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Discovery of Apparent Inconsistency Between Saltstone Waste Acceptance Criteria And The Saltstone Performance Assessment

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Summary

It was recently discovered that the Saltstone WAC contained no radionuclide disposal limits derived from the PA. This evaluation concludes that such limits are not needed in the WAC, so long as the waste being processed and disposed in Saltstone arises from the In-Tank Precipitation Process (ITP) or the Effluent Treatment Process (ETF) and does not exceed the total radionuclide inventory analyzed in the PA for radionuclides significant to PA results. All of the waste processed and emplaced in the Saltstone vaults to date arose from ITP and ETF and did not exceed the PA inventory of significant radionuclides; thus, the waste processed and disposed to date is bounded by the PA. The current contents of Tank 50 also arose from ITP and ETF and, if treated according to the PA and emplaced in the Saltstone vaults, will be bounded by the PA.

Introduction

One intent of DOE Order 435.1[1], as expressed in the performance assessment/composite analysis guidance[2], is to ensure that proposed or discovered changes in wasteforms, containers, radionuclide inventories, facility design, and operations are reviewed to ensure that the assumptions, results, and conclusions of the DOE approved performance assessment[3] (PA), and composite analysis[4] (CA), as well as any Special Analyses (SA) that might have been performed, remain valid (i.e., that the proposed change is bounded by the PA and CA) and the changes are within the bounds of the Disposal Authorization Statement[5]. The goal is to provide flexibility in day-to-day operation 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 discovered condition (apparent inconsistency between the Saltstone Waste Acceptance Criteria (WAC) and the Saltstone PA) is within the assumptions, parameters, and bases of the approved PA[3] and CA[4]. If it is, then this document serves as the technical basis for authorizing the condition. If not, then, in order to authorize the activity, 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 Discovered Condition

While preparing to update the Saltstone WAC to accommodate new pre-treatments of the waste (e.g., Low-Curie Salt), it was discovered that the WAC[6] contained no limits derived from the PA.

Background

The Saltstone facility, located in Z-Area, contains a wastewater treatment facility to mix salt waste from the high-level waste tank farms and the Effluent Treatment Facility (ETF) with cementitious solids to form a grout and a disposal facility to receive the grout into large concrete vaults for disposal as low-level waste (LLW). The Saltstone facility was designed to treat and dispose of decontaminated salt solution produced by the in-tank precipitation process (ITP) and aqueous residues from the ETF[3]. The Saltstone facility began processing radioactive waste in June 1990. Operations continued through August 1998. A total of 2.6 million gallons of salt solution was processed[7]. At that time, the facility was put in standby mode because of difficulties encountered with ITP. Subsequently, it was decided that ITP was unacceptable and other

processes were considered. Eventually, an Environmental Impact Statement on Salt Processing Alternatives was published[8]. The Record of Decision (ROD)[9], indicated that Caustic Side Solvent Extraction was the preferred technology for salt decontamination. However, the ROD also indicated that some salt waste may be acceptable for disposal in Saltstone with little pre-treatment.

The performance assessment for the Saltstone disposal facility[3] analyzed the potential impact to public health and the environment from the disposal of the entire amount of ITP salt solution expected. The PA assumed that a total of 7.3×10^8 liters of mixed wastewater (i.e., the "nominal blend" of ITP and ETF wastes) containing the average concentration of radionuclides as listed in Table 2.6-2 of reference 3 would be disposed in Z-Area. The total radionuclide inventory of the radionuclides considered in the PA is also stated in Table 2.6-2 of reference 3. The PA concluded that none of the DOE performance objectives would be exceeded.

DOE Headquarters reviewed the PA. The review resulted in additional information being developed, which was documented in an addendum to the PA[10]. The additional information resulted in modification to the PA results. The revised results also meet all the performance objectives. Therefore, processing and disposal of all of the projected ITP salt solution, blended with ETF waste, will meet all DOE requirements for protection of public health and the environment. Thus, the Waste Acceptance Criteria (WAC) for Saltstone, from the perspective of meeting DOE LLW disposal performance objectives, need only restrict the source of the waste being processed and disposed to that arising from ITP and ETF. The PA analysis included one vault cell filled with waste from the Naval Fuels Facility; all of that waste has been emplaced in cell A of vault 4.

Waste Acceptance Criteria were needed to ensure that other objectives (e.g., Toxicity Characteristic Leaching Procedure (TCLP), operational health physics) were met. The WAC are documented in the SRS Waste Acceptance Criteria Manual 1S, Procedure WAC 4.01[6]. The original (i.e., Revision 0) WAC was published in December 1995. The radionuclide limits from Revision 2, issued June 19, 1998, are shown in Table 1.

Presently, High-Level Waste Tank 50, the tank from which salt solution is transferred to Z-Area, contains salt solution derived from ITP, which is mixed with salt solution from ETF. Thus, the material now in Tank 50 is the same as that analyzed in the PA.

SRS is now considering whether salt waste from selected HLW tanks can be transferred to Z-Area for processing and disposal as Saltstone with the only pretreatment being washing of the solid salt to remove much of the cesium. Such salt waste is termed "low-curie salt (LCS)" waste. Since the waste would not be processed through ITP prior to transfer to Z-Area, specific radionuclide limits derived from the PA are needed to ensure that performance objectives will not be compromised.

Limits for disposal of low-curie salt in Saltstone were developed from the PA results[11]. These limits can be used for salt solution arising from the LCS program or other programs. Subsequently, these limits were incorporated into Revision 3 of the Saltstone WAC. The radionuclide limits from Revision 3, issued February 25, 2002, are shown in Table 2.

When the Saltstone WAC were updated, it became apparent that Revision 2, as well as previous revisions, contained no limits derived from the PA. It was also apparent that the updated limits in Table 2 are much lower, for some radionuclides, than the corresponding values in Table 1. As a consequence, this Unreviewed Disposal Question (UDQ) evaluation was prepared to consider

whether Saltstone now in vaults one and four, as well as the current contents of Tank 50, are within the bounds of the Saltstone PA.

Supporting Analysis

As stated above, the Saltstone PA was performed to show that the total quantity of ITP salt solution, blended with ETF waste, would meet DOE performance objectives when disposed as Saltstone. Since the Saltstone PA showed that all performance objectives would be met, all of the Saltstone produced from ITP and ETF waste, which has been emplaced in vaults one and four, is within the bounds of the PA. Also, the salt solution presently in HLW Tank 50, which contains salt waste arising from ITP, is within the bounds of the PA. In support of this rationale, the following analysis is presented.

The total radionuclide inventory disposed in Z-Area to date and the volumes of salt solution processed are shown in Table 3 in comparison with the total radionuclide inventory and salt waste volume assumed in the PA[3]. Cell A of Vault 4 is not shown because that cell contains only drummed waste from the Naval Fuel Facility[3]. The disposed inventory of two of the radionuclides, $^{233/234}U$ and ^{238}U, exceed the total inventory assumed in the PA. These radionuclides, however, were screened from the PA analysis. Thus, their exceedance is inconsequential. The groundwater pathway screening methodology described in Section 3.2.3.4 of the PA[3] is very simple and conservative. The radionuclide concentration of saltstone, presented in Table 2.6-2 of the PA was decayed for 100 years, because of the presumed 100-year period of institutional control. The resulting radionuclide content was apportioned between the solid and liquid phases, assuming the saltstone to be saturated with liquid, according to a radionuclide distribution coefficient (i.e., K_d). Then, it was assumed that a person would drink 2 liters of the liquid phase each day for a year. If the resulting dose was less than 4 mrem/year, the groundwater protection performance objective assumed in the PA, the radionuclide was screened out as being a negligible contributor to the analysis. For the inventory already disposed of $^{233/234}U$ and ^{238}U, the dose calculated using the screening methodology is 0.9 mrem/year assuming the 0.49 curies reported as $^{233/234}U$ is either ^{233}U or ^{234}U and 0.3 mrem/year for the 0.0164 Ci of ^{238}U. Thus, the larger inventory disposed to date would also have screened out (i.e., the dose calculated in the screening analysis is less than the 4 mrem/year screening limit).

For five other radionuclides, ^{14}C, ^{59}Ni, ^{63}Ni, ^{237}Np, and $^{239/240}Pu$, the fraction of the PA inventory disposed to date exceeds the fraction of the PA volume disposed to date. All of these radionuclides were either screened from the PA analysis (i.e., ^{59}Ni and ^{237}Np) or were insignificant contributors to the total calculated PA impacts (i.e., ^{14}C, ^{63}Ni, $^{239/240}Pu$). Thus, their exceedance is inconsequential.

The average radionuclide concentration processed into each of the vault cells (i.e., total cell inventory divided by total salt solution volume processed into the cell), the average concentration assumed in the PA, and corresponding radionuclide limits from Saltstone WAC revisions 2 and 3 are shown in Table 4.

All of the average concentrations for ^{14}C shown in Table 4 exceed the average value assumed in the PA; however, none of the average concentrations for ^{14}C exceed the WAC limits. Three of the values shown for ^{59}Ni exceed the average concentration assumed in the PA; however, all of the ^{59}Ni values are derived from analytical results that are “less than” the value stated. Values for ^{59}Ni in Vault 1, cells B and C and cell G of Vault 4 exceed the WAC revision 3 limit. However, this limit was derived from the intruder screening analysis in the PA[11]. Since the PA screened ^{59}Ni as having no impact in the analysis, this exceedance is inconsequential. One of the values

for ^{63}Ni exceeds the average concentration assumed in the PA; that value is nearly two orders of magnitude less than the PA-derived limit. One of the values shown for ^{129}I exceeds the average value assumed in the PA; that value is about 20 times less than the PA-derived limit. Two of the values for $^{233/234}$U exceed the average value assumed in the PA. These values also exceed the PA-derived limit; however, the limit is derived from the intruder screening analysis. Since the PA screened $^{233/234}$U as having no impact in the analysis, this exceedance is inconsequential. Three of the values shown for ^{237}Np exceed the average value assumed in the PA; none exceed the PA-derived limit. Since the PA screened ^{237}Np as having no impact in the analysis, this exceedance is inconsequential. One of the values for $^{239/240}$Pu exceeds the average concentration assumed in the PA for ^{239}Pu. This value also exceeds the PA-derived limit for ^{240}Pu. . However, this limit was derived from the intruder screening analysis in the PA[11]; the PA-derived limit for ^{239}Pu was derived from the groundwater screening analysis. Since the PA screened these radionuclides as having no impact in the analysis, this exceedance is inconsequential. None of the average concentrations shown in Table 4 exceed the limits in WAC revision 2.

Evaluation

1. Does the proposed activity involve a change to the Performance Assessment or exceed PA performance measures/conclusions?

   No. Per the evaluation above, the discovered activity (i.e., Saltstone WAC not derived from PA) neither changes the PA nor exceeds performance measures/conclusions. The Saltstone PA analyzed the entire radionuclide inventory of salt solution that could arise from ITP, blended with ETF salt waste, and concluded that no performance measure would be exceeded. Therefore, the WAC for disposal of ITP and ETF waste in Saltstone need not contain any radionuclide limits derived from the PA, so long as the source of salt waste transferred to Saltstone is restricted to that arising from ITP and ETF.

2. Does the proposed activity involve a:

   a. change to the basic disposal concept as described in the PA?

      No. Disposal of salt solution arising from ITP and ETF was analyzed in the Saltstone PA. All of the salt solution that has been processed into Saltstone to date and the present contents of Tank 50 arose from ITP and ETF. All of the salt solution disposed to date was mixed with cementitious materials per the formulation assumed in the PA and was disposed into the Saltstone vaults that were analyzed in the PA. The present contents of Tank 50 will be mixed with cementitious materials per the same formulation and disposed into the same vaults. Therefore, there is no change to the disposal concept described in the PA.

   b. change to the analyses or radionuclide limits as described in the PA?

      No. The analyses described in the PA are for salt waste arising from ITP and ETF, mixed with cementitious materials per the formulation assumed in the PA. Since all of the salt solution disposed to date and the present contents of Tank 50 is the same as that analyzed in the PA, there are no changes to the analyses as described in the PA. The PA was performed to validate that disposal of the entire projected inventory of salt waste arising from ITP, blended with ETF

waste, would not exceed performance objectives. Because of this approach, no radionuclide limits were derived from the PA for ITP salt solution. Therefore, there are no changes to the radionuclide limits as described in the PA.

c. change in the disposal authorization that leads to a significant change in projected dose?

No. The disposal authorization is based on the PA for Saltstone. Since all the salt solution disposed to date and the present contents of Tank 50 are the same as that analyzed in the PA, there is no change in the disposal authorization.

d. change in the results in the approved PA that is greater than 10%?

No. Since all the salt solution processed to date and the present contents of Tank 50 are the same as that analyzed in the PA, there is no change in the PA results.

e. change of greater than 10% in the dose calculated in the approved PA?

No. Since all the salt solution processed to date and the present contents of Tank 50 are the same as that analyzed in the PA, there is no change in the dose calculated in the PA.

f. Does the proposed activity modify the analysis or conclusions provided in the Composite Analysis?

No. Since all the salt solution processed to date and the present contents of Tank 50 are the same as that analyzed in the PA, the discovered activity does not modify the analysis or conclusions provided in the Composite Analysis.

g. change to the Disposal Authorization Statement?

No. Since all the salt solution processed to date and the present contents of Tank 50 are the same as that analyzed in the PA, there need be no change in the Disposal Authorization Statement.

Conclusion

The Saltstone PA analyzed the disposal of the entire projected amount of salt waste arising from the ITP process, blended with salt waste arising from ETF. The PA concluded that no performance measures would be exceeded. Therefore, as long as the waste being disposed in Saltstone arose from ITP and ETF, there was no need to develop radionuclide disposal limits for Saltstone. Since all the waste disposed in Saltstone to date arose from ITP and ETF and the salt solution was mixed with cementitious materials per the formulation assumed in the PA, all the Saltstone presently in the Saltstone vaults is bounded by the Saltstone PA. The current contents of Tank 50 are also within the bounds of the PA because it arose from ITP and ETF and will be processed per the PA. The current contents of Tank 50 should be analyzed versus revision 2 of the Saltstone WAC to ensure that considerations other than the PA are not compromised. The current Saltstone WAC (i.e., revision 3) was developed for waste not arising from ITP.

Table 1 Radionuclide Acceptance Limits from WAC 4.01, Rev. 2, 6/19/98.		
Radioactive Contaminant	Acceptance Limit (nCi/g)	Basis
^{3}H	1800	Resp. Prot.
^{14}C	800	NRC Class A
^{59}Ni	23,000	NRC Class A
^{63}Ni	3,700	NRC Class A
^{60}Co	6.8	Shielding
^{79}Se	12	Groundwater
$^{90}Sr/Y$	40	NRC Class A
^{94}Nb	20	NRC Class A
^{99}Tc	320	NRC Class A
$^{106}Ru/Rh$	128	Shielding
^{125}Sb	76	Shielding
^{126}Sn	14	Shielding
^{129}I	1	Groundwater
^{137}Cs	45	Shielding
^{154}Eu	16	Shielding
^{237}Np	0.03	Groundwater
^{241}Pu	200	Hazard Analysis
RCG (See note 1)	1	Shielding
Total Alpha (See note 2)	20	NRC Class A
Total Beta/Gamma	7500	AB Source Term

Note 1: Radionuclides that emit high-energy gamma radiation must be monitored to assure radiation exposure to Z Area personnel will not exceed RC&O guidelines. Based on process knowledge and waste tank histories, the 6 isotopes shown in the equation below have been identified as the principal gamma-emitting species in salt solution from ITP and ETF operations (concentrations expressed in nCi/g) that are used to calculate the Radiation Control Guide (RCG):

$$RCG = 0.145 \times [^{60}Co] + 0.0078 \times [^{106}Ru] + 0.013 \times [^{125}Sb] + 0.0705 \times [^{126}Sn] + 0.022 \times [^{137}Cs] + 0.061 \times [^{154}Eu]$$

The effect of other gamma-emitting isotopes, if present in significant concentrations in the waste, shall also be incorporated into the RCG calculation to estimate the total effect of all gamma emitting species. A USQE is required whenever RCG > 1.

Note 2: The Total Alpha limit of 20 nCi/gm protects the Authorization Basis limit of 50 nCi/gm.

Table 2 Radionuclide Acceptance Limits from WAC 4.01, Rev. 3, 2/25/02.		
Radioactive Contaminant	Acceptance Limit (nCi/g)	Basis
^{3}H	1800	Resp. Prot.
^{14}C	3.6	PA*
^{59}Ni	0.0013	PA*
^{63}Ni	10.4	PA*
^{60}Co	6.8	Shielding
^{79}Se	3.5	PA*
$^{90}Sr/Y$	120	AB Source Term
^{93}Zr	0.0013	PA*
^{94}Nb	20	NRC Class A
^{99}Tc	1000	AB Source Term
$^{106}Ru/Rh$	128	Shielding
^{107}Pd	0.0013	PA*
^{121m}Sn	8.8	PA*
^{125}Sb	76	Shielding
^{126}Sn	0.21	PA*
^{129}I	0.43	PA*
^{135}Cs	0.0013	PA*
^{137}Cs	45	Shielding
^{151}Sm	60	PA*
^{152}Eu	0.10	PA*
^{154}Eu	1.8	PA*
^{232}U	0.0033	PA*
^{233}U	0.0013	PA*
^{234}U	0.0013	PA*
^{238}U	0.0013	PA*
^{237}Np	0.0013	PA*
^{238}Pu	20	PA*
^{239}Pu	0.68	PA*
^{240}Pu	0.0013	PA*
^{241}Pu	0.15	PA*
^{242m}Am	0.0020	PA*
^{243}Am	0.0013	PA*
^{243}Cm	0.014	PA*
^{244}Cm	0.058	PA*
RCG (See note 1)	1	Shielding
Total Alpha (See note 2)	20	NRC Class A
Total Beta/Gamma	7500	AB Source Term

Note 1: Same as Table 1

Note 2: Same as Table 1

* Limits derived from the PA[3], many of which are based on screening analyses, which are very conservative. These limits will likely be revised via Special Analyses.

Table 3 Total Radionuclide Inventory Disposed in Saltstone to Date and Comparison to That Analyzed in the Saltstone Performance Assessment

Radionuclide	Vault 1-Cell A, Ci[a]	Vault 1-Cell B, Ci[a]	Vault 1-Cell C, Ci[a]	Vault 4-Cell G, Ci[a]	Total Disposed to date, Ci	Total Assumed in PA, Ci[b]	Fraction of Total Assumed Disposed to date
H-3	13	16.1	7.5	10.8	4.74E+01	1.90E+04	2.49E-03
C-14	0.5	0.5	0.3	7.90E-02	1.38E+00	6.50E+00	2.12E-01
Ni-59	5.00E-04[c]	3.80E-03[c]	3.00E-02	8.90E-03[c]	4.32E-02	2.00E-01	2.16E-01
Co-60	1.10E-03	1.90E-03	2.60E-03	1.30E-03	6.90E-03	2.00E+02	3.45E-05
Ni-63	1.90E-03	1.10E-02	9.60E-01	8.40E-03[c]	9.81E-01	2.00E+01	4.91E-02
Se-79	0.1	7.20E-02	1.30E-01	9.30E-03	3.11E-01	3.20E+02	9.73E-04
Sr-90	5.90E-03[c]	6.50E-03	8.40E-03	4.60E-03	2.54E-02	6.80E+02	3.74E-05
Nb-94	8.00E-04[c]	6.80E-04[c]	1.00E-03[c]	5.20E-04[c]	3.00E-03	NR[d]	
Tc-99	40	35.7	32.7	16.5	1.25E+02	6.50E+04	1.92E-03
Ru-106	1.10E-03	1.10E-03	4.20E-01	1.80E-01	6.02E-01	3.30E+04	1.82E-05
Sb-125	0.1	8.50E-03	4.8	1.1	6.01E+00	6.50E+03	9.24E-04
Sn-126	0.3	0.2	5.10E-01	4.10E-02	1.05E+00	1.30E+02	8.08E-03
I-129	1.00E-02	1.80E-02	8.40E-02	6.00E-02	1.72E-01	2.00E+01	8.60E-03
Ba-133	NR[d]	NR[d]	3.60E-03[c]	2.90E-03[c]	6.50E-03	NR[d]	
Cs-137	1.7	2.3	5.1	3.3	1.24E+01	2.00E+04	6.20E-04
Sm-151	NR[d]	3.60E-02[c]	1.40E-03	9.70E-04[c]	3.84E-02	2.00E+03	1.92E-05
Eu-152	NR[d]	3.20E-04[c]	8.80E-03[c]	6.40E-03[c]	1.55E-02	5.80E+00	2.68E-03
Eu-154	4.20E-04[c]	5.90E-04[c]	2.10E-03[c]	9.60E-04[c]	4.07E-03	6.50E+02	6.26E-06
Eu-155	NR[d]	2.80E-03[c]	7.80E-03[c]	3.30E-04[c]	1.09E-02	3.20E+02	3.42E-05
U-233/234	NR[d]	NR[d]	2.90E-01	2.00E-01	4.90E-01	2.60E-03	1.88E+02
U-235/236	NR[d]	NR[d]	3.20E-03	4.80E-03	8.00E-03	NR[d]	
Np-237	3.00E-05	6.40E-04[c]	3.80E-03	7.10E-04	5.18E-03	5.80E-02	8.93E-02
U-238	NR[d]	NR[d]	7.40E-03	9.00E-03[c]	1.64E-02	2.00E-03	8.20E+00
Pu-238	NR[d]	2.60E-04	7.50E-03	3.60E-03	1.14E-02	4.90E+01	2.32E-04
Pu-239/240	NR[d]	7.50E-04	1.20E-02	2.70E-03	1.55E-02	1.20E+00	1.29E-02
Pu-241	2.80E-04[c]	4.40E-03	4.10E-02	6.60E-03	5.23E-02	3.20E+01	1.63E-03
Am-241	NR[d]	NR[d]	5.00E-04	1.10E-03	1.60E-03	1.30E+02	1.23E-05
Pu-242	NR[d]	NR[d]	9.00E-04	3.70E-04[c]	1.27E-03	NR[d]	
other alpha	0.2	0.1	NR[d]	NR[d]	3.00E-01	1.30E+02	2.31E-03
Total Volume							
Salt Solution							
Gallons	550286	553980	935562	551680	2.59E+06	1.93E+08	1.34E-02

a. Data from Table 2-3 of reference 11.
b. Data from Table 2.6-2 of reference 3.
c. Value in reference 11 is stated as less than the value here (i.e., the value for ^{59}Ni in Vault 1 – Cell A in reference 11 is < 5E-04. For the purposes of this report, the value used is 5.00E-04).
d. NR means that a value was not reported.

Table 4 Average Radionuclide Concentration in Salt Solution Disposed to Date in Z-Area

Radionuclide	Average Salt Solution Concentration, Ci/L[a]				Average Concentration Assumed in PA, Ci/L[b]	WAC Rev.2	WAC Rev.3
	Vault 1-Cell A	Vault 1 – Cell B	Vault 1-Cell C	Vault 4-Cell G			
H-3	6.24E-06	7.68E-06	2.12E-06	5.17E-06	2.6E-05	2.25E-03	2.25E-03
C-14	2.40E-07	2.38E-07	8.47E-08	3.78E-08	8.9E-09	1.00E-03	4.50E-06
Ni-59	2.40E-10[c]	1.81E-09[c]	8.47E-09[c]	4.26E-09[c]	2.7E-10	2.88E-02	1.63E-09
Co-60	5.28E-10	9.06E-10	7.34E-10	6.23E-10	2.8E-07	8.50E-06	8.50E-06
Ni-63	9.12E-10	5.25E-09	2.71E-07	4.02E-09	2.7E-08	4.63E-03	1.30E-05
Se-79	4.80E-08	3.43E-08	3.67E-08	4.45E-09	4.4E-07	1.50E-05	4.38E-06
Sr-90	2.83E-09[c]	3.10E-09	2.37E-09	2.20E-09	9.3E-07	5.00E-05	1.50E-04
Nb-94	3.84E-10[c]	3.24E-10[c]	2.82E-10[c]	2.49E-10[c]	NA[d]	2.50E-05	2.50E-05
Tc-99	1.92E-05	1.70E-05	9.23E-06	7.90E-06	8.9E-05	4.00E-04	1.25E-03
Ru-106	5.28E-10	5.25E-10	1.19E-07	8.62E-08	4.5E-05	1.60E-04	1.60E-04
Sb-125	4.80E-08	4.05E-09	1.36E-06	5.27E-07	8.9E-06	9.50E-05	9.50E-05
Sn-126	1.44E-07	9.54E-08	1.44E-07	1.96E-08	1.8E-07	1.75E-05	2.63E-07
I-129	4.80E-09	8.58E-09	2.37E-08	2.87E-08	2.7E-08	1.25E-06	5.38E-07
Ba-133	NR[e]	NR[e]	1.02E-09[c]	1.39E-09[c]	NA[d]	NA[d]	NA[d]
Cs-137	8.16E-07	1.10E-06	1.44E-06	1.58E-06	2.7E-05	5.63E-05	5.63E-05
Sm-151	NR[e]	1.72E-08[c]	3.95E-10	4.64E-10[c]	2.7E-06	NA[d]	7.50E-05
Eu-152	NR[e]	1.53E-10[c]	2.48E-09[c]	3.06E-09[c]	8.0E-09	NA[d]	1.25E-07
Eu-154	2.02E-10[c]	2.81E-10[c]	5.93E-10[c]	4.60E-10[c]	8.9E-07	2.00E-05	2.25E-06
Eu-155	NR[e]	1.34E-09[c]	2.20E-09[c]	1.58E-10[c]	4.4E-07	NA[d]	NA[d]
U-233/234	NR[e]	NR[e]	8.19E-08	9.58E-08	3.6E-10[f]	NA[d]	1.63E-09[g]
U-235/236	NR[e]	NR[e]	9.04E-10	2.30E-09	NA[d]	NA[d]	NA[d]
Np-237	1.44E-11	3.05E-10[c]	1.07E-09	3.40E-10	8.0E-11	3.75E-08	1.63E-09
U-238	NR[e]	NR[e]	2.09E-09	4.31E-09[c]	2.7E-12	NA[d]	1.63E-09
Pu-238	NR[e]	1.24E-10	2.12E-09	1.72E-09	6.7E-08	NA[d]	2.50E-05
Pu-239/240	NR[e]	3.58E-10	3.39E-09	1.29E-09	1.7E-09	NA[d]	1.63E-09[g]
Pu-241	1.34E-10[c]	2.10E-09	1.16E-08	3.16E-09	4.4E-08	2.50E-04	1.88E-07
Am-241	NR[e]	NR[e]	1.41E-10	5.27E-10	1.8E-07	NA[d]	NA[d]
Pu-242	NR[e]	NR[e]	2.54E-10	1.77E-10[c]	NA[d]	NA[d]	NA[d]
other alpha	9.60E-08	4.77E-08	NR[e]	NR[e]	1.8E-07	2.50E-05	2.50E-05

a. Derived from Table 3.
b. From Table 2.6-2 of Reference 3
c. Value derived from analysis result of less than the stated quantity in Table 3.
d. NA means that the PA or WAC has no value for that radionuclide.
e. NR means that a value was not reported in the original analysis.
f. The larger of the values for the two radionuclides (i.e., ^{234}U) is shown.
g. The most restrictive of the WAC values for the two radionuclides is shown.

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9. *Record of Decision Savannah River Site Salt Processing Alternatives*, Federal Register Volume 66, Number 201, pages 52752-52756, October 17, 2001.

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