution strength for salt near the bottom of a tank.

2.0 Introduction

Radioactive waste is stored in 44 carbon steel tanks at the Savannah River Site. To economize on waste tank space, evaporators are utilized to reduce the volume of waste. After the hot waste is returned from the evaporator to the waste tank, soluble salts precipitate as the waste cools. The mixture of soluble salts, insoluble metal oxides, and soluble fission products and insoluble fissile material is commonly referred to as saltcake. Twenty of the waste tanks are currently utilized for salt storage.

The In-Tank Precipitation (ITP) process precipitates the soluble fission products in order to prepare feed for the Defense Waste Processing Facility. The feed for ITP is the saltcake and its interstitial liquid. The saltcake must be dissolved in order to provide the feed. SRS is currently evaluating safe, efficient and cost effective means for salt dissolution and waste removal. Given the schedule for waste processing, the rate at which the saltcake is removed is not a determining factor.

As part of this evaluation, the Office of Science and Technology (OST) is sponsoring a demonstration of salt removal from Tank 41H. The salt dissolution techniques which will be demonstrated are: (i) single slurry pump method, (ii) modified density gradient method, and (iii) a yet to be determined technique for removal of insoluble solids.

Removal of saltcake was performed in Type IV tanks and Tank 10H, a Type I tank, during the late 1970's and early 1980's. The techniques utilized then were steam jet circulation, density gradient and slurry pump agitation [1]. This demonstration is designed to test improvements to these processes. For example one of the concerns from the earlier salt removal operations was non-homogeneous vertical saltcake dissolution resulting in mounds of inaccessible salt at the edges of the tank [2]. It is desirable to develop a method in which the saltcake profile remained horizontal. Another problem that occurred was perched saltcake [3]. This phenomenon occured when the liquid did not completely cover the salt. An upper crust of saltcake dries and remained clinging to the cooling coils, while the salt below continued to dissolve. To dissolve a crust similar to this in Tank 10H, the tank was refilled with additional dissolution water.

One of the primary concerns during salt removal is that the salt solutions produced are potentially corrosive [4]. The soluble salts which precipitated have a high concentration of nitrate anion (an aggressive corrosion species) and relatively low concentrations of hydroxide and nitrite anions (corrosion inhibitor species). Therefore, corrosion inhibitors must be added to the dissolution water to ensure that the salt solutions produced are not corrosive towards the tank walls or cooling coils. The options being considered are inhibiting the dissolution water so that the sodium hydroxide concentration is between

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DPST-79-269

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TO: A. S. JENNINGS

FROM: D. L. KISER

         DLK.

BENCH-SCALE INVESTIGATION OF DIFFERENT
CONCEPTS FOR WASTE TANK SALT DISSOLUTION

INTRODUCTION AND SUMMARY

For the Phase I waste removal program, salt will be removed from waste tanks
to be retired by batching in fresh water as a layer above the salt cake.
Steam agitation jets are to be mined into the salt cake and will recirculate
the supernate to dissolve the salt. Steam, used to power the agitation jets,
will add condensate to the supernate and heat the contents of the tank. After
four volume turnovers of the supernate and a cool-down period, the supernate will
be jetted to an evaporator feed tank. The process will be repeated until most
of the salt is dissolved. The major disadvantage with this method is the long
cool down periods (totaling up to 10 months/tank for Type I and 24 months/tank
for Type IV) required for liquid transfer due to elevated temperature in the
tank.

Two alternate salt dissolution concepts have been investigated on a bench-scale.
The density driven concept appears to have about the same dissolution rate as a
steam agitation jet without the elevated temperature condition. Mechanical
agitation is superior to the density driven concept in both dissolution rate and
fresh water requirements. Both the density driven concept and mechanical agitation
are proposed to be tested in the Phase I program, to confirm the bench scale test
results with radioactive waste salt.

INFORMATION ONLY

BACKGROUND

Introduction

The only previous salt dissolution in the plant was done in Tank 22. The best results from Tank 22 dissolution became the basis for the Phase I waste removal program. Because of problems (extra funding uncertainty and waste tank temperature elevation) with the steam agitation jet method proposed for Phase I, the Waste Management group of SRL was asked by Separations Technology to investigate alternate concepts.

Phase I Method as Proposed by the Engineering Department

Tank 22 experiments revealed that mining a steam agitation jet into the salt cake using small liquid batches gave the best dissolution with a rate during circulation of about 0.01 ft/hr. For Phase I, two steam agitation jets in each tank are to be mined into the salt cake to a depth of about 8 feet initially. Approximately 90,000 gallons of water and steam will be added to each batch, and each jet will circulate the liquid at 75 gpm until four volume turnovers or less are completed. The objective is to dissolve 30,000 gallons of salt cake. After a cool-down period, a transfer jet, already installed in the tank, will be used to remove the concentrated supernate to the evaporator feed tank.

SIMULATED SALT CAKE EXPERIMENTS

Preparation of Salt Cake

Salt cake was prepared by mixing chemicals together in a beaker to form a concentrated synthetic waste of composition given below. The solution temperature was raised to the boiling point and the mixture boiled until about 20% of the original volume evaporated. The solution was then cooled slowly to room temprature, resulting in the nitrate, nitrite, and aluminate salts crytallizing to form a solid salt cake.

Component	Molar Concentration
NO_3^-	3.0
NO_2^-	2.9
OH	9.0
$CO_3^=$	0.1
Na^+	11.6
Al^{+++}	1.0
Cl^-	0.05
$SO_4^=$	0.01

Mechanisms Affecting Dissolution Rate in Waste Tanks

Previous work with simulated salt cake suggested dissolution rates in the range of 0.1 to 6 ft/hr.   Results of Tank 22 salt dissolution showed clearly that the salt dissolution rate was much higher for salt cake in a vertical plane than for salt cake in a horizontal plane. Bench-scale salt dissolution tests were conducted in a 6-inch diameter beaker to determine separate salt dissolution rates for horizontal and vertical salt surfaces. Figure 1 and 2 show the experimental set up. In Figure 1, salt was crystalized in the bottom of the beaker. In Figure 2, an annular ring of salt was formed on the beaker wall. For the experiment shown in Figure 1, tap water was added as batch. After a 20-30 minute lag time for initial dissolution, supernate was continuously pumped from the beaker until dry. Then another batch of water was added and the cycle repeated. In Figure 2, tap water was added as a batch just filling the center of the salt annulus. Water was then continuously added and supernate continuously removed from the bottom at the same rate as the salt cake dissolved.

Because dissolution in the waste tanks using the steam agitation jet will be at elevated temperature (80-90°C),[1] a dissolution experiment was performed in a 6-inch diameter beaker with hot tap water (Figure 3). In Figure 3, a vertical hole was made in the salt cake for supernate removal. Water and supernate were continuously added and removed, respectively. The dissolution rate was increased about 17% with the 20°C increase in temperature. Water vapor losses from the glass vessel at 40°C were considered large enough so as to preclude meaningful salt dissolution experiments at a higher temperature with the open system.

The last mechanism to be investigated was mechanical agitation. In a 6-inch diameter beaker, all the tap water was added as a batch and the liquid was agitated with a laboratory stirrer (Figure 4). The stirrer was a 2-blade propeller, 2 inches in diameter. The supernate was monitored for specific gravity, and agitation was terminated when a specific gravity of 1.3 was reached.

Density Driven Concept

The density driven concept involves mining a vertical well into the salt cake like that proposed for the steam agitation jet method. Higher density supernate flows into the well by the force of gravity and is removed by pumping or jetting from the bottom of the well. In Tank 22, a mined well proved to be the significant factor in reaching an acceptable dissolution rate in the horizontal plane. With a steam agitation jet, dissolution rates in Tank 22 without the vertical well were 0.003 to 0.006 ft/hr compared to 0.013 ft/hr with the well. With the density driven concept, water may be continuously added while the concentrated supernate from the bottom of the well may be continuously removed. The concept utilizes primarily equipment already installed in waste tanks. Since steam is not required for the density driven concept, long cool down periods** and additional ventilation system requirements for salt dissolution are eliminated.

Results of a comparative test of the density driven concept with simulation of the circulation pattern of a steam agitation jet are given in Table I. For the test, salt cake was formed in 12" nominal diameter jar to a depth of 3" (Figure 5). A roll-flex tubing pump was used to remove or circulate the liquid. To create a well, a hole was jetted into the salt cake with water using a stainless steel rod attached to the pump tubing. Operating and structural conditions were scaled in proportion to a full size waste tank as much as possible. For the circulation

(1) DP-1135, C. B. Goodlett, Concentration of Radioactive Wastes
**   totaling up to 10 months/tank for Type I and 24 months/tank for Type IV

jet concept, discharge of the pump circulation with the withdrawal point was proportioned relative to that of a steam agitation jet planned for Phase I. As a terminating point in the dissolution, a specific gravity target of 1.3 was selected. The specific gravity of 1.3 was the best achievable in Tank 22 at a satisfactory dissolution rate in the horizontal plane. The dissolution rates achieved indicate the circulation jet has no advantage over density driven flow at room temperature. If the circulation jet dissolution rate was adjusted for the temperature elevation expected in salt tank from the steam agitation jets, a dissolution rate of 0.015 would be anticipated. The change in the rate is calculated based on the previous temperature effect work (see "Mechanism Affecting Dissolution Rate in Waste Tanks") and salt solubility data.

Mechanical Agitation

Table I also includes comparative tests with mechanical agitation. For these tests, three laboratory stirrers were placed in the 12" nominal diameter jar at scaled positions of Type IV tank risers (Figure 5). The 2-blade impellers were positioned at 1/4" to 1/2" above the initial surface of the salt cake. Blade diameter (1") were scaled in proportion to 1/12 scale model that was tested at the Engineering Test Center.

Specific gravity was used to monitor the agitation time. Tests were run with the agitation terminated at 1.3 and 1.4 specific gravity. Results tabulated in Table I show that at 1.3 s.g., the dissolution rate was about 4 times higher than the dissolution rate for the density driven method. By extending the agitation period to get 1.4 s.g., the rate was reduced to about 2 times the density driven method, but in addition the water requirements were cut in half.

CONCLUSIONS AND RECOMMENDATIONS

Dissolution rate with the density driven concept is about the same as with a steam agitation jet. Due to the advantages of the density driven concept over the steam agitation jet method, the density driven concept should be tested in a waste tank at the beginning of the Phase I waste removal program.

Mechanical agitation has shown considerable promise in both increasing the dissolution rate and reducing water consumption. In addition, vigorous agitation in a salt tank may be required to remove sludge which may settle out of the salt and blanket the salt surface, thus significantly reducing the dissolution rate. It is recommended that testing the mechanical agitation concept be included as part of Phase I salt removal. A test could be conducted with a spare Tank 16 Bingham pump suspended above a salt heel in a Type IV salt tank.

Density Driver Concept was Recommended.

DLK/aa

TABLE 1

RESULTS OF THREE DIFFERENT SALT DISSOLUTION CONCEPTS

	Density Driven	Circulation Jet	Mechanical Agitation	Mechanical Agitation
Cake Dissolution Rate (ft/hr)	0.0100	0.011	0.042	0.019
Supernate Removed sp.gr. Target (+0.01)	1.30	1.30	1.30	1.40
Supernate Removal Rate (ml/min)	11*	35**	***	***
Water Added to Salt Cake Dissolved Ratio (gram:gram)	1.1	1.2	1.1	0.6

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*   Continuous, equivalent to 23 gpm on an 85-foot diameter waste tank.
**   Recirculated, equivalent to 75 gpm on an 85-foot diameter waste tank.
***   Pumped at termination as a batch.

Figure 1:   HORIZONTAL PLANE DISSOLUTION

Dissolution Rate: 0.002 ft/hr
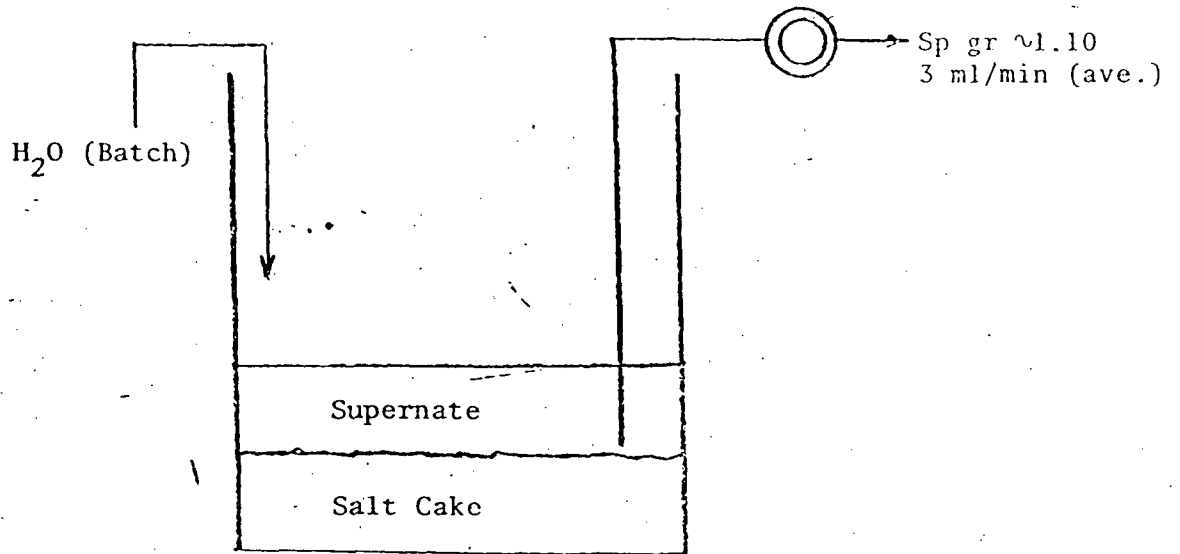
Figure 2:   VERTICAL PLANE DISSOLUTION

Dissolution Rate: 0.1 ft/hr
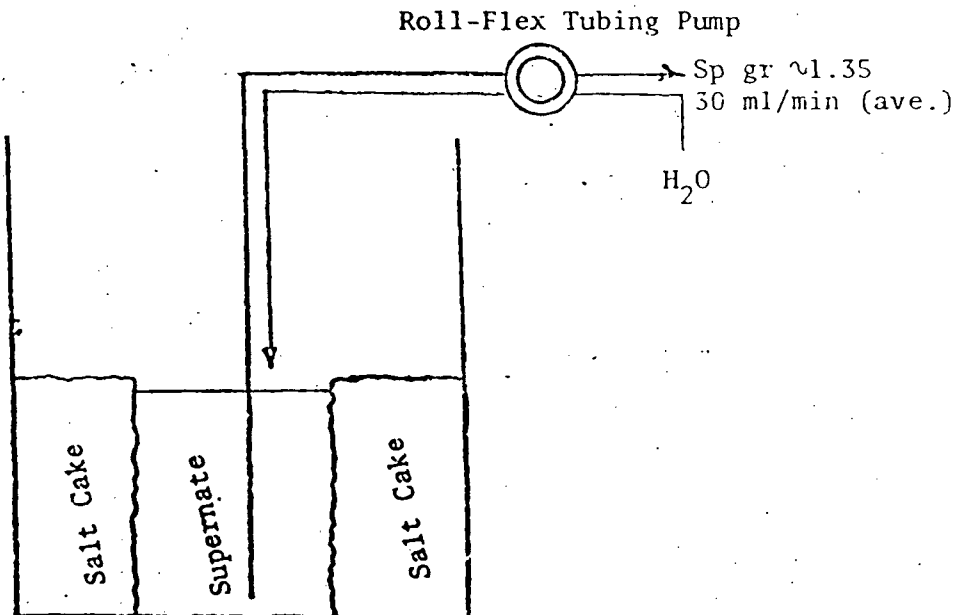
Figure 3:   HOT WATER DISSOLUTION

Dissolution Rate: +0.004 ft/hr of a control at
                    ambient temperature (20°C)
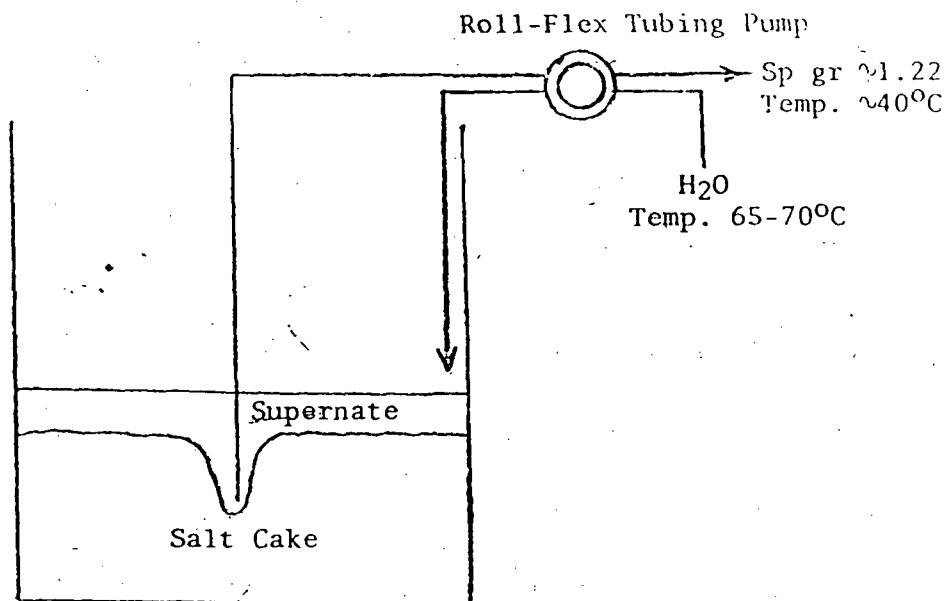
Figure 4:   MECHANICAL AGITATION DISSOLUTION
            Dissolution Rate: 0.04 → 0.09 ft/hr
            Stirrer Speed: 425 → 1200 rpm
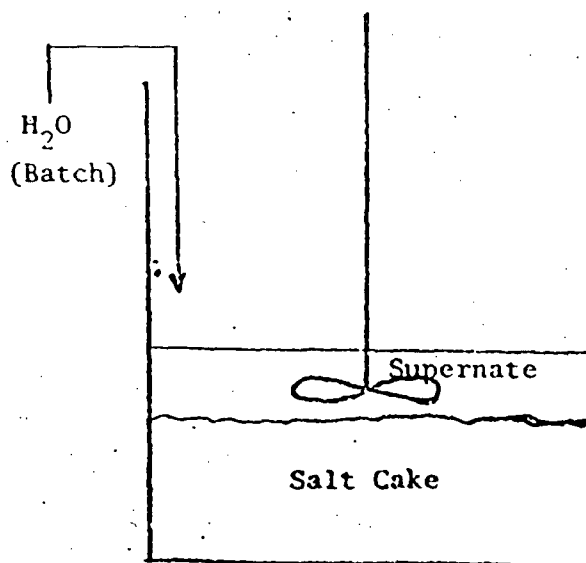
Figure 5: WASTE TANK SALT DISSOLUTION CONCEPTS

Density Driven

H_2O

Supernate

Salt Cake

Circulation Jet

H_2O
(Batch)

Supernate

Salt Cake

Mechanical Agitation

Three Stirrers
Stirrer Speed: 1000 ± 100 rpm
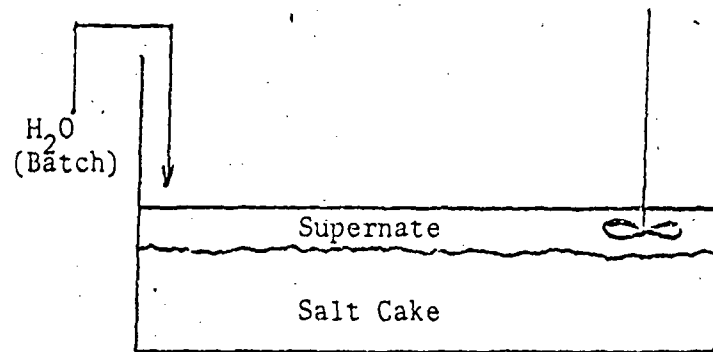
H_2O
(Batch)

Supernate

Salt Cake

Evaulation and Ranking of the Tank Focus Area Solid Liquid Separation Needs

by

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SUMMARY

Proposed treatment processes for High Level Waste at Hanford
and Oak Ridge National Laboratory (ORNL) include pretreatment
to separate insoluble solids from the aqueous waste.
Crossflow filtration, dead-end filtration, and settling are
methods applicable to these separations. Testing is needed
for selection of the appropriate technique for each
application. Crossflow filtration is most applicable to the
ORNL Radiochemical Engineering Development Center (REDC)
waste and the ORNL Newly Generated Low Level Liquid Waste
(NGLLLW) treatment. In addition, the Hanford Tank Waste
Remediation System (TWRS), ORNL Melton Valley Storage Tank
(MVST) TRU Processing Facility, and the ORNL Gunite Tank
program may require crossflow filtration. Backflushable
cartridge filters are probably the best treatment method for
the MVST supernate cesium removal program and part of the
NGLLLW treatment process. Some of the streams require a
combination of methods, such as a coarse settling and a
polishing filtration stage.

Solid/liquid separation work conducted at the Savannah River
Site (SRS), ORNL, Hanford, and West Valley is summarized and
the applicability to current needs discussed. General
information about crossflow filtration and design of
filtration experiments is also included.

WESTINGHOUSE SAVANNAH RIVER COMPANY
INTEROFFICE MEMORANDUM

Flowsheet and Physical Property Estimation for SRS Tank 25 Salt Dissolution

By

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Author: ~~J. A. Pike~~   5/6/05

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Technical Review: 5/10/05

M. D. Drumm, Salt Disposition Planning

Approval:

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Approval: R. M. Mobley by E. Bran per Telecon w/ D. T. Conrad

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DAS w/F&B process requires the following steps:

1. Create a well in the saltcake to the bottom of the tank.
2. Place pump suction at or near the bottom of the well.
3. Remove the free liquid above the saltcake.
4. Drain the interstitial liquid from the saltcake to the maximum extent practical.
5. Refill the pores by adding water/inhibited water in a location opposite the well until liquid level reaches the top of saltcake.
6. Add enough liquid over the saltcake to create thin layer of liquid, what ever depth is practical, but thinner is better.
7. Simultaneously add water and remove dissolved salt solution.
8. Adjust the water addition rate and dissolved salt solution removal rate such that liquid level remains at the same level above the saltcake, and that the highest saturation levels are maintained.
9. Stop water addition when desired saltcake is dissolved.
10. Drain free liquid.

The continuous feed and removal creates flow over the surface of the saltcake without increasing the surface area contacted by the dissolution water beyond the top of the saltcake. The pump out rate is maintained low enough so that most of the liquid flow into the well is primarily from flow through the saltcake. Accounting for some variability in actual volume production, the DAS with F&B process should readily achieve about 30 to 50 gpm pump out rate. Hydraulic flow calculations would provide a more detailed estimation of flow through the saltcake.[5]

This process limits the interaction between the undrained layer of interstitial liquid and the created salt solution by directing the majority of the flow across the top or through the upper layer of the saltcake. If the liquid is allowed to sit in the tank an extended period of time, greater than two weeks, the risk of the high activity interstitial liquid mixing with the lower activity salt solution becomes higher.

2.4   Tank 25 Saltcake Deliquification Simulation Summary

Savannah River National Laboratory (SRNL) created a deliquification simulation of the Tank 25 saltcake deliquification operation. The physical properties of the saltcake were assumed the same as those found for Tank 41. The liquid properties were estimated based on current supernate composition data. In addition to the nominal case of the best-estimated property values, a few select cases were simulated that represent known variability in the properties. Five properties were varied, but the cases selected represent the largest expected change to the results rather than run all 243 possible variations. These case runs provide an indication of what could reasonably be expected from variability already known to exist. Table 5 shows what each case run represents relative to the nominal base case. Table 6 shows the actual values used in each case.[7]

Table 5: Available Drain Simulation Runs

Case Run	Temperature	Intrinsic Permeability	Porosity	Well Height	Liquid Retention
1	Nominal	Nominal	Nominal	Nominal	Nominal
2	Low	Nominal	Nominal	Nominal	Nominal
3	High	Nominal	Nominal	Nominal	Nominal
4	Low	Low	Nominal	High	Nominal
5	High	High	Nominal	Low	Nominal
6	Low	Nominal	Low	Nominal	Low
7	High	Nominal	High	Nominal	High
8*	Low	Nominal	Low	Nominal	High
9*	High	Nominal	High	Nominal	Low

* Additional runs created for this summary.

Table 6 Drain Simulation Conditions for each Case

Case	Temperature (°C)	Intrinsic Permeability (m²)	Porosity	Well Height (ft)	Liquid Retention Curve
1 (nominal)	50	3.51×10^{-11}	0.30	2	Sandy Loam
2	30	3.51×10^{-11}	0.30	2	Sandy Loam
3	60	3.51×10^{-11}	0.30	2	Sandy Loam
4	30	2.5×10^{-11}	0.30	5	Sandy Loam
5	60	5.0×10^{-11}	0.30	1	Sandy Loam
6	30	3.51×10^{-11}	0.25	2	Loamy Sand
7	60	3.51×10^{-11}	0.40	2	Loam
8*	30	3.51×10^{-11}	0.25	2	Loam
9*	60	3.51×10^{-11}	0.40	2	Loamy Sand

* Additional runs created for this summary.

The simulations were run such that the well pump operated continuously until approximately 1 gpm interstitial drain rate is reached. The well pump was cycled to continue draining until a

minimum volume is drained. The simulation results show a wide range of times to reach this volume depending on how the drain properties vary relative to the nominal properties and, possibly, never reach the desired volume. Alternatively, draining can continue until reaching a hydraulically equivalent endpoint, i.e., equivalent hydraulic pressure. Hydraulically equivalent endpoint occurs at equivalent interstitial flow rates. The completed simulations provide data to readily compare an end point of 1 gpm.

Figure 3 and Figure 4 provide some structure of the relationship between each possible variation and the choice of case runs. These figures also show the key results of each case. The cases were split into two groups that varied three parameters in order to simplify interpretation. The cube represents the three parameters varied between each set of cases. The axis for each dimension of the cube represents the range of variation expected or known for each parameter. The orientation of the range of values, i.e., high to low, was arranged such that the bottom front left corner represents the least aggressive, least favorable property combination. This combination would be expected to be the least well drained, the slowest rate, and, perhaps, the least volume drained. The upper back right corner represents the most aggressive, most favorable property combination.

The results depicted in Figure 3 and Figure 4 show comparable hydraulically equivalent endpoints, i.e., equivalent hydraulic pressure. The results depicted in the figures were pulled from the case runs when approximately 1 gpm interstitial liquid flow rate is achieved. Continuing to drain in any case will produce a better-drained saltcake, but this part of the drain curve also represents the least productive portion of the operation. Considering that the best process outcome is the lowest amount of residual liquid, not necessarily the most removed liquid, the figures show both values along with estimated time to reach the end state.

Figure 3 shows that the time to reach the end state changes relatively little compared to the dramatic variation in drained and residual volumes. Two of the three axis parameters, well height and temperature, can be controlled to some degree by the design of the operation. The variation in intrinsic permeability results in the most variation. Notice that the case with the least residual will take longer and produce more drained liquid volume even though the same stopping point is achieved.

Figure 4 shows the variation caused by properties that change the initial liquid volume and volume of retained residual. The cases 1 – 7 were chosen as most physically likely. The analyst considered the combination represented by case 8 and 9 as unlikely combinations that could not readily exist. Cases 8 and 9 were run to make this summary more complete. This figure shows that there is considerably more variability inherent in saltcake physical properties than in controlled properties.

>500 hrs
V_d >180 kgal
V_r <150 kgal

500 hrs
V_d 110 kgal
V_r 230 kgal

500 hrs
V_d 140 kgal
V_r 210 kgal

500 hrs
V_d 140 kgal
V_r 200 kgal

430 hrs
V_d 75 kgal
V_r 260 kgal

Permeability
High

Data Key:
drain time
V_d volume drained from saltcake
V_r volume of residual liquid in saltcake

Well Height
Low

Low   High

Low   Temperature   High
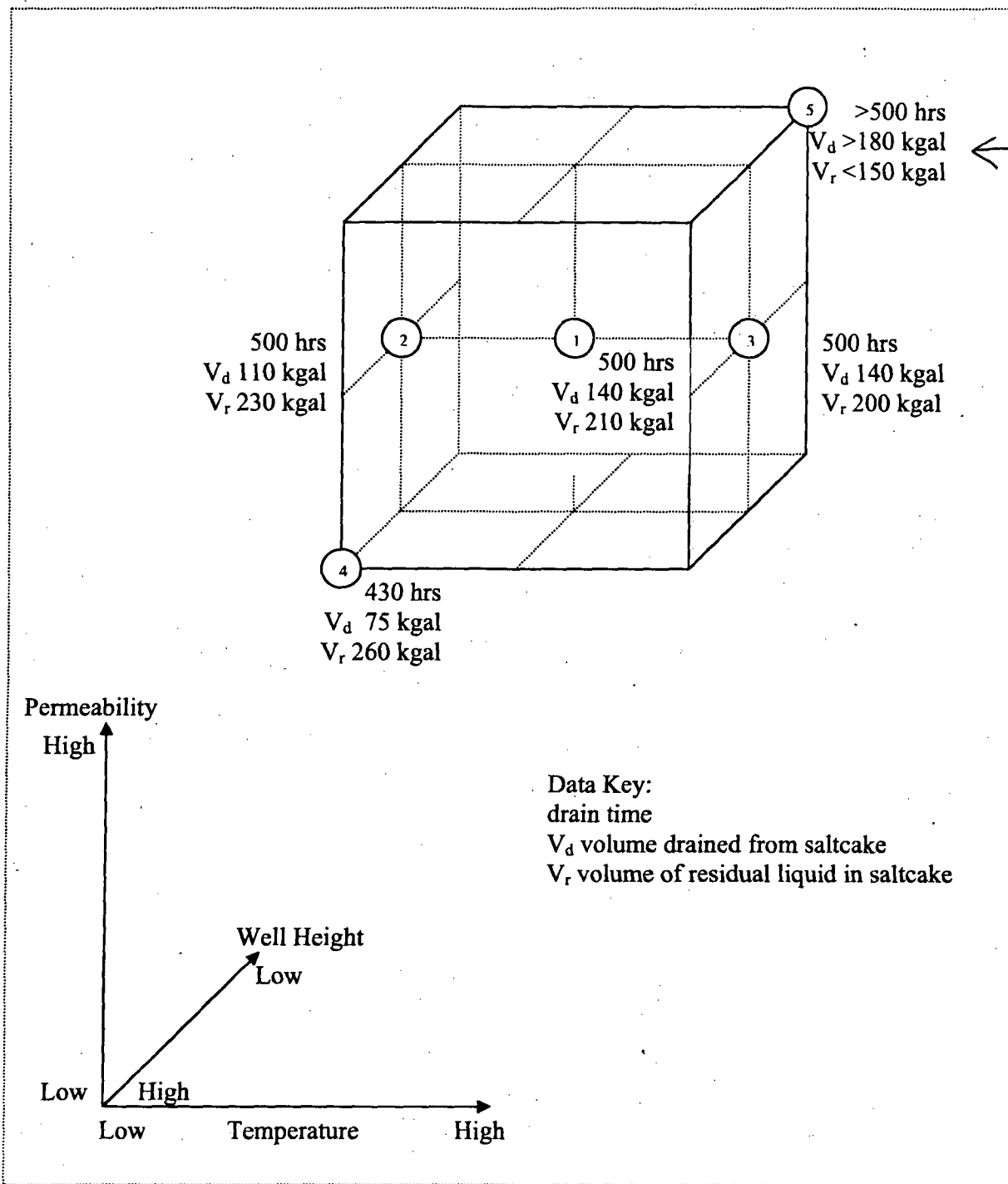
Figure 3: Representation of Drain Cases 1 Through 5
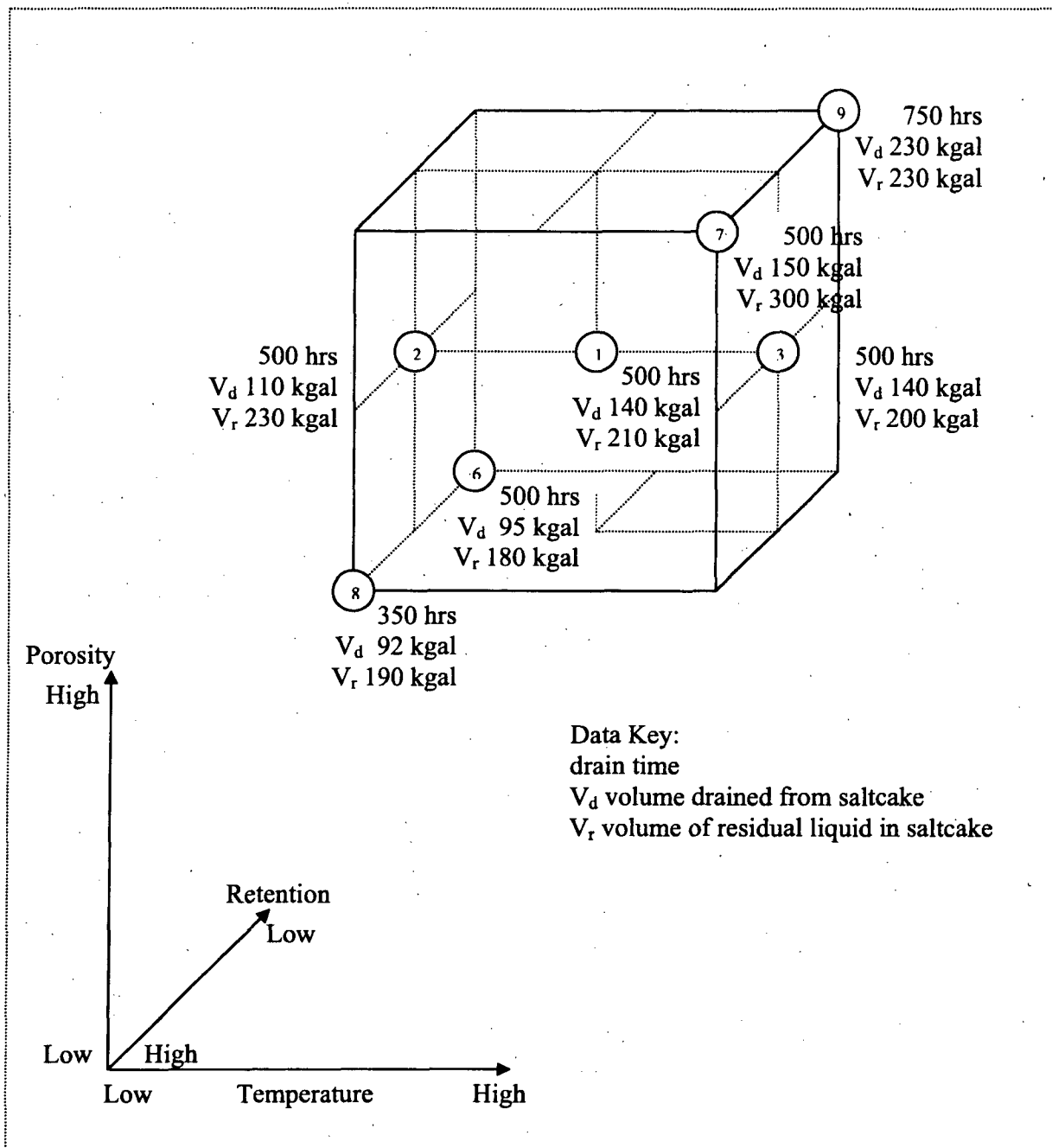
Figure 4: Representation of Drain Cases 6 Through 9 and 1 Through 3

The figure shows a cube diagram with data points. The axes are labeled:
- Porosity (Low to High, vertical)
- Temperature (Low to High, horizontal)
- Retention (High to Low, diagonal)

Data points:

Point 9:
750 hrs
V_d 230 kgal
V_r 230 kgal

Point 7:
500 hrs
V_d 150 kgal
V_r 300 kgal

Point 2:
500 hrs
V_d 110 kgal
V_r 230 kgal

Point 1:
500 hrs
V_d 140 kgal
V_r 210 kgal

Point 3:
500 hrs
V_d 140 kgal
V_r 200 kgal

Point 6:
500 hrs
V_d 95 kgal
V_r 180 kgal

Point 8:
350 hrs
V_d 92 kgal
V_r 190 kgal

Data Key:
drain time
V_d volume drained from saltcake
V_r volume of residual liquid in saltcake

EVALUATION OF SOLID-LIQUID SEPARATION TECHNOLOGIES TO REMOVE SLUDGE AND MONOSODIUM TITANATE FROM SRS HIGH LEVEL WASTE

Michael R. Poirier

August 16, 2000

SUMMARY

The author conducted a review of solid-liquid separation technologies as possible replacements for the MOTT crossflow filters in the crystalline silicotitanate (CST) ion exchange and solvent extraction flowsheets. The review used the Tanks Focus Area (TFA) funded solid-liquid separation study conducted in 1995 reviewing the technical literature as a starting point. The review also included discussions with vendors, as well as soliciting guidance from researchers at the Savannah River Technology Center (SRTC) and within the DOE complex who possess extensive experience in solid-liquid separation. Finally, the author coordinated a workshop with representatives from SRTC, Savannah River Site (SRS) High Level Waste, SRS Solid Waste, and the academic community on the specific application of interest.

Based on the findings, SRTC recommends the following work to evaluate alternative solid-liquid separation processes for removing sludge and MST from high level waste salt solution.
- Focus on identifying flocculants and additives that will improve the performance of the crossflow filters.
- Investigate settling and decanting followed by polishing filtration (both crossflow and dead-end). This testing needs to use a large volume of continuous fresh feed. It should also examine improvements that could be achieved by the addition of flocculants and additives.
- If flocculation with crossflow filtration proves ineffective, investigate high shear filtration (using a centrifugal filter or VSEP filter).
- If flocculation and crossflow filtration proves ineffective, investigate flocculation in combination with centrifugation.

INTRODUCTION

The Salt Disposition Systems Engineering Team selected three cesium removal technologies for further development to replace the In Tank Precipitation (ITP) process: small tank tetraphenyl-borate (TPB) precipitation, crystalline silicotitanate (CST) ion exchange, and caustic solvent extraction.

As a pretreatment step for the CST and solvent extraction flowsheets, the incoming salt solution that contains entrained sludge is contacted with monosodium titanate (MST) to adsorb strontium and plutonium. The resulting slurry is filtered to remove the sludge and MST. The filtrate is either contacted with CST in an ion exchange column or processed through a solvent extraction system to remove cesium.

The high level waste salt solution that feeds this process is approximately 5.6 M sodium and contains small levels of insoluble sludge (up to 600 mg/L).[1] The sludge particles are micron/submicron sized. The mean particle size in a sludge sample from Tank 41H was 15 μ. The mean particle size in samples of simulated sludge have varied from 3 – 14 μ. The MST concentration in the filter feed will be ~ 0.55 g/L and its mean particle size is ~ 10 μ. The specification for MST is < 1% less than 1 μ and < 1% greater than 35 μ.[2] The expected viscosity of the supernate is 2.3 – 2.7 cp. at 30° C.[3] The goal of this solid-liquid separation is to remove insoluble solids from the waste stream in order to meet the Z-area waste acceptance criteria for

alpha contamination, to prevent the insoluble solids from plugging the ion exchange columns, and to prevent insoluble particles from reducing the solvent extraction process efficiency.

Testing performed by SRTC and the University of South Carolina showed the filtration rates were less than desired for simulated salt solution containing various concentrations of MST and sludge solids (0.02 – 0.08 gpm/ft^2 versus a target of 0.25 gpm/ft^2).[4,5,6] To achieve the desired production rates, the current design has a 3000 ft^2 crossflow filter and a 5000 gpm filter feed pump. The large filter and pump needed for the process will significantly increase the size of the shielded cell needed.

HLW-PE requested SRTC to investigate methods to improve the separation of sludge and MST solids from high level waste salt solution.[7] This work includes investigating flocculants and additives, changing filter operating parameters, and investigating alternate solid-liquid separation technologies. This report describes the evaluation of alternate solid –liquid separation technologies. The other tasks will be described in separate reports.

SRTC conducted a meeting on June 29, 2000 with Professor Baki Yarar (Colorado School of Mines), Professor Vince Van Brunt (University of South Carolina), and representatives from the SRS High Level Waste Division, the SRS Solid Waste Division, and SRTC to discuss alternative solid-liquid separation technologies to replace for the Mott crossflow filters which are used in the current design bases for the ion exchange and solvent extraction flowsheets.[8]

The recommendations from the meeting were the following:
- One should not filter without prior settling and decanting.
- Without flocculation, solid-liquid separation options are limited.
- The filtration rate can be improved by making the solid particles hydrophobic.
- If these recommendations are followed, other solid-liquid separation technologies are viable.
- If these recommendations are not followed, solid-liquid separation is limited to membrane filtration.

The remainder of this report discusses alternative solid-liquid separation technologies that could be employed to separate MST and sludge from SRS high level waste salt solution. The technologies are grouped into the following categories:
- Crossflow filtration
- Dead-end filtration
- Gravity settling
- Centrifugal methods
- Other methods

APPROACH

The evaluation was conducted in the following manner: Previous SRTC studies of solid-liquid separation processes were reviewed. The author conducted a literature search, and contacted vendors and colleagues at DOE sites to identify plausible solid-liquid separation processes.

The Tanks Focus Area (TFA) funded an investigation of solid-liquid separation technologies to recommend the ones that would be most applicable to separating solids from DOE Site high level waste streams.[9] The study found that although many solid-liquid separation techniques are available (i.e., centrifuges, settling, dead-end filters, depth filters, etc.), crossflow filtration has a number of advantages over these technologies for use in solid-liquid separations in DOE Site waste:

- Crossflow filters have minimal maintenance requirements compared with centrifuges.
- Crossflow filters generally do not require additives that increase waste volume and change process chemistry. Dead-end filters and depth filters generally require additives.
- Crossflow filters can rapidly concentrate slurries to high insoluble solids levels. Settling and clarification typically require long times.
- Crossflow filters can be employed for continuous washing of slurries.
- Crossflow filtration space requirements are typically less than other solid-liquid separation technologies.

The study reviewed a number of solid-liquid separation tests conducted by the DOE complex with simulated and actual DOE site waste. The results from those studies will be discussed later.

DISCUSSION

During the review and literature search, SRTC identified the following potential alternatives to the 0.5 μ Mott crossflow filter for removing insoluble solids from SRS high level waste:

- Other crossflow filters
  - Smaller pore size Mott filters
  - Graver filter
  - Centrifugal filter
  - Vibratory Shear Enhanced Processing (VSEP) Filter
  - Vacco filter
- Dead-end filtration
- Settling and decanting
  - No polishing step
  - Dead-end filtration polishing step
  - Crossflow filtration polishing step
- Centrifugal methods
- Other solid-liquid separation technologies
  - Magnetic filter

Crossflow Filtration

Filters separate solids from liquid with a semi-permeable barrier. The barrier contains pores which allow liquids and dissolved solids to pass, but which block insoluble solids that are larger than the pore. As the filter rejects particles, they can accumulate on the surface forming a filter cake. The filter cake provides an additional layer that can remove insoluble particles and

increases the removal efficiency of the filter. The filter cake also increases the resistance of the filter. The filter flux can be described by equation [1]

$$J = \Delta P/R_f \hspace{6cm} [1]$$

where J is flux, ΔP is the pressure differential or driving force, and R_f is the filter resistance. With a filter cake, the flux is described by equation [2]

$$J = \Delta P/(R_f + R_c) \hspace{5cm} [2]$$

where R_c is the filter cake resistance. The filter cake can be removed and the cake resistance reduced by periodic backpulsing.[10] If the filter cake thickness can be reduced, the filter flux will increase. The shear generated by crossflow filtration sweeps particles away from the filter, reduces the cake thickness, and increases filter flux.

Fine particles that are smaller than the pore opening can become trapped in the filter pores. These particles would decrease the porosity of the filter and filter flux. Pore fouling is generally not alleviated by backpulsing. Filter flux can be increased by reducing the number of particles that become trapped in the filter pores.

Smaller Pore Size Mott Filter

In addition to the 0.5 μ porous metal, crossflow filter, Mott manufactures 0.1 μ and 0.2 μ porous metal, crossflow filters. By having a smaller pore size filter, small particles are less likely to become trapped within the filter pores. If filter fouling by particles becoming trapped within the filter pores could be reduced, the overall filter flux might be increased. Charles Nash, a researcher associated with the River Protection Program research effort for treating Hanford waste, indicates that a 0.1 μ Mott porous metal crossflow filter is the baseline solid-liquid separation technology for BNFL's program to treat high level waste at the DOE's Hanford Site.

The smaller pore size would increase the membrane resistance and could reduce filter flux. The effect of pore size on filter flux can be modeled with a modified Hagen Poiseuille equation

$$J = \frac{\varepsilon d^2 \Delta P}{32\mu l} \hspace{5cm} [3]$$

where J is filter flux, ε is porosity, d is pore diameter, ΔP is differential pressure, μ is viscosity, and l is pore length.[10] If all solid particles are stopped by the filter, filter flux should decrease with pore size.

SRTC tested 0.2, 0.5, and 2.0 μ porous metal Mott filters for the ITP process. The 0.5 μ filter gave adequate decontamination and had a higher flux than the 0.2 μ filter.[9] However, the ITP feed is different from the feed for this process in that it contains tetraphenylborate solids known to improve filter performance relative to that observed for slurries of sludge and MST. Previous filter testing for the SRS Effluent Treatment Facility found 100,000 nominal molecular weight

cutoff (NMWC) ultrafilters (~ 0.05 μ) performed better than 0.2 μ ceramic microfilters.[11] That feed was different from the feed in this process (50 mg/L insoluble solids, 1500 mg/L dissolved solids). If the pore fouling can be reduced, the smaller pore size Mott filter may still produce high filter flow rates. Based on previous SRTC testing of a 0.1 μ Graver filter that is discussed below, no testing of the 0.1 μ Mott filter is recommended.

Graver Filter

The Graver Separation Systems produces a combination ceramic/stainless-steel filter. The ceramic is composed of titania and is bonded to the stainless-steel substrate by sintering. A pore size of 0.1 μ is available. The filter has a very fine pore at the filter surface to block small particles, but has a more open structure within the filter to reduce its resistance. The smaller pore size will reduce pore fouling. The more open structure will reduce filter resistance and the decrease in filter flux from the smaller pores at the surface. Since some forms of titania extract strontium and actinides from alkaline solutions, this phenomenon would need to be evaluated before placing the filters in radioactive service.

SRTC tested the 0.1 μ Graver filter and the 0.5 μ Mott filter with Hanford and Oak Ridge simulated sludge.[9,12,13] The mean particle size was 1.9 – 6.5 μ. With the Mott filter, the axial velocity varied from 2 – 12 ft/sec, and the transmembrane pressure varied from 5 – 45 psi. With the Graver filter, the axial velocity varied from 2 – 9 ft/sec, and the transmembrane pressure varied from 10 – 65 psi. The measured filter flux varied as a function of axial velocity, transmembrane pressure, and insoluble solids concentration. The Graver filter performed slightly better (~ 20%) with 0.1 wt.% sludge, while the 0.5 μ Mott filter performed better (~100%) with 5 wt.% sludge.

The 20% improvement in filter flux observed is much less than what is needed for this process. Similar filtration results would be expected with the 0.1 μ Mott crossflow filter. No tesing is recommended.

Centrifugal Filter

The centrifugal system combines centrifugation with membrane filtration. Solids are removed from the liquid at the membrane surface, and the centrifugal force acts to keep the surface clean, minimizing the formation of a polarization layer. The centrifugal force is used to slough off any buildup on the surface, rather than to separate the solids from the liquid.

The centrifugal filter could be combined with most commercially available filter media (i.e., it could be equipped with 0.1, 0.2, or 0.5 μ porous metal filter sheets that are similar to the Mott crossflow filters in the current design bases). The centrifugal motion increases shear at the filter surface and reduces cake buildup. The effect is the same as increasing the axial velocity without increasing system pressure requirements.

SRTC tested a centrifugal filter as a replacement to the ceramic microfilters at the Effluent Treatment Facility.[11] The filter ran for over 10 hours and showed no significant fouling. That feed stream was different than the feed stream for this process, and it contained a low

concentration (43 mg/L) of small, colloidal particles. Centrifugal filters are commercially available (Spintek, Pall) and have been used in radioactive service at LANL.

The manufacturer's experience with commercial units shows they require regular maintenance to balance the rotor. Frequent maintenance is undesirable for solid-liquid separation equipment in the Alternative Salt Disposition Process.

The centrifugal filter should be considered as a backup to flocculation combined with crossflow filtration and settling/decanting combined with polishing filtration. This filter is likely to achieve the desired filter flux rates. Issues of maintenance and reliability need to be addressed if further development is warranted.

Vibratory Shear Enhanced Processing (VSEP) Filter

The VSEP filter, manufactured by New Logic, is similar to a plate and frame or disk stack filter. It could be fitted with a variety of filter elements. The filter pack consists of parallel disks. The feed moves slowly between the disks. A pressure differential forces fluid through the filters. The filter elements vibrate vigorously to create shear. The shear is equivalent to 200 Gs.

A VSEP filter vibrating at 1 inch peak-to-peak displacement and 60 Hz produces a shear rate of 150,000 s^{-1} which is about four times the shear rate attainable with crossflow filters.[14] The system is suitable for concentrating submicron particles and colloids.

The VSEP filter is commercially available, but has not been demonstrated in radioactive service. The manufacturer recently sold a unit for use in low level radioactive service. If one of these systems were to be used in high level radioactive service, SRS would need to evaluate the system parts for radioactive service and minimize the maintenance needed.

The vendor could test this filter on SRS simulated waste at their facility for approximately $1200. SRS could rent a pilot unit for approximately $6000/month or procure a pilot unit for $90,000.

The VSEP filter should be considered as a backup to flocculation combined with crossflow filtration and settling/decanting combined with polishing filtration. This filter is likely to achieve the desired filter flux rates. Issues of maintenance, reliability, and use in radioactive service need to be addressed if further development is warranted.

VACCO Filter

The VACCO filter is another crossflow filter. It is composed of a series of stacked disks. The disks contain micro-channels or pores. As the fluid flows through the disks, a differential pressure drives liquid and soluble solids through the pores. It has a more structured packing than the Mott or Graver filters, but the smallest pore size available is 3 μ. It was previously tested by SRTC with 3 wt.% ORNL Radiochemical Engineering Development Center (REDC) simulant and fouled very rapidly.[15] The filter flux was about an order of magnitude less than the filter

flux with a 0.5 μ Mott crossflow filter using the same simulant. No testing is recommended unless he manufacturer can produce a filter with a smaller pore size.

Dead-End Filtration

Another plausible technology for removing sludge and MST from high level waste is dead-end porous metal filters (e.g., such as those manufactured by Fundabac or Pall). The filter surface would be similar to the filter surface of the Mott crossflow filters, but it would be the outside surface of a cylinder approximately 2 inches in diameter and 10 – 50 inches long. The fluid would flow from the outside to the inside of the filter at a constant flow rate. As the solids are rejected by the filter, they form a filter cake and increase the pressure drop across the filter. When the pressure drop reaches a certain value, the filter is back washed. If the time between back washes is long, this filter is a viable option. If the time between backwashes is short, the dead-end filter is not desirable.

Previous SRTC testing evaluated a Pall porous metal filter as a replacement for the ceramic crossflow filters at the ETF.[11] During testing, the filter fouled vary rapidly and the time between back-washes was typically 5-6 minutes and about 50% of the filtrate was needed to back-wash the filter. The filter had a pore size of 5 μ and was fouled by small, colloidal particles. If a filter with smaller pore size could be found, it might operate longer between back-washes.

The performance of the dead-end filter might be improved with the addition of a filter aid.[9] Diatomaceous earth is commonly used, but would not be suitable for this waste stream. Any filter aid would need to be evaluated for compatibility with high pH, high ionic strength, radioactive stream, as well as compatibility with down stream processes (e.g., DWPF).

In the previous TFA investigation of solid-liquid separation technologies, the author found dead-end filtration to work best with low concentrations of large particles. In a study to treat Hanford Cladding Removal Waste, the authors investigated crossflow and dead-end filtration. The simulated waste contained 1000 – 2100 ppm solids with a mean particle size of 1.2 μ. The 0.5 μ Mott crossflow filter performed better than the 0.5 μ dead-end filter tested.

This type of filter should be examined in combination with settling and decanting.

Settling and Decanting

No Polishing Step

With this technique, the insoluble solids would settle, and the supernate would be decanted and processed through the ion exchange or solvent extraction systems without any additional treatment.

In theory, the sludge solids in this waste stream settle very slowly (i.e., they did not settle out in the waste tanks and were carried forward with the salt solution). Very long settling times could be required to achieve the solids removal required. Every day of settling time required adds 25,000 gallons of storage capacity to the facility, which will increase the footprint and cost of the

building dramatically. SRTC measured settling rates of insoluble solids in an actual Tank 41H sample.[16] Tank 41H was to feed the ITP process, so it should be similar to the feed for the this process. Table 1 shows the particle size measured and Table 2 shows the settling rate measured. The mean particle size is ~ 15 microns. The measured settling rates for the smallest particles (< 4 μ) are less than 4 in/day. If the particles in the feed to this process have similar settling rates, settling and decanting is unlikely to be effective at removing a significant fraction of particles. The settling rates could be improved by the addition of flocculants and additives.

Table 1. Particle Size of Particles in Tank 41H Sample

Size (μ)	Volume %	Cumulative Volume %		Size (μ)	Volume %	Cumulative Volume %
0.97	0.29	0.29		31.11	7.61	66.0
1.38	1.51	1.80		44.00	7.47	73.5
1.94	2.42	4.23		62.23	7.23	80.7
2.75	4.54	8.76		88.00	6.61	87.4
3.89	6.05	14.8		124.4	5.86	93.2
5.50	8.80	23.6		176.0	3.65	96.9
7.78	8.54	32.2		248.0	3.14	100
11.00	9.73	41.9		352.0	0	100
15.56	8.97	50.9		497.8	0	100
22.00	7.58	58.4		704.0	0	100

Table 2. Measured Settling Rate for Tank 41H Particles

PSD % Value	Settling Rate (in/day) ρ=1.194 g/ml	Settling Rate (in/day) ρ=1.399 g/ml
50% ≥ 15.13 μ	15	25
90% ≥ 2.98 μ	3.6	3.2
95% ≥ 2.08 μ	2.1	1.7
98% ≥ 1.40 μ	1.3	1.1

The Colorado Mineral Research Institute evaluated a counter-current de-cantation system for use in SRS high level waste sludge processing.[17] They performed settling studies on simulated Purex and Hanford sludge. Without the addition of flocculants, the settling rates were very low (0.17 – 2.2 in/h). With the addition of flocculants (e.g., Alcar W23, Alcar 662, Alcar 600, and Percol 600), the settling rates increased dramatically (to as high as 92 in/h).

The settling tank size is related to particle settling velocity by equation [4]

$$A = Q/v_s F_b \qquad\qquad [4]$$

where A is the tank cross section area, Q is the desired processing rate, v_s is the particle settling rate, and F_b is a fraction of the bulk settling rate (0.5 is commonly used with circular clarifiers of good design).[18,19] Equation 4 can be solved for v_s to determine the required settling rate as a function of tank diameter. Table 3 shows the estimated required settling velocity as a function of tank diameter.

Table 3. Required Settling Rate

Tank Diameter (ft)	Cross Section Area (ft^2)	Required Settling Rate (in/day)
14	154	630
16	201	484
18	254	382
20	314	308
22	380	256
24	452	214
26	531	182
28	616	158

The tank diameter values in Table 3 approximate the expected values for the facility. The table shows that even with the addition of flocculants, settling and decanting without a polishing step is unlikely to perform the solids removal needed for this process.

Crossflow Filter Polishing Step

In this technique, the insoluble solids (sludge and MST) would settle and the supernate would be decanted and filtered. This technique was tested at the University of South Carolina's Filtration Research Engineering Demonstration (FRED) in 1998.[6] After settling for two days and decanting, the filter flux with the decanted supernate was $1.3 - 2.1$ gpm/ft^2 depending on operating conditions and approached the clean water flux (2.25 gpm/ft^2). When settling and decanting was not used as a pretreatment, filter flux varied between $0.02 - 0.12$ gpm/ft^2. Because of the small batch size, the decanted supernate may not have had enough solids to significantly foul the filter. In a full-scale process, the decanted supernate could contain more very fine particles that could foul the filter more severely than the simulated sludge feeds. Additionally, the process would need to be designed so the settled solids could be re-suspended. Testing performed by ORNL in 1999 to evaluate re-suspension of settled sludge and MST showed this could be difficult.[20]

SRTC should investigate settling and decanting, followed by polishing with a crossflow filter to treat this waste stream. This technique needs to be evaluated at the pilot-scale with a large volume of continuous fresh feed. This work should also include flocculation/additive addition to improve the settling step.

Dead-End Filtration Polishing Step

In this technique, the insoluble solids would settle and the supernate would be decanted and filtered. By using settling and decantation as a pretreatment step to the dead-end filter, the solids loading on the filter will be decreased which should lead to a longer operating time between back-pulses. If 90% of the solid particles could be removed by settling, the improvement in operating time could be as much as 10X. If 99% of the solid particles could be removed by settling, the operating time between back-pulses could be as much as 100X. Additionally, the process would need to be designed so the settled solids could be re-suspended.

SRTC should investigate settling and decanting, followed by polishing with a dead-end filter to treat this waste stream. This technique needs to be evaluated at the pilot-scale with a large volume of continuous fresh feed. This work should also include flocculation/additive addition to improve the settling step.

Centrifugal Methods

The centrifuge relies on centrifugal force to exaggerate the density difference between the particles in a liquid, so the solids will "settle" more quickly. Thus, the centrifuge can theoretically be expected to completely remove even small, colloidal solids, given a long enough period of operation. There is no separation by a barrier, and therefore, no place for solids to become trapped. Centrifuges work best with fast settling solids.

The particle settling velocity can be estimated from the following equation

$$V_s = \left(\frac{\Delta\rho}{18\mu}\right)gd^2\left(\frac{\Omega_b^2 R_b}{g}\right)$$

where V_s is the settling velocity, $\Delta\rho$ is the density difference between the particle and the fluid, μ is viscosity, g is the gravitational constant, d is particle diameter, Ω_b is the rotational speed of the centrifuge bowl, and R_b is the bowl radius. The required settling rate is described by

$$V_{s,req} = \frac{1}{2}\frac{h}{L}\frac{Q}{A}$$

where $V_{s,req}$ is the required settling rate, h is the distance between internal surfaces of the centrifuge, L is the centrifuge length, Q is flow rate, and A is cross-sectional area of the centrifuge. Combining these equations gives the following expression for centrifuge flow rate

$$Q = 2V_{s(1g)}\left(\frac{\Omega_b^2 R_{av}}{g}\right)\left(\frac{LA}{h}\right)$$

where $V_{s(1g)}$ is settling rate under gravity settling, and R_{av} is the average radius of the bowl and the pool.[21]

Using the above equations, the throughput of a centrifuge can be estimated. Table 4 shows the results.

Hobbs measured the settling rate of the insoluble solids in Tank 41H salt cake.[16] The measured settling rates varied between 1 in/day and 25 in/day. The facility design requires a minimum flow rate, on average, of 21 gpm. If the settling rate is 1 in/day, a centrifuge would likely be impractical for this solid-liquid separation need.

Hanford evaluated centrifuges for separating solids in Purex sludge, Redox sludge, Cladding removal waste, and Neutralized Current Acid Waste streams. In simulant testing performed, the centrifuge was ineffective unless polymeric flocculants were added to the waste. In a test

performed with actual NCAW, large volumes of water were required to removed the separated solids from the centrifuge bowl.[9]

Table 4. Estimated Centrifuge Throughput

$V_{s(1g)}$	10 in/day	1 in/h	1 in/min
Ω_b	500 rpm	500 rpm	500 rpm
R_{av}	1 ft	1 ft	1 ft
G	9.8 m/s^2	9.8 m/s^2	9.8 m/s^2
L	10 ft	10 ft	10 ft
A	3.14 ft^2	3.14 ft^2	3.14 ft^2
H	1 ft	1 ft	1 ft
Flow Rate	0.56 gpm	1.34 gpm	80.6 gpm

Professor Yarar advised SRS against using centrifuges without flocculants.[8]

Centrifuges have been used successfully in the SRS Separations canyons. The centrifuges used there are standard milk centrifuges. The motors are remoted from the bowls so they can receive periodic maintenance. The bowls have not required replacement.

A centrifuge is the baseline technology for separating insoluble solids in Hanford K-basin. However, that design uses a centrifuge in combination with a polishing filter.

Centrifuge manufacturers have small portable centrifuges that can perform quick scoping tests. SRTC should coordinate one of these tests. If that test shows promising results, a centrifuge could be rented (~ $7500/3 weeks) to perform laboratory-scale tests with simulated salt solution. For centrifuges to be effective in this application, they will most likely need to be used in combination with flocculants and/or polishing filters.

Centrifugation should be considered as a backup technology if the desired processing rates cannot be achieved with flocculation combined with filtration or settling/decanting combined with polishing filtration. Centrifugation will likely require a flocculant to work effectively.

Other Solid-Liquid Separation Technologies

Magnetic Filter

High gradient magnetic separation (HGMS) removes magnetic particles that cannot be separated by other traditional magnetic separation processes because of their lower paramagnetic properties and smaller size. The process consists of a fine ferromagnetic wire matrix inserted in the bore of a magnet, which is energized by an externally applied magnetic field. The external magnetic field creates large magnetic field gradients around the wires, thereby improving the removal efficiency of small and weakly magnetic particles. As the wires become loaded with particles, the magnet can be turned off and the particles drop off of the wires.

HGMS can only remove magnetic particles. Non-magnetic particles (e.g., MST) would need to be adsorbed onto magnetic particles in order to be removed by this process.

The process has been tested with simulated SRS high level waste sludge by Professor James Ritter at the University of South Carolina. In those tests, the removal efficiency was very good, but the solids loading was less than desirable. The system was only able to concentrate solids to 16 g/L insoluble solids.[22] HLW-PE has stated a goal to concentrate this stream to 5 wt.% insoluble solids.

Additional testing would be needed to more thoroughly evaluate this technology, including testing with MST. No testing is recommended.

RECOMMENDATIONS

SRTC recommends the following work to further evaluate alternative solid-liquid separation processes for removing sludge and MST from high level waste salt solution:

- Focus on identifying flocculants and additives that will improve the performance of the crossflow filters.
- Investigate settling and decanting followed by polishing filtration (both crossflow and dead-end). This testing needs to use a large volume of continuous fresh feed. It should also examine improvements that could be achieved by the addition of flocculants and additives.
- If flocculation with crossflow filtration proves ineffective, investigate high shear filtration (using a centrifugal filter or VSEP filter).
- If flocculation and crossflow filtration proves ineffective, investigate flocculation in combination with centrifugation.

ACKNOWLEDGMENTS

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EVALUATING CENTRIFUGES FOR SOLID-LIQUID SEPARATION IN THE SRS SALT PROCESSING PROGRAM

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SUMMARY

As a pretreatment step for the caustic side solvent extraction (CSSX) flowsheet, the incoming salt solution that contains entrained sludge is contacted with monosodium titanate (MST) to adsorb strontium and actinides. The resulting slurry is filtered to remove the sludge and MST. Testing performed by the Savannah River Technology Center (SRTC) and the University of South Carolina showed cross-flow filtration rates lower than desired for simulated salt solution containing various concentrations of MST and sludge solids (i.e., 0.02 – 0.08 gpm/ft^2). Because of the low filtration rates measured during simulant and real waste testing, the authors investigated centrifugation as potential replacement for the cross-flow filters.

These tests used a pilot-scale decanter centrifuge. The centrifuge generated approximately 4100 Gs during the tests. The feed solutions for the test consisted of 5.6M sodium, average salt solution with insoluble solids. The insoluble solids in the tests included the following: (1) simulated Tank 8F sludge, (2) simulated Tank 8F sludge and MST, (3) simulated Tank 8F sludge, strontium nitrate, and sodium permanganate, and (4) simulated Tank 8F sludge, MST, and Cytec HX-400 flocculant. The insoluble solids concentration for the tests measured 0.06 wt %, 0.29 wt %, 1.29 wt %, and 6.0 wt % (measured values < 0.5 wt % to 6.5 wt %).

The conclusions from this work follow.
- The decanter centrifuge did not remove sufficient insoluble solids to meet the target clarified liquid turbidity of 5 – 10 NTU.
- The product from the tests with Tank 8F simulated sludge had a turbidity of 91 ± 41 NTU.
- The product from the tests with Tank 8F simulated sludge plus MST had a turbidity of 271 ± 105 NTU.
- The product from the tests with Tank 8F simulated sludge plus strontium nitrate and sodium permanganate had a turbidity of 267 ± 130 NTU.
- The product from the tests with Tank 8F simulated sludge plus MST and a polymeric flocculant had a turbidity of 50 ± 18 NTU.

The testing does suggest that a centrifuge could be employed for solid-liquid separation under the following options.
- Using a combination of a centrifuge and polishing filter. Previous testing suggests centrifugation as a pretreatment could increase filter flux to 0.25 gpm/ft^2, but more thorough testing is needed to quantify the improvement.
- Insoluble solids removal could improve with a two-stage centrifugation system. The first stage would use a decanter type centrifuge, such as the one used in this testing. The second stage would use a disk centrifuge, which proves more effective at removing small, slow settling particles.
- A third option would allow the treated liquid with low solids content to pass directly to the solvent extraction process. The authors recommend a set of scouting tests be performed to examine whether the solids collect in the contactors.

INTRODUCTION

The Department of Energy selected Caustic Side Solvent Extraction (CSSX) as the preferred cesium removal technology for Savannah River Site waste.

As a pretreatment step for the CSSX flowsheet, the process contacts the incoming salt solution that contains entrained sludge with monosodium titanate (MST) to adsorb strontium and actinides. The resulting slurry is filtered to remove the sludge and MST with the filtrate processed through the solvent extraction system. Testing performed by SRTC and the University of South Carolina with simulated and real waste showed filtration rates of 0.02 – 0.08 gpm/ft[2].[1,2,3,4,5] Because of the low filtration rates measured during simulant and real waste testing, SRTC identified alternative solids-liquid separation technologies as potential replacement for the crossflow filters.[6] One technology identified as a possible replacement is the centrifuge.[7]

The centrifuge relies on centrifugal force to exaggerate the density difference between the particles in a liquid, so the solids will "settle" more quickly. Thus, the centrifuge can, theoretically, completely remove even small, colloidal solids, given a long enough period of operation. Separation occurs without a physical barrier, and therefore, no place exists for trapping of the solids. Centrifuges work best with fast settling solids.

The particle settling velocity can be estimated from the following equation

$$V_s = \left(\frac{\Delta\rho}{18\mu}\right) g d^2 \left(\frac{\Omega_b{}^2 R_b}{g}\right)$$

where V_s is the settling velocity, $\Delta\rho$ is the density difference between the particle and the fluid, μ is viscosity, g is the gravitational constant, d is particle diameter, Ω_b is the rotational speed of the centrifuge bowl, and R_b is the bowl radius.[8] The required settling rate is described by

$$V_{s,req} = \frac{1}{2}\frac{h}{L}\frac{Q}{A}$$

where $V_{s,req}$ is the required settling rate, h is the distance between internal surfaces of the centrifuge, L is the centrifuge length, Q is flow rate, and A is cross-sectional area of the centrifuge. Combining these equations gives the following expression for centrifuge flow rate

$$Q = 2V_{s(1g)}\left(\frac{\Omega_b{}^2 R_{av}}{g}\right)\left(\frac{LA}{h}\right)$$

where $V_{s(1g)}$ is settling rate under gravity settling, and R_{av} is the average radius of the bowl and the pool.[8]

As the equation shows, centrifuges work best with fast settling solid particles. We expect slow settling sludge particles in the feed to the Salt Processing Facility since the waste comes primarily from evaporator operations that allowed settling and removal of the larger sludge

particles. Hobbs measured settling rates of solid particles in a such a sample from Tank 41H as 1 – 25 in/day.[9]

Centrifuges successfully treat streams in the SRS Separations canyons. They have operated since 1953. The bowls rotate at 1740 rpm and produce 1730 Gs.[10,11] The canyon centrifuges have a residence time of 3 – 6 minutes[10] versus 7 minutes in this test. The centrifuges separate MnO_2, which precipitates to remove fission products, and silicates, which are flocculated with a gelatin strike. The feed for the centrifuges is acid rather than basic, which could affect the particle morphology. The centrifuges used there are standard milk centrifuges with the motors remoted from the bowls for ease of periodic maintenance. The bowls have not required replacement.

Jacobs estimated the required target removal efficiency for this test as 99.5% (see Attachment 1 for details).[12] Since the baseline feed solution contains 1.15 g/L insoluble solids, the clarified product stream should contain less than 0.006 g/L insoluble solids. An insoluble solids concentration of 0.006 g/L corresponds to a turbidity of 5 – 10 NTU.[13]

EXPERIMENTAL

Apparatus

The Pilot Centrifuge Test Facility centered on an Alfa-Laval Sharples P600 series decanter centrifuge. The facility included a 100-gallon polypropylene slurry feed tank with a Lightnin'® Model EV5P50M ½ HP mixer. Clarified liquid product collected in a 150-gallon polypropylene tank, and the solids product collected in a modified 25-L polypropylene carboy. Figure 1 contains a photograph of the pilot test facility. Test slurry was fed to the centrifuge by way of a 3 HP Teel centrifugal pump with variable recycle back to the feed tank. Flow of the feed slurry was controlled manually by a 3/8" metering valve and monitored by a Fischer-Porter 3/8" magnetic flow meter. Data were logged by a computerized data acquisition system (DAS) that consisted of a Dell Dimension XPS T700r desktop computer running LabView version 5.1.

The operating principle behind the decanter centrifuge (see Figure 2) is that denser solids sediment against the rotating bowl wall. The less dense liquid phase forms a concentric inner layer. Personnel can vary the liquid or "pond" depth, with a maximum pond depth preferred for maximum liquid clarification.

The sedimented solid particles continuously exit from the centrifuge bowl by virtue of the action of a helical screw conveyor or "scroll". The scroll rotates at a slower speed than the bowl. The gearbox establishes the differential speed between the scroll and bowl. The solids are pushed out of the pond by the scroll and up the conical "beach". The centrifugal force generated by the rotating centrifuge compacts the solids and expels excess liquid. The concentrated solids discharge from the feed end of the centrifuge and the clarified liquid discharges from the opposite end.
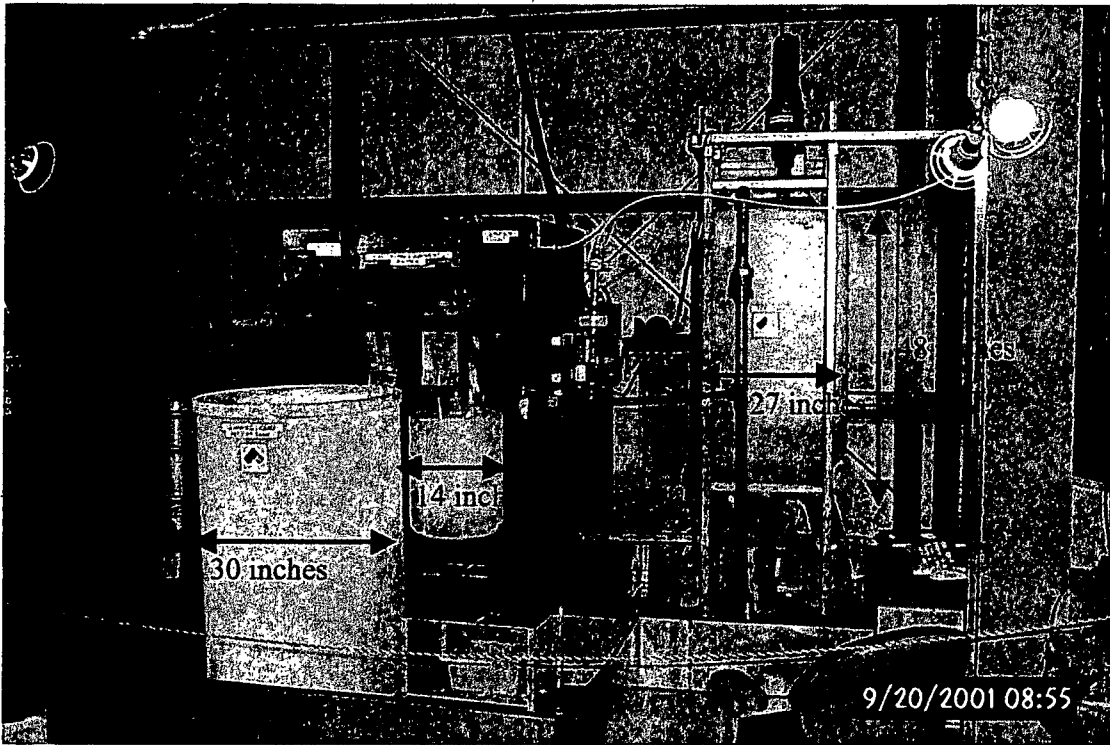
FIGURE 1- Pilot Centrifuge Test Facility
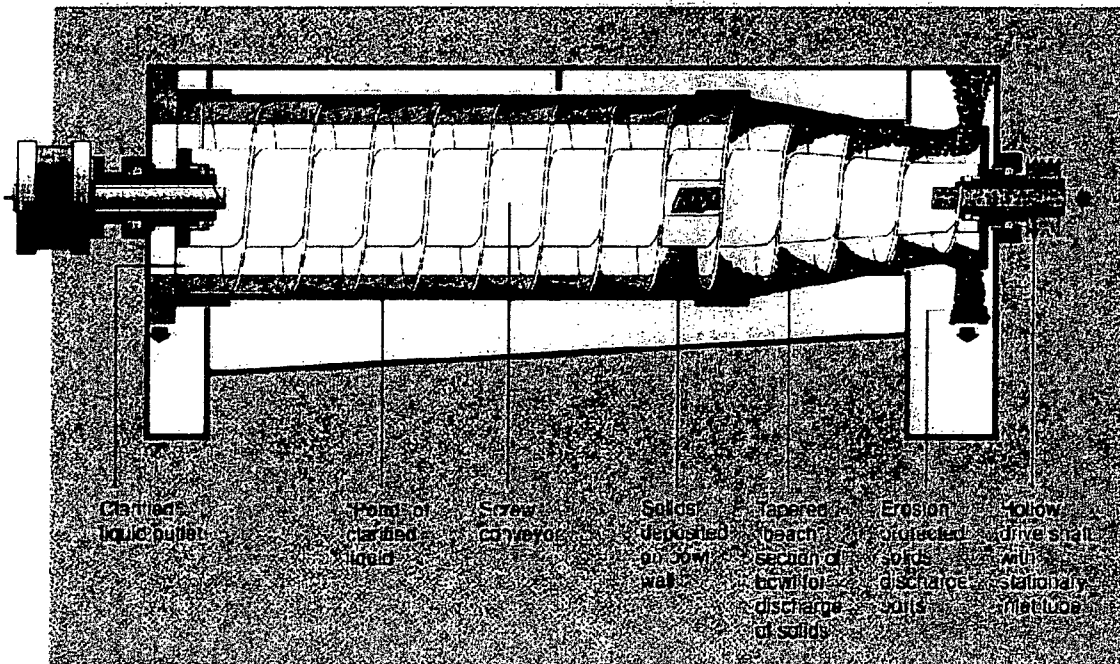
FIGURE 2 – Schematic of Decanter Centrifuge

Feed Slurries

All slurries fed to the centrifuge were made from a stock 5.6 M sodium simulated average SRS High-Level Waste (see Table 1). We omitted sodium chloride and sodium fluoride from the feed at the vendor's request, to prevent corrosion. Personnel added insoluble solids to the solution in varying amounts. The insoluble solids for the various tests consisted of the following: (1) simulated Tank 8F sludge, (2) simulated Tank 8F sludge and MST, (3) simulated Tank 8F sludge, strontium nitrate and sodium permanganate, and (4) simulated Tank 8F sludge, MST, and Cytec HX-400 flocculant. Table 2 shows the insoluble solids concentration for each test. In the tests with sodium permanganate, researchers added sodium formate as the reducing agent (4.5 moles of formate per mole of manganese). In the flocculant tests, personnel added the flocculant at 15 mg of flocculant per gram of insoluble solids (i.e., 1.5 wt % solids basis).

Previously, SRTC found that the addition of strontium nitrate and sodium permanganate improved strontium and actinide removal from Hanford High Level Waste solutions.[14] In addition, they found strontium nitrate and sodium permanganate addition improved cross-flow filtration rates. The researchers performed tests with those additives to evaluate the solid-liquid separation by centrifuge for this alternate process chemistry.

In other testing, SRTC found the addition of flocculants, such as Cytec HX-400, improved particle settling and filtration.[13,15] Tests included this additive to evaluate its impact on centrifugation.

Table 1. Centrifuge Test Supernate Composition

Species	Concentration
Na	5.6 (M)
K	0.015 (M)
Cs	0.00014 (M)
OH	1.93 (M)
NO_3	2.16 (M)
NO_2	0.53 (M)
AlO_2	0.31 (M)
CO_3	0.16 (M)
SO_4	0.15 (M)
PO_4	0.01 (M)
C_2O_4	0.004 (M)
SiO_3	0.004 (M)
MoO_4	0.0002 (M)
Tri-n-butyl phosphate	0.5 mg/L
Di-n-butyl phosphate	25 mg/L
Mono-n-butyl phosphate	25 mg/L
n-butanol	2 mg/L
Formate	1500 mg/L (0.033 M)

Table 2. Insoluble Solids Concentration for Centrifuge Tests

Sludge + MST	Sludge Only	Sludge + MnO4	Sludge + MST + Flocculant
0.031 wt % sludge 0.029 wt % MST	0.06 wt % sludge	0.031 wt % sludge 0.0065 M MnO_4 0.0065 M Sr	0.031 wt % sludge 0.029 wt % MST HX-400
0.15 wt % sludge 0.14 wt % MST	0.29 wt % sludge	0.15 wt % sludge 0.031 M MnO_4 0.031 M Sr	0.15 wt % sludge 0.14 wt % MST HX-400
0.67 wt % sludge 0.62 wt % MST	1.29 wt % sludge	0.67 wt % sludge 0.14 M MnO_4 0.14 M Sr	0.67 wt % sludge 0.62 wt % MST HX-400
3.1 wt % sludge 2.9 wt % MST	6.0 wt % sludge		3.13 wt % sludge 2.87 wt % MST HX-400

Experimental Operations

Each experiment began by combining simulated supernate solution and the appropriate amount of solids in the feed tank and agitating the mixture for a minimum of 15 minutes. Then, personnel collected a sample (~ 50 mL) of the feed for later turbidity measurement. Operators started the centrifuge in accordance with EDS Field Procedure FP-904. To achieve maximum liquid clarification, we operated the centrifuge at a maximum differential speed between the bowl and scroll by running the scroll at its minimum speed (approximately 1670 rpm) and running the bowl at its maximum safe operating speed of approximately 5000 rpm (approximately 4100 Gs). According to the following equation, with a gear ratio of 98:1, this condition yields a differential of approximately 34.

$$\Delta(differential) = [Bowl\ speed - Scroll\ speed]/Gear\ ratio$$

Once the centrifuge reached the appropriate speed, personnel activated the DAS and then introduced feed. The initial tests used a slurry feed rate of approximately 0.5 gpm, but we later reduced the rate to 0.1 gpm to increase residence time in the centrifuge. Slurry feed to the centrifuge continued for two hours, during which time personnel collected samples of the clarified liquid product (~50 mL) every 15 minutes and analyzed them for turbidity. When the feed was consumed operators closed the feed valve, and stopped the Teel pump.

At the end of each test, personnel collected a concentrated solids product sample. Operators then rinsed the centrifuge with process water until the liquid product stream discharge appeared clear. Personnel shut down the centrifuge according to Field Procedure FP- 904.

RESULTS

Table 3 shows the turbidity of the clarified liquid stream and the estimated insoluble solids concentration calculated from the equation developed by Martino et. al.[13] The results show the product turbidity significantly exceeds the target of 5 – 10 NTU. The product from the tests with Tank 8F simulated sludge had a turbidity of 91 ± 41 NTU.

Table 3. Centrifuge Product Turbidity

Sludge (wt%)	MST (wt%)	Floc (wt%)	Sr(NO₃)₂ (M)	MnO₄ (M)	Insol. Solids meas. (wt%)	Turbidity (NTU)	Samples (#)	Insol. Solids (mg/L)
0.06	-	-	-	-	< 0.5	101.5 ± 21.8	9	28
0.29	-	-	-	-	< 0.5	68.7 ± 26.3	8	41
1.29	-	-	-	-	< 0.5	103.6 ± 68.0	6	42
6.0	-	-	-	-		Not measured		
0.031	-	-	0.0065	0.0065	< 0.5	227.3 ± 74.5	15	92
0.15	-	-	0.031	0.031	0.54	154.8 ± 24.1	9	63
0.67	-	-	0.14	0.14	5.5	445.1 ± 68.6	9	180
0.031	0.029	-	-	-	< 0.5	164.6 ± 19.6	9	67
0.15	0.14	-	-	-	< 0.5	392.4 ± 71.1	9	159
0.67	0.62	-	-	-	< 0.5	255.0 ± 29.5	9	103
0.031	0.029	0.0009	-	-	< 0.5	32.2 ± 6.9	9	13
0.15	0.14	0.0044	-	-	< 0.5	67.3 ± 10.4	9	27
0.67	0.62	0.019	-	-	0.80	48.1 ± 21.0	9	19
3.1	2.9	0.09	-	-	3.3	51.9 ± 12.5	9	21

The product from the tests with Tank 8F simulated sludge plus MST had a turbidity of 271 ± 105 NTU. The product turbidity with only sludge feed proved lower than the product turbidity with sludge and MST feed.

The product from the tests with Tank 8F simulated sludge plus strontium nitrate and sodium permanganate had a turbidity of 267 ± 130 NTU. The addition of strontium and permanganate led to higher product turbidity.

The product from the tests with Tank 8F simulated sludge plus MST and a polymeric flocculant had a turbidity of 50 ± 18 NTU. The addition of the flocculant improved product quality, but not to the level desired. In previous testing, flocculants showed significant improvement in particle settling rate.[13,15] However, in this test, shear from the recirculation pump and agitator probably tended to break down the flocculated solids as also observed in previous cross-flow filter tests.

According to the vendor (Alfa Laval), a 20 gpm decanter centrifuge (model CHNX-418) would be 3.5 m x 1.0 m x 2 m high. A 20 gpm disk centrifuge (model CHPX-513) would be 1.3 m x 1.5 m x 2 m high. The vendor provided a list of 62 units in nuclear service in Europe (see Attachment 2). The units are in research laboratories, power plants, and waste disposal facilities.

Even though it did not achieve the target solids removal, the testing does suggest that a centrifuge could be employed for solid-liquid separation under the following options: a

combination of a centrifuge and polishing filter, a two-stage centrifugation system, or no additional treatment following the centrifuge.

The centrifuge reduced the insoluble solids level in the feed stream to 13 – 180 mg/L. This reduction in insoluble solids would slow cake buildup in a cross-flow filter and could increase filter flux. Previous SRTC testing investigated settling and decanting followed by polishing filtration. The tests showed that reducing the insoluble solids in the filter feed could increase cross-flow filter flux significantly.[13] Based on the settling study and the results from these tests, we estimate centrifugation as a pretreatment could increase filter flux to 0.25 gpm/ft^2. However, a firm estimate requires more thorough testing to quantify the improvement.

Insoluble solids removal could improve with a two-stage centrifugation system. The first stage would use a decanter type centrifuge, such as the one used in this testing. The second stage would use a disk centrifuge, which is more effective at removing small, slow settling particles. To evaluate this option, we could supply product samples from these tests to the vendor to evaluate the feasibility of the two-stage centrifugation process. The vendor recommended this approach.

The level at which insoluble solids adversely impact the centrifugal contactors has not been determined. Hence, another option would feed product samples from these tests to the 2 cm centrifugal contactors to determine whether the solids levels observed in these tests adversely impact them. The authors recommend a set of scouting tests to examine whether the solids collect in the contactors.

CONCLUSIONS

The conclusions from this work follow.
- The decanter centrifuge did not remove sufficient insoluble solids to meet the target clarified liquid turbidity of 5 – 10 NTU.
- The product from the tests with Tank 8F simulated sludge had a turbidity of 91 ± 41 NTU.
- The product from the tests with Tank 8F simulated sludge plus MST had a turbidity of 271 ± 105 NTU.
- The product from the tests with Tank 8F simulated sludge plus strontium nitrate and sodium permanganate had a turbidity of 267 ± 130 NTU.
- The product from the tests with Tank 8F simulated sludge plus MST and a polymeric flocculant had a turbidity of 50 ± 18 NTU.

OPTIONS

The testing does suggest that a centrifuge could be employed for solid-liquid separation under the following options:
- Use a combination of a centrifuge and polishing filter. Previous testing suggests centrifugation as a pretreatment could increase filter flux to 0.25 gpm/ft^2, but more thorough testing is needed to quantify the improvement.
- Insoluble solids removal could improve with a two-stage centrifugation system. The first stage would use a decanter type centrifuge, such as the one used in this testing. The second

stage would use a disk centrifuge, which is more effective at removing small, slow settling particles.

- A third option would allow the treated liquid with low solids content to pass directly to the solvent extraction process. The authors recommend a set of scouting tests be performed to examine whether the solids collect in the contactors.

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ATTACHMENT 1
DETERMINATION OF REQUIRED INSOLUBLE SOLIDS REMOVAL

Roy Jacobs

To: Steve Subosits, Herbert Elder, Joe Carter, Samuel Fink/WSRC/Srs,
Michael Poirier/WSRC/Srs

cc:

Subject: Estimate of required sludge removal efficiency

01/17/01 08:13 AM

Gentlemen (and I mean that sincerely),

While considering alternatives to crossflow filtration (like flocculation
followed by settle/decant or centrifuging), I again asked the question "what
removal efficiency is required to prevent busting the Saltstone alpha limit?"
Since I've never heard an answer to that question, I decided to have a go at
it myself. (See attached Excel file).

Assuming 600 mg sludge/L salt solution and alloting 25% of the alpha limit for
sludge particles (5 nCi out of 20), I get a required efficiency of 99.5%. John
Fowler suggested that a dead-end polishing filter might be needed downstream
of a centrifuge or a settler.

Caution: This is all based on nominal conditions and the calc has not been
reviewed. On the other hand, I did not include dilution from 6.4 to 5.6 M.

Roy

- SLUDGE ~ 1.XLS

Estimate of Nominal Sludge Removal Efficiency

Bases:

Ci alpha/gal @13 wt% sluge solids in slurry

U232	1.20E-05
U234	3.00E-05
Np237	1.00E-05
Pu238	1.3

From BDR 138*

1.95 lbs sludge/gal slurry @ 19 wt% solids
600 mg sludge/L salt soln
density of 6.4 M salt soln = 1300 g/L

*average 5 year old sludge
feed to DWPF

Calculations:

13/19 x 1.95 lbs sludge/gal slurry x 453.6 g/lb = 605 g sludge/gal slurry

1.3 Ci/gal x 1E9 nCi/Ci / (1000 mg/g x 605 g/gal) = 2148 nCi alpha/mg sludge

600 mg sludge/L salt soln x 2148 nCi alpha/mg sludge / 1300 g soln/L =   991 nCi alpha/g salt solution

Saltstone limit is 20 nCi/g. Assume that 25% of that can be alloted to alpha from sludge.
Then the required removal efficiency is

(991 - 5) / 991 * 100 = 99.5 % efficiency

ATTACHMENT 2
ALFA LAVAL CENTRIFUGES IN EUROPEAN NUCLEAR FACILITIES

Customer	Country	Contractor	Machine	Qty.	Year	Location	Type
Euroatom	Italy		CRPX 207 SGV	1	1966	RL	Disk
RCN, Petten	Netherlands		BRPX 213 SFD		1971	WD	Disk
Toshiba	Japan		BRPX 213 SGV	2	1975	PP	Disk
Coarso	Italy		CRPX 207 SGP	2	1978	PP	Disk
EIR	Switzerland		BRPX 213 SGV	1	1978	RL	Disk
Mühlenberg	Switzerland		BRPX 207 SGV	2	1978	PP	Disk
Benznau	Switzerland		BRPX 207 SGV	1	1978	PP	Disk
Toshiba	Japan		BRPX 213 SGV	2	1980	PP	Disk
Toshiba	Japan		BRPX 417 SGV	2	1980	PP	Disk
Toshiba	Japan		BRPX 213 SGV	2	1981	PP	Disk
Toshiba	Japan		BRPX 417 SGV	2	1981	PP	Disk
Toshiba	Japan		BRPX 417 SGV	1	1981	PP	Disk
KKW Isar 1	Germany	Siemens	BRPX 213 SGV-34	1	1981	PP	Disk
KKW Brunsbüttel	Germany	Siemens	BRPX 213 SGV-34	1	1982	PP	Disk
Nersa	France		BRPX 213 SGV	1	1983	PP	Disk
KKW Phillipsburg 1	Germany	Siemens	BRPX 207 SGV-34	1	1983	PP	Disk
KKW Phillipsburg 1	Germany	Siemens	BRPX 213 SGV-34	1	1983	PP	Disk
KKW Phillipsburg 1	Germany	Siemens	KWNX 416 S-31G	1	1983	PP	Decanter
KKW Phillipsburg 2	Germany	Siemens	BRPX 213 SGV-34	1	1983	PP	Decanter
KKW Phillipsburg 2	Germany	Siemens	KWNX 416 S-31G	1	1983	PP	Decanter
Toshiba	Japan		BRPX 413 SGD	2	1983	PP	Disk
KKW Isar 2	Germany	Siemens	BRPX 213 SGV-34	1	1985	PP	Disk
KKW Isar 2	Germany	Siemens	KWNX 416 S-31G	1	1985	PP	Decanter
Idreco	Italy		BRPX 213 SGV	3	1985	PP	Disk
KKW Neckarwestheim	Germany	Siemens	BRPX 213 SGV-34	1	1985	PP	Disk
KKW Neckarwestheim	Germany	Siemens	KWNX 416 S-31G	1	1985	PP	Decanter
KKW Brockdorf	Germany	Siemens	BRPX 213 SGV-34	1	1985	PP	Disk
KKW Brockdorf	Germany	Siemens	KWNX 416 S-31G	1	1985	PP	Decanter
KKW Emsland	Germany	Siemens	BRPX 213 SGV-34	1	1985	PP	Disk
KKW Emsland	Germany	Siemens	KWNX 416 S-31G	1	1985	PP	Decanter
KKW Obrigheim	Germany		BRPX 213 SGV-34	1	1986	PP	Disk
KKW Obrigheim	Germany		KWNX 416 S-31G	1	1986	PP	Decanter
KKW Würgassen	Germany		BRPX 213 SGV-34	2	1987	PP	Disk
KKW Karlstein	Germany		NX 309	1	1987	RL	Decanter
KKW Grohnde	Germany		CHPX 510 SGD-34 CG	1	1990	PP	Disk
KKW Grohnde	Germany		KWNX 416 S-31G	1	1990	PP	Decanter
KKW Phillipsburg 1	Germany	Siemens	CHPX 510 SGD-34 CG	1	1990	PP	Disk
KKW Phillipsburg 1	Germany	Siemens	KWNX 416 S-31G	1	1990	PP	Decanter
KKW Karlstein	Germany		KWNX 409 S-31G	1	1992	RL	Decanter
KKW Isar 1	Germany		BRPX 213 SGV-34 CG	1	1992	PP	Disk
ABB Atom	Sweden		KWNX 416	1	1994	PP	Decanter
KKW Rheinsberg	Germany	Siemens	KWNX 416	1	1995	PP	Decanter
Teollisuuden Voima Oy	Finnland		KWNX 416	1	1995	PP	Decanter
Teollisuuden Voima Oy	Finnland		CHPX 510	1	1995	PP	Disk
Sage Brno (Temelin)	Czech Rep.		KWNX 418	1	1996	PP	Decanter
Sage Brno (Temelin)	Czech Rep.		CHPX 513	1	1996	PP	Disk
Teollisuuden Voima Oy	Finnland		KWNX 416	1	1996	PP	Decanter
Teollisuuden Voima Oy	Finnland		CHPX 510	1	1996	PP	Disk
Zwilag Würenlingen	Switzerland	BWB	BTPX 205 SGD-34 CDP	1	1997	WD	Disk
Yonggwang 5	Korea	HPA	CHPX 517 SGV-34 CGR	1	1997	PP	Disk
Yonggwang 5	Korea	HPA	KWNX 418 S-31	1	1997	PP	Decanter
Yonggwang 6	Korea	HPA	CHPX 517 SGV-34 CGR	1	1997	PP	Disk
Yonggwang 6	Korea	HPA	KWNX 418 S-31	1	1997	PP	Decanter

62

Key to location:	RL	Research lab
	PP	Power Plant
	WD	Waste disposal

APPROVALS:

Author:

(signature)

M. R. Poirier, WPT

11/15/01
Date

(signature)

M. A. Norato,

11/15/2001
Date

Design Check:

(signature)

F. F. Fondeur, WPT

11.15.01
Date

Customer:

(signature) _(initials)_

R. E. Edwards, HLW-PE Manager

11/28/01
Date

(signature)

J. T. Carter, SPP Engineering Director

12/5/01
Date

(signature)

H. D. Harmon, TFA SPP Technology
Development Manager

11/20/01
Date

(signature) BILL CLARK

J. W. McCullough, DOE-SR
Technical Support Manger

11/20/01
Date

Management:

(signature)

S. D. Fink, TFA System Lead and LWP Manager

11-29-2001
Date

(signature)

W. L. Tamosaitis, WPT Manager

12/10/01
Date

WESTINGHOUSE SAVANNAH RIVER SITE
INTEROFFICE MEMORANDUM

SRT-EDS-970022

(handwritten: 165.4 circled)

April 2, 1997

CC: M. A. Ebra, 773-42A
           D. R. Muhlbaier, 786-5A

TO: B. L. Lewis, 703-H, 8-1638

FROM: R. A. Dimenna, 773-42A, 5-8203
            G. A. Taylor, 703-H, 8-8934

SUBJECT: **Flow Pattern Calculations for Eductor Flow in Tank 43**

Summary

An analysis has been completed to determine the liquid region in Tank 43 which is influenced by flow through the eductor intake while transfering liquid to the evaporator. The purpose of the analysis was to find how high the Tank 43 eductor inlet should be so that excess solids are not entrained to the evaporator. The results are based on an assumption that a reference level in the tank can be established, either by the observation of a "clear" liquid interface or by a turbidity measurement indicating a tank elevation at which the solids concentration is acceptably low to allow transfer to the evaporator. The analysis shows that if the eductor inlet is 12 inches above this reference level, then there will be no appreciable entrainment of solids into the evaporator.

Introduction

Tank 43 is the feed tank for the 2H evaporator, and as such is in a continually active state. There is feed coming into the tank from the H-canyon, DWPF, and Tank 38. The recycle flow from Tank 38 comes into the tank from a high, side wall penetration, while the canyon and DWPF feeds enter through a downcomer. Because of the continual activity in the tank, no settling of recent flows can be assumed. Nevertheless, the operation of Tank 43 as the evaporator feed tank since the early 1980's has resulted in a settled sludge layer building up over the years. This settled sludge layer occupies the bottom region of the tank, above which there is a region of suspended solids. Slow settling of the solids will result in a solids concentration gradient upward through the liquid region, such that at some point above this suspended solids region there will be a region in which the liquid is relatively free of solids. It is in this region that it is desired to place the eductor inlet.

The eductor inlet is currently set at 65 inches above the tank bottom, and well away from both the center of the tank and the tank outer wall. The settled sludge "interface" is at about 60 inches, but the downcomer extends to 44 inches. The downcomer is located well away from the eductor suction. Injecting new waste into the settled sludge region will cause a portion of this region to remain in an agitated state. Because of the agitated state and the very slow settling rate, it is reasonable to assume that there will be no clear interface between the sludge and a "clear" liquid region. Because of the lack of both characterization data of the sludge and analytical tools, the height of the suspended sludge region must be determined is by an in situ measurement.

DR. BAKI YARAR CONSULTATION ON SALT ALTERNATIVES SOLID-LIQUID SEPARATIONS

W.B. Van Pelt

September 15, 2000

WSRC contracted with Dr. Baki Yarar of the Colorado School of Mines to consult on solid-liquid separation techniques potentially applicable to sludge/monosodium titanate (MST) slurries. Dr. Yarar attended a working meeting on this topic at SRS on June 29, 2000. Attendees included representatives from SRTC, Solid Waste, and High Level Waste (Salt Disposition Engineering).

Pursuant to his visit, Dr. Yarar supplied the attached letter report, which provides both supplemental information regarding the June 29 meeting key points and requested literature references. The key points from both Dr. Yarar's visit and his letter are:

- Filtration should be preceded by sedimentation whenever possible.
- Compositionally consistent feed slurries is key to optimum solid-liquid separation performance.
- Clear solids removal requirements must be well defined to select the appropriate removal technique.
- Without flocculation, solids removal options become very limited.
- Flocculation would be more effective in slurries with >1% solids loading.
- Flocculation/sedimentation followed by a polishing filter should be considered.
- Flotation may be another viable technique to consider.
- Hydrophobization of solids will enhance filtration and likely explains why slurries with tetraphenyborate (TPB) have higher filtration fluxes than those without TPB.
- Another additive in place of the TPB may restore the hydrophobic quality to the slurry and thereby improve filtration rates over the sludge/MST slurry alone.

APPROVALS

Author:
_____   _____
W.B. Van Pelt, Waste Processing Technology   Date

Management:
_____   _____
S.D. Fink, TFA System Lead   Date

Customer:
_____   _____
J.T. Carter, HLW-PE Manager   Date

_____   _____
K.J. Rueter, HLW-Eng Manager   Date

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Research Manager,
Waste Handling and Mechanical
Westinghouse Savannah River Company
Building 773-42 A
Aiken, SC 29808

Jul. 19, 2000

Re: MEETING REPORT : 6-29-2000 AT WSRC

Dear Dr. Van Pelt:

Based on our meeting at your facilities and in response to your e-letter dated 7-6-2000
I am pleased to present the following report regarding the solid/liquid separation
procedures of the slurries under discussion.

First of all, allow me to note that the statements in the "key points list" we have generated
during the meeting, are perfectly valid as we have discussed them in detail; and the
material below is a supplement to this list.

Needs:

1) The first need is that the processing plant which may have the flowsheet steps of
   flocculation $\Rightarrow$ sedimentation $\Rightarrow$ decantation $\Rightarrow$ filtration $\Rightarrow$ etc , should be
   assured to receive feed with a uniform composition. Solids content, pH, ionic-
   composition etc., should always be the same, within the limits of design
   tolerances. This can be accomplished by continuously or batchwise blending of
   the feed material from various storage tanks. The blending should follow an exact
   recipe in perpetuity.
2) The above is true for whatever process is chosen among your candidate
   solid/liquid separation procedures, i.e.: Solvent extraction or flocculation/filtration
   or ion exchange etc.
3) Among the process options presented, by Dr. Poirier my perception is that the one
   entitled *"TBP Precipitation"* is the most likely to succeed in such a system as
   yours, though, it needs to be slightly modified or perhaps the figure presented
   should be expanded to include more detail. I would be happy to work with you on
   a modified version of this flowsheet for the process.

4) Since the solids content of the original feedstock is low, flocculation would be more efficient if its solids content is raised to 1 % or more. I note from the report by Dr. P. A. Taylor that such a slurry has been taken in model studies. This is a correct type of slurry that can flocculate better.

5) For the choice of applicable filtration technology my feeling is that one should use regular drum or belt filtration to remove the flocculated solids which have gone through a sedimentation/decantation procedure, followed by a polishing filtration step applied to the clarified thickener overflow.

6) The polishing step could possibly use a high-capacity-small pore size filter such as the *"Mott Filter"* cited during our meeting. Naturally, testing should define the ultimate choice.

7) We have noted That a hydrophobic filter-cake will likely be much easier to dewater. Most likely NaTBP would render the cake with monosodium titanate quite hydrophobic. Your researchers will know the true answer to this from their own experience. Hydrophobic particles will also aggregate in water as if they were treated with a low molecular weight flocculant solution. The phenomenon is known as 'hydrophobic bonding" or "hydrophobic aggregation".

8) The advantage of an aggregated-solids-containing slurry is what we know as "blanketing effect" where the aggregates drag all other solids down to the bottom of the sedimentation tank.

There is good chance that hydrophobic aggregates with the help of entrapped air-bubbles will partly float to the top of the sedimentation tank. In that case I wonder if "flotation" has ever been considered by your team as a solid/liquid separation approach. *"Dissolved air flotation"* or *"micro-bubble flotation"* are off-the-shelf technologies used by water treatment technologists for the preparation of potable water. Such systems work exceptionally-well with slurries that contain only small concentrations of solids. The following reference might be a worthwhile source for an introductory preview of this widespread technology. [F. N. Kemmer (editor) "The NALCO Water Handbook, second edition, McGraw-Hill, New York, 1987, p. 9.18)

I would be happy to deliver a lecture or brief a group on flotation technology, if/when you choose. Modesty notwithstanding, I am internationally known in the areas of flotation and flocculation. Similarly in the ultimate, I would be happy to consider running hydrophobization and flotation tests in my laboratory at CSM (Colorado School of Mines), if your decisions lead to it.

9) In response to your question on "research needs" I propose the following approach prior to undertaking a new research program:
    a) Form a small committee consisting of two or maximum three persons, from your own research team together with Dr. Taylor and myself if you like.
    b) The committee should generate questions and sift the information available on your shelves, in a short period like a few days.

c) The questions should be seeking what is available for the your TBP
Precipitation process and remain strictly germane to it. Then if necessary
undertake a research program to fill in the gaps of information that will
serve this model (hypothetical flowsheet).

10) A special note on flocculation and flocculants:

I remember that I participated in the early stages of a research project that was
commissioned by WSRC to a company named Colorado Minerals Research Institute,
in Golden, Colorado. Its manager is still Bob Cutriss. I think they produced a report
which must be in your records. If it is not confidential (I was part of the team, and I
can sign a confidentiality agreement if needed) I would like to look at this report.
As part of my contribution, I produced a chapter (if they included it in the submittal)
where I briefly explained how flocculant molecules would behave in such a high pH
and high ionic strength aqueous medium. In outline I explained two potential events
that flocculant molecules would experience, viz.,:

a) Flocculant molecules will be screened away from adsorption sites due to high
ionic concentrations and
b) They will likely coil up and exhibit a reduced effective molecular weight.

My suggestion is: let me look at this report, then make a few phone calls to flocculant
vendors and write you a few pages on the likely approach you can adopt for the
choice of a flocculant/surfactant system to render the flocs hydrophobic.

11) Note on Hydrophobicity and filtration (your question-2 in e-letter)

Filtration is a process of fluid (water) motion in capillary media. Thus a filter-cake is a
network of capillaries. The following references for example provide an insight into the
relationship of hydrophobic solids and the flow of water in a single capillary made of
such solids. *(see appendix-1 for full reference and photocopies of relevant pages)*

Reference-1: Adamson, Physical Chemistry of Surfaces; particularly p. 435 and equation
XIII-3 .

Reference-2: Ross, on p. 110 has equation A-82, which is the same of Adamson's. The
accompanying quotation is: *"The pressure, ΔP, required to blow a liquid out of a
capillary tube of radius (r) is also a result of the curvature of the liquid surface"*.

$$\Delta P = (2 \, \sigma \cos \theta) / r$$

Since θ is the contact angle (se Reference–1 Adamson for concept) and in aqueous media
θ = 0 indicates a hydrophilic solid and θ > 0 indicates a hydrophobic solid, it follows
from inserting values into this equation that ΔP will get smaller for increased
hydrophobicity of the solid, i.e.: easier filtration.

Reference-3: Klimpel, which is the proceedings of a conference, on p. 213 has a paper by S.K.Mishra. On p.215 of this paper, equation (9) is given as follows:

$$S_r \propto \text{constant} \left(1/ \log \gamma \cos \theta \right)$$

Here, γ is the surface tension of the aqueous phase.
This expression, again, is in agreement with the one cited above (Adamson and Ross).
This expression integrates the expressions cited with the filtration equation (Darcy) which is more familiar to chemical engineers.

As Mishra points out elsewhere in the paper, and references he quotes, surfactants and oils are used to make the solid hydrophobic and they need to be compatible with the solids filtered, since filtration systems are subject to the combined influence(s) of numerous variables.
We also know that in many applications hydrophobic solids are commercially available to be used as filtration aids (slurry additives). Similarly, some patents disclose the use of fiber-like additives as filtration aids. Such additives, create micro-channels (capillaries) between their wall and the solid being filtered, thus enhancing the overall filtration process.

Overall, therefore, we have fundamental and practical evidence that hydrophobicity aids filtration.

Please let me know if I can assist your efforts in any other way.

With kind regards.

Yours sincerely.

Baki Yarar

addition: Appendix-1

DU PONT

INTER-OFFICE MEMORANDUM

SAVANNAH RIVER PLANT

August 30, 1982

TO: J. H. HERSHEY, 704-F

FROM: W. L. WEST, 704-3F

TANK 20 DENSITY DRIVEN SALT REMOVAL

SUMMARY

Recent SRL corrosion inhibitor tests indicated the previous corrosion inhibitor limits in TA 2-970A for tank 20 salt removal are inadequate.[1] SRL has recommended that dissolved salt solutions up to 5.5M nitrate be kept within existing DPSTS 241-5.01 Technical Standard (TS) corrosion inhibitor limits. Limits for salt solutions above 5.5M nitrate are being evaluated by SRL.

Continued salt removal in tank 20 would require substantial addition of corrosion inhibitors to keep the salt solution within Technical Standard limits. Existing tank 20 facilities cannot provide the mixing required to ensure that dissolved salt solutions are maintained inside TS limits.

The demonstration of density driven salt removal in tank 20 should be considered complete. The salt remaining in tank 20 can be removed by mechanical agitation with the slurry pumps provided by Project 9S-2081. Pumps and chemical addition facilities provided by the Project will permit adequate control of solution chemistry during the remainder of tank 20 salt removal.

DISCUSSION

The tank 20 density driven salt removal demonstration began in June 1980 and has removed about 714,000 gallons of salt (66.2% of original salt) in 1,557,000 gallons of solution at an average specific gravity of 1.4.

Corrosion inhibitors (NO_2^- and OH^-) in the dissolved salt
solutions were permitted to be outside TS corrosion inhibitor
limits under TA 2-970A (see Table 1).

The TA inhibitor limits for salt removal were based on
electropotential measurements made by SRL on unstressed steel.[2]
Recent SRL corrosion results from slow strain rate and
fracture mechanics tests on steel in high nitrate (>5.5M) salt
solutions indicate the TA corrosion inhibitor limits were not
as conservative as originally thought.[1] Based on these recent
test results, SRL has recommended conducting salt removal
within existing TS corrosion inhibitor limits.[3] SRL is
evaluating the inhibitor limits required for dissolved salt
solutions above 5.5M [NO_3^-]. Presently, tank 20 solution is
within TS limits.

During salt removal from tank 20, about 30% of the dissolved
salt solution was outside TS corrosion inhibitor limits.
Large volumes of inhibitors (up to 10,000 gallons per batch)
could be required to maintain tank 20 chemistry within Technical
Standard limits. Thorough mixing of inhibitors with the salt
solution would be necessary to control solution chemistry.
No facilities are currently available to agitate and mix the
tank 20 solution. Use of slurry pumps for final salt removal
had already been planned. Project 9S-2081 will provide slurry
pumps and inhibitor addition facilities to remove the remaining
366,000 gallons of salt in tank 20.

Discontinuing density-gradient driven salt removal in tank
20 means that removal of an additional 100,000 gallons of
salt in tank 20 containing about 280,000 Ci (about 6% of
the total activity currently in type IV tanks) will be delayed
about 4 years until completion of Project 9S-2081 in 1986.

WLW/se

CC: R. Maher, 703-A
     O. M. Morris, 704-H
     R. R. Beckmeyer, 704-8F
     E. J. O'Rourke, 704-8F
     G. M. Johnson, 704-8F
     O. W. Mowry, 704-8F
     J. E. Black, 704-8H
     R. A. Scaggs, 704-8H
     J. E. Owen, 704-2H
     E. B. Snell, 704-2H
     D. J. Coon, 704-1H
     A. Q. Goslen, 704-2H

REFERENCES

1. Memo, Meeting Report-Recent Stress Corrosion Cracking
    Data and Tank 20, R. S. Ondrejcin to R. L. Folger,
    May 7, 1982.

2. TA 2-970A, DPSOX 9289, Salt Removal From Tank 20,
    May 2, 1980.

3. Personal Communication, R. S. Ondrejcin to W. L. West,
    July 19, 1982.

TABLE I

CORROSION INHIBITOR LIMITS
TA 2-970A VERSUS TECHNICAL STANDARDS

TA 2-970A		Technical Standards	
		Nitrate Range	Minimum Limit
For all NO_3^- concentrations:		Above 5.5M NO_3^-	SRL Evaluating Limits
Temperature Range	Minimum Concentration	3.0 to 5.5M NO_3^-	$[NO_2^-]+[OH^-] = 1.2M$ $[OH^-] = 0.3M$
below 50°C	$[OH^-] \geq 0.25M$	1.0 to 3.0M NO_3^-	$[NO_2^-]+[OH^-] = 0.4 \times [NO_3^-]$ $[OH^-] = 0.1 \times [NO_3^-]$
50°C - 75°C	$[OH^-] \geq 0.50M$	Less than 1M NO_3^-	$OH^- = 0.01M$ or pH 12

Savannah River Site
High Level Waste Salt Disposition
Systems Engineering Team

HLW Salt Disposition Alternatives Identification
Preconceptual Phase I
Summary Report

WSRC-RP-98-00162

APPROVED for Release for
Unlimited (Release to Public)

April 17, 1998

Alternative #/Title	Disposition	Comments
ML6 - KBI centrifugal bioreactor technology (CBR)	Reject-Science 4b, Theoretical application only and no micro-organisms identified for Cs. Reject-Safety 1b, Does not meet final disposal form.	
ML7 – Grout disposal of salt solution containing Cs 137 in Saltstone	Reject-Safety 1b, Does not meet final disposal form.	Hybrid to address waste form, – renumber as ML7.1
ML7.1 - Disposal of salt solution containing Cs 137.	ACCEPT	including requirements to meet Class C Cs loading if saltstone is used
ML8 - Interstitial fluid displacement for preferential recovery of Cs from saltcake	Reject-Safety 1b, Does not meet final disposal form. Reject-Science 4b, Inadequate basis. Reject-Process 5d, Inadequate eng basis.	
ML9 - Salt dehydration and encapsulation.	Reject-Safety 1b, Does not meet final disposal form.	Hybrid to add TRU to WIPP, renumber as ML9.1
ML9.1 - Salt dehydration and encapsulation with TRU going to WIPP	ACCEPT	
ML10 - In situ grouting of salt within the HLW tanks	Reject-Schedule 2b, Need for additional tanks to support current missions Reject-Process 5a1, Space availability	Need to check volume reduction /increase from grouting. Hybrid disposition included under ML7.1
ML11 - Bio-denitrification in large tanks	Reject-Science 4b, Availability of biological agent Reject-Process 5d, Engineering maturity for Process scale & material	Hybrid rejected based on no benefit from this denitrification approach even considered with other technology
ML12 - Direct processing of interstitial liquid from crystallized salt solutions	ACCEPT	No restriction from DF requirement
ML13 – Grout the salt solution in a purpose built facility without worrying about separating the Cs-137, etc.	ACCEPT	

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ALTERNATIVES FOR HIGH-LEVEL WASTE SALT PROCESSING AT THE SAVANNAH RIVER SITE

Committee on Cesium Processing Alternatives for High-Level Waste at the Savannah River Site
Board on Radioactive Waste Management
Board on Chemical Sciences and Technology
National Research Council

NATIONAL ACADEMY PRESS
Washington, D.C.

8

Barriers to Implementation of HLW Salt Processing Options

The focus of this concluding chapter is on the third charge of the committee's statement of task (see Chapter 1): "Are there significant barriers to the implementation of any of the preferred alternatives, taking into account their state of development and their ability to be integrated into the existing Savannah River Site (SRS) high-level waste (HLW) system?" Many of the research and development (R&D) barriers to implementation of salt processing alternatives were addressed in Chapter 3, Chapter 4, Chapter 5, Chapter 6, Chapter 7, and will not be repeated here. Instead, this chapter focuses on what the committee considers to be two "global" challenges for selecting and implementing a salt processing alternative: (1) systems integration, and (2) program management.

SYSTEMS INTEGRATION

HLW salt processing is a single but key component of a much larger HLW processing system (see Figure 1.2), and the option(s) selected for processing the HLW salts must be fully compatible with the other system components. The U.S. Department of Energy (DOE) recognized the importance of *systems engineering*[1] to the success of the salt processing program when it asked the committee to comment on systems integration in this third charge. Given the abbreviated schedule for this project, the committee did not have an opportunity to perform a detailed analysis of the HLW system at SRS or the potential for integration of the candidate processing options into that system. The committee did, however, gather much information on this issue through its oral and written communications with SRS staff throughout the course of this study. Much of this information is presented in Chapter 3, Chapter 4, Chapter 5, Chapter 6, Chapter 7, of this report. **Based on this information, the committee concludes that**

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[1] See *Systems Analysis and Systems Engineering in Environmental Remediation Programs at the Department of Energy Hanford Site* (National Research Council, 1998) for a good discussion of systems engineering concepts.

systems integration is not adequately implemented in the HLW saltprocessing options program at SRS.

The committee observed that cesium, strontium, and actinide processing are being treated as individual issues rather than as components of a fully integrated engineered system. The committee did see some evidence that systems engineering approaches are being used at SRS—for example, SRS is considering the potential impacts of waste stream feeds to the Defense Waste Processing Facility (DWPF) in its evaluation of processing options. However, other essential aspects of a systems engineering approach were lacking, especially the integrated consideration of alternative flowsheets for processing the HLW salt solutions.

The following example perhaps best illustrates the committee's conclusion on this point: As the committee became better acquainted with the HLW system at SRS, the members realized that the tank wastes were more variable in chemical and radionuclide compositions than they had been led to believe initially. The compositional differences are caused by differing inputs to the tanks from reprocessing operations (see Chapter 1) over the years and by subsequent tank transfers and processing operations. In fact, the tanks contain variable quantities of sludge, saltcake, and supernate, and radionuclide concentrations can vary from tank to tank by several orders of magnitude. This variability is illustrated for some key radionuclides in Figure 8.1, Figure 8.2, Figure 8.3.
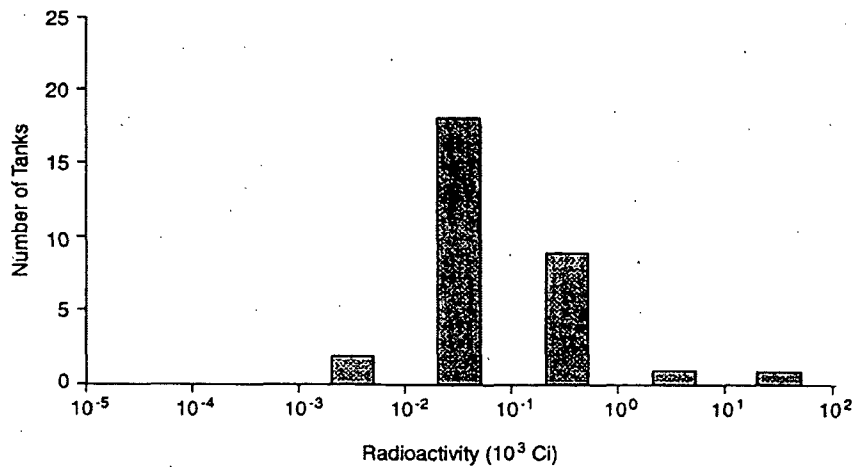
FIGURE 8.1 Histogram showing the variation of strontium-90 activity in soluble radionuclides in the high-level waste tanks at Savannah River. SOURCE: Data from Fowler (2000); see Table 1.2 of this report.
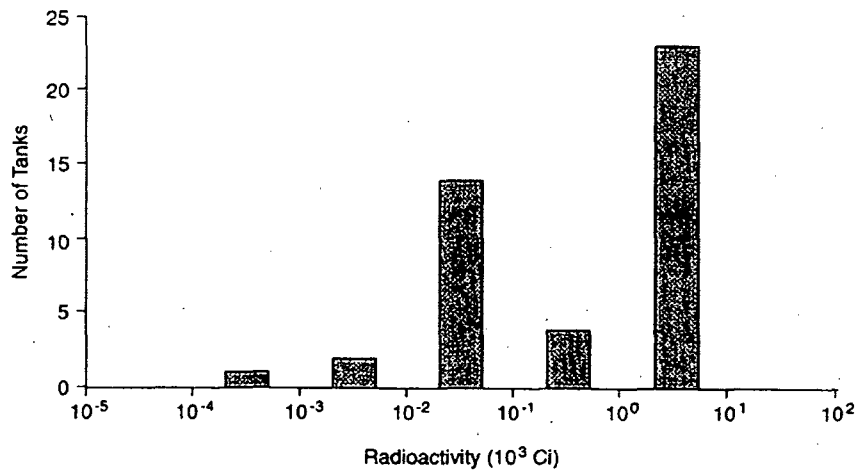
FIGURE 8.2 Histogram showing the variation of alpha activity in soluble radionuclides in the high-level waste tanks at Savannah River. SOURCE: Data from Fowler (2000); see Table 1.2 of this report.
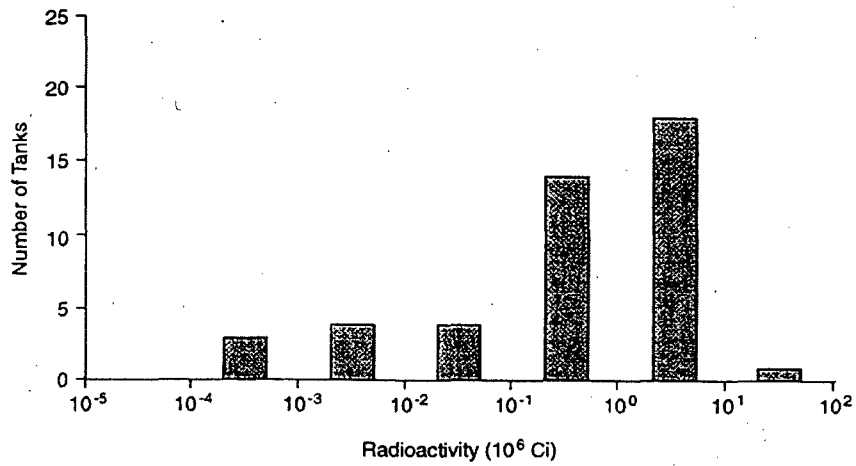
FIGURE 8.3 Histogram showing the variation of cesium-137 activity in soluble radionuclides in the high-level waste tanks at Savannah River. SOURCE: Data from Fowler (2000); see Table 1.2 of this report.

These observed intertank variabilities led the committee to pose the following questions about implementation of each salt processing option:

1) Should all tanks be subject to the same processing operations? Although the average concentrations of soluble radionuclides in the tank waste appear to be above saltstone limits (see Table 7.1), the concentrations of soluble radionuclides in some of the tanks (see Table 1.2) fall below the supernate waste stream limits for the saltstone facility—suggesting that the contents of some tanks could be sent directly to grout with little or no radio-nuclide removal. Thus, instead of blending tank wastes to produce a feed that might allow all tank contents to be treated by a single process, as is now planned, would it be advantageous to tailor processing based on chemical and radionuclide contents of individual tanks? For example, could tank wastes with little or no cesium be processed only to remove strontium and actinides—essentially, the direct grout option discussed in Chapter 7? Alternatively, could tank wastes with low strontium and actinide concentrations be processed only to remove cesium? Indeed, could tank wastes with low actinide, strontium, and cesium concentrations be sent directly to the saltstone facility after minor waste conditioning (e.g., filtration)?[2] Although this tailored approach might require additional regulatory approval and perhaps some facility modifications (e.g., the construction of additional waste transfer lines), it might allow the tank wastes to be processed on a faster schedule, thereby reducing costs and freeing up tank storage space for ongoing HLW processing operations. Indeed, using this approach, SRS might be able to process some tank wastes before a cesium processing option is selected and implemented, because the saltstone facility already exists. Once a cesium processing option is implemented, SRS could focus its processing efforts on the high-cesium tanks, which would produce the cesium feed stream that may be used later this decade to immobilize excess weapons plutonium.

2) Should the actinide and strontium processing step be performed prior to cesium removal? Only for the tetraphenylborate (TPB) process have advantages been presented by SRS for the MST operation as a front-end step (see Chapter 3). For other process options, the committee sees no advantages in removing these radionuclides in a front-end operation and believes that there may, in fact, be a significant disadvantage: shielding requirements are higher, thereby increasing the hazard, cost, and time of processing. The removal of cesium in a front-end step could result in much reduced radiation fields, allowing strontium and actinides to be removed in a smaller, less expensive facility. As noted previously, the MST processing step might be skipped altogether for tank wastes with low strontium and cesium concentrations.

Recommendation: SRS should implement a more fully inte grated systems engineering approach for processing HLW salt solu

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[2] Dilution of the supernate during salt processing operations could further reduce the radionuclide concentrations shown in Table 1.2, thereby allowing more tank waste to be sent to the saltstone facility without the need for extensive radionuclide separations.

tions. To this end, SRS should consider (1) tailoring the processing op erations to tank waste contents, with the goal of reducing processingtime and costs and freeing up tank space, and (2) changing the order of processing to remove radionuclides from the HLW salt solutions, withthe goal of reducing processing hazards, costs, and time.

PROGRAM MANAGEMENT

As noted in several chapters of this report, the committee has concluded that SRS generally appears to have a good understanding of the technical uncertainties that must be resolved before a HLW salt processing option can be implemented, and the committee has recommended approaches to and conduct of R&D work to resolve these uncertainties. Assuming that this R&D work is adequately funded and the appropriate people are identified to perform this work, the main barrier to the successful implementation of a salt processing option then involves two management issues: (1) ensuring that the R&D work stays properly focused on the right problems; and (2) ensuring that the salt processing program uses the information gathered by the R&D program to make appropriate selection and implementation decisions. The remainder of this chapter addresses these issues.

The experience with in-tank precipitation (see Chapter 4) illustrates how unanticipated technical "surprises" can upset even seemingly well-planned projects. Given the large volume and the chemical complexity of tank waste at SRS, such unexpected problems are possible and perhaps even likely in the future. Consequently, a primary objective of the R&D program on alternatives should be to bound performance over the range of waste and operating conditions likely to be encountered during future processing operations. This will enable engineers to design and implement a process that can accommodate such future surprises without major upsets to the high-level waste system.

The R&D program will likely be conducted at several sites across the country, and competent technical leadership will be required to ensure that this program is properly focused and coordinated. **The committee does not believe that the R&D program management should reside solely at SRS, because that site does not have the full range of technical capabilities required to direct and evaluate the required R&D work.** The committee had contact with many technical staff at SRS during the course of this review and found them, on the whole, to be a capable and dedicated group. Nevertheless, considerable experience is present at several other DOE sites (e.g., experience with cesium removal at Hanford Site and Oak Ridge National Laboratory). SRS should take greater advantage of this knowledge and experience. This work has been and will most likely continue to be carried out at a number of companies, universities, and national laboratories. The committee believes that the personnel who oversee and evaluate this work should have the similar range of technical expertise as the experts who actually perform the R&D. DOE Order 435.1 specifies that leadership and responsibility for

S-112 Retrieval Progress

Blaine Barton

Savannah River – Hanford Technical Exchange

March 22-24, 2005
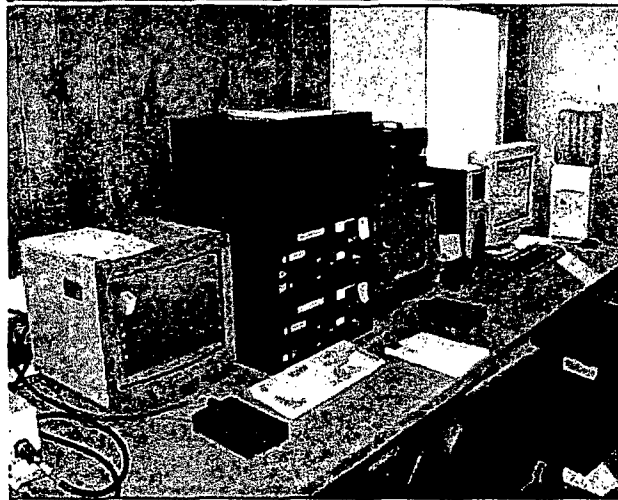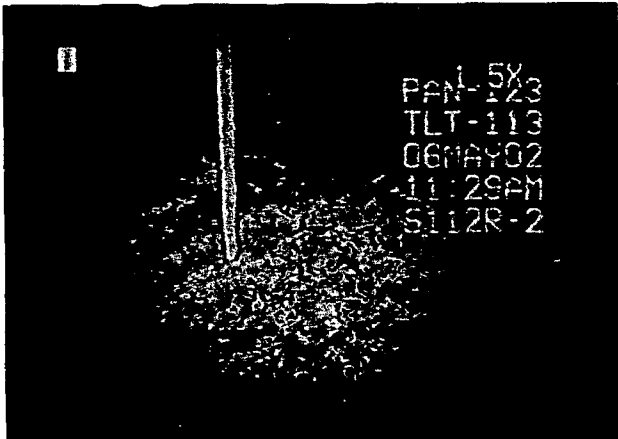
Office of River Protection

CH2M HILL
Hanford Group, Inc.

Progress of Retrieval at Hanford

W. B. Barton
CH2M HILL Hanford Group, Inc.

Approved for Public Release;
Further Dissemination Unlimited

A. D. Aardal 3/18/2005
Release Approval   Date

CH2M-25332-VA
Revision 0

S-112 Retrieval Progress

- What is in the tank
- Retrieval Method
- Progress
- Surprises
- Lessons Learned

CH2MHILL
Hanford Group, Inc.
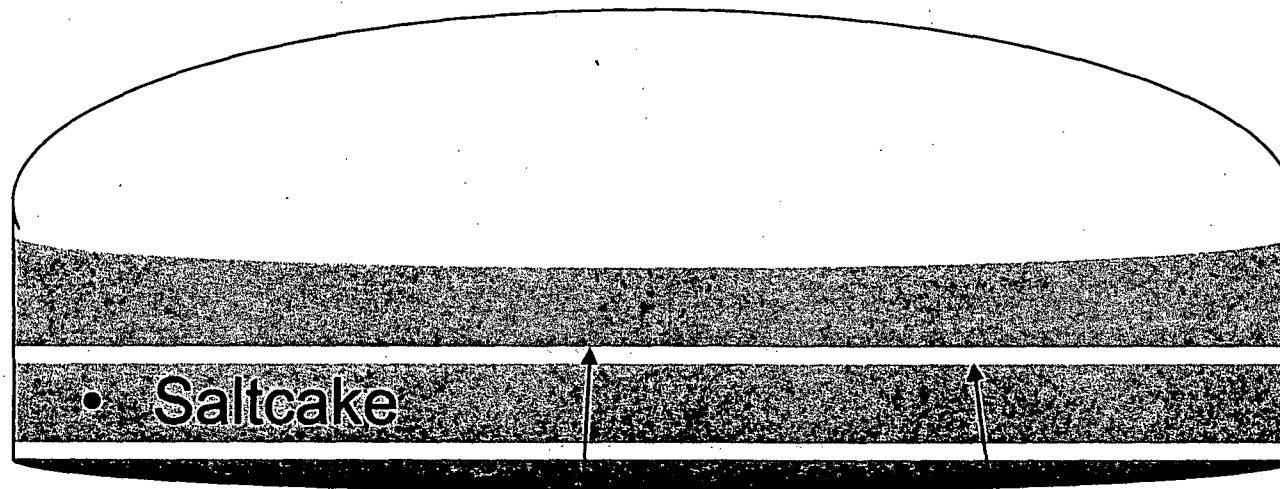
- Saltcake

- Sludge

- Interstitial Liquid Level

Hard Salt Layers

CHG0503-13.5

What is in S-112?

S112 Riser 8 LOW 23 Jul 2003 01:19:00

Reference Scan Taken On: 1/13/1999 8:18:88 AM

Depth From Bottom Of Tank (ft)

Office of River Protection

CHG0503-13.6
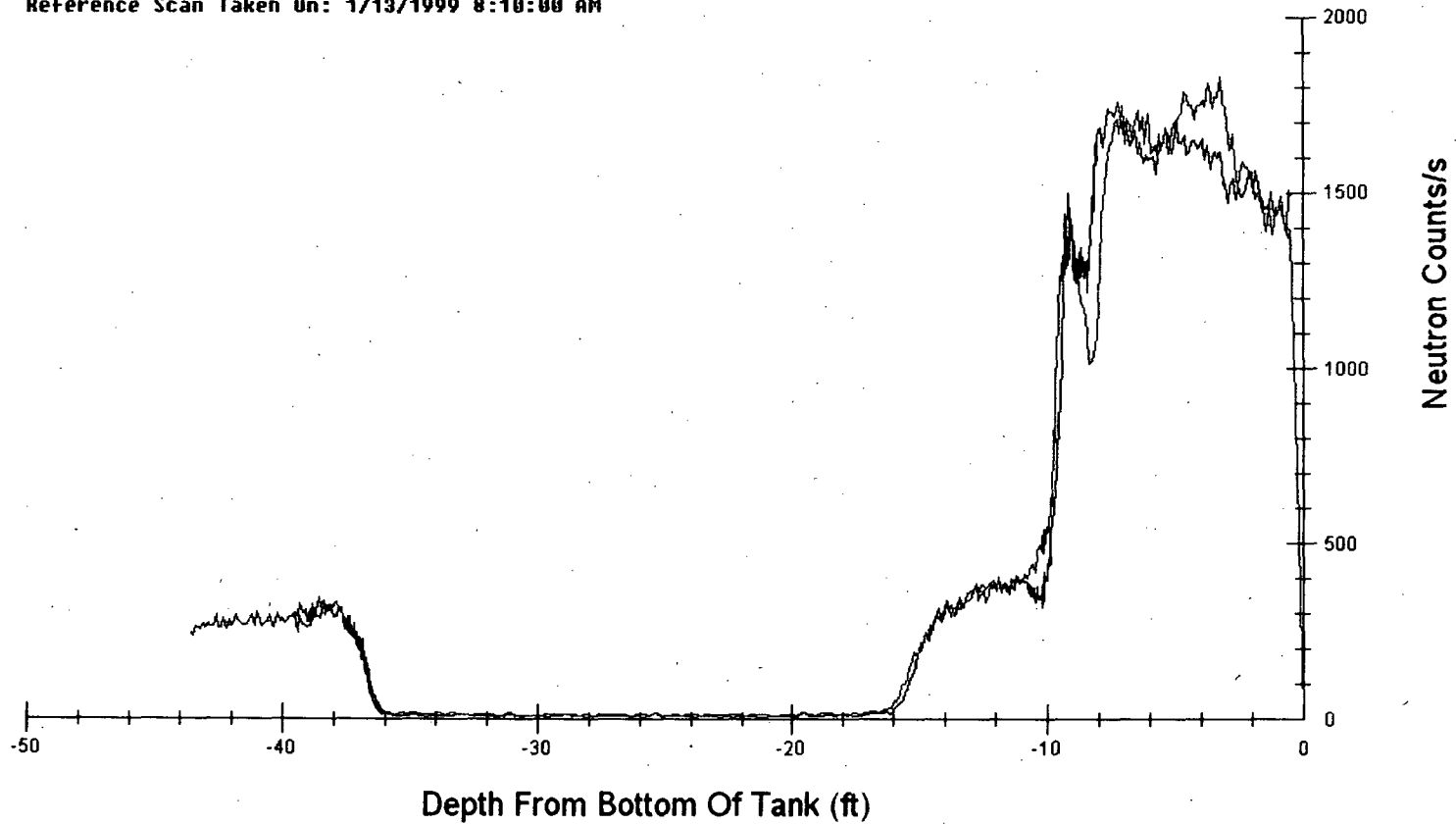
What is in S-112?

S112 Riser 8 LOW 06 Jan 1999 08:49:00

Reference Scan Taken On: No Reference Scan Found

CHG0503-13.7
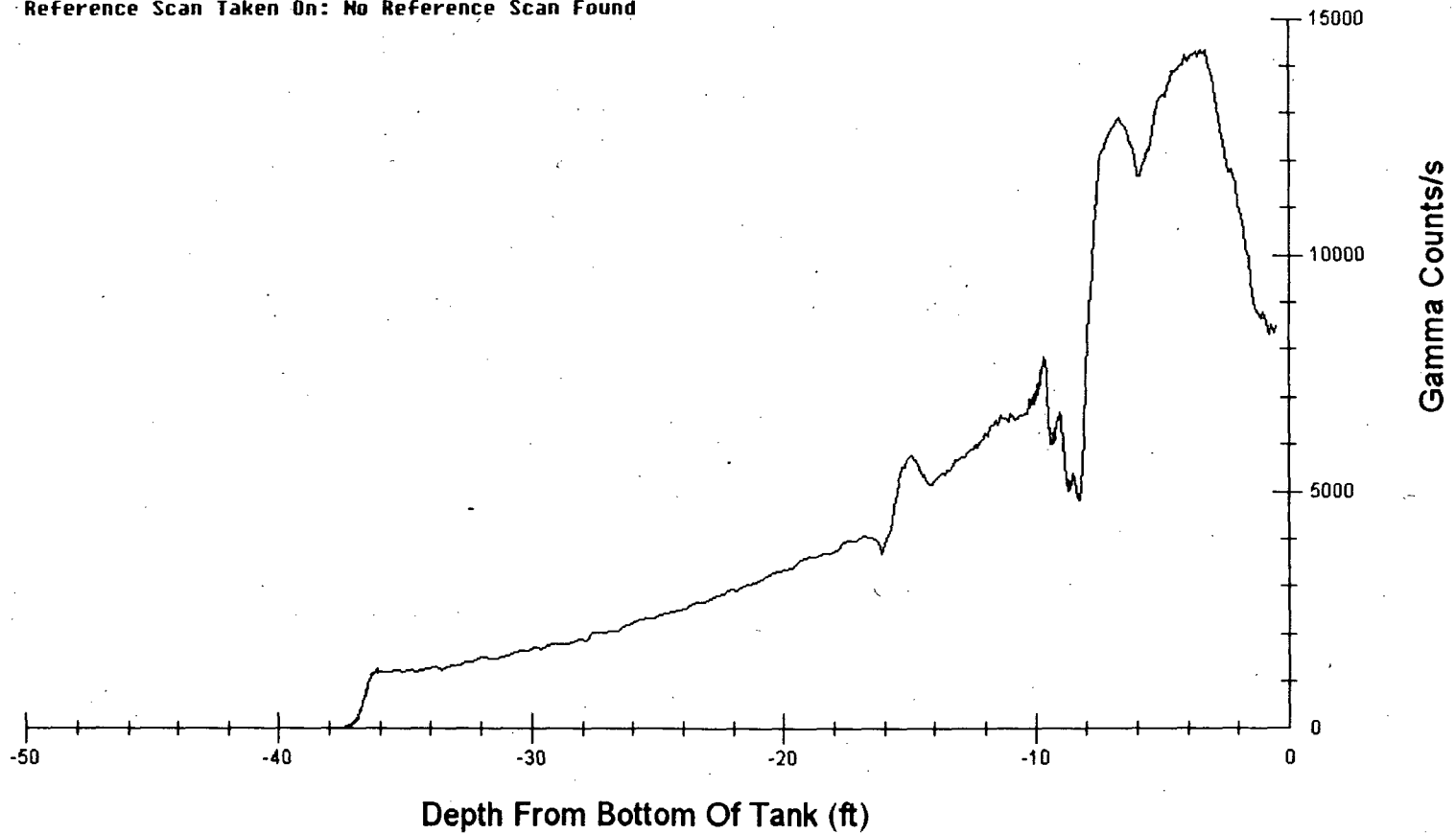
Retrieval Method

- Dissolve salts in warm water

- Sluice insoluble waste with water

- No provisions to remove solids larger than 3/8 inch.

- Pump suction fixed at bottom of tank

- Water introduced through four remote water distribution devices (RWDD)
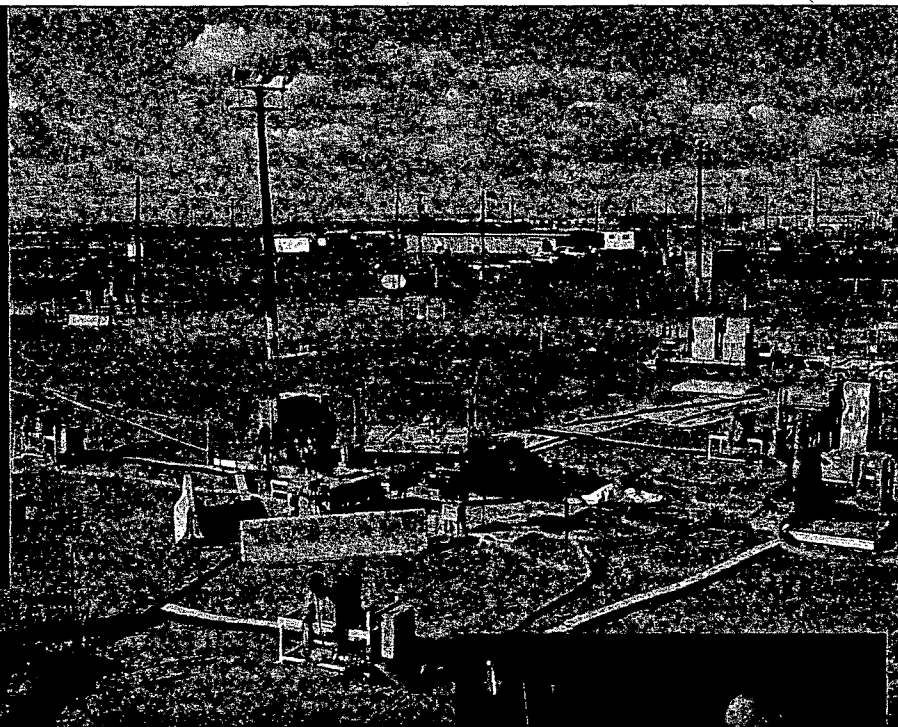
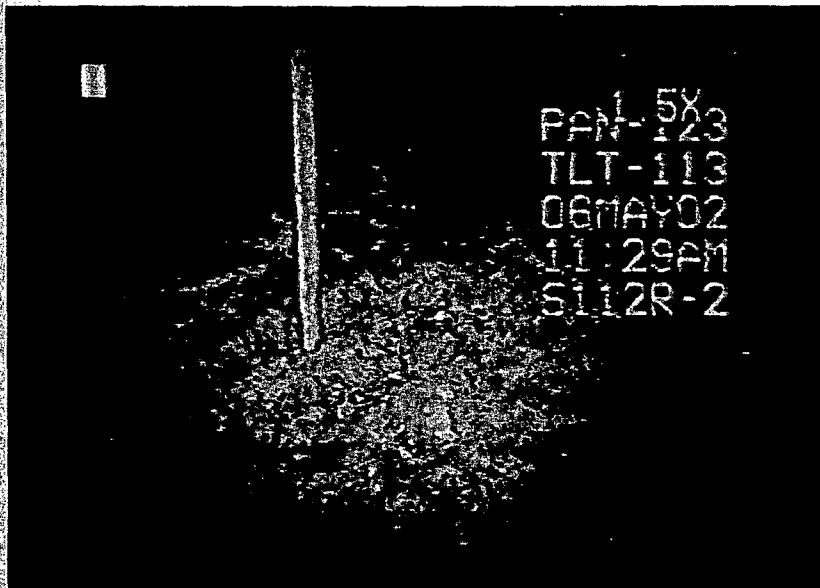- Diagrams and pictures in last year's talk

Retrieval Method

241-SY-102
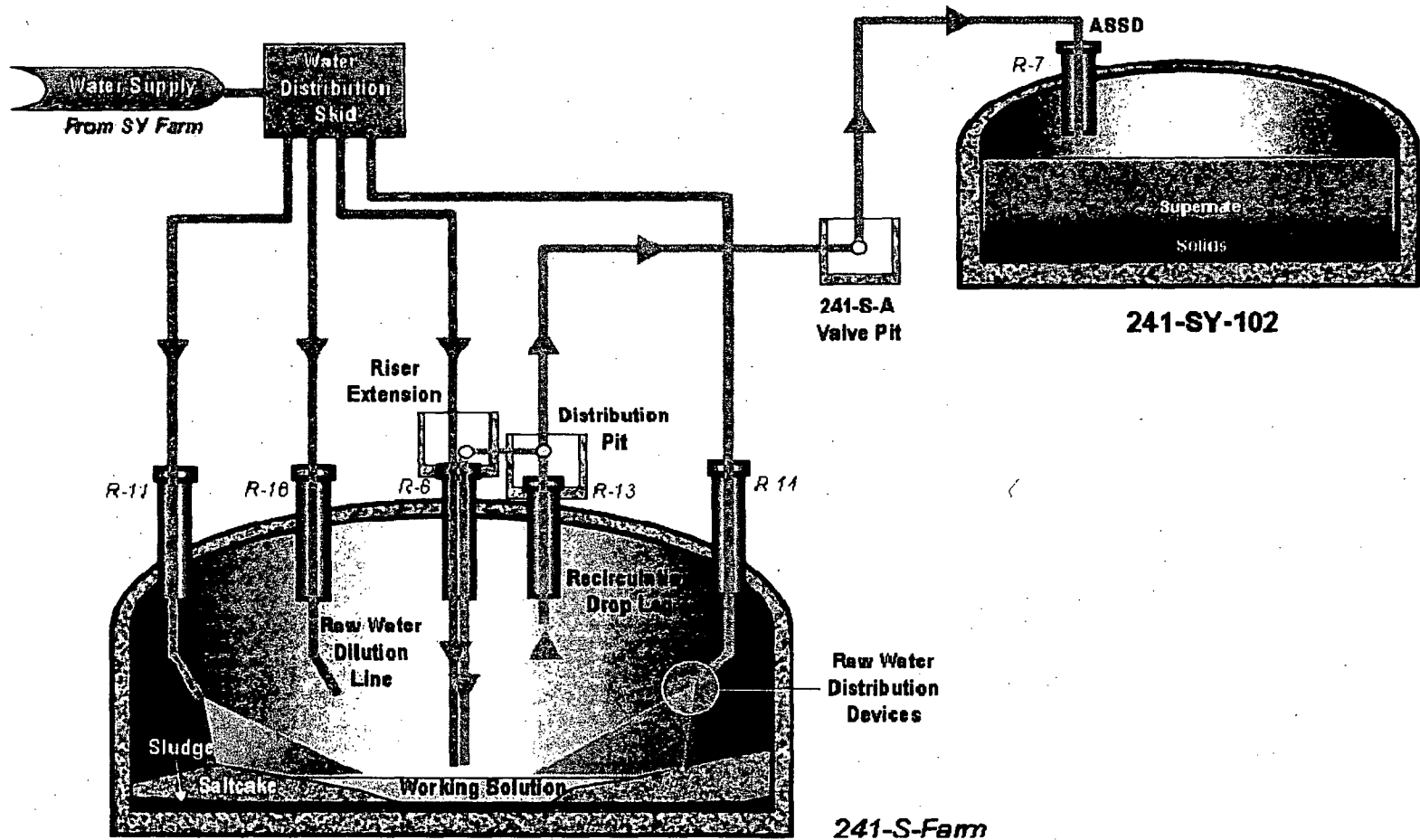
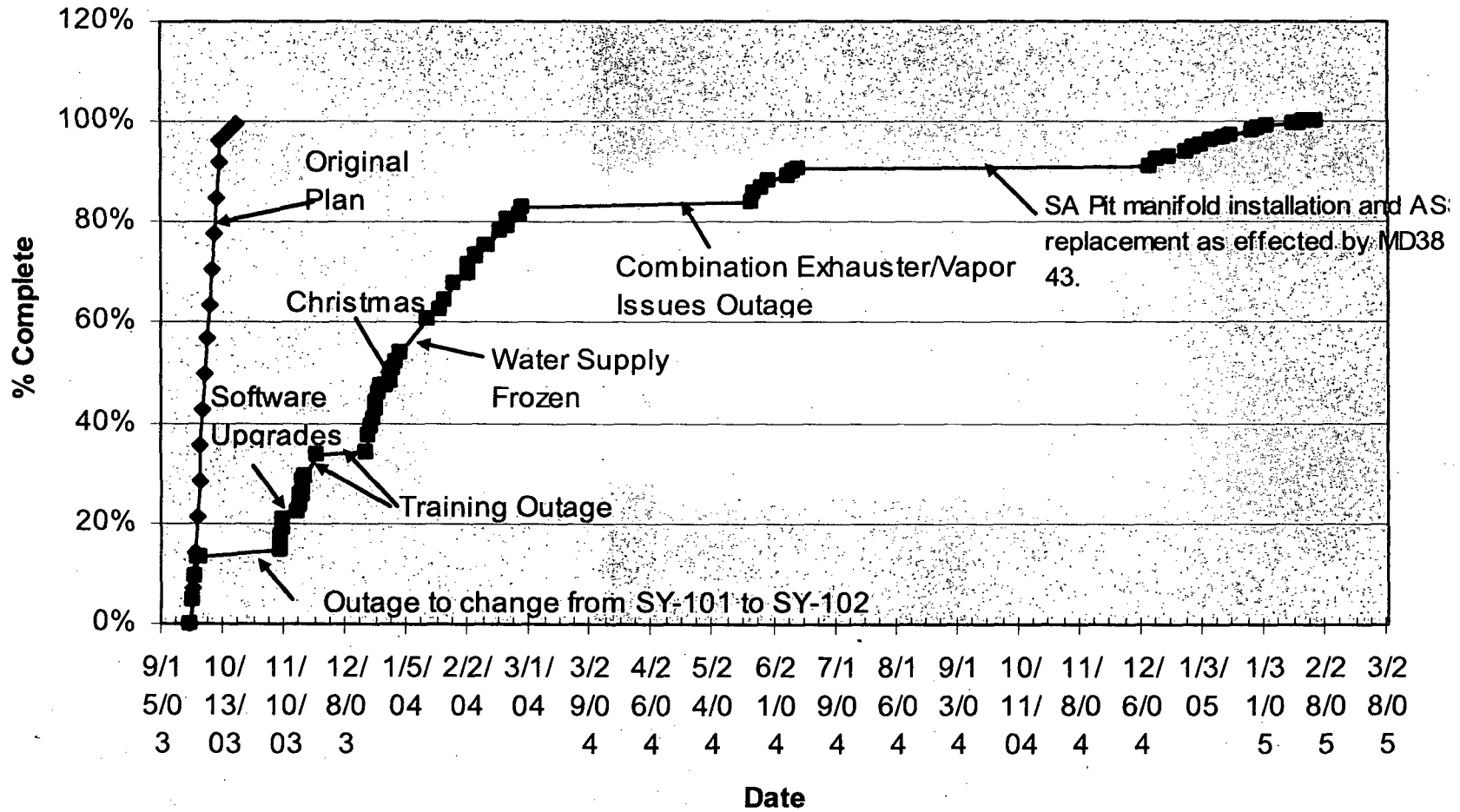241-S-112 and 241-S-102 Phase 1

241-S-Farm
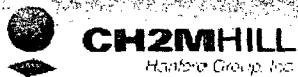
Retrieval Method

Progress

S-112 Schedule Progress

Original Plan

Christmas

Software Upgrades

Training Outage

Water Supply Frozen

Combination Exhauster/Vapor Issues Outage

SA Pit manifold installation and AS replacement as effected by MD38 43.

Outage to change from SY-101 to SY-102

% Complete

Date

CHG0503-13.11

Progress

S-112 Retrieval Progress

Chart titled "S-112 Retrieval Progress" with y-axis labeled "% Completion" ranging from 0% to 120%, and x-axis labeled "Operating Days" ranging from 0 to 130. Two curves are shown: "Original Plan" and "Actual".

Progress

S-112 Retrieval Progress

Chart showing % Retrieved vs Volume Transferred to SY Farm (kgal). Y-axis labeled "% Retrievec" from 0% to 120%. X-axis labeled "Volume Transferred to SY Farm (kgal)" from 0 to 2500. Two data series labeled "Actual Progress" and "Predicted Progress" rise together to approximately 100% near 1500 kgal, then level off. A vertical line near 2000 kgal marks "Allocated DST Space".

CHG0503-13.13

S-112 Retrieval Efficiency based on DST space usage
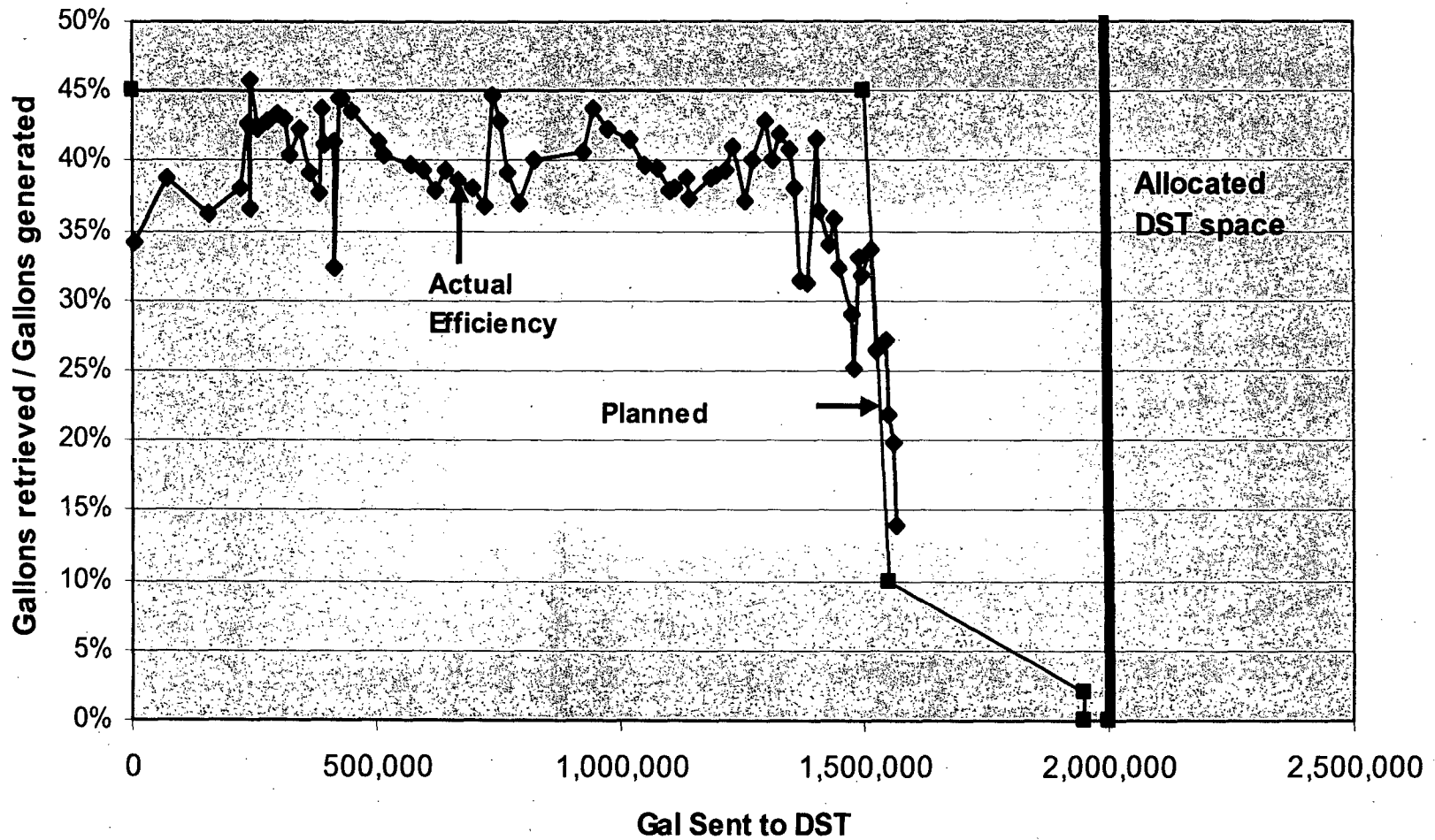
Progress

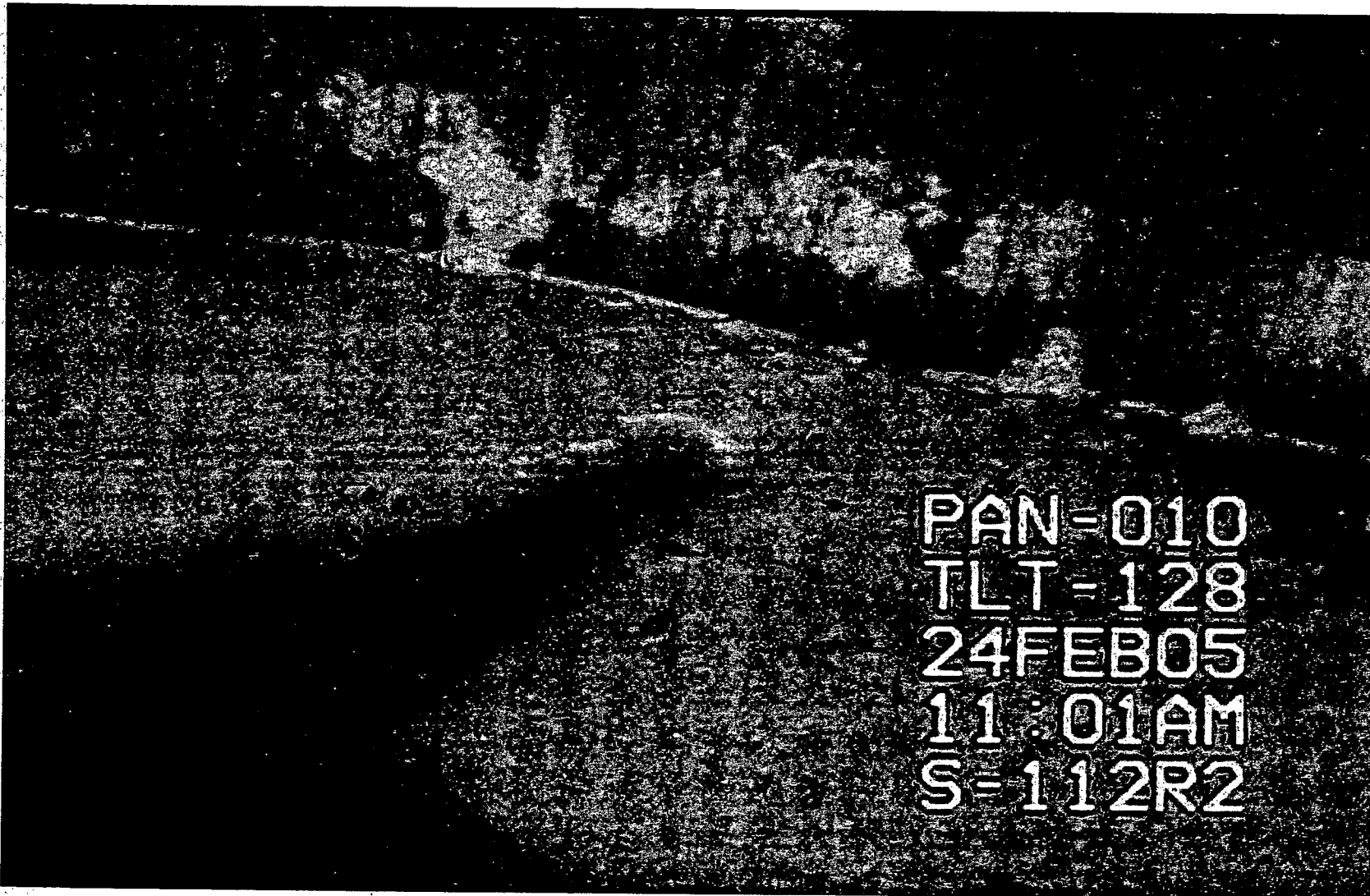- Discuss Sample results

PAN=257
TLT-135
23FEB05
08:31PM
S-112R2
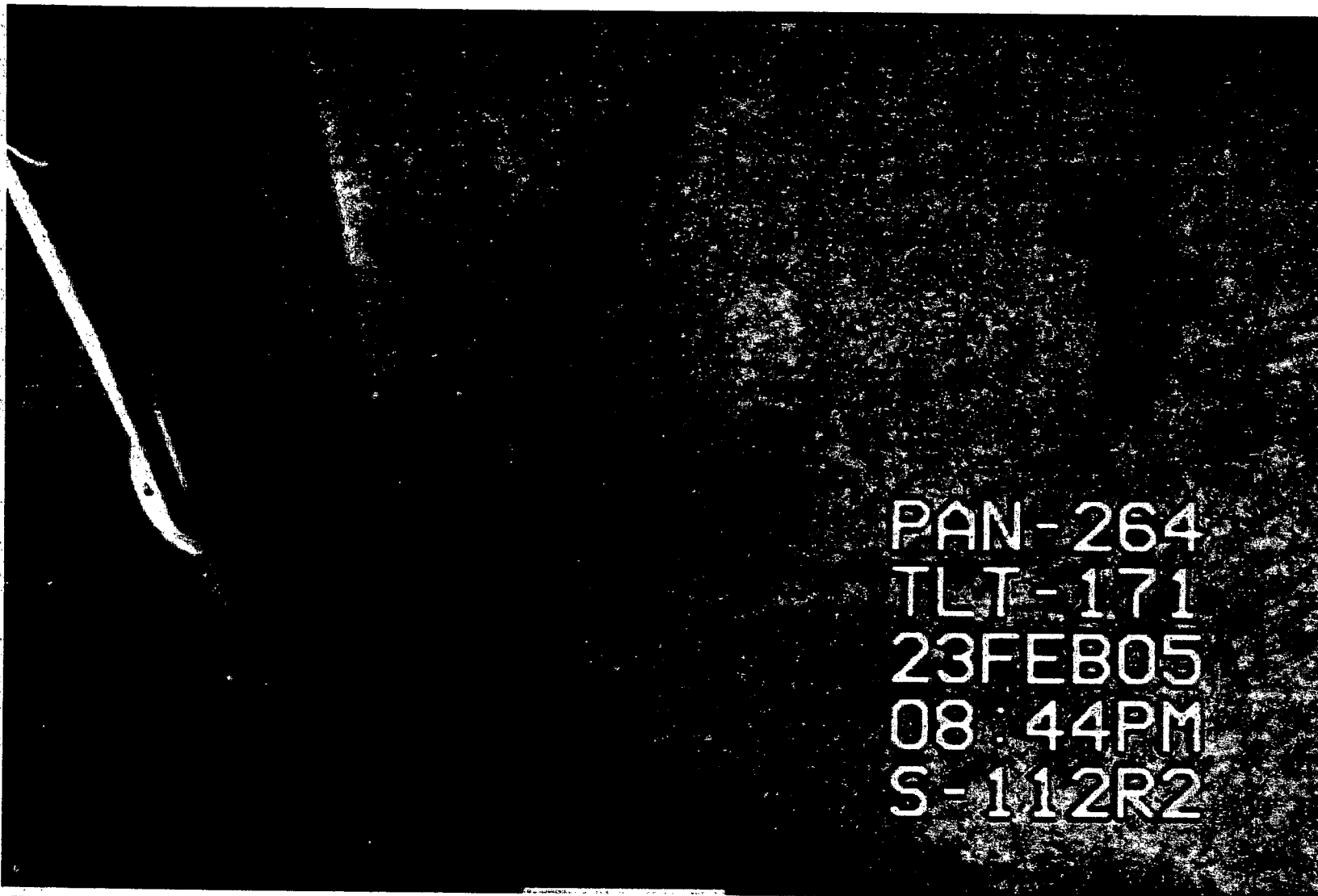
PAN-264
TLT-171
23FEB05
08:44PM
S-112R2

Progress

PAN=010
TLT=128
24FEB05
11:01AM
S=112R2

CHG0503-13.18

CH2MHILL
Hanford Group, Inc.

- ~35 kgal waste remaining.
- 100 % retrieved based on beginning inventory.
- Solid monolith of salt 75 ft diameter and ~ 20 inches thick.
- Pump screen is blinding.

Office of River Protection

Surprises

- We did not anticipate the length of time to dissolve the salt.
  - Dissolution is controlled by surface area, temperature, degree of saturation at interface
- Degree of density stratification in brine
- Impact of endothermic reaction
- Hard nonporous salt layers and their impact on progress
- Plugging of pump screen

Lessons Learned

- Use the hottest water the system can tolerate.

- Some method of stirring is essential, especially near the end.

- Screens on pump suction may cause more problems than they protect against.

- Train and retrain the operating staff.

CHG0503-13.21

Lessons Learned

- Design equipment and processes for relatively long use.

- Most of the mobile radionuclides wash out of the salt early in the process.

- Treat software with the same rigor as hardware during design, installation, and operation

- If it works don't fix it.