SRNS-J2100-2008-00004
Revision 2
June 2009

ESTIMATED CLOSURE INVENTORY FOR THE SALTSTONE DISPOSAL FACILITY

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Prepared for U.S. Department of Energy Under Contract No. DE-AC09-08SR22470

APPROVALS

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ACRONYMS/ABBREVIATIONS

ARP/MCU	Actinide Removal Project/Modular Caustic Side Solvent Extraction
CSSX	Caustic Side Solvent Extraction
DDA	Deliquification, Dissolution and Adjustment
DF	Decontamination Factor
DOE	Department of Energy
DWPF	Defense Waste Processing Facility
ETP	Effluent Treatment Project
MST	Monosodium Titanate
PA	Performance Assessment
SDF	Saltstone disposal Facility
SRS	Savannah River Site
SWPF	Salt Waste Processing Facility
WCS	Waste Characterization System

1.0 INVENTORY INPUTS

1.1 Purpose

The purpose of this report is to provide estimated radiological and chemical inventories of the Saltstone Disposal Facility (SDF) at the time of closure to support Performance Assessment (PA) modeling.

1.2 Scope

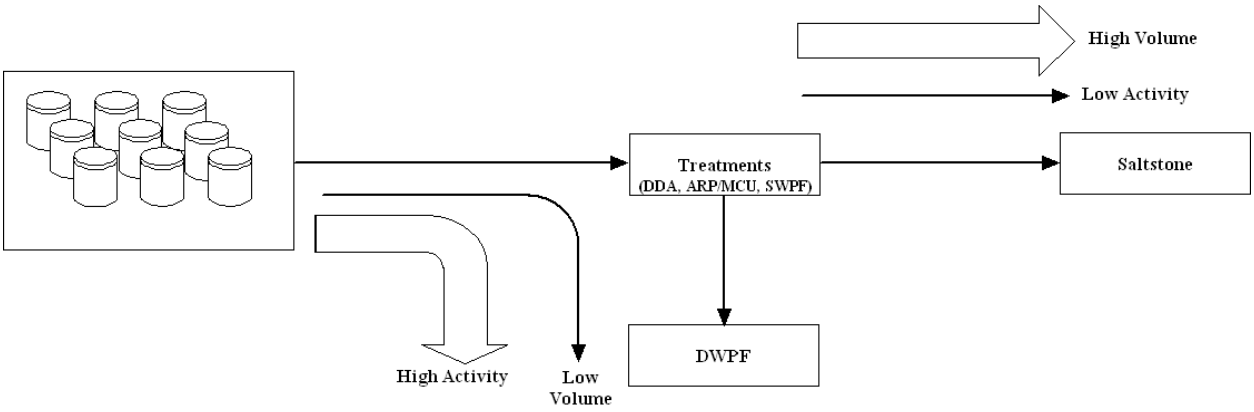
Estimated radiological and chemical inventories are to be provided for each of the existing and future SDF disposal units.

1.3 Approach

To estimate the material to be disposed within the SDF, the source of the disposed material needs to be identified. The only sources are the material currently stored in the Tank Farm waste tanks and additional material received from the H-Canyon. The material in the Tank Farms will be disposed in one of three locations, Saltstone, a permanent repository via the Defense Waste Processing Facility (DWPF), or remain in the tanks. The Department of Energy's (DOE's) plans call for stabilizing and disposing of the high-level waste in a deep geologic repository for spent nuclear fuel and high-level radioactive waste. This will be done by stabilizing the high-level waste in a borosilicate glass matrix through vitrification in a facility known as the DWPF. This process has been on-going since 1996. As for the salt waste, DOE plans to remove fission products and actinides from these materials using a variety of technologies, combine the removed fission products and actinides with the metals being vitrified in DWPF, and solidify the remaining low-activity salt stream into a grout matrix, known as saltstone, suitable for disposal at the SDF at the Savannah River Site (SRS). [DOE-WD-2005-001]

To determine the material for disposal at Saltstone, a starting point of the Tank Farm inventory was used. Then the radionuclides were adjusted based on treatment processes. The remaining inventory was assumed to be sent to Saltstone and was distributed among the actual and future disposal units. Figure 1-1 illustrates the basic flow of material disposal.

Figure 1-1 Basic Flow Diagram



Saltstone presently has material disposed in two vaults (1 and 4). Vault 4 is being used to dispose material and plans include continuing to add material. The current system plan was used to estimate the additional material to be added to Vault 4. [LWO-PIT-2007-00062]

Once the Vault 4 inventory was estimated, it was removed from the Tank Farm inventory. The remaining inventory was adjusted to account for the removal of treated radionuclides. This resulting inventory was distributed amongst the future disposal units.

2.0 TANK FARM INVENTORY

2.1 Initial Inventory

The starting point for determining the material to be disposed in Saltstone was the inventory of material in the Tank Farms with a basis date of August 14, 2007. [LWO-LWE-2007-00250] This inventory reported material in three separate phases; supernate, salt/insolubles, and sludge. The supernate is further separated into three areas; free supernate, salt interstitial, and sludge interstitial. This report contains two estimates of actinide soluble concentrations, one conservative and the other not conservative. The higher, or conservative, of the two was used for this input.

2.1.1 Initial Inventory Adjustment

There are several radionulcides that required further estimating for the purpose of the PA model, Cl-36, K-40, Zr-93, Nb-93m, Pd-107, Sn-126, Pt-193, Ra-226, Th-229, Th-230, U-233, U-234, and U-236. Table 2-1 is a description of how these constituents were estimated beyond that of the current Tank Farm inventory. [LWO-LWE-2007-00250]

Table 2-1 Constituents Estimate Method

Constituent	Estimate Method
Cl-36	The Cl-36 inventory was estimated from the expected detection limit since a minimal amount is expected. Another estimate has been made which assumes all the Cl in the tanks is Cl-36. [LWO-PIT-2006-00039] This estimate is an over estimate. Given the most likely source, neutron activation of Cl-35 within the waste tanks (no Cl-35 was within the reactor fuel or targets), minimal, if any, Cl-36 is expected. For this reason, waste tank samples have not been tested, over time, for Cl-36. The detection of Cl-36 is hindered by other beta emitters. The detection ability is increased as the major beta emitters (Sr-90, Y-90, and Cs-137) are removed from the sample analysis. With the removal of these isotopes, a detection limit four orders of magnitude below the major isotopes is assumed to be easily determined. The current inventories are based on the future disposal cell concentration estimated to be four orders of magnitude below the maximum beta emitter (Sr-90) inventory. Since a minimal amount of Cl-36 is expected, choosing the future disposal cell concentration (lowest of the three disposal unit designs) was appropriate.
Cm-242	Set equal to the Am-242m inventory (Vault 4)
Cm-247	Grown in 70 years from the decay of Cf-251 (Vault 4)
K-40	The K-40 inventory was estimated from the expected detection limit since a minimal amount is expected. Another estimate has been made which determined an upper bounding limit. [CBU-PIT-2005-00178] This estimate is an over estimate. Given the most likely source, neutron activation of K-39 within the waste tanks (no K-39 was within the reactor fuel or targets), minimal, if any, K-40 is expected. For this reason, waste tank samples have not been tested, over time, for K-40. Since, a minimal amount if expected, similar to the Cl-36 estimate; the same inventory was used for K-40.
Nb-93m	The Nb-93m activity was estimated from the activation of Nb-93 and decay of Zr-93. [LWO-PIT-2006-00039] The total inventory is assumed to be slightly soluble (0.1%) and distributed across the vaults by relating it to the Nb-94 inventory.
Pd-107	The Pd-107 activity was estimated from irradiated reactor assemblies. [CBU-PIT-2005-00178] The total inventory is assumed to be slightly soluble (0.1%) and distributed across the vaults by relating it to the Ni-59 and Ni-63 inventories.
Pt-193	The Pt-193 activity was estimated from the activation of Pt-192. [LWO-PIT-2006-00039] The total inventory is assumed to be slightly soluble (0.1%) and distributed across the vaults by relating it to the Ni-59 and Ni-63 inventories.

Table 2-1 Estimate Constituents Estimate Method (Continued)

Constituent	Estimate Method
Ra-226	The Ra-226 activity is assumed to be in transient equilibrium with U-234 and a waste age of 50 years. [CBU-PIT-2005-00040].
Sn-126	The Sn-126 activity is based current tank farm inventory plus an estimate of future additions from the canyon(s). The canyon(s) future additions concentration is based on the current concentration of Sn-126 within the tank farm by the projected canyon(s) volume.
Th-229	The Th-229 activity is assumed to be in transient equilibrium with U-233. [CBU-PIT-2005-00040] This is a conservative assumption given the relatively long half lives and the relative young age of our waste.
Th-230	The Th-230 activity is assumed to be in transient equilibrium with U-234. [CBU-PIT-2005-00040] This is a conservative assumption given the relatively long half lives and the relative young age of our waste.
U-233	U-233 is the daughter of Np-237. An assumption of equal activities would be acceptable, although, reviewing Tanks 18 and Tank 19 sample results indicates a larger U-233 activity than Np-237, ~40 times larger in Tank 18 and ~80 times in Tank 19. Closer examination of Tank 18 and 19 sample results shows all the U-233 measurements to be less than detection limits. One sample analysis of "crusty" solids from Tank 19 that contained a detectable amount of U-233 shows a ratio of ~3 times. Although, based on U-233 being the daughter of Np-237, secular equilibrium was chosen as the estimate method. Therefore, the U-233 activity is set equal to Np-237 activity. [U-TR-F-00005, WSRC-TR-2002-00052]
U-234	The activity of U-234 in natural uranium is approximately equal to that of U-238. Uranium, in a fission reaction burns U-234 as well as U-235. So, it is assumed if U-235 is depleted, U-234 is also depleted. Reviewing the ratio of U-235 to U-238 in FTF waste indicates uranium content ranging from slightly to highly depleted. Therefore it is reasonable to assume the U-234 activity is equal to that of U-238.
U-236	Reviewing highly depleted to highly enriched uranium indicates U-236 activity ranging from 0-1% of the total activity of U-234, U-235 and U-238. So, the U-236 content is assumed to be equal to 1% of the total activity of U-234, U-235 and U-238.
Zr-93	The Zr-93 activity was estimated from irradiated reactor assemblies. [CBU-PIT-2005-00178] The total inventory is assumed to be slightly soluble (0.1%) and distributed across the vaults by relating it to the Nb-94 inventory.

2.2 Tank Farm Additions

2.2.1 Canyon Influents

The H-Canyon will continue to transfer material to the Tank Farms. This material will add to the current inventory of the Tank Farms. To account for this, projected volumes and concentrations were used to estimate the added material from the future canyon transfers. The projected volume was used from the system plan, while the estimated concentration was predicted using the estimate from the Tank 50 material balance within Waste Characterization System (WCS) 1.5. [LWO-PIT-2007-00062, LWO-PIT-2008-00013]

2.2.2 Effluent Treatment Project

The Effluent Treatment Project (ETP) primarily treats material that originated from the Tank Farm, and returns the contaminated portion to the Tank Farm. This material is already accounted for in the starting inventory, and therefore, is addressed with respect to material that will be disposed within Saltstone. The ETP also receives material from outside the Tank Farm. This is assumed to be negligible relative to the total amount of material disposed, and is not included in the final amount. Therefore, no ETP additions were added to the inventory to be disposed in Saltstone.

2.2.3 Defense Waste Processing Facility Recycle

In treating sludge material from the Tank Farm, DWPF returns a recycle stream to the Tank Farm. This recycle stream contains a fraction of the material originally sent to DWPF. Since there are no outside additions to the treatment of this material, the recycle stream contains components already accounted for in the initial inventory. Therefore, there were no additions to the inventory from DWPF.

3.0 TREATMENTS

For the purpose of this inventory estimate, the treatments that will be considered are in sludge removal, Deliquification, Dissolution and Adjustment (DDA), Actinide Removal Process/Modular Caustic Side Extraction Unit (ARP/MCU), and Salt Waste Processing Facility (SWPF).

3.1 Sludge Removal

Most of the highly radioactive radionuclides reside within the sludge at the bottom of a number of the waste tanks. The current Tank Farm inventory has a separate accounting for dry sludge contents (sludge minus sludge interstitial liquid). This entire dry sludge inventory is assumed to be disposed within DWPF or left as residuals within closed waste tanks.

3.2 Deliquification, Dissolution and Adjustment

The DDA process involves the following:

1. removing the supernate from above the saltcake;
2. extracting interstitial liquid within the saltcake matrix;
3. dissolving the saltcake and transferring the resulting salt solution to a settling tank; and
4. transferring the salt solution to the Saltstone Facility feed tank where, if required, the salt solution is aggregated with other Tank Farm waste to adjust batch chemistry. [DOE-WD-2005-001]

Because of the relatively high solubility of cesium, a majority of the Cs-137 in the salt waste is found in the concentrated supernate solution and interstitial liquid. Cesuim-137 is the predominant radionuclide found in salt waste. Therefore, removal of supernate and interstitial liquid from the waste tank, prior to dissolution of the saltcake, will significantly reduce the total cesium inventory sent to SDF. The supernate and interstitial liquid removed during the DDA process, will be transferred to another waste tank, and stored for future processing through SWPF, except for the DDA material that was removed from Tank 41, which is accounted for in the Tank 49 contents.

3.3 Actinide Removal Project/Modular Caustic Side Solvent Extraction Unit

The ARP/MCU will treat liquid and dissolved salt, prior to disposal at Saltstone. Actinide Removal Project will use modified facilities at SRS to remove strontium and actinides from saltcake waste. Modular Caustic Side Solvent Extraction Unit will utilize the caustic side solvent extraction (CSSX) process to remove cesium from the salt waste. These treatments will remove solid (non-soluble) material by filtering, and cesium by solvent extraction, to be sent to DWPF. Although Monosodium Titanate (MST) strikes will be performed, for the purposes of inventory estimates, only filtering of material was assumed except for Cs-137. In other words, no strikes were assumed (no removal of soluble material). These are significant conservatisms; no MST strikes (no soluble material removal) and a minimum Cs decontamination factor. The ARP/MCU process assumptions are listed in Table 3-1. [LWO-PIT-2008-00013]

Table 3-1 ARP/MCU Process Assumptions

Component	Decontamination Factor
Cesium	12
Solids Filter	99%

3.4 Salt Waste Processing Facility

Salt Waste Processing Facility will also treat liquid and dissolved salt prior to disposal at Saltstone. The SWPF will utilize the same technologies as ARP and MCU to remove strontium, actinides, and cesium from the salt waste. However, due to process differences (larger number of contractors), SWPF will provide a much higher decontamination factor (DF) for cesium than

the MCU. The solids portion was assumed to be 99%, and removed by filtering, the same as ARP/MCU, while the DFs (applied to the soluble portion) were assumed to be quite different (although still conservative estimates). The DFs are listed in the Table 3-2. [CBU-PIT-2006-00056]

Table 3-2 Salt Waste Processing Facility Process Assumptions

Component	Decontamination Factor
Strontium	20
Cesium	40,000
Uranium	1.35
Neptunium	2.4
Plutonium	5.5
Americium	4.6
Curium	1.7
Solids Filter	99%

4.0 SALTSTONE DISPOSAL UNITS

For disposal, material from the Tank Farms will be sent to Saltstone, to be mixed with dry grout materials for permanent disposal. There are projected to be 64 disposal cells beyond the existing two disposal vaults. Currently Vaults 1 and 4 have been built, and contain disposed material. An example for Pu-238 inventory calculation in Vaults 1 and 4, and a future disposal cell, is included in Section 4.6.

4.1 Decay Date

The system plan calls for the last disposal cell to be filled at the end of the fiscal year 2030. [LWO-PIT-2007-00062] Therefore, all the inventories have been decayed corrected to October 1, 2030.

4.2 Vault 1

There are no plans to add additional salt material to Vault 1. Therefore, the current estimated inventory was used. [WSRC-RP-2006-00568] The chemical inventory was estimated using the highest value measured in the salt solution feed tank. [SWD-SWE-99-0056] The Vault 1 inventories are listed in Tables 4-1 and 4-2.

Table 4-1 Saltstone Vault 1 Radionuclide Inventory Estimate at Closure

Radionuclide	Inventory at Closure (Ci)
Am-241	4.7E-04
Ba-137m	4.1E+00
C-14	1.3E+00
Cl-36	7.6E-04
Co-60	8.2E-05
Cs-137	4.3E+00
Eu-152	1.8E-03
Eu-154	2.3E-04
H-3	6.1E+00
I-129	1.1E-01
K-40	7.6E-04
Nb-93m	2.5E-01
Nb-94	2.5E-03
Ni-59	3.5E-02
Ni-63	7.8E-01
Np-237	4.5E-03
Pd-107	1.9E-03
Pt-193	3.7E-01
Pu-238	7.8E-03
Pu-239	1.2E-02
Pu-240	1.2E-02
Pu-241	9.8E-03
Pu-242	9.0E-04
Ra-226	6.4E-07
Rh-106	1.5E-10
Ru-106	1.5E-10
Sb-125	1.6E-01
Sb-126	1.4E-01
Sb-126m	1.0E+00
Se-79	3.0E-01
Sn-126	1.0E+00
Sr-90	6.9E-03
Tc-99	1.1E+02
Te-125m	3.8E-02
Th-229	3.0E-01
Th-230	4.1E-01
U-233	2.8E-01
U-234	2.8E-01
U-235	3.2E-03
U-236	3.2E-03
U-238	7.4E-03
Y-90	6.9E-03
Zr-93	2.5E-01
Total	**1.3E+02**

Note: Due to the smaller number of radionuclides analyzed, the Vault 1 inventory contains fewer radionuclides than the other disposal units.

Table 4-2 Saltstone Vault 1 Chemical Inventory Estimate at Closure (10/1/2030)

Chemical	Inventory at Closure (kg)
Silver	2.0E+01
Arsenic	4.4E+01
Barium	2.0E+01
Cadmium	3.9E+02
Chromium	1.8E+02
Mercury	1.5E+00
Nitrite	7.4E-01
Nitrate	3.4E+01
Lead	2.3E+01
Selenium	6.1E+01

4.3 Vault 4

Vault 4's projected inventory was calculated by adding the inventory, prior to DDA processing, to the recent processing inventory. [WSRC-RP-2006-00568, SRS-REG-2007-00033] The system plan was used to determine which material will be added to Vault 4. [LWO-PIT-2007-00062] Table 4-3 lists the tanks that are projected to send material to Vault 4, and the amount of material planned to be sent.

Table 4-3 Tanks Planned to Send Material to Vault 4

Tank	Amount of Current Inventory Planned for Vault 4 Disposal	Portion of Vault 4 Inventory to be Treated by ARP/MCU
Tank 22	All	Half
Tank 23	All	None
Tank 24	All	All
Tank 25	All Liquid, Half Salt	All
Tank 41	All Liquid, Quarter Salt	None
Tank 49	All	Half
Tank 50	All	None

Revision 2 Update

A discrepancy was found in the Tank 50 inventory used to develop the Vault 4 inventory. The Tank 50 inventory used in prior revisions was based on a sample of a settled tank contents. To resolve this discrepancy, an analysis based on a sample of mixed tank contents, that was appropriate to the baseline date of 8/14/2007, was used and the Tank 50 inventory updated to reflex this change. [WSRC-STI-2007-00554]

The updated Vault 4 inventory (based on a mixed Tank 50 sample) was compared to the previous revision inventory. To provide conservatism, only those radionuclides' inventories that increased were adjusted. For those radionuclides' inventories that did not increase (i.e. the Revision 1 inventory was greater than Revision 2), the previous revision inventory was used as an estimate.

For added conservatism, a factor of five (5) was multiplied to those radionuclides whose inventories increased (not to the radionuclides' inventories that remain unchanged). This factor was a bounding estimate of the material variability within the waste tanks.

Furthermore, the chemical inventory from the Naval Fuel Material Facility Saltcrete drums disposed in Vault 4 cell A was added, the radionuclide inventory was already accounted within the annual Saltstone inventory updates. [WSRC-RP-90-638, WSRC-RP-2008-00390]

The Vault 4 inventory to be used in modeling is included in Tables 4-4 and 4-5. Note, this is not necessarily the projected inventory but a conservative inventory for modeling.

Table 4-4 Saltstone Vault 4 Radionuclide Inventory Estimate at Closure (10/1/2030)

Radionuclide	Inventory at Closure (Ci)
Ac-227	1.6E-05
Al-26	3.4E-01
Am-241	1.3E+02
Am-242m	6.7E-02
Am-243	1.8E+00
Ba-137m	2.8E+05
Bk-249	1.8E-28
C-14	2.7E+01
Ce-144	1.8E-09
Cf-249	6.5E-13
Cf-251	1.2E+00
Cf-252	1.8E-18
Cl-36	3.0E-03
Cm-242	6.7E-02
Cm-243	2.1E-01
Cm-244	1.3E+02
Cm-245	9.2E-01
Cm-247	3.9E-06
Cm-248	1.2E-13
Co-60	4.6E-01
Cs-134	5.2E-01
Cs-135	5.4E+00
Cs-137	3.0E+05
Eu-152	9.7E-02
Eu-154	1.2E+01
Eu-155	6.8E-01
H-3	2.6E+02
I-129	2.8E-01
K-40	3.0E-03
Na-22	1.5E-01
Nb-93m	8.4E+00
Nb-94	8.7E-02
Ni-59	4.0E-01
Ni-63	2.2E+01
Np-237	6.1E-01

Radionuclide	Inventory at Closure (Ci)
Pa-231	9.3E-05
Pd-107	5.0E-02
Pm-147	4.1E-01
Pr-144	1.8E-09
Pt-193	1.0E+01
Pu-238	9.1E+03
Pu-239	3.8E+02
Pu-240	1.2E+02
Pu-241	2.4E+03
Pu-242	8.1E-01
Pu-244	1.6E-02
Ra-226	4.1E+00
Ra-228	1.6E-06
Rh-106	9.1E-07
Ru-106	9.1E-07
Sb-125	5.7E+00
Sb-126	9.0E-01
Sb-126m	6.4E+00
Se-79	4.6E+01
Sm-151	4.2E+01
Sn-126	6.4E+00
Sr-90	2.4E+05
Tc-99	5.8E+02
Te-125m	1.4E+00
Th-229	2.5E+01
Th-230	7.5E+00
Th-232	3.2E-04
U-232	4.4E-02
U-233	2.4E+01
U-234	2.6E+01
U-235	4.7E-01
U-236	7.7E-01
U-238	5.9E-01
Y-90	2.4E+05
Zr-93	8.4E+00
Total	**1.1E+06**

Table 4-5 Saltstone Vault 4 Chemical Inventory Estimate at Closure

Chemical	Inventory at Closure (kg)
Arsenic	1.0E+03
Barium	2.6E+01
Cadmium	3.8E+01
Chromium	3.5E+03
Copper	8.8E+02
Fluorine	3.5E+03
Iron	6.9E+02
Lead	4.5E+02
Manganese	4.1E+02
Mercury	9.8E+02
Nickel	8.9E+01
Nitrate	4.3E+06
Nitrite	4.0E+05
Selenium	5.2E+03
Silver	2.7E+01
Uranium	2.8E+02
Zinc	1.3E+03

4.4 Remaining Disposal Units

There are a projected total of 64 additional disposal cells within the SDF beyond Vaults 1 and 4. Vault 1 and 4 current inventories have been accounted for, as well as the additional material to be added to Vault 4. The remaining 64 cells will receive the remaining Tank Farm inventories. Although the system plan calls for 34 units (36 units minus two vaults), or 68 disposal cells; 64 disposal cells were used to develop these estimates based on the facility layout plan. [LWO-PIT-2007-00062] The material was averaged over the remaining cells. The average disposal cell inventories for a single cell are included in Tables 4-6 and 4-7.

Table 4-6 Future Saltstone Disposal Cell Radionuclide Inventory Estimate at Closure (10/1/2030)

Radionuclide	Inventory per cell at Closure (Ci)
Ac-227	1.7E-07
Al-26	1.9E-01
Am-241	1.4E+00
Am-242m	5.9E-04
Am-243	3.7E-02
Ba-137m	2.2E+01
Bk-249	1.8E-28
C-14	2.0E+00
Ce-144	3.6E-10
Cf-249	6.7E-13
Cf-251	2.3E-14
Cf-252	1.8E-18
Cl-36	4.2E-04
Cm-242	6.3E-19
Cm-243	2.1E-04
Cm-244	9.5E-01
Cm-245	2.4E-04
Cm-247	7.1E-14
Cm-248	7.4E-14
Co-60	5.4E-02
Cs-134	1.5E-05
Cs-135	1.3E-04
Cs-137	2.3E+01
Eu-152	9.8E-02
Eu-154	1.8E+00
Eu-155	1.3E-01
H-3	3.0E+01
I-129	3.8E-01
K-40	4.2E-04
Na-22	6.9E-02
Nb-93m	3.7E-01
Nb-94	3.8E-03
Ni-59	8.4E-02
Ni-63	2.4E+00
Np-237	5.0E-02
Pa-231	9.8E-07
Pd-107	5.6E-03
Pm-147	7.7E-02
Pr-144	3.6E-10
Pt-193	1.1E+00
Pu-238	1.7E+02
Pu-239	1.5E+01
Pu-240	4.1E+00
Pu-241	4.2E+01
Pu-242	3.9E-03
Pu-244	1.6E-05
Ra-226	7.8E-07
Ra-228	8.7E-05
Rh-106	1.2E-06
Ru-106	1.2E-06
Sb-125	2.4E-01
Sb-126	1.2E+00
Sb-126m	8.2E+00
Se-79	1.4E+00
Sm-151	5.9E+01
Sn-126	8.2E+00
Sr-90	3.7E+01
Tc-99	5.4E+02
Te-125m	5.8E-02
Th-229	3.9E-02
Th-230	1.9E-01
Th-232	1.4E-03
U-232	3.1E-04
U-233	3.7E-02
U-234	1.3E-01
U-235	3.0E-03
U-236	1.6E-02
U-238	1.0E-01
Y-90	3.7E+01
Zr-93	3.7E-01
Total	**1.0E+03**

Table 4-7 Future Saltstone Disposal Cell Chemical Inventory Estimate at Closure

Chemical	Inventory per cell at Closure (kg)
Arsenic	1.6E+02
Barium	1.7E+00
Cadmium	8.0E+00
Chromium	4.0E+02
Copper	2.9E+02
Fluorine	2.0E+02
Iron	3.5E+01
Lead	9.7E+01
Manganese	1.1E+02
Mercury	1.5E+02
Nickel	2.2E+01
Nitrate	2.2E+05
Nitrite	7.7E+04
Selenium	5.5E+02
Silver	3.4E-01
Uranium	9.6E+00
Zinc	4.0E+02

4.5 Total Saltstone Disposal Inventory

The modeled total inventory of material to be disposed of in all saltstone disposal units is listed in Tables 4-8 and 4-9.

Table 4-8 Total Saltstone Disposal Units' Radionuclide Inventory Estimate at Closure (10/1/2030)

Radionuclide	Inventory at Closure (Ci)
Ac-227	2.7E-05
Al-26	1.3E+01
Am-241	2.2E+02
Am-242m	1.0E-01
Am-243	4.2E+00
Ba-137m	2.8E+05
Bk-249	1.2E-26
C-14	1.6E+02
Ce-144	2.5E-08
Cf-249	4.4E-11
Cf-251	1.2E+00
Cf-252	1.2E-16
Cl-36	3.1E-02
Cm-242	6.7E-02
Cm-243	2.2E-01
Cm-244	1.9E+02
Cm-245	9.4E-01
Cm-247	3.9E-06
Cm-248	4.9E-12
Co-60	3.9E+00
Cs-134	5.2E-01
Cs-135	5.4E+00
Cs-137	3.0E+05
Eu-152	6.4E+00
Eu-154	1.3E+02
Eu-155	9.0E+00
H-3	2.2E+03
I-129	2.5E+01
K-40	3.1E-02
Na-22	4.6E+00
Nb-93m	3.2E+01
Nb-94	3.3E-01
Ni-59	5.8E+00
Ni-63	1.8E+02
Np-237	3.8E+00
Pa-231	1.6E-04
Pd-107	4.1E-01
Pm-147	5.3E+00
Pr-144	2.5E-08
Pt-193	8.1E+01
Pu-238	2.0E+04
Pu-239	1.3E+03
Pu-240	3.8E+02
Pu-241	5.1E+03
Pu-242	1.1E+00
Pu-244	1.7E-02
Ra-226	4.1E+00
Ra-228	5.6E-03
Rh-106	7.8E-05
Ru-106	7.8E-05
Sb-125	2.1E+01
Sb-126	7.8E+01
Sb-126m	5.3E+02
Se-79	1.4E+02
Sm-151	3.8E+03
Sn-126	5.3E+02
Sr-90	2.4E+05
Tc-99	3.5E+04
Te-125m	5.2E+00
Th-229	2.8E+01
Th-230	2.0E+01
Th-232	9.0E-02
U-232	6.4E-02
U-233	2.7E+01
U-234	3.5E+01
U-235	6.7E-01
U-236	1.8E+00
U-238	7.0E+00
Y-90	2.4E+05
Zr-93	3.2E+01
Total	**1.1E+06**

Table 4-9 Total Saltstone Disposal Units' Chemical Inventory Estimate at Closure

Chemical	Inventory at Closure (kg)
Arsenic	1.1E+04
Barium	1.5E+02
Cadmium	9.4E+02
Chromium	2.9E+04
Copper	1.9E+04
Fluorine	1.6E+04
Iron	2.9E+03
Lead	6.7E+03
Manganese	7.5E+03
Mercury	1.1E+04
Nickel	1.5E+03
Nitrate	1.8E+07
Nitrite	5.3E+06
Selenium	4.0E+04
Silver	6.9E+01
Uranium	8.9E+02
Zinc	2.7E+04

4.6 Example Calculation

Sections 4.6.1 – 4.6.3 are some example calculations to determine Pu-238 inventory for all disposal units.

4.6.1 Vault 1 Pu-238 Inventory

$A_{Vault1} = A_{Vault112/31/05}$

$A_{Vault112/31/05}$ = 9.5E-03 Ci – Inventory as of 12/31/2005, decayed to 10/1/2030 using a half life of 8.77E+01 years yields 7.8E-03 Ci

4.6.2 Vault 4 Pu-238 Inventory

$$A_{Vault4} = A_{<2007} + A_{Tk22l} + A_{Tk23l} + A_{Tk24l} + A_{Tk25l} + A_{Tk41l} + A_{Tk49l} + A_{Tk50l} + \left(A_{Tk25s} \times \frac{1}{2} \times 1\% \right) + \left(A_{Tk41s} \times \frac{1}{4} \right)$$

$A_{<2007}$ = 1.2E+00 Ci - Inventory as of September 2007

A_{Tk22l} = 1.9E+02 Ci – Tank 22 liquid (supernate and sludge interstitial) inventory

A_{Tk23l} = 9.1E-01 Ci – Tank 23 liquid (supernate and sludge interstitial) inventory

A_{Tk24l} = 5.7E+03 Ci – Tank 24 liquid (supernate and sludge interstitial) inventory

A_{Tk25l} = 2.0E+03 Ci – Tank 25 liquid (supernate and salt interstitial) inventory

The Tank 25 salt inventory was adjusted because half of the salt inventory is planned to be sent to Vault 4.

A_{Tk41l} = 1.8E+03 Ci – Tank 41 liquid (salt and sludge interstitial) inventory

The Tank 41 salt inventory was adjusted because a quarter of the salt inventory is planned to be sent to Vault 4.

A_{Tk49l} = 1.5E+02 Ci – Tank 49 liquid (supernate and sludge interstitial) inventory

A_{Tk50l} = 1.8E+02 Ci – Tank 50 liquid (supernate) inventory

A_{Tk25s} = 5.0E+03 Ci – Tank 25 salt inventory

The Tank 25 salt inventory was adjusted because half of the salt inventory is planned to be sent to Vault 4. In addition this inventory will be treated by ARP/MCU which, with respect to Pu-238, means the inventory will be filtered to 99% efficiency.

A_{Tk41s} = 4.0E+03 Ci – Tank 41 salt inventory

The Tank 41 salt inventory was adjusted because a quarter of the salt inventory is planned to be sent to Vault 4. This inventory will not be treated by ARP/MCU, therefore no filtering adjustment was made.

A_{Vault4} = 8.8E+03 Ci (dated 8/14/2007), decayed to 10/1/2030 using a half life of 8.77E+01 years yields 7.3E+03 Ci

Since the inventory from revision 1 (9.1E+03 Ci) was greater than the one calculated above, the previous revision's inventory was used for modeling.

4.6.3 Single Disposal Cell Pu-238 Inventory

$$A_{Cell} = \frac{V_{Can} \times c_{Can} + 1\% \times \left(\sum^{1-16,21,26-40,42-48,51} A_{Tkis} + \frac{1}{2} A_{Tk25s} + \frac{3}{4} A_{Tk41s} \right) + \sum^{1-16,21,26-40,42-48,51} A_{Tkil}}{DF_{SWPF} \times T_{Cells}}$$

V_{Can} = 4389 kgal – Volume of canyon additions

c_{Can} = 6.13E+03 pCi/ml – Concentration in canyon additions

$\sum^{1-16,21,26-40,42-48,51} A_{Tkis}$ = 6.6E+04 Ci – Sum of salt inventories for Tanks 1-16, 21, 26-40, 42-48, 51

A_{Tk25s} = 5.0E+03 Ci – Tank 25 salt inventory

A_{Tk41s} = 4.0E+03 Ci – Tank 41 salt inventory

The salt inventories were adjusted because they will be treated by SWPF which, with respect to Pu-238, means the inventories will be filtered to 99% efficiency.

$\sum^{1-16,21,26-40,42-48,51} A_{Tkil}$ = 7.0E+04 Ci – Liquid (supernate, salt interstitial and sludge interstitial) inventory for Tanks 1-16, 21, 26-40, 42-48, 51

DF_{SWPF} = 5.5 – Decontamination fraction

T_{Cells} = 64 – Total number of remaining disposal cells after Vaults 1 and 4

A_{Cell} = 2.0E+02 Ci (dated 8/14/2007), decayed to 10/1/2030 using a half life of 8.77E+01 years yields 1.7E+02 Ci

5.0 DRILL CUTTINGS INVENTORY

A scenario was developed in which a person drills a well for the purpose of retrieving drinking water. It was assumed the well would completely penetrate a vault or disposal cell. The inventory of the drilled segment was calculated for use in the proposed scenarios.

The calculated inventory was developed for the column of grout that would be removed during drilling activities. Each vault and disposal cell concentration for each radionuclide and chemical was calculated using the closure inventories (Tables 4.1, 4.2, and 4.4 – 4.7) and the final grout volume (i.e., Vault 1 was only assumed to be half full). The final grout volume was determined by using each vault's/cell's height and inside cell dimensions. The drill cutting column was assumed to be eight inches in diameter for the complete depth of each vault or disposal cell. To add conservatism, the maximum inventory of each isotope and chemical, independent of which disposal cell, was determined and assembled into one mass of drill cuttings. For example, the Pu-238 drill cuttings inventory was based on the Vault 4 concentration while the Tc-99 drill cuttings inventory was based on the future disposal cell concentration.

Table 5-1 Maximum Drill Cuttings' Radionuclide Inventory Estimate at Closure (10/1/2030)

Radionuclide	Inventory at Closure (Ci)
Ac-227	4.8E-11
Al-26	3.8E-06
Am-241	4.0E-04
Am-242m	2.0E-07
Am-243	5.3E-06
Ba-137m	8.5E-01
Bk-249	3.6E-33
C-14	8.1E-05
Ce-144	7.2E-15
Cf-249	1.3E-17
Cf-251	3.7E-06
Cf-252	3.6E-23
Cl-36	9.1E-09
Cm-242	2.0E-07
Cm-243	6.5E-07
Cm-244	3.8E-04
Cm-245	2.8E-06
Cm-247	1.2E-11
Cm-248	1.5E-18
Co-60	1.4E-06
Cs-134	1.6E-06
Cs-135	1.6E-05
Cs-137	9.0E-01
Eu-152	1.9E-06
Eu-154	3.7E-05
Eu-155	2.6E-06
H-3	8.0E-04
I-129	7.6E-06
K-40	9.1E-09
Na-22	1.4E-06
Nb-93m	2.6E-05
Nb-94	2.6E-07
Ni-59	1.7E-06
Ni-63	6.6E-05
Np-237	1.8E-06
Pa-231	2.8E-10
Pd-107	1.5E-07
Pm-147	1.5E-06
Pr-144	7.2E-15
Pt-193	3.0E-05
Pu-238	2.8E-02
Pu-239	1.2E-03
Pu-240	3.6E-04
Pu-241	7.3E-03
Pu-242	2.5E-06
Pu-244	5.0E-08
Ra-226	1.2E-05
Ra-228	1.7E-09
Rh-106	2.3E-11
Ru-106	2.3E-11
Sb-125	1.6E-05
Sb-126	2.3E-05
Sb-126m	1.6E-04
Se-79	1.4E-04
Sm-151	1.2E-03
Sn-126	1.6E-04
Sr-90	7.2E-01
Tc-99	1.1E-02
Te-125m	3.9E-06
Th-229	7.6E-05
Th-230	2.3E-05
Th-232	2.8E-08
U-232	1.3E-07
U-233	7.2E-05
U-234	7.9E-05
U-235	1.4E-06
U-236	2.3E-06
U-238	2.0E-06
Y-90	7.2E-01
Zr-93	2.6E-05
Total	**3.2E+00**

Table 5-2 Maximum Drill Cuttings' Chemical Inventory at Closure

Chemical	Inventory at Closure (kg)
Arsenic	3.1E-03
Barium	2.4E-04
Cadmium	4.7E-03
Chromium	1.1E-02
Copper	5.7E-03
Fluorine	1.1E-02
Iron	2.1E-03
Lead	1.9E-03
Manganese	2.1E-03
Mercury	3.0E-03
Nickel	4.3E-04
Nitrate	1.3E+01
Nitrite	1.5E+00
Selenium	1.6E-02
Silver	2.4E-04
Uranium	8.5E-04
Zinc	7.9E-03

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