SRS-REG-2007-00041
Revision 1

KEYWORDS:
Performance Assessment
Saltstone
UDQE

RETENTION: PERMANENT

UNREVIEWED DISPOSAL QUESTION EVALUATION:

Evaluation of Liquid Weeping from Saltstone Vault 4 Exterior Walls

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REVIEWS AND APPROVALS

Authors

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Summary

Upon restart of grout production in the Saltstone Production Facility and subsequent disposal in the Saltstone Disposal Facility, wet spots on the exterior of the Vault 4 walls have been observed during grout pouring operations. The wet spots are a result of liquid weeping from minor hairline cracks in the vault walls or at construction joints. This analysis concludes that vault cracking was a situation that was acknowledged and accounted for in the performance assessment analyses and there is an insignificant impact due to the radionuclide inventory associated with the weeping. Therefore, the discovery of the vault weeping is within the bounds of the existing Performance Assessment[1] and Composite Analysis[2].

Introduction

One intent of DOE Order 435.1[3], as expressed in the performance assessment/composite analysis guidance[4], is to ensure that proposed or discovered changes in waste forms, containers, radionuclide inventories, facility design, and operations, are reviewed to ensure that the assumptions, results, and conclusions of the DOE approved performance assessment (PA), and composite analysis (CA), as well as any Special Analyses (SA) that might have been performed, remain valid (i.e., that the proposed change or discovery is bounded by the PA and CA) and the changes are within the bounds of the Disposal Authorization Statement. The goal is to provide flexibility in day-to-day operations and to require those issues with a significant impact on the PA's conclusions, and therefore the projected compliance with performance objectives/measures, to be identified and brought to the proper level of attention. It should be noted that the term performance measure is used to describe site specific adaptations of the DOE Order 435.1 Performance Objectives and requirements (e.g., performance measures such as applying drinking water standards to the groundwater impacts assessment).

The intent of this document is to provide an evaluation to determine if the presence of minor hairline cracks in the vault walls and cracks at construction joints in the Saltstone Disposal Facility (SDF) Vault 4 concrete walls, indicated by observed wet spots on the exterior surface of the vault walls, are within the assumptions, parameters, and bases of the approved PA, including applicable Special Analyses, and CA and that the risk associated with the weeping radionuclide inventory is insignificant. If not, then, according to the SRS Disposal Authorization Statement[5] (DAS), the PA and CA would need to be updated as appropriate and DOE approval sought of the update (special analysis or revision of the PA or CA).

Description of the Discovery

Upon restart of grout production in the Saltstone Production Facility and subsequent disposal in the Saltstone Disposal Facility, wet spots on the exterior of the Vault 4 walls have been observed during grout pouring operations. The wet spots are a result of liquid weeping from minor cracks in the vault walls.

Supporting Analysis

Liquids, in the form of bleed water or flush water present, in and around the saltstone grout occur from transient conditions encountered during saltstone operations. Bleed/process water resulting from the grout curing process, flush water used to clean portions of the transfer lines after grout runs, condensation from temperature changes, along with small amounts of rainwater intrusion can fill the gap that forms between the grout and the vault wall. During active grout pouring in a cell, the water column can create enough hydrostatic pressure to cause liquid to weep from the cracks in the vault walls. The liquids present from processing operations are removed on a routine basis using the Vault 4 Drain Water Return System. The Drain Water Return System collection pipe within the vault cells will be emptied and flushed prior to final closure to ensure any radionuclide inventory in the pipe is minimized. The requirement to flush the system will be added to a future revision to the Saltstone Disposal Facility Closure Plan. The vault will not contain free liquid at the time of closure.

The issue of SDF vault cracking has been addressed in the 2005 SA[6], the response to the Nuclear Regulatory Commission's (NRC) Request for Additional Information[7] (RAI), and in the responses to public meeting Action Items[8] during the NRC consultation process on the SRS Salt Waste Disposal Section 3116 Waste Determination[9]. Cracking of SDF concrete vaults is expected to occur. In addition, the SA assumes that degradation of the vault concrete over time will increase the water flux through the vault concrete.

<u>2005 Special Analysis</u>

The 2005 SA addressed the issue of cracking in Section A.4. The analysis indicates that vertical cracks or fractures are anticipated due to static settlement and seismic effects (i.e., earthquakes). The analysis concluded that during the evaluation period of 0 to 10,000 years the conditions predicted by anticipated low water infiltration and saltstone suction head of roughly 1200 cm result in an insignificant impact from any cracks.

> *Vertical cracks or fractures spanning the entire Saltstone Vault 4 width and height are predicted to occur at 30 ft intervals, coinciding with construction joints, in response to static settlement and earthquakes. For the assumed properties of saltstone, the literature indicates cracks can be neglected when the suction head exceeds approximately 200 cm in saltstone. Such conditions are predicted to occur during the 0-10,000 year period. This conclusion applies regardless of crack geometry, i.e., open at top, open at bottom, or through-crack.* [2005 SA[6], page 83]

<u>Responses to NRC RAI</u>

The responses to the NRC's RAI[7] elaborated on the cracking evaluations in RAIs 32, 36, 39, and 42. The responses indicated that the occurrence of cracks in the vault over the 10,000 year evaluation period are expected and were accounted for by increasing the hydraulic conductivity (i.e., increased water flow potential) through the vault structure rather than by explicitly modeling physical cracks in the vaults. RAI 32 states:

> *In the 2005 SA, two aspects of concrete degradation were considered: 1) cracking caused by differential settlement and seismic events and 2) internal and external mechanisms/processes which led to an increase in hydraulic conductivity over time. These processes include rebar corrosion, ettringite formation (sulfate attack), carbonation, and calcium hydroxide leaching.*
>
> *A structural analysis predicted that cracks will develop from differential settlement and seismic events over a 10,000 year period and their apertures will increase with increasing time [Peregoy, 2003]. However, that analysis showed that the cracks will open either at the top or at the bottom and will be pinched closed at the opposite end. The 2005 Special Analysis, Section A.4, concluded that cracks of any geometry have very little effect on contaminant transport rate. Based on this finding, large-scale cracks [from seismic events and settlement] were not explicitly modeled in the 2005 SA [Cook et al. 2005, Section A.4].*
>
> ****. The saturated hydraulic conductivity of the saltstone vault concrete was increased from 1.0E-12 to 1.0E-9 cm/sec over 10,000 years [Cook et al. 2005, Section A.4, p. A-9]. This approach was intended to address the consequences of degradation (cracks) regardless of the mechanism and to eliminate: 1) numerical difficulties associated with modeling fracture networks in a groundwater computer code and 2) large uncertainties associated with inputs such as timing, frequency, and size of fractures in the concrete vault. ***.* [NRC RAI[7], page 217]

RAI 42 provides information relative to the anticipated crack sizes and acknowledges that cracks will form in the first 100 years of evaluation and were accounted for in the performance assessment.

> *An extensive structural analysis was performed for Vault 4 to assess the potential for large-scale cracking in response to forecast static settlement and earthquakes (Peregoy 2003). Approximately vertical cracks or fractures spanning the entire vault width and height are predicted to form at multiple construction joints, which occur at 30 ft. intervals. Cracks form within the first 100 years and gradually open with time (Figure 42-1).*
>
> ***
>
> *The influence of small-scale cracking, together with other phenomena degrading the hydraulic performance of saltstone, is implicitly addressed by an increasing saturated*

hydraulic conductivity with time (Cook et al. 2005, Section A.2.3.1)…. [NRC RAI[7] , page 253]

For the analysis, cracks ranging in size from .01 in. open at the top to .06 in. open at the bottom were evaluated beginning at year 100 and increasing in size to .62 in. and 2.18 in. respectively at year 10,000.

<u>Response to Public Meeting Action Items</u>

The responses to Action Items from NRC consultation public meetings[8] further clarified and evaluated the potential impact of vault cracking in Action Items 7 and 8 from the July 27, 2005 public meeting and Action Items 7 and 10 from the August 17, 2005 public meeting. Action Item 7 (7/27/05) summarizes the varied mechanisms that could potentially cause cracking, the additional sensitivity cases evaluated and the final conclusion that after extending the potential range of degradation impacts the calculated doses were still below the performance objectives.

Degradation mechanisms qualitatively considered for the concrete vault and the Saltstone waste form include:

- *Cracking from seismic events and settlement*
- *Cracking due to external static loading (weight of overburden and cap)*
- *Chemical reactions involving the waste components in Saltstone which could result in expansion and cracking.*
- *Chemical reactions involving ions in the soil which could result in expansion and cracking*
- *Chemical reactions involving corrodents in the soil which could cause leaching and an increase in porosity and/or cracking in the vault*
- *Physical process such as freeze-thaw cycles*

[NRC Action Items[8], page 22]

The responses to Action Items 8 (7/27/05), 7 (8/17/05) and 10 (8/17/05) detail new sensitivity cases, Scenarios 31 and 32 and an in-filled crack scenario, modeling vault and saltstone cracking in which the cracks are fully saturated (i.e., increased water content over anticipated case) or filled with granular material. In both cases, the calculated doses are below the performance objective of 25 mrem/year. Action Item 8 (7/27/05) states:

In sensitivity case 31, discussed in the response to U.S. Nuclear Regulatory Commission (NRC) Action Items 10 (8/17/05) contained within this document, the vault and saltstone are assumed to exhibit large-scale through-cracks at a 30 ft. spacing. For comparison, the best-estimate settlement/seismic crack spacing averages 200 ft, i.e. three transverse cracks over a 600 ft. length (Ref. 6). Cracks are represented by two feet wide columns of *gravel in the numerical model. Secondly, the vault, saltstone, and cracks are assumed to be fully saturated. The latter is implemented in the numerical model by setting the water retention and relative permeability curves to 1.0 regardless of suction (i.e., the saturated conductivity value is used under both saturated and unsaturated conditions). These pessimistic assumptions maximize advective flow through porous saltstone, and, more importantly, force flow through the postulated fractures. The result is an increase in dose from 0.05 mrem/year to 3.5 mrem/year in comparison to the base case (sensitivity case 1 from the response to NRC Action Item 10 (8/17/05)….*

A second sensitivity case was run to specifically explore the impact of fractures in-filled with granular material. This sensitivity case is not one of the 33 cases discussed in the response to NRC Action Item 10 (8/17/05). In this case, a conservative 30 ft. crack spacing was assumed as in sensitivity case 31 discussed above. In-filled cracks were represented by two foot columns of native soil in the numerical model to accommodate the existing mesh resolution, and result in a conservative representation of physical cracks with a nominal aperture of roughly one inch. The baseline moisture curves were used for all materials. The result is an increase in dose from 0.05 mrem/yr to 1.1 mrem/yr in comparison to the base case (sensitivity case 1 from the response to NRC Action Item 10 (8/17/05)…. [NRC Action Items[8], page 24]

In order to address the risk associated with the radionuclide inventory from weeping, calculation N-CLC-Z-00020[10] was completed and is included as Attachment 2. The calculation concludes that there is no significant risk associated with the radiological inventory resulting from weeping from the Vault 4 walls.

Evaluation

To complete this UDQE, the following questions, which must be addressed in any UDQE[11], are answered with respect to the information concerning liquid weeping from the Saltstone Vault 4 exterior walls.

1a. *Is the proposed activity or new information outside the bounds of the approved PA/CA (e.g., does the proposed activity or new information involve a change to the basic disposal concept as described in the PA/CA such as critical inputs/assumptions or an increase in inventory analyzed in the CA)?*

NO. Based on the discussion above concerning the basic disposal concept, critical inputs and assumptions bound the condition of minor hairline cracking of the vault wall and cracking at construction joints. Therefore the discovery of wet spots on the exterior of the vault walls is within the bounds of the existing PA/CA.

1b. *Does the proposed activity or new information cause the PA/CA performance measures to be exceeded?*

NO. Based on the discussion above concerning the basic disposal concept, critical inputs and assumptions bound the condition of minor hairline cracking of the vault wall. Therefore the performance measures would not be exceeded.

1c. *Would the radionuclide disposal limits in the approved PA need to be changed to implement the proposed activity?*

NO. Since this is new information rather than a proposed activity, the issue of disposal limits is addressed in Question 1d.

1d. *Does the new information involve a change in the radionuclide disposal limits in the approved PA?*

NO. Based on the analysis above, the 2005 SA bounds the observed conditions. Changes to the radionuclide disposal limits are not required.

1e. *Does the proposed activity or new information involve a change to the DAS?*

NO. Based on the discussion above concerning the basic disposal concept, critical inputs and assumptions in the existing PA bound the condition of minor hairline cracking of the vault wall and cracking at construction joints. Therefore, a change to the Disposal Authorization Statement is not required.

Conclusion

The depth of the initial evaluations presented in the 2005 SA and additional information supplied as part of the NRC consultation process illustrates that vault cracking was a situation that was acknowledged and accounted for in the performance assessment analyses. Therefore, the discovery of the vault weeping is within the bounds of the existing PA/CA. In addition, the radionuclide inventory in the Vault 4 walls and surrounding soil associated with weeping does not present a significant risk based on the calculations in Attachment 2. Furthermore, because backfilling around the vaults and final closure of the SDF will not occur for a number of years, observation of vault conditions, including additional spotting or cracking, can be performed and the facility maintained until the time of closure or addressed in subsequent revisions of the PA as required.

References

[1] WSRC-RP-92-1360, Radiological Performance Assessment for the Z-Area Saltstone Disposal Facility, Revision 0, December 1992.

[2] WSRC-RP-97-311, Composite Analysis E-Area Vaults and Saltstone Disposal Facilities, Revision 0, September 1997.

[3] DOE Order 435.1, Radioactive Waste Management, Change 1, August 2001.

[4] Maintenance Guide for U.S. Department of Energy Low Level Waste Disposal Facility Performance Assessments and Composite Analysis, November 1999.

[5] Disposal Authorization Statement for the Department of Energy Savannah River Site E-Area Vaults and Saltstone Disposal Facilities, September 1999.

[6] WSRC-TR-2005-00074, Special Analysis: Revision of Saltstone Vault 4 Disposal Limits (U), Revision 0, May 2005.

[7] CBU-PIT-2005-00131, Response To Request for Additional Information on the Draft Section 3116 Determination For Salt Waste Disposal at the Savannah River Site, Revision 1, July 2005.

[8] CBU-PIT-2005-00203, Response To Action Items From Public Meetings Between NRC and DOE To Discuss RAI For the Savannah River Site, Revision 1, September 2005.

[9] DOE-WD-2005-001, Section 3116 Determination for Salt Waste Disposal at the Savannah River Site, January 2006.

[10] N-CLC-Z-00020, Saltstone Vault 4 Weeping Radionuclide Screening, Revision 0, April 2008.

[11] WSRC Manual SW24, Section SSF-ENG-2002, Saltstone Facility Unreviewed Disposal Question (U), Revision 1, February 2006.

ATTACHMENT 1

Technical Review Package # TRC-WD-2008-00451

Technical Review Package
Contents Sheet

TRP #: TRC-WD-2008-00451 Rev. 0

Technical Review Package Title

APPROVAL OF SRS-REG-2007-00041, REV. 1 AND N-CLC-Z-00020, REV. 0

Functional Classification: NA

Documents included in package

☐ Yes	☒ No	**DATR**
☒ Yes	☐ No	**DATR Summary**
☐ Yes	☒ No	**USQS**
☐ Yes	☒ No	**USQE**
☒ Yes	☐ No	**CHAPS**

Other Documents Included (List)

THE PA DOES NOT IMPACT TRAPPED HYDROGEN LOCATIONS IN Z AREA. THE ASSOCIATED UDQE IS SRS-REG-2007-00041, REV. 1- UDQE: EVALUATION OF LIQUID WEEPING FROM SALTSTONE VAULT 4 EXTERIOR WALLS.

CLASSIFICATION REVIEW

UNCLASSIFIED - Does Not Contain Unclassified Controlled Nuclear Information

DC/RO: N/A N/A

Date: NA

Guide: N/A

OSR 14-398 (Rev 5-14-2007)

UNCLASSIFIED

Consolidated Hazard Analysis Process (CHAP) Screening

CHAP Screening No.	Rev No.	Functional Classification**	Building/Location	System
TRC-WD-2008-00451		NA		

Title

APPROVAL OF SRS-REG-2007-00041, REV. 1 AND N-CLC-Z-00020, REV. 0

Brief Description of Proposed Activity

THE PA INVOLVES THE APPROVAL OF SRS-REG-2007-00041, REV. 1 AND N-CLC-Z-00020, REV. 0. THIS PA COVERS THE REVISION OF A PREVIOUSLY APPROVED UDQE RELATED TO WEEPING THROUGH CRACKS IN THE WALLS OF SALTSTONE VAULT 4. THE UDQE IS BEING REVISED TO ATTACH N-CLC-Z-00020, REV. 0. THE CALCULATION IS A QUANTITATIVE DOSE ASSESSMENT FROM THE WEEPING WHICH SUPPORTS THE ORIGINAL CONCLUSIONS OF THE UDQE. THE BASIC ASSUMPTIONS AND RESULTS OF THE ORIGINAL UDQE ARE NOT IMPACTED BY THE PA.

1. Is this a routine activity (i.e., an activity judged to be of sufficiently low risk from a process hazards perspective based on past experience, or is a routinely performed activity based on the facility/project schedule or other definitive facility/project document)? [Routine is defined as regular, repetitive, customary, normal operations; if routine activity is changed, check NO and continue.]

   ☒ Yes — If Yes, a CHA is NOT required. The activity should be controlled using AHA, RWP, operating procedures, and/or maintenance/test procedures. N/A remaining questions and obtain required signatures.
   ☐ No — If NO, continue . . .

2. Is this work activity categorized as a complex, high risk activity per Manual 8Q, Procedure 122 and are the primary drivers process hazards? [Complex Activity - activity involving extensive interaction between Operations and other work groups, complex or numerous system alignment changes, complex communication or coordination, unusual hazards to personnel or equipment, or work performed in RBA where dose rate exceeds 100 mr/hr; High Risk - based on hazard categorization detemination as part of AHA procedure.]

   ☐ Yes — If YES, a CHA is required. N/A remaining questions and obtain required signatures.
   ☐ No — If NO, continue..
   ☒ N/A — If N/A, continue..

3. Does the process change, proposed activity or modification only involve laboratory bench scale (small lab quantity) activities?

   ☐ Yes — If YES, a CHA is NOT required. The activity should be controlled using AHA, RWP, operating procedures, laboratory procedures, and/or Conduct of R&D manual (SRNL). N/A remaining questions and obtain required signatures.
   ☐ No — If No, continue..
   ☒ N/A — If N/A, continue..

4. Does the process change, proposed activity or modification potentially increase workers, or co-located workers, exposure to EXISTING hazardous materials (chemical/radiological), or an EXISTING process unique energy source that has not been evaluated and documented? [Are there potentially new initiating events or are new controls potentially needed for existing hazards? Will there be an impact on existing controls or the functional classification of controls? If the answer to either or both of the foregoing questions is Yes, check Yes below. Refer to existing CHA hazard assessment table as necessary to answer this question.]

   ☐ Yes — If Yes, a CHA is required. The CHAP team and CHA activity should be tailored to the complexity of the operation and magnitude of the hazards.
   ☐ No — If No, continue..
   ☒ N/A — If N/A, continue..

5. Does the process change, proposed activity or modification introduce any NEW hazardous materials (chemical/radiological), a NEW process unique energy source, or NEW equipment? [Refer to the existing CHA hazard asessment table as necessary or answer this question.]

   ☐ Yes — If Yes, a CHA is required. The CHAP team and CHA activity should be tailored to the complexity of the operation and magnitude of the hazards.
   ☐ No — If No, continue..
   ☒ N/A — If N/A, continue..

6. Does the process change, proposed activity, or modification involve a change to existing design and operating parameters (i.e. temperature, pressure, pH, specific gravity, liquid level, H_2 generation rate, flow rate, shielding, concentration, change from manual operation to DCS, etc.) that have not been evaluated and documented?

   ☐ Yes — If Yes, a CHA is required.
   ☐ No — If No, a CHA is NOT required.
   ☒ N/A — If N/A, a CHA is NOT required.

Reviewer Preparer	Date
AARON STAUB	2008/04/03
Design Authority Engineer	**Date**
AARON STAUB	2008/04/03
Regulatory/Safety Documentation Manager or Designee	**Date**
PHILLIP NORRIS	2008/04/03

Note: Preparer and Design Authority Engineer can be the same. If CHAP screening is negative, Design Authority Manager may substitute for Regulatory Program/Safety Documentation Manager. If CHAP screening is positive, obtain Regulatory Program Manager/Safety Documentation approval.
*Numbers should be of the form: Brief Facility Designator-Year-Sequential Number (e.g., HTANK-2007-0001).
**This form is intended to address unmitigated process hazards for any system/unit operation, regardless of functional classification.

DATR Summary Sheet for Referencing Previous Technical Reviews

Date 2008/04/03 | TRP# TRC-WD-2008-00451 | Building | System

Title: APPROVAL OF SRS-REG-2007-00041, REV. 1 AND N-CLC-Z-00020, REV. 0

Section 1.0 - Description

Description of Proposed Activity

Functional Classification: NA

THE PA INVOLVES THE APPROVAL OF SRS-REG-2007-00041, REV. 1 AND N-CLC-Z-00020, REV. 0. THIS PA COVERS THE REVISION OF A PREVIOUSLY APPROVED UDQE RELATED TO WEEPING THROUGH CRACKS IN THE WALLS OF SALTSTONE VAULT 4. THE UDQE IS BEING REVISED TO ATTACH N-CLC-Z-00020, REV. 0. THE CALCULATION IS A QUANTITATIVE DOSE ASSESSMENT FROM THE WEEPING WHICH SUPPORTS THE ORIGINAL CONCLUSIONS OF THE UDQE. THE BASIC ASSUMPTIONS AND RESULTS OF THE ORIGINAL UDQE ARE NOT IMPACTED BY THE PA.

Screening Questions:

Question			
Is the modification being developed a duplicate of a previous modification, or revision or implementation of the same modification?	☒ Yes	☐ No	
Was a review that meets the criteria contained in the Technical Review procedure previously conducted?	☒ Yes	☐ No	
Were the results of that review incorporated into the documents under review?	☒ Yes	☐ No	☐ N/A
Do the USQ, SPHR, EPHA, and/or EEC, if required, remain valid? If YES, for a prior-use USQS or USQE exclusion, both a USQ process qualified individual and a USQ qualified reviewer are required to sign below.	☒ Yes	☐ No	☐ N/A

Section 2.0 - Review of Previous Work

This proposed Activity was addressed in Technical Review Package TRC-WD-2007-01563

Identify applicable previous reviews and document references.

WSRC-SA-2003-00001, REV. 5 WSRC-IM-94-10 SECTION 4.0, REV. 101 S-TSR-Z-00002, REV. 5 SF-CRF-070001 SF-CRF-080001

Identify any Minor Differences Between Previous Modification and Current Modification, or between this revision/implementation and the documents original modification documents

THE SCOPE OF THE PA IS IDENTICAL. THIS PA ADDS AN ENGINEERING CALCULATION WHICH FURTHER SUPPORTS THE CONCLUSIONS OF THE ORIGINAL REVIEW. SINCE THE PREVIOUS TECH REVIEW, WSRC-IM-94-10 SECTION 4 HAS BEEN RAISED TO REVISION 102 TO INCLUDE SF-CRF-080001 THAT ALLOWS INSTALLATION OF FILTERS IN THE VAULT ROOF PASSIVE VENTS. THE PA HAVE NO AFFECT ON THIS CHANGE TO THE AB AND THE PREVIOUS TECH REVIEW BASIS AND CONCLUSIONS REMAIN VALID.

Conclusion

THE PRESENCE OF WEEPING ON THE VAULT 4 WALLS DOES NOT IMPACT THE CONCLUSIONS OF THE EXISTING SALTSTONE PA/SA. THE OUTCOME OF THE RESULTS OF THE UDQE ARE NOT CHANGED BY THIS PA.

Section 3.0 - Preparer

Preparer (Date)
AARON STAUB 2008/04/03

Design Authority (Date)
AARON STAUB 2008/04/03

Security Classification
UNCLASSIFIED

Contributing Reviewers

NA

Reviewer	Date
AARON STAUB	2008/04/03
CHRISTOPHER BEHELER	2008/04/03

ATTACHMENT 2

Calculation N-CLC-Z-00020 Revision 0

OSR 45-24 (Rev 9-12-2007)

Calculation Cover Sheet

Project/Task Saltstone	Calculation No. N-CLC-Z-00020	Project/Task No. N/A
Title Saltstone Vault 4 Weeping Radionuclide Screening	Functional Classification GS	Sheet 1 of 15
	Discipline Nuclear	
Calc Type ☒ Type 1 ☐ Type 2	Type 1 Calc Status ☐ Preliminary ☒ Confirmed	
Computer Program No. Microsoft Excel ☐ N/A	Version/Release No. Microsoft Excel 2002 SP3	

Purpose and Objective

The purpose of this calculation is to determine which, if any, radionuclides might pose a significant risk if released into the soil surrounding Saltstone Vault 4 from within the Vault 4 walls for existing Vault 4 weeping. The list of radionuclides that might pose a significant risk are determined by applying a groundwater pathway screening factor based on the NCRP-123 groundwater pathway screening methodology. Radionuclides with a screening dose of less than 2.5 mrem/yr do not have a significant risk and would be removed from further consideration based on the conservative screening methodology employed.

DC/RO Date

UNCLASSIFIED
DOES NOT CONTAIN
UNCLASSIFIED CONTROLLED
NUCLEAR INFORMATION
DC & Reviewing Official: David K. Cheng
Date: April 2, 2008

Summary of Conclusion

Based on the initial and additional radionuclide screenings performed in this document, there is no significant risk associated with the radionuclides that might be released into the soil surrounding the vaults from within the Saltstone vault walls for Saltstone Vault 4 weeping. This risk assessment is based on a conservative risk calculation methodology and a reasonable water ingestion screening dose of less than 2.5 mrem/yr.

TRC-WD-2008-00451

Revisions

Rev No.	Revision Description
0	Original Issue

Sign Off

Rev No.	Originator (Print) Sign/Date	Verification/Checking Method (See Form Instructions)	Verifier/Checker (Print) Sign/Date	Manager (Print) Sign//Date
0	Mark H. Layton Mark H Layton 4/2/08	Document Review	Kent H. Rosenberger Kent H. Rosenberger 4/2/2008	Tom Robinson T. A. Robinson 4/2/2008

Additional Reviewer (Print)	Signature	Date
Design Authority (Print) AARON STAUB	Signature	Date 4/3/08
Release to Outside Agency (Print)	Signature	Date

Security Classification of the Calculation

Table of Contents

1.0 Calculation Purpose

The purpose of this calculation is to determine which, if any, radionuclides might pose a significant risk if released into the soil surrounding Saltstone Vault 4 from within the Vault 4 walls for existing Vault 4 weeping. The list of radionuclides that might pose a significant risk are determined by applying a groundwater pathway screening factor based on the NCRP-123 groundwater pathway screening methodology. Radionuclides with a screening dose of less than 2.5 mrem/yr do not have a significant risk and would be removed from further consideration based on the conservative screening methodology employed.

2.0 Open Items

None

3.0 References

1. NCRP-123, *Screening Models for Releases of Radionuclides to Atmosphere, Surface Water and Ground,* NCRP Report 123, Volumes I and II, January 1996.
2. WSRC-STI-2007-00184, Phifer, M. A., *FTF Closure Cap Concept and Infiltration Estimates*, Savannah River National Laboratory, Aiken, S.C., Revision 2, October 15, 2007.
3. CBU-PIT-2005-00131, *Response To Request for Additional Information on the Draft Section 3116 Determination For Salt Waste Disposal at the Savannah River Site*, Revision 1, July 14, 2005.
4. X-ESR-H-00123, *Evaluation of DDA Batch 3 material in Tank 50 for Transfer to Saltstone*, Revision 0, January 31, 2008.
5. WSRC-STI-2006-00198, Phifer, M. A., Millings, M. R. and Flach, G. P., *Hydraulic Property Estimation for the E-Area and Z-Area Vadose Zone Soils, Cementitious Materials, and Waste Zones*, Washington Savannah River Company, Aiken, SC, Revision 0, September 2006.
6. WSRC-TR-2006-00004, Kaplan, D. I, *Geochemical Data Package for Performance Assessment Calculations Related to the Savannah River Site (U)*, Washington Savannah River Company, Aiken, SC, Revision 0, February 28, 2006.
7. ISSN 0146-6453, International Commission on Radiological Protection (ICRP) Publication 72, Annals of the ICRP, (1996), *Age-dependent Doses to Members of the Public form Intake of Radionuclides: Part 5 Compilation of Ingestion and Inhalation Dose Coefficients*, Volume 26 No. 1, March 1996.
8. WSRC-TR-2005-00131, Hiergesell, R.A., *Saltstone Disposal Facility: Determination of the probable maximum water table elevation*, Revision 0, June 28, 2005.
9. W828992, *Savannah River Plant DWPF – 200Z Area Saltstone Vaults 6 and 7 plan, Sections and Details Concrete*, Revision B1.
10. WSRC-TR-2008-00001, *Z-Area Groundwater Monitoring Report for 2007 (U)*, Revision 0, January 9, 2008.
11. Tuli, J. K. *Nuclear Wallet Cards*, 7th Edition, National Nuclear Data Center Brookhaven National Laboratory, April 2005.

4.0 Inputs and Assumptions

4.1 Inputs

Input	Parameter	Value	Reference
1	Soil Infiltration Rate (no closure cap present)	16.45 inches/yr (41.78 cm/yr)	WSRC-STI-2007-00184, Rev. 2, Table 1 page 2.
2	Water table darcy average velocity	20.82 ft/yr (0.057 ft/d)	CBU-PIT-2005-00131, Rev. 1, Table 47-1.
3	Lower vadose zone porosity value	0.39 (unitless)	WSRC-STI-2006-00198, Rev. 0, Table 5-9.
4	Lower vadose zone average dry bulk density	1.62 g/mL	WSRC-STI-2006-00198, Rev. 0, Table 5-9.
5	Saturation value	0.72 (unitless)	WSRC-STI-2006-00198, Rev. 0, Page 141.
6	Saltstone Vault 4 Radionuclide concentrations	See Table 1	X-ESR-H-00123, Rev. 0, pages 7 and 9.
7	Internal Ingestion DCFs	See Table 1	ISSN 0146-6453 (ICRP Publication 72).
8	Lower vadose zone distribution coefficients	See Table 1	WSRC-TR-2006-00004, Rev. 0, Table 10.
9	Maximum water table elevation near Vault 4	246 ft	WSRC-TR-2005-00131, Rev. 0, Figure 8, page 14.
10	Saltstone Vault 4 bottom elevation	269 ft	W82992, Rev. B1.
11	Annual consumption rate of drinking water	800 L/yr	NCRP-123, page 72.
12	Thickness of soil layer between Vault and water table	700 cm	Assumption 2.
13	Cs-137 half life	30.3 years	Tuli, *Nuclear Wallet Cards.*
14	Sr-90 half life	28.8 years	Tuli, *Nuclear Wallet Cards.*
15	Pu-238 half life	87.8 years	Tuli, *Nuclear Wallet Cards.*

Table 1 – Saltstone Vault Inputs by Radionuclide

	Vault 4 Radionuclide Concentrations pCi/mL	Vault 4 Radionuclide Concentrations Ci/L	Internal Ingestion DCFs (rem/µCi)	Internal Ingestion DCFs (mrem/Ci)	Kd Values (mL/g)
H3	1.24E+03	1.24E-06	6.66E-05	6.66E+04	0
C14	1.35E+03	1.35E-06	2.15E-03	2.15E+06	0
Ni63	7.86E+01	7.86E-08	5.55E-04	5.55E+05	7
Sr90	6.90E+04	6.90E-05	1.04E-01	1.04E+08	5
Tc99	1.84E+04	1.84E-05	2.37E-03	2.37E+06	0.1
I129	3.86E+00	3.86E-09	4.07E-01	4.07E+08	0
Cs137	3.72E+07	3.72E-02	4.81E-02	4.81E+07	50
Th230	3.27E+03	3.27E-06	7.77E-01	7.77E+08	900
Th232	1.70E-02	1.70E-11	8.51E-01	8.51E+08	900
U232	4.41E-01	4.41E-10	1.22E+00	1.22E+09	200
U233	4.69E+01	4.69E-08	1.89E-01	1.89E+08	200
U234	1.38E+02	1.38E-07	1.81E-01	1.81E+08	200
U235	2.45E-01	2.45E-10	1.74E-01	1.74E+08	200
U236	6.37E+00	6.37E-09	1.74E-01	1.74E+08	200
U238	3.19E+00	3.19E-09	1.67E-01	1.67E+08	200
Np237	1.56E+01	1.56E-08	4.07E-01	4.07E+08	0.6
Pu238	7.30E+04	7.30E-05	8.51E-01	8.51E+08	270
Pu239	4.67E+03	4.67E-06	9.25E-01	9.25E+08	270
Pu240	4.67E+03	4.67E-06	9.25E-01	9.25E+08	270
Pu241	3.42E+03	3.42E-06	1.78E-02	1.78E+07	270
Pu242	8.45E+01	8.45E-08	8.88E-01	8.88E+08	270
Pu244	3.92E-01	3.92E-10	8.88E-01	8.88E+08	270
Am241	9.44E+03	9.44E-06	7.40E-01	7.40E+08	1100
Am242m	5.07E+00	5.07E-09	7.03E-01	7.03E+08	1100
Am243	1.35E+02	1.35E-07	7.40E-01	7.40E+08	1100
Cm242	4.20E+00	4.20E-09	4.44E-02	4.44E+07	1100
Cm244	2.25E+04	2.25E-05	4.44E-01	4.44E+08	1100
Cm245	7.16E+01	7.16E-08	7.77E-01	7.77E+08	1100

4.2 Assumptions

Assumption 1: The total radionuclide inventory released into the soil surrounding the Saltstone Vault 4 from within the Vault walls is less than or equal to 1000 liters of undiluted salt solution.
Assumption 1 Basis: The initial inventory used in the screening is 1000 liters of salt solution situated directly below the saltstone vault. This inventory is assumed to bound the quantity of salt solution that could be contained in the vault walls and located in the soil right outside the vault.
Assumption 1 Conservatisms: Based on Vault 4 visual inspections, the current Vault 4 weep locations are less than 100 linear feet on the vault surface. Conservatively assuming weep locations are represented by cracks that run the full 18 inch wall depth and are 0.01 inches in width, the crack volume for the entire Vault 4 would be less than 4 liters. The 0.01 inches accounts for the actual crack width and any migration into the concrete, assuming a concrete hydraulic conductivity of 1.0E-10 cm/sec (3.2E-3 cm/yr). The soil surveys around Vault 4 do not indicate that liter volumes of vault contents were released to the surrounding soil. Assuming that 1000 liters inventory is released into the soil surrounding Saltstone Vault 4 is therefore conservative by greater than an order of magnitude. The assumption of 1000 liters of undiluted salt solution also ignores any retardation within the Vault 4 walls.

Assumption 2: The minimum travel distance from Saltstone Vault 4 to the top of the water table is 23 feet (7.0 meters).
Assumption 2 Basis: The minimum travel distance from Vault 4 to the top of the water table is assumed to be bound by the distance based on the maximum water table elevation near Vault 4 of 246 ft and the Vault 4 bottom elevation of 269 ft.
Assumption 2 Conservatisms: The maximum water table elevation used near Vault 4 (246 ft) used is higher than the water table elevation of 220 ft recorded during 2007 well tests (WSRC-TR-2008-00001), which would indicate that there is potentially an additional 26 feet of vadose soil between the Vault 4 bottom and the top of the water table that is not utilized in this calculation.

Assumption 3: A dilution water volume of 4.48E+07 liters is assumed.
Assumption 3 Basis: The exposure scenario from NCRP-123 assumes the entire waste inventory is susceptible to leaching over a period of one year into a dilution water volume. The dilution water volume of 4.48E+07 liters is based on a mixing cell defined by the vault length (600 ft from ref. W828992), vault width (200 ft from ref. W828992), and a 13.2 ft mixing cell depth (See Figure 1). The 13.2 ft mixing cell depth was determined assuming the contaminant volume transported downward through the soil (defined by the 600 ft length times the 200 ft width times the infiltration rate darcy velocity of 16.45 in/yr) is equal to the contaminant volume transported laterally through the water table (defined by the 600 ft length times the unknown depth times the water table darcy velocity of 20.82 ft/yr). The two equations are used to solve for the unknown depth: (600 ft)*(200 ft)*(16.45 in/yr) = (600 ft)*(mixing depth in ft)*(20.82 ft/yr).

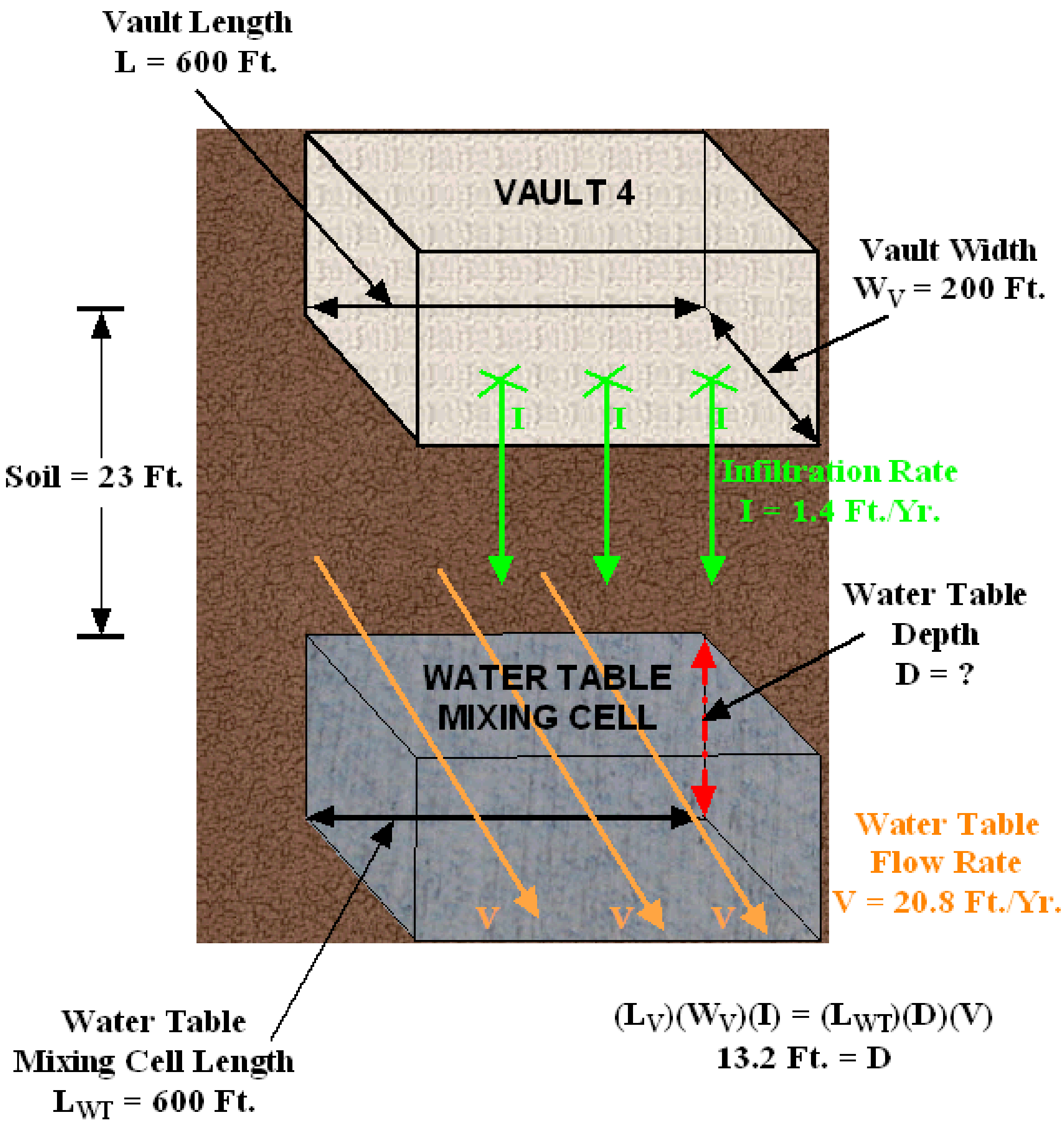
Assumption 3 Conservatisms: The assumed 13.2 ft mixing cell depth is less than the well screen length (approximately 20 ft) that would be used to extract the ingestion water from the aquifer. No additional downstream mixing or water dilution (due to the well screen) was assumed beyond the mixing provided by the 13.2 ft mixing cell depth.

Assumption 4: Radionuclides with a total water ingestion pathway screening dose of less than 2.5 mrem/yr are not considered to pose a significant risk.
Assumption 4 Basis: Past PA modeling has shown the water ingestion pathway to be dominant over other pathways. The 2.5 mrem/yr screening limit is $1/10^{th}$ of the All-Pathways annual performance objective of 25 mrem/yr adopted for SRS PAs, which is based on the DOE 435-1 and 10CFR61 All-Pathways performance objectives. Setting the limit at $1/10^{th}$ of the All-Pathways annual performance objective is sufficient to account for other lesser pathways (e.g., vegetable ingestion, milk ingestion) and unquantified daughter products given the gross conservatisms inherent in the screening methodology.
Assumption 4 Conservatisms: None

Assumption 5: The soil underneath Saltstone Vault 4 has the properties of SRS lower vadose zone soil.
Assumption 5 Basis: The vadose zone properties are based on a geotechnical report on Z-Area hydraulic properties. The soil under the vaults is relatively undisturbed and is characterized as vadose zone soil, in contrast to “backfill soil” which will be used during closure cap installation.
Assumption 5 Conservatisms: None

Assumption 6: The risk contribution of daughter products is not quantified.
Assumption 6 Basis: Based on past groundwater modeling experience, in growth of daughter products during transport is not expected to cause a significant increase in any of the initial inventories nor add an additional radionuclide that would pose an independent risk. The daughter product inventories are by their nature a percentage of the parent inventories. A review of the list of the decay chains and potential daughter products associated with the Saltstone Vault 4 inventory resulted in identification of two decay chains of concern (Am-241 to Np-237 and Th-230 to Ra-226). The daughters in these chains will be produced in measurable quantities within 10,000 years, have appreciable DCFs, and have distribution coefficients smaller than their parents (0.6 mL/g for Np-237 vs. 1100 mL/g for Am-241 and 5 mL/g for Ra-226 vs. 900 mL/g for Th-230). Np-237 production from Am-241 decay is not a concern because even assuming all the Am-241 in the initial inventory (9.44E-3 Ci or 2.75 E-3 g) decays to Np-237, an insignificant additional amount of Np-237 (1.94E-6 Ci vs. the initial 1.56E-05 Ci inventory) would be produced. The production of Ra-226 from Th-230 is not a concern because while the Ra-226 has a higher DCF than Th-230 and will travel through the soil more quickly (due to its low K_d value), the Th-230 parent has a large half-life (75,400 years) which will cause the Ra-226 to be produced very slowly over time. In addition, the low K_d value will result in the Ra-226 that is produced being dispersed, with the Ra-226 dose contribution spread out over time.
Assumption 6 Conservatisms: None

Figure 1 – Saltstone Vault 4 and Water Table Mixing Cell



5.0 Analytical Methods and Computations

5.1 Groundwater Pathway Screening Factor Determination

The groundwater pathway screening factor is based on formula 8.21 from NCRP-123 Section 8.2.3.2. The groundwater screening factor (SF_{gw}) will be calculated for the groundwater ingestion pathway as follows:

$$SF_{gw} = \lambda_L * A_0 * (U_{dw}/V) \sum X_i (DF_{ing})_i$$

where:

λ_L = Leach rate of the parent nuclide (yr^{-1})
A_0 = Factor to account for parent decay and daughter ingrowth (dimensionless)
U_{dw} = Consumption rate of drinking water (L/yr)
V = dilution volume (L/yr)
X_i = parent inventory assuming no ingrowth (Ci)
DF_{ing} = ingestion dose factor for parent (mrem/Ci)

With X_i = parent inventory concentration (Ci/L) * initial volume (L)

The leach rate of the parent nuclide is based on formula 4.3 from NCRP-123 Section 4.2.2:

$$\lambda_L = I / (R * H * n)$$

where:

I = groundwater infiltration rate (cm/yr)
R = Retardation factor (dimensionless)
H = thickness of soil layer (cm)
n = soil porosity (dimensionless)

With $R = 1 + (\rho_b * k_d) / n * S$

where:

ρ_b = soil density (g/mL)
k_d = soil partition (distribution) coefficient (mL/g)
S = saturation (dimensionless)

The screening dose can be calculated by multiplying the radionuclide inventory by the groundwater screening factor and totaling the contributions from the individual radionuclides. Radionuclides with a screening dose of less than 2.5 mrem/yr would be removed from further consideration. The 2.5 mrem/yr screening limit is discussed further in Assumption 4 of Section 4.2.

5.2 Initial Groundwater Pathway Screening Factor Determination

The initial groundwater pathway screening factor will not take into consideration effect of decay on inventory. The entire 1000 liters of salt solution will conservatively be assumed to travel to the water table where it is available for ingestion. This approach removes temporal considerations from the screening. The groundwater screening factor (SF_{gw}) is therefore calculated and summed for each radionuclide as follows:

$$SF_{gw} = \lambda_L * A_0 * (U_{dw}/V) \sum X_i \, (DF_{ing})_i$$

where:

λ_L = Leach rate of the parent nuclide (yr^{-1}) = I / (R * H * n)
A_0 = Factor to account for parent decay and daughter ingrowth (dimensionless) = 1 (no decay or ingrowth assumed in initial screening)
U_{dw} = Consumption rate of drinking water (L/yr) = 800 L/yr
V = dilution volume (L/yr) = 4.48E+07 L/yr
X_i = parent inventory assuming no ingrowth (Ci/L) = concentration * 1000L
DF_{ing} = ingestion dose factor for parent (mrem/Ci) = variable
I = the groundwater infiltration rate (cm/yr) = 41.78 cm/yr
R = Retardation factor (dimensionless) = R = 1 + (ρ_b*k_d) / n * S
H = thickness of soil layer (cm) = 700 cm
n = soil porosity (dimensionless) = 0.39
S = saturation (dimensionless) = 0.72
ρ_b = the soil density (g/mL) = 1.62 g/mL
k_d = the soil partition (distribution) coefficient (mL/g) = based on radionuclide

The Leach rate (λ_L) calculations for each radionuclide are shown in Table 2. The groundwater screening factor (SF_{gw}) calculations for each radionuclide and the total groundwater screening factor are shown in Table 3. The initial groundwater pathway screening factor of 17.8 mrem/yr exceeds the 2.5 mrem/yr screening limit. As can be seen from Table 3, one radionuclide (Cs-137) exceeds the 2.5 mrem/yr screening limit individually and two other radionuclides (Pu-238 and Sr-90) have contributions greater than 0.1 mrem/yr. Based on these results and the fact that these three radionuclides all have relatively short half lives, further screening is warranted for Cs-137, Sr-90 and Pu-238.

Table 2 – Leach Rate Calculation by Radionuclide

	infiltration rate	porosity	saturation	pore velocity	bulk density	distribution coefficient	retardation factor	solute velocity	soil thickness	Leach Rate
	cm/yr	unitless	unitless	cm/yr	g/mL	mL/g	unitless	cm/yr	cm	yr-1
	I	n	S	v	ρ_b	K_d	R	v_R	H	λ_L
H3	41.78	0.39	0.72	148.80	1.62	0	1.00	148.80	700	1.53E-01
C14	41.78	0.39	0.72	148.80	1.62	0	1.00	148.80	700	1.53E-01
Ni63	41.78	0.39	0.72	148.80	1.62	7	41.38	3.60	700	3.70E-03
Sr90	41.78	0.39	0.72	148.80	1.62	5	29.85	4.99	700	5.13E-03
Tc99	41.78	0.39	0.72	148.80	1.62	0.1	1.58	94.36	700	9.71E-02
I129	41.78	0.39	0.72	148.80	1.62	0	1.00	148.80	700	1.53E-01
Cs137	41.78	0.39	0.72	148.80	1.62	50	289.46	0.51	700	5.29E-04
Th230	41.78	0.39	0.72	148.80	1.62	900	5193.31	0.03	700	2.95E-05
Th232	41.78	0.39	0.72	148.80	1.62	900	5193.31	0.03	700	2.95E-05
U232	41.78	0.39	0.72	148.80	1.62	200	1154.85	0.13	700	1.33E-04
U233	41.78	0.39	0.72	148.80	1.62	200	1154.85	0.13	700	1.33E-04
U234	41.78	0.39	0.72	148.80	1.62	200	1154.85	0.13	700	1.33E-04
U235	41.78	0.39	0.72	148.80	1.62	200	1154.85	0.13	700	1.33E-04
U236	41.78	0.39	0.72	148.80	1.62	200	1154.85	0.13	700	1.33E-04
U238	41.78	0.39	0.72	148.80	1.62	200	1154.85	0.13	700	1.33E-04
Np237	41.78	0.39	0.72	148.80	1.62	0.6	4.46	33.35	700	3.43E-02
Pu238	41.78	0.39	0.72	148.80	1.62	270	1558.69	0.10	700	9.82E-05
Pu239	41.78	0.39	0.72	148.80	1.62	270	1558.69	0.10	700	9.82E-05
Pu240	41.78	0.39	0.72	148.80	1.62	270	1558.69	0.10	700	9.82E-05
Pu241	41.78	0.39	0.72	148.80	1.62	270	1558.69	0.10	700	9.82E-05
Pu242	41.78	0.39	0.72	148.80	1.62	270	1558.69	0.10	700	9.82E-05
Pu244	41.78	0.39	0.72	148.80	1.62	270	1558.69	0.10	700	9.82E-05
Am241	41.78	0.39	0.72	148.80	1.62	1100	6347.15	0.02	700	2.41E-05
Am242m	41.78	0.39	0.72	148.80	1.62	1100	6347.15	0.02	700	2.41E-05
Am243	41.78	0.39	0.72	148.80	1.62	1100	6347.15	0.02	700	2.41E-05
Cm242	41.78	0.39	0.72	148.80	1.62	1100	6347.15	0.02	700	2.41E-05
Cm244	41.78	0.39	0.72	148.80	1.62	1100	6347.15	0.02	700	2.41E-05
Cm245	41.78	0.39	0.72	148.80	1.62	1100	6347.15	0.02	700	2.41E-05

Table 3 –Initial Screening Factor Calculation by Radionuclide

	Inventory Concentration	DCFs	Annual Water Ingestion U_{dw}	Dilution Water Volume V	Inventory Volume	Inventory Curies	Leach Rate	SF_{gw}	
	Ci/L	mrem/Ci	L/yr	L/yr	L	Ci	λ_L	mrem	> 2.5 mrem
H3	1.24E-06	6.66E+04	800	4.48E+07	1.00E+03	1.24E-03	1.53E-01	2.26E-04	
C14	1.35E-06	2.15E+06	800	4.48E+07	1.00E+03	1.35E-03	1.53E-01	7.94E-03	
Ni63	7.86E-08	5.55E+05	800	4.48E+07	1.00E+03	7.86E-05	3.70E-03	2.88E-06	
Sr90 (initial)	6.90E-05	1.04E+08	800	4.48E+07	1.00E+03	6.90E-02	5.13E-03	6.58E-01	
Tc99	1.84E-05	2.37E+06	800	4.48E+07	1.00E+03	1.84E-02	9.71E-02	7.57E-02	
I129	3.86E-09	4.07E+08	800	4.48E+07	1.00E+03	3.86E-06	1.53E-01	4.30E-03	
Cs137 (initial)	3.72E-02	4.81E+07	800	4.48E+07	1.00E+03	3.72E+01	5.29E-04	1.69E+01	Yes
Th230	3.27E-06	7.77E+08	800	4.48E+07	1.00E+03	3.27E-03	2.95E-05	1.34E-03	
Th232	1.70E-11	8.51E+08	800	4.48E+07	1.00E+03	1.70E-08	2.95E-05	7.62E-09	
U232	4.41E-10	1.22E+09	800	4.48E+07	1.00E+03	4.41E-07	1.33E-04	1.27E-06	
U233	4.69E-08	1.89E+08	800	4.48E+07	1.00E+03	4.69E-05	1.33E-04	2.10E-05	
U234	1.38E-07	1.81E+08	800	4.48E+07	1.00E+03	1.38E-04	1.33E-04	5.92E-05	
U235	2.45E-10	1.74E+08	800	4.48E+07	1.00E+03	2.45E-07	1.33E-04	1.01E-07	
U236	6.37E-09	1.74E+08	800	4.48E+07	1.00E+03	6.37E-06	1.33E-04	2.63E-06	
U238	3.19E-09	1.67E+08	800	4.48E+07	1.00E+03	3.19E-06	1.33E-04	1.26E-06	
Np237	1.56E-08	4.07E+08	800	4.48E+07	1.00E+03	1.56E-05	3.43E-02	3.89E-03	
Pu238	7.30E-05	8.51E+08	800	4.48E+07	1.00E+03	7.30E-02	9.82E-05	1.09E-01	
Pu239	4.67E-06	9.25E+08	800	4.48E+07	1.00E+03	4.67E-03	9.82E-05	7.58E-03	
Pu240	4.67E-06	9.25E+08	800	4.48E+07	1.00E+03	4.67E-03	9.82E-05	7.58E-03	
Pu241	3.42E-06	1.78E+07	800	4.48E+07	1.00E+03	3.42E-03	9.82E-05	1.07E-04	
Pu242	8.45E-08	8.88E+08	800	4.48E+07	1.00E+03	8.45E-05	9.82E-05	1.32E-04	
Pu244	3.92E-10	8.88E+08	800	4.48E+07	1.00E+03	3.92E-07	9.82E-05	6.11E-07	
Am241	9.44E-06	7.40E+08	800	4.48E+07	1.00E+03	9.44E-03	2.41E-05	3.01E-03	
Am242m	5.07E-09	7.03E+08	800	4.48E+07	1.00E+03	5.07E-06	2.41E-05	1.54E-06	
Am243	1.35E-07	7.40E+08	800	4.48E+07	1.00E+03	1.35E-04	2.41E-05	4.31E-05	
Cm242	4.20E-09	4.44E+07	800	4.48E+07	1.00E+03	4.20E-06	2.41E-05	8.04E-08	
Cm244	2.25E-05	4.44E+08	800	4.48E+07	1.00E+03	2.25E-02	2.41E-05	4.31E-03	
Cm245	7.16E-08	7.77E+08	800	4.48E+07	1.00E+03	7.16E-05	2.41E-05	2.40E-05	
Initial SF_{gw}	NA	NA	NA	NA	NA	NA	NA	17.8	Yes
Cs137 (decayed)	NA	4.81E+07	800	4.48E+07	NA	1.20E-06	5.29E-04	5.46E-07	
Sr90 (decayed)	NA	1.04E+08	800	4.48E+07	NA	2.40E-03	5.13E-03	2.29E-02	
Pu238 (decayed)	NA	8.51E+08	800	4.48E+07	NA	2.70E-05	9.82E-05	4.03E-05	
Final SF_{gw} with decay	NA	NA	NA	NA	NA	NA	NA	0.14	No

5.3 Further Evaluation of Cs-137, Sr-90 and Pu-238

The only radionuclides determined in Section 5.2 as requiring further investigation are Cs-137, Sr-90 and Pu-238. These radionuclides have relatively short half lives (Cs-137 half life is 30.03 years, Sr-90 half life is 28.8 years, Pu-238 half life is 87.8 years), which means ignoring the fraction of the parent decayed during transport (in the term A_0) during the initial screening is overly conservative. The amount of time it would take for these radionuclides to travel the 7 meter distance from the vault to the water table can be calculated assuming the infiltration rate through the soil (assuming no closure cap) is driving the radionuclide transport. The solute (retarded) velocity of the radionuclides can be solved by the equation

$$v_R = v/R$$

where:

- v_R = the solute (retarded) velocity (cm/yr)
- v = the pore velocity (cm/yr) = U/(nS)
- R = Retardation factor (dimensionless) = R = 1 + (ρ_b*k_d) / n
- U = the darcy velocity (cm/yr) = I
- I = the groundwater infiltration rate (cm/yr) = 41.78 cm/yr
- ρ_b = the soil density (g/mL) = 1.62 g/mL
- k_d = the soil partition (distribution) coefficient (mL/g) = based on radionuclide
- n = soil porosity (dimensionless) = 0.39

Solving for the solute (retarded) velocity gives values of v_R = 0.514 cm/yr (0.0169 ft/yr) for Cs-137, v_R = 4.986 cm/yr (0.1636 ft/yr) for Sr-90, and v_R = 0.095 cm/yr (0.0031 ft/yr) for Pu-238. Sr-90 has a higher travel transport velocity because it has a significantly lower distribution coefficient in soil (5 mL/g vs. 50 mL/g).

5.3.1 Decayed Inventory Determination for Cs-137

Given a v_R = 0.514 cm/yr for Cs-137, it will take 1364 years for the Cs-137 to travel the 7 meters from Vault 4 to the water table. Given a Cs-137 half life of 30.03 years, the initial Cs-137 inventory of 37.2 curies will be decayed to 1.2E-06 curies after only 750 years (less than the 1364 years it will take to travel 7 meters).

5.3.2 Inventory Decay Time Determination for Sr-90

Given a v_R = 4.986 cm/yr for Sr-90, it will take 140 years for the Sr-90 to travel the 7 meters from Vault 4 to the water table. Given a Sr-90 half life of 28.8 years, the initial Sr-90 inventory of 0.069 curies will be decayed to 0.0024 curies after 140 years (the time it will take to travel 7 meters).

5.3.3 Inventory Decay Time Determination for Pu-238

Given a v_R = 0.095 cm/yr for Pu-238, it will take 7343 years for the Pu-238 to travel the 7 meters from Vault 4 to the water table. Given a Pu-238 half life of 87.8 years, the initial Pu-238 inventory of 0.073 curies will be decayed to 2.7E-05 curies after 1000 years (less than one seventh of the time it will take to travel 7 meters).

5.3.4 Groundwater Pathway Screening with decayed Cs-137, Sr-90 & Pu-238

The Cs-137, Sr-90, and Pu-238 inventories after decay time consideration (1.2E-06 curies of Cs-137, 0.0024 curies of Sr-90, and 2.7E-05 curies of Pu-238) can be substituted back into the groundwater pathway screening process described in Section 5.2 to determine a groundwater screening factor (SF_{gw}) for Cs-137 and Sr-90, and Pu-238 with the decayed inventories. The results of this final screening for Cs-137, Sr-90, and Pu-238 are shown in the last four rows of Table 3, with the results being that groundwater screening factor (SF_{gw}) of 0.14 mrem/yr no longer exceeds the 2.5 mrem/yr screening limit.

5.4 Results

Conservatively assuming 1000 liters of radionuclide inventory is released into the soil surrounding the Saltstone Vault 4 (from within the Vault walls), the groundwater screening factor (SF_{gw}) of 0.14 mrem/yr calculated for this inventory does not exceed the 2.5 mrem/yr screening limit.

6.0 Conclusion

Based on the initial and additional radionuclide screenings performed in this document, there is no significant risk associated with the radionuclides that might be released into the soil surrounding Saltstone Vault 4 from within the vault walls for Saltstone Vault 4 weeping. This risk assessment is based on a conservative risk calculation methodology and a reasonable water ingestion screening dose of less than 2.5 mrem/yr.

7.0 Attachments and Appendices

None.