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Initial Performance Assessment of the Disposal of Spent Nuclear Fuel and High-Level Waste Stored at Idaho National Engineering Laboratory

Volume 2: Appendices

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Abstract

This performance assessment characterized plausible treatment options conceived by the Idaho National Engineering Laboratory (INEL) for its spent fuel and high-level radioactive waste and then modeled the performance of the resulting waste forms in two hypothetical, deep, geologic repositories: one in bedded salt and the other in granite. The results of the performance assessment are intended to help guide INEL in its study of how to prepare wastes and spent fuel for eventual permanent disposal. This assessment was part of the Waste Management Technology Development Program designed to help the U.S. Department of Energy develop and demonstrate the capability to dispose of its nuclear waste, as mandated by the *Nuclear Waste Policy Act of 1982*.

The waste forms comprised about 700 metric tons of initial heavy metal (or equivalent units) stored at the INEL: graphite spent fuel, experimental low enriched and highly enriched spent fuel, and high-level waste generated during reprocessing of some spent fuel. Five different waste treatment options were studied; in the analysis, the options and resulting waste forms were analyzed separately and in combination as five waste disposal groups. When the waste forms were studied in combination, the repository was assumed to also contain vitrified high-level waste from three DOE sites for a common basis of comparison and to simulate the impact of the INEL waste forms on a moderate-sized repository. The performance of the waste form was assessed within the context of a whole disposal system, using the U.S. Environmental Protection Agency's *Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes*, 40 CFR 191, promulgated in 1985.

Although numerous caveats must be placed on the results, the general findings were as follows: Though the waste form behavior depended upon the repository type, all current and proposed waste forms provided acceptable behavior in the salt and granite repositories. In the salt repository, the behavior was acceptable even without a moderately corrosion-resistant canister or chemically adsorptive backfill. Furthermore, in

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the salt repository, differences among the individual waste forms were less pronounced than in the granite repository and the differences nearly disappeared when the waste forms were combined as a disposal group with vitrified high-level waste. In contrast, in the granite repository, the differences among waste forms were greater and these differences, though small, remained for the waste forms when combined as a disposal group. These findings suggest that treatment of the relatively small mass of INEL fuel and waste cannot substantially improve the performance of a moderate- or large-sized repository. Although waste treatment generally improved performance, an exception occurred for the graphite spent fuel from Fort St. Vrain, which was assumed to contain intact fuel particles. Moderate treatment showed degraded performance in comparison to whole block disposal; only extensive treatment improved performance.

Observations on other factors important to the performance assessment are that the diffusive transport of radionuclides from the waste parcel holes into the drifts and then the advective transport through the drifts were very important phenomena to model—their importance was comparable to the effect of the waste treatment options. Also, because of the requirements of 40 CFR 191, direct release of radionuclides to the ground surface from exploratory drilling into the repository dominated the releases from the salt repository.

An important assumption for the analysis was that critical conditions were to be avoided in the repository when a waste parcel completely degraded; thus only a small amount of fissionable uranium was permitted in each container. The result of this assumption was that a large number of containers was required (especially for the highly enriched spent fuel), which, because of the cost implications, suggested that future work should attempt to demonstrate that critical noncatastrophic conditions in a repository would be technically acceptable. Furthermore, other performance criteria such as dose to an individual (pending promulgation of new regulations on required repository performance) need to be examined to substantiate the initial guidance from this performance assessment, which indicates that the current waste forms are acceptable for long-term environmental compliance. Finally, the second phase of the performance assessment should be pursued and should examine the behavior of the waste forms in an unsaturated tuff repository.

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Appendix A: Spent Fuel and Waste Information

PERFORMANCE ASSESSMENT APPENDIX A SPENT FUEL AND WASTE INFORMATION

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Prepared By:

**Westinghouse Idaho Nuclear Company, Inc. (WINCO)
Spent Fuel and Waste Management Technology Development Program
System Analysis/Performance Assessment**

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A.1 Current Storage Forms

The Department of Energy (DOE) has received spent nuclear fuel at the Idaho Chemical Processing Plant (ICPP) for interim storage and reprocessing since 1953. Reprocessing of these fuels for uranium and other resource recovery has resulted in the production of 3800 m³ calcine (a granular solid) [Ref. 1] and an inventory of 1,100 m³ of high-level waste (HLW) [Ref. 2] converted to 120 m³ calcine and 5,700 m³ of low-level radioactive, sodium-bearing liquid waste [Ref. 3]. Also, 343 metric tons (27 metric tons of heavy metal(MTHM)) [Ref. 6,16] graphite fuels and 474 metric tons (300 MTHM) special fuels [Ref. 4] are in inventory. Continued receipt of fuels and decontamination activities will add to these inventories.

To date, a major activity of ICPP has been the reprocessing of spent nuclear fuels to recover fissile uranium. With the changes in world events, the demand to recover and recycle this material has decreased, and DOE has discontinued reprocessing fuels for the recovery of uranium. The DOE Spent Fuel and Waste Management Technology Development Program is investigating, testing, and selecting technologies to prepare spent fuels and high level waste for disposal in a geologic repository.

There are over ninety types of fuels being stored at the Idaho National Engineering Laboratory (INEL), most of them at ICPP. Over 90 wt percent of these fuels have been shipped to INEL for storage, with the balance of these fuels located at their generation site, such as the Fort St. Vrain facility. The volumes and masses of the spent fuels are described in Attachment 4 & 4A.

Since regulations prohibit the disposal of high-level wastes other than in a solid form [Title 10 Code of Federal Regulations (CFR) Part 60.137.c.1], all the liquid waste must be converted into a solid form before it can be disposed of in a repository. In addition, the waste form must meet the current Best Demonstrated Available Technology (BDAT) requirement of 40 CFR 268.42(a). For this draft performance assessment (PA), the liquid waste will be assumed converted into calcine form. Thus, in the sections below, minimally-conditioned spent fuels and calcines will be the base repository form to be considered for disposal in the repository. The calcine will not meet the BDAT requirement of 40 CFR 268.42(a), but it will provide a basis that all other waste forms could be measured against. If the calcine waste form meets the release limits of 40 CFR 191, an application could be submitted to the Administrator demonstrating that the calcine alternative can achieve a measure of performance equivalent to the BDAT under 40 CFR 268.48(b). The Performance Assessment results will show which disposal form is most stable in a repository environment. The volumes and masses of the calcine (including the present liquid waste) are also described in Attachment 4 & 4A.

The total amount of spent fuels and high-level waste (HLW) located at INEL [666 MTHM or equivalent waste units] is very small in comparison with the capacity of the Yucca Mountain Project repository (with a designed capacity of 70,000 MTHM, of which 10,000 MTHM have been designated for the defense program). To represent a repository of reasonable size, this PA will include the immobilized HLW from the other DOE complexes, such as West Valley Demonstration Project (WVDP), Savannah River Site (SRS), and Hanford (HANF).

Due to time constraints and the fact that data are not readily available on spent fuels located at these other DOE complexes, this draft PA evaluation includes only INEL spent fuels. However, the spent fuels from these DOE complexes could be easily incorporated into future

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PA studies if desired and most of the attributes of these fuels will be analyzed as this PA evaluates the disposal of the 90 INEL fuels.

The data on the immobilized HLW from other DOE complexes are extracted from the Characteristics of Potential Repository Wastes (DOE/RW-0184-R1), dated July 1992. The immobilized HLW mass from these other facilities is 12,640 metric tons, with a volume of 4706 m³.

A.2 Contaminants Independent of Spent Fuels and Waste Forms

A.2.1 Radionuclide Inventory

The radionuclide inventories for the INEL spent fuels, calcine, and liquid wastes are shown in Tables A.2.1.1 through A.2.1.4, and A.2.1.8 through A.2.1.13. The tables show the current, anticipated, and design radionuclide inventories. The anticipated radionuclide inventory is the projected radionuclide inventory of the spent fuel or HLW that needs to be disposed of in a repository by the year 2030. The design radionuclide inventory considers any uncertainty in the projection and includes a margin for the uncertainty. Note that for the anticipated HLW, all the liquid wastes will be converted into calcine. Thus, no liquid waste is included in the anticipated or design radionuclide inventory.

The current radionuclide inventories for the immobilized HLW from the other DOE complexes are shown in Tables A.2.1.5 through A.2.1.7. Inventories have been decayed to 2030 in the the PA calculations.

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Table A.2.1.1 Current Radionuclide Inventory by Spent Fuel and Waste Type (Curies in 1992)
for Graphite Fuels (See note a)

Radionuclide	Half Life (yrs)	40CFR191 Controls	> 20 yr Half Life (w/ alpha)	> 20 yr Half Life (other)	Total Curies
C-14	5730	Y		Y	1.297E+02
Cl-36	3.01E+05			Y	1.450E+00
Ni-59	7.50E+04			Y	1.011E+01
Ni-63	96			Y	2.842E+02
Se-79	6.50E+04			Y	1.571E+01
Rb-87	4.70E+10			Y	9.350E-04
Sr-90	29.12	Y		Y	2.575E+06
Zr-93	1.53E+06			Y	6.221E+02
Tc-99	2.10E+05	Y		Y	3.933E+02
Pd-107	6.50E+06			Y	4.617E-01
Cd-113	9.00E+05			Y	2.130E-12
Sn-126	1.00E+05	Y		Y	6.852E+00
I-129	1.57E+07	Y		Y	1.177E+00
Cs-135	2.30E+06	Y		Y	7.482E+00
Cs-137	30	Y		Y	2.693E+06
Ce-142	5.00E+10		Y		5.170E-05
Nd-144	2.00E+15		Y		1.051E-03
Sm-147	1.06E+11		Y		3.388E+00
Sm-151	90			Y	4.048E+04
Th-232	1.41E+10	Y	Y		2.720E+00
U-232	72		Y		1.680E+03
U-233	1.59E+05	Y	Y		2.304E+03
U-234	2.45E+05	Y	Y		2.941E+02
U-235	7.04E+08	Y	Y		1.371E+00
U-236	2.34E+07	Y	Y		1.134E+01
U-238	4.47E+09	Y	Y		3.156E-02
Np-237	2.14E+06	Y	Y		8.161E+00
Pu-238	87.74	Y	Y		4.974E+04
Pu-239	2.41E+04	Y	Y		1.145E+02
Pu-240	6537	Y	Y		2.067E+02
Am-241	432.2	Y	Y		1.989E+03
Am-242m	152		Y		2.838E-01
Cm-242	4.471E-01				6.092E+01
Am-243	7380	Y	Y		1.454E+01
Cm-243	28.5		Y		2.078E+00
Cm-244	1.812E+01				1.184E+03
Cm-245	8500		Y		1.295E-01
Cm-246	4730		Y		3.758E-02
Cm-247	1.56E+07		Y		1.110E-07
TOTAL					5.367E+06

a) The daughters of actinides not included; short lived daughters included with parents on fission products;
Nuclides under 20 years not included.

b) Based on ORIGEN-S (Ref. 6, 16)

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Table A.2.1.2 Current Radionuclide Inventory by Spent Fuels and Waste Type (Curies in 1992)
for Special Fuels (See Note a)

Radionuclide	Half Life (yrs)	40CFR191 Controls	> 20 yr Half Life (w/ alpha)	> 20 yr Half Life (other)	Total Curies
C-14	5.730E+03	Y		Y	3.092E+02
Cl-36	3.010E+05			Y	2.284E+00
Ni-59	7.600E+04			Y	1.039E+03
Ni-63	1.000E+02			Y	1.229E+05
Se-79	6.500E+04			Y	9.798E+01
Rb-87	4.800E+10			Y	1.084E-03
Sr-90	2.910E+01	Y		Y	1.522E+07
Y-90	7.310E-03			Y	1.522E+07
Zr-93	1.500E+06			Y	4.443E+02
Zr-95	1.753E-01			Y	1.215E+05
Nb-93m	1.610E+01				1.587E+02
Nb-94	2.000E+04			Y	8.983E-04
Nb-95	9.574E-02				2.793E+05
Tc-99	2.130E+05	Y		Y	3.160E+03
Ru-106	1.020E+00				1.996E+06
Rh-106	9.475E-07			Y	1.996E+06
Pd-107	6.500E+06			Y	2.307E+01
Cd-113m	1.410E+01				4.780E+02
Sn-126	1.000E+05	Y		Y	1.699E+02
Sb-125	2.758E+00				4.377E+05
Sb-126m	3.613E-05				1.699E+02
Sb-126	3.395E-02				2.378E+01
Te-125m	1.588E-01				1.068E+05
I-129	1.570E+07	Y		Y	7.216E+00
Cs-135	2.300E+06	Y		Y	7.323E+01
Cs-137	3.017E+01	Y		Y	2.040E+07
Ba-137m	4.852E-06				1.930E+07
Ce-141	8.898E-02				5.770E+01
Ce-144	7.792E-01				1.922E+07
Pr-144m	1.369E-05				2.302E+05
Pr-144	3.285E-05				1.922E+07
Pm-147	2.623E+00				8.970E+06
Sm-147	1.060E+11		Y		1.336E-04
Sm-151	9.000E+01			Y	8.125E+04
Ho-166m	1.200E+03			Y	1.192E-03
Tl-207	9.069E-06				8.303E-04
Tl-208	5.805E-06				5.202E-03
Tl-209	4.183E-06				2.337E-09
Pb-209	3.708E-04				1.410E-05
Pb-210	2.230E+01			Y	7.858E-05
Pb-211	6.864E-05				8.328E-04
Pb-212	1.214E-03				1.448E-02
Pb-214	5.134E-05				6.728E-05
Bi-210	1.372E-02				7.859E-06
Bi-211	4.069E-06				8.328E-04
Bi-212	1.151E-04				1.448E-02
Bi-213	8.670E-05				1.410E-05
Bi-214	3.784E-05				6.728E-05
Po-210	3.789E-01				7.858E-05
Po-211	1.635E-08				2.231E-08
Po-212	9.443E-15				9.279E-03
Po-213	1.268E-13				1.379E-05
Po-214	5.187E-12				6.726E-05
Po-215	5.641E-11				8.328E-04

Table A.2.1.2 Current Radionuclide Inventory by Spent Fuels and Waste Type (Curies in 1992)
for Special Fuels (Continued)

Radionuclide	Half Life (yrs)	40CFR191 Controls	> 20 yr Half Life (w/ alpha)	> 20 yr Half Life (other)	Total Curies
Po-216	4.595E-09				1.448E-02
Po-218	5.894E-06				6.728E-05
At-217	1.014E-09				1.410E-05
Rn-219	1.255E-07				8.328E-04
Rn-220	1.762E-06				1.448E-02
Rn-222	1.047E-02				6.728E-05
Fr-221	9.126E-06				1.410E-05
Fr-223	4.145E-05				1.099E-07
Ra-223	3.131E-02				8.328E-04
Ra-224	1.002E-02				1.448E-02
Ra-225	4.079E-02				1.410E-05
Ra-226	1.600E+03	Y	Y		6.728E-05
Ac-225	2.738E-02				1.410E-05
Ac-227	2.177E+01		Y		8.320E-04
Th-227	5.125E-02				8.213E-04
Th-228	1.913E+00				1.446E-02
Th-229	7.300E+03		Y		1.410E-05
Th-230	7.540E+04	Y	Y		2.543E-02
Th-231	2.910E-03				7.916E+00
Th-234	6.598E-02				6.358E+01
Pa-231	3.280E+04		Y		3.430E-03
Pa-233	7.392E-02				7.406E+01
Pa-234m	2.225E-06				6.358E+01
Pa-234	7.632E-04				1.069E-04
U-232	7.000E+01		Y		4.213E-02
U-233	1.592E+05	Y	Y		6.110E-03
U-234	2.460E+05	Y	Y		2.381E+02
U-235	7.040E+08	Y	Y		7.916E+00
U-236	2.342E+07	Y	Y		6.970E+01
U-237	1.848E-02				2.691E-02
U-238	4.470E+09	Y	Y		6.358E+01
U-240	1.609E-03				9.156E-08
Np-237	2.140E+06	Y	Y		7.406E+01
Np-239	6.448E-03				3.414E+03
Np-240m	1.373E-05				9.156E-08
Pu-238	8.770E+01	Y	Y		4.687E+05
Pu-239	2.410E+04	Y	Y		6.319E+04
Pu-240	6.560E+03	Y	Y		1.057E+05
Pu-241	1.410E+01				1.569E+07
Pu-242	3.750E+05	Y	Y		3.521E+02
Pu-244	8.000E+07		Y		9.166E-08
Am-241	4.327E+02	Y	Y		3.388E+05
Am-242m	1.410E+02		Y		6.365E-01
Am-242	1.828E-03				6.331E-01
Am-243	7.370E+03	Y	Y		3.414E+03
Cm-242	4.457E-01				7.391E+02
Cm-243	2.910E+01		Y		3.536E-01
Cm-244	1.810E+01				2.083E+05
Cm-245	8.500E+03		Y		4.383E-03
Cm-246	4.760E+03		Y		3.020E-04
TOTAL					1.398E+08

a) The total radionuclide inventory is based on ORIGEN-2 calculations of one HEU and one LEU fuel type [Ref. 21,22].

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Table A.2.1.3 Current Radionuclide Inventory by Spent Fuels and Waste Type (Curies in 1992)
for Liquid Waste (See Note a)

Radionuclide	Half Life (yrs)	40CFR191 Controls	> 20 yr Half Life (w/ alpha)	> 20 yr Half Life (other)	Total Curies
H-3	1.236E+01	Y		Y	1.133E+03
Co-60	5.273E+00				1.923E+03
Sr-90	29.12	Y		Y	1.058E+06
Tc-99	2.100E+05	Y		Y	8.461E+01
Ru-106	1.009E+00				2.391E+03
Sb-125	2.772E+00				2.059E+03
I-129	1.57E+07	Y		Y	4.775E+02
Cs-134	2.06				1.944E+04
Cs-137	30	Y		Y	1.162E+06
Ce-144	7.788E-01				2.551E+03
Eu-154	8.61				9.750E+03
Eu-155	4.963E+00				4.769E+03
U-233	1.59E+05	Y	Y		4.28E+00
U-234	2.45E+05	Y	Y		1.05E+02
U-235	7.04E+08	Y	Y		6.43E-01
U-236	2.34E+07	Y	Y		1.44E+00
U-238	4.47E+09	Y	Y		3.61E-02
Np-237	2.14E+06	Y	Y		2.580E+01
Pu-238	87.74	Y	Y		4.63E+03
Pu-239	2.41E+04	Y	Y		3.91E+02
Pu-240	6537	Y	Y		1.66E+02
Pu-242	3.76E+05	Y	Y		8.15E-02
Am-241	432.2	Y	Y		2.182E+04
Am-243	7380	Y	Y		3.278E+02
TOTAL					2.310E+06

- a) The radionuclide inventories listed here are based on samples of the various waste tanks. [Ref. 3]
b) Actinide ratios are based on alumina calcine data. [Ref 5]

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Table A.2.1.4 Current Radionuclide Inventory by Spent Fuels and Waste Type (Curies in 1992)
for Calcine Waste (See Note a)

Radionuclide	Half Life (yrs)	40CFR191 Controls	> 20 yr Half Life (w/ alpha)	> 20 yr Half Life (other)	Total Curies
Sr-90	29.12	Y		Y	1.061E+07
Y-90	7.31E-03				1.061E+07
Ru-106	1.01				1.162E+02
Cs-134	2.06				1.509E+04
Cs-137	30	Y		Y	1.204E+07
Ba-137m	--				1.131E+07
Ce-144	7.79E-01				1.020E+02
Pm-147	2.63				1.727E+05
Eu-154	8.61				3.860E+04
Th-230	7.700E+04	Y		Y	6.286E-02
Th-232	1.405E+10	Y	Y		1.886E-07
U-233	1.59E+05	Y	Y		2.17E+00
U-234	2.45E+05	Y	Y		5.33E+01
U-235	7.04E+08	Y	Y		3.26E-01
U-236	2.34E+07	Y	Y		7.28E-01
U-238	4.47E+09	Y	Y		1.83E-02
Np-237	2.14E+06	Y	Y		6.286E+02
Pu-238	87.74	Y	Y		1.018E+04
Pu-239	2.41E+04	Y	Y		8.617E+02
Pu-240	6537	Y	Y		3.661E+02
Pu-242	3.76E+05	Y	Y		1.817E-01
Am-241	432.2	Y	Y		1.257E+03
Am-243	7380	Y	Y		1.886E+01
TOTAL					4.512E+07

a) The radionuclide inventories listed here are based on activities of actual calcine samples (aluminum and zirconium) retrieved from the second bin set (Ref. 5) and a total calcine volume of 3800 m³. Additional inventory information contained in (Ref. 1).

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Table A.2.1.5 Current Estimated Radionuclide Inventory by Spent Fuels and Waste Type (Curies in 1992)
for West Valley Demonstration Project HLW Glass (see note b)

Radionuclide	Half Life (yrs)	40CFR191 Controls	> 20 yr Half Life (w/ alpha)	> 20 yr Half Life (other)	Total Curies
Fe-55	2.600E+00				7.590E+02
Co-60	5.273E+00				8.333E+02
Ni-59	7.500E+04			Y	1.144E+02
Ni-63	9.600E+01			Y	8.305E+03
Se-79	6.500E+04			Y	3.795E+00
Sr-90	2.912E+01	Y		Y	7.233E+06
Y-90	7.306E-03				7.233E+06
Zr-93	1.530E+06			Y	2.943E+02
Nb-93m	1.361E+01				1.966E+02
Tc-99	2.100E+05	Y		Y	1.177E+02
Ru-106	1.009E+00				1.524E+01
Rh-106	9.481E-07				1.524E+01
Pd-107	6.500E+06			Y	1.191E+01
Cd-113m	1.460E+01				2.294E+03
Sn-121m	5.001E+01			Y	1.887E+01
Sn-126	1.001E+05	Y		Y	1.125E+02
Sb-125	2.772E+00				7.865E+03
Sb-126	3.396E-02				1.576E+01
Sb-126m	3.615E-05				1.125E+02
Te-125m	1.589E-01				1.925E+03
Cs-134	2.063E+00				5.583E+03
Cs-135	2.300E+06	Y		Y	1.744E+02
Cs-137	3.000E+01	Y		Y	7.783E+06
Ba-137m	4.852E-06				7.370E+06
Ce-144	7.788E-01				7.040E-01
Pr-144	3.288E-05				7.040E-01
Pm-146	5.500E+00				1.172E+01
Pm-147	2.625E+00				9.488E+04
Sm-151	9.000E+01			Y	9.103E+04
Eu-152	1.361E+01				3.933E+02
Eu-154	8.606E+00				1.031E+05
Eu-155	4.963E+00				2.577E+04
Tl-207	9.069E-06				8.855E+00
Tl-208	5.835E-06				3.493E+00
Pb-209	3.767E-04				2.269E-01
Pb-211	6.868E-05				8.883E+00
Pb-212	1.214E-03				9.708E+00
Bi-211	4.059E-06				8.883E+00
Bi-212	1.152E-04				9.708E+00
Bi-213	8.685E-05				2.269E-01
Po-212	9.513E-15				6.215E+00
Po-213	1.332E-13				2.162E-01
Po-215	5.644E-11				8.883E+00
Po-216	4.756E-09				9.708E+00
At-217	1.024E-09				2.269E-01
Rn-219	1.256E-07				8.883E+00
Rn-220	1.763E-06				9.708E+00
Fr-221	9.132E-06				2.269E-01
Fr-223	4.148E-05				1.188E-01

Table A.2.1.5 Current Estimated Radionuclide Inventory by Spent Fuels and Waste Type (Curies in 1992)
for West Valley Demonstration Project HLW Glass (Continued) (See Note b)

Radionuclide	Half Life (yrs)	40CFR191 Controls	> 20 yr Half Life (w/ alpha)	> 20 yr Half Life (other)	Total Curies
Ra-223	3.133E-02				8.883E+00
Ra-224	1.003E-02				9.708E+00
Ra-225	4.056E-02				2.269E-01
Ra-228	60.700E+00				1.642E+00
Ac-225	2.740E-02				2.269E-01
Ac-227	2.177E+01		Y	Y	8.883E-02
Ac-228	6.998E-04				1.642E+00
Th-227	5.127E-02				8.745E+00
Th-228	1.914E+00				9.708E+00
Th-229	7.340E+03		Y		2.269E-01
Th-230	70.700E+04	Y		Y	6.490E-02
Th-231	2.913E-03				9.735E-02
Th-232	1.405E+10	Y	Y		1.774E+00
Pa-234	7.648E-04				8.635E-01
Pa-231	3.276E+04		Y		1.642E+01
Pa-233	7.398E-02				2.525E+01
Pa-234m	2.226E-06				8.635E-01
U-232	7.200E+01		Y		7.480E+00
U-233	1.585E+05	Y	Y		9.763E+00
U-234	2.445E+05	Y	Y		4.538E+00
U-235	7.038E+08	Y	Y		9.735E-02
U-236	2.342E+07	Y	Y		3.025E-01
U-238	4.468E+09	Y	Y		8.635E-01
Np-236	1.151E+05			Y	1.023E+01
Np-237	2.140E+06	Y	Y		2.525E+01
Np-239	6.453E-03				3.740E+02
Pu-236	2.853E+00				9.075E-01
Pu-238	8.774E+01	Y	Y		8.965E+03
Pu-239	2.407E+04	Y	Y		1.757E+03
Pu-240	6.537E+03	Y		Y	1.287E+03
Pu-241	1.441E+01				8.718E+04
Pu-242	3.763E+05	Y	Y		1.752E+00
Am-241	4.322E+02	Y	Y		5.775E+04
Am-242	1.829E-03				3.190E+02
Am-242m	1.520E+02		Y		3.218E+02
Am-243	7.380E+03	Y	Y		3.740E+02
Cm-242	4.471E-01				2.648E+02
Cm-243	2.850E+01		Y		1.449E+02
Cm-244	1.812E+01				8.250E+03
Cm-245	8.500E+03		Y		9.515E-01
Cm-246	4.730E+03		Y		1.081E-01

Total 3.014E+07

a) This table represents the radionuclide content of 275 canisters, each containing 1,900 kg of HLW glass having the radionuclide composition described in the WVDP Mass Balance Revision 7 (Crocker 1989). Radioactivity shown is as of the end of year 1989. Extracted from Characteristics of Potential Repository Wastes DOE/RW-0184-R1 Dated July 1992.

b) For this PA evaluation, radionuclide inventory assumed to be in 1992.

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Table A.2.1.6 Current Estimated Radionuclide Inventory by Spent Fuels and Waste Type (Curies in 1992)
for Savannah River Site HLW Glass (see note b)

Radionuclide	Half Life (yrs)	40CFR191 Controls	> 20 yr Half Life (w/ alpha)	> 20 yr Half Life (other)	Total Curies
Cr-51	7.591E-02				4.919E-13
Co-60	5.273E+00				8.974E+05
Ni-59	7.500E+04			Y	1.266E+02
Ni-63	9.600E+01			Y	1.571E+04
&Se-79	6.500E+04			Y	8.974E+02
Rb-87	4.699E+10			Y	4.605E-03
Sr-89	1.381E-01				2.254E-01
Sr-90	2.912E+01	Y		Y	2.469E+08
Y-90	7.306E-03				2.528E+08
Y-91	1.603E-01				3.997E+00
Zr-93	1.530E+06			Y	5.900E+03
Zr-95	1.753E-01				5.308E+01
Nb-94	2.031E+04			Y	5.095E-01
Nb-95	9.630E-02				1.117E+02
Nb-95m	9.887E-03				6.587E-01
Tc-99	2.100E+05	Y		Y	1.626E+04
Ru-103	1.076E-01				8.895E-05
Ru-106	1.009E+00				1.190E+07
Rh-103m	1.068E-04				8.641E-05
Rh-106	9.481E-07				1.193E+07
Pd-107	6.500E+06			Y	7.780E+01
Ag-110m	6.846E-01				6.645E+02
Cd-113	9.000E+05				2.646E-10
Cd-115m	1.222E-01				6.407E-06
Sn-121m	5.001E+01			Y	4.174E+02
Sn-123	3.539E-01				1.346E+03
Sn-126	1.001E+05	Y		Y	2.332E+03
Sb-124	1.649E-01				3.762E-04
Sb-125	2.772E+00				4.488E+06
Sb-126	3.396E-02				3.253E+02
Sb-126m	3.615E-05				2.332E+03
Te-125m	1.589E-01				1.458E+06
Te-127	1.060E-03				6.349E+02
Te-127m	2.986E-01				6.486E+02
Te-129	1.324E-04				1.613E-08
Te-129m	9.205E-02				2.508E-08
Cs-134	2.063E+00				1.781E+06
Cs-135	2.300E+06	Y		Y	5.252E+02
Cs-136	3.590E-02				4.140E-36
Cs-137	3.000E+01	Y		Y	2.293E+08
Ba-136m	9.767E-09				4.546E-35
Ba-137m	4.852E-06				2.195E+08
Ba-140	3.504E-02				5.409E-33
La-140	4.592E-03				2.273E-33
Ce-141	8.907E-02				1.897E-07
Ce-142	1.050E+11			Y	5.075E-02
Ce-144	7.788E-01				5.213E+07
Pr-143	3.716E-02				6.328E-31
Pr-144	3.288E-05				5.213E+07
Pr-144m	1.370E-05				6.270E+05

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Table A.2.1.6 Current Estimated Radionuclide Inventory by Spent Fuels and Waste Type (Curies in 1992)
for Savannah River Site HLW Glass (Continued)(see note b)

Radionuclide	Half Life (yrs)	40CFR191 Control	> 20 yr Half Life (w/ alpha)	> 20 yr Half Life (other)	Total Curies
Nd-144	2.100E+15		Y		2.567E-05
Nd-147	3.030E-02				6.661E-41
Pm-147	2.625E+00				1.278E+08
Pm-148	1.471E-02				3.684E-07
Pm-148m	1.131E-01				5.330E-06
Sm-147	1.071E+11		Y		1.056E-02
Sm-148	8.007E+15		Y		3.057E-08
Sm-149	1.000E+16		Y		9.407E-09
Sm-151	9.000E+01			Y	1.309E+06
Eu-152	1.361E+01				1.948E+04
Eu-154	8.606E+00				3.273E+06
Eu-155	4.963E+00				2.508E+06
Eu-156	4.160E-02				2.763E-28
Tb-160	1.981E-01				5.916E-03
Tl-208	5.835E-06				5.958E+00
U-232	7.200E+01		Y		7.073E+01
U-233	1.585E+05	Y	Y		8.367E-03
U-234	2.445E+05	Y	Y		1.811E+02
U-235	7.038E+08	Y	Y		8.309E-01
U-236	2.342E+07	Y	Y		5.958E+00
U-238	4.468E+09	Y	Y		5.546E+01
Np-236	1.151E+05			Y	9.212E-05
Np-237	2.140E+06	Y	Y		4.703E+01
Pu-236	2.853E+00				6.449E+02
Pu-237	1.249E-01				4.723E-08
Pu-238	8.774E+01	Y	Y		7.838E+06
Pu-239	2.407E+04	Y	Y		6.819E+04
Pu-240	6.537E+03	Y	Y		4.585E+04
Pu-241	1.441E+01				8.821E+06
Pu-242	3.763E+05	Y	Y		6.465E+01
Am-241	4.322E+02	Y	Y		5.821E+04
Am-242	1.829E-03				7.585E+01
Am-242m	1.520E+02		Y		7.643E+01
Am-243	7.380E+03	Y	Y		3.057E+01
Cm-242	4.471E-01				1.846E+02
Cm-243	2.850E+01		Y		2.939E+01
Cm-244	1.812E+01				5.683E+05
Cm-245	8.500E+03		Y		3.547E-02
Cm-246	4.730E+03		Y		2.822E-03
Cm-247	1.561E+07		Y		3.488E-09
Cm-248	3.393E+05		Y		3.626E-09

Totals

1.238E+09

a) Quantities shown are for sludge-precipitate glass and are extracted from Characteristics of Potential Repository Wastes DOE/RW-0184-R1 Dated July 1992, assuming sludge aged an average of 5 years and supernate aged an average of 15 years, with a total of 5282 canisters each containing 3,710 lb of glass (1,682 kg). Radionuclide contents are at time of filling canister.

b) For this PA evaluation, radionuclide inventory assumed to be in 1992.

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Table A.2.1.7 Current Estimated Radionuclide Inventory by Spent Fuels and Waste Type (Curies in 1992)
for Hanford HLW Glass (see note b)

Radionuclide	Half Life (yrs)	40CFR191 Controls	> 20 yr Half Life (w/ alpha)	> 20 yr Half Life (other)	Total Curies
Fe-55	2.600E+00				2.764E+05
Ni-59	7.500E+04			Y	2.666E+02
Co-60	5.273E+00				8.408E+03
Ni-63	9.600E+01			Y	3.077E+04
Se-79	6.500E+04			Y	7.644E+00
Sr-89	1.381E-01				1.278E+02
Sr-90	2.912E+01	Y		Y	8.193E+07
Y-90	7.306E-03				8.193E+07
Y-91	1.603E-01				1.423E+03
Nb-93m	1.361E+01				1.131E+03
Zr-93	1.530E+06			Y	2.178E+03
Zr-95	1.753E-01				5.410E+03
Nb-95	9.630E-02				1.111E+04
Tc-99	2.100E+05	Y		Y	1.833E+04
Ru-103	1.076E-01				5.841E-01
Rh-103m	1.068E-04				5.272E-01
Ru-106	1.009E+00				9.780E+06
Rh-106	9.481E-07				9.780E+06
Pd-107	6.500E+06			Y	7.977E+01
Ag-110m	6.846E-01				3.116E+03
Cd-113m	1.460E+01				2.862E+04
In-113m	1.893E-04				4.939E+01
Sn-113	3.154E-01				4.939E+01
Cd-115m	1.222E-01				1.721E-02
Sn-119m	6.713E-01				1.062E+04
Sn-121m	5.001E+01			Y	2.078E+02
Sn-123	3.539E-01				5.664E+03
Sn-126	1.001E+05	Y		Y	9.016E+02
Sb-124	1.649E-01				6.860E-02
Sb-126	3.396E-02				1.270E+02
Sb-126m	3.615E-05				9.016E+02
Sb-125	2.772E+00				3.450E+06
Te-125m	1.589E-01				8.408E+05
Te-127	1.060E-03				5.782E+03
Te-127m	2.986E-01				5.880E+03
Te-129	1.324E-04				7.350E-04
Te-129m	9.205E-02				1.131E-03
I-129	1.571E+07			Y	3.195E-02
Cs-134	2.063E+00				2.352E+06
Cs-135	2.300E+06	Y		Y	4.920E+02
Cs-137	3.000E+01	Y		Y	9.996E+07
Ba-137m	4.852E-06				9.447E+07
Ce-141	8.907E-02				2.215E-02
Ce-144	7.788E-01				5.841E+07
Pr-144	3.288E-05				5.841E+07
Pr-144m	1.370E-05				7.017E+05
Pm-147	2.625E+00				7.781E+07
Pm-148m	1.131E-01				2.587E-02
Sm-151	9.000E+01			Y	1.639E+06
Eu-152	1.361E+01				5.370E+03
Gd-153	6.631E-01				2.254E+01
Eu-154	8.606E+00				6.586E+05
Eu-155	4.963E+00				8.056E+05

Table A.2.1.7 Current Estimated Radionuclide Inventory by Spent Fuels and Waste Type (Curies in 1992)
for Hanford HLW Glass (Continued)(see note b)

Radionuclide	Half Life (yrs)	40CFR191 Controls	> 20 yr Half Life (w/ alpha)	> 20 yr Half Life (other)	Total Curies
Tb-160	1.981E-01				2.156E-01
U-234	2.445E+05	Y	Y		9.447E+00
U-235	7.038E+08	Y	Y		3.861E-01
U-236	2.342E+07	Y	Y		9.310E-01
U-238	4.468E+09	Y	Y		7.291E+00
Np-237	2.140E+06	Y	Y		3.900E+02
Pu-238	8.774E+01	Y	Y		1.505E+03
Pu-239	2.407E+04	Y	Y		2.764E+03
Pu-240	6.537E+03	Y	Y		1.062E+03
Pu-241	1.441E+01				5.057E+04
Pu-242	3.763E+05	Y	Y		2.568E-01
Am-241	4.322E+02	Y	Y		1.131E+06
Am-242	1.829E-03		Y		8.114E+02
Am-243	7.380E+03	Y	Y		1.325E+02
Cm-242	4.471E-01				9.780E+02
Cm-244	1.812E+01				2.450E+04
Total					5.841E+08

- a) This table identifies the maximum expected activity of 1,960 HWVP canisters at the time of verification. The maximum is principally based on close-coupling the final accumulated tank of NCAW (21 months from fuel discharge- to HWVP). Canister contains 1,650 kg of HLW glass (85 percent fill). Extracted from Characteristics of Potential Repository Wastes DOE/RW-0184-R1 Dated July 1992.
- b) For this PA evaluation, radionuclide inventory assumed to be in 1992.

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Table A.2.1.8 Anticipated Radionuclide Inventory by Spent Fuel and Waste Type (Curies in 2030)
for Graphite Fuels

Radionuclide	Half Life (yrs)	40CFR191 Controls	> 20 yr Half Life (w/ alpha)	> 20 yr Half Life (other)	Total Curies
C-14	5730	Y		Y	1.294E+02
Cl-36	3.01E+05			Y	1.453E+00
Ni-59	7.50E+04			Y	1.014E+01
Ni-63	96			Y	2.190E+02
Se-79	6.50E+04			Y	1.575E+01
Rb-87	4.70E+10			Y	9.380E-04
Sr-90	29.12	Y		Y	1.025E+06
Zr-93	1.53E+06			Y	6.238E+02
Tc-99	2.10E+05	Y		Y	3.943E+02
Pd-107	6.50E+06			Y	4.629E-01
Cd-113	9.00E+05			Y	2.140E-12
Sn-126	1.00E+05	Y		Y	6.865E+00
I-129	1.57E+07	Y		Y	1.180E+00
Cs-135	2.30E+06	Y		Y	7.502E+00
Cs-137	30	Y		Y	1.123E+06
Ce-142	5.00E+16		Y		5.180E-05
Nd-144	2.00E+15		Y		1.054E-03
Sm-147	1.06E+11		Y		3.396E+00
Sm-151	90			Y	3.028E+04
Th-232	1.41E+10	Y	Y		2.725E+00
U-232	72		Y		1.149E+03
U-233	1.59E+05	Y	Y		2.310E+03
U-234	2.45E+05	Y	Y		2.948E+02
U-235	7.04E+08	Y	Y		1.373E+00
U-236	2.34E+07	Y	Y		1.137E+01
U-238	4.47E+09	Y	Y		3.163E-02
Np-237	2.14E+06	Y	Y		8.187E+00
Pu-238	87.74	Y	Y		3.702E+04
Pu-239	2.41E+04	Y	Y		1.147E+02
Pu-240	6537	Y	Y		2.064E+02
Am-241	432.2	Y	Y		1.877E+03
Am-242m	152		Y		2.362E-01
Am-243	7380	Y	Y		1.452E+01
Cm-242	4.471E-01				4.526E+01
Cm-243	28.5		Y		8.277E-01
Cm-244	1.812E+01				2.772E+02
Cm-245	8500		Y		1.294E-01
Cm-246	4730		Y		3.744E-02
Cm-247	1.56E+07		Y		1.110E-07
TOTAL					2.223E+06

Table A.2.1.9 Anticipated Radionuclide Inventory by Spent Fuels and Waste Type (Curies in 2030)
for Special Fuels

Radionuclide	Half Life (yrs)	40CFR191 Controls	> 20 yr Half Life (w/ alpha)	> 20 yr Half Life (other)	Total Curies
C-14	5.730E+03	Y		Y	3.084E+02
Cl-36	3.010E+05			Y	2.284E+00
Ni-59	7.600E+04			Y	1.039E+03
Ni-63	1.000E+02			Y	1.057E+05
Se-79	6.500E+04			Y	1.075E+02
Rb-87	4.800E+10			Y	1.726E-03
Sr-90	2.910E+01	Y		Y	1.008E+07
Y-90	7.310E-03			Y	1.008E+07
Zr-93	1.500E+06			Y	4.935E+02
Zr-95	1.753E-01			Y	8.417E-27
Nb-93m	1.610E+01				3.725E+02
Nb-94	2.000E+04			Y	1.430E-03
Nb-95	9.574E-02				1.869E-26
Tc-99	2.130E+05	Y		Y	3.480E+03
Ru-106	1.020E+00				4.813E+00
Rh-106	9.475E-07			Y	4.813E+00
Pd-107	6.500E+06			Y	2.344E+01
Cd-113m	1.410E+01				1.992E+02
Sn-126	1.000E+05	Y		Y	1.785E+02
Sb-125	2.758E+00				2.923E+03
Sb-126m	3.613E-05				1.785E+02
Sb-126	3.395E-02				2.600E+01
Te-125m	1.588E-01				7.133E+02
I-129	1.570E+07	Y		Y	7.746E+00
Cs-135	2.300E+06	Y		Y	7.573E+01
Cs-137	3.017E+01	Y		Y	1.353E+07
Ba-137m	4.852E-06				1.280E+07
Ce-141	8.898E-02				4.571E-60
Ce-144	7.792E-01				1.242E+00
Pr-144m	1.369E-05				1.490E-02
Pr-144	3.285E-05				1.242E+00
Pm-147	2.623E+00				4.602E+04
Sm-147	1.060E+11		Y		4.856E-04
Sm-151	9.000E+01			Y	7.549E+04
Ho-166m	1.200E+03			Y	1.866E-03
Tl-207	9.069E-06				4.290E-03
Tl-208	5.805E-06				4.473E-02
Tl-209	4.183E-06				1.406E-07
Pb-209	3.708E-04				3.627E-05
Pb-210	2.230E+01			Y	1.262E-04
Pb-211	6.864E-05				4.302E-03
Pb-212	1.214E-03				1.245E-01
Pb-214	5.134E-05				4.812E-04
Bi-210	1.372E-02				1.262E-04
Bi-211	4.069E-06				4.302E-03
Bi-212	1.151E-04				1.245E-01
Bi-213	8.670E-05				3.627E-05
Bi-214	3.784E-05				4.812E-04
Po-210	3.789E-01				1.262E-04
Po-211	1.635E-08				5.527E-06
Po-212	9.443E-15				7.978E-02
Po-213	1.268E-13				3.549E-05
Po-214	5.187E-12				4.810E-04
Po-215	5.641E-11				4.302E-03

Table A.2.1.9 Anticipated Radionuclide Inventory by Spent Fuels and Waste Type (Curies in 2030)
for Special Fuels (Continued)

Radionuclide	Half Life (yrs)	40CFR191 Controls	> 20 yr Half Life (w/ alpha)	> 20 yr Half Life (other)	Total Curies
Po-216	4.595E-09				1.245E-01
Po-218	5.894E-06				4.812E-04
At-217	1.014E-09				3.627E-05
Rn-219	1.255E-07				4.302E-03
Rn-220	1.762E-06				1.245E-01
Rn-222	1.047E-02				4.812E-04
Fr-221	9.126E-06				3.627E-05
Fr-223	4.145E-05				2.719E-05
Ra-223	3.131E-02				4.302E-03
Ra-224	1.002E-02				1.245E-01
Ra-225	4.079E-02				3.627E-05
Ra-226	1.600E+03	Y	Y		4.812E-04
Ac-225	2.738E-02				3.627E-05
Ac-227	2.177E+01		Y		4.297E-03
Th-227	5.125E-02				4.242E-03
Th-228	1.913E+00				1.244E-01
Th-229	7.300E+03		Y		3.627E-05
Th-230	7.540E+04	Y	Y		7.104E-02
Th-231	2.910E-03				1.057E+01
Th-234	6.598E-02				6.363E+01
Pa-231	3.280E+04		Y		9.607E-03
Pa-233	7.392E-02				8.396E+01
Pa-234m	2.225E-06				6.363E+01
Pa-234	7.632E-04				1.703E-04
U-232	7.000E+01		Y		1.219E-01
U-233	1.592E+05	Y	Y		1.445E-02
U-234	2.460E+05	Y	Y		2.647E+02
U-235	7.040E+08	Y	Y		1.057E+01
U-236	2.342E+07	Y	Y		8.069E+01
U-237	1.848E-02				1.103E-02
U-238	4.470E+09	Y	Y		6.363E+01
U-240	1.609E-03				1.459E-07
Np-237	2.140E+06	Y	Y		8.400E+01
Np-239	6.448E-03				3.407E+03
Np-240m	1.373E-05				1.459E-07
Pu-238	8.770E+01	Y	Y		4.097E+05
Pu-239	2.410E+04	Y	Y		6.352E+04
Pu-240	6.560E+03	Y	Y		1.589E+03
Pu-241	1.410E+01				5.993E+06
Pu-242	3.750E+05	Y	Y		3.524E+02
Pu-244	8.000E+07		Y		1.460E-07
Am-241	4.327E+02	Y	Y		6.474E+05
Am-242m	1.410E+02		Y		8.829E-01
Am-242	1.828E-03				8.786E-01
Am-243	7.370E+03	Y	Y		3.407E+03
Cm-242	4.457E-01				5.419E+02
Cm-243	2.910E+01		Y		2.756E-01
Cm-244	1.810E+01				9.686E+04
Cm-245	8.500E+03		Y		6.965E-03
Cm-246	4.760E+03		Y		4.789E-04
TOTAL					5.395E+07

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Table A.2.1.10 Anticipated Radionuclide Inventory by Spent Fuels and Waste Type (Curies in 2030)
for Calcine Waste

Radionuclide	Half Life (yrs)	40CFR191 Controls	> 20 yr Half Life (w/ alpha)	> 20 yr Half Life (other)	Total Curies
Sr-90	29.12	Y		Y	4.710E+06
Y-90	7.31E-03				4.710E+06
Ru-106	1.01				1.320E-15
Cs-134	2.06				4.280E-07
Cs-137	30	Y		Y	5.490E+06
Ba-137	--				5.190E+06
Ce-144	7.79E-01				2.120E-18
Pm-147	2.63				7.330E-05
Eu-154	8.61				1.940E+03
Th-230	70.700E+04	Y		Y	6.915E-02
Th-232	1.405E+10	Y	Y		2.074E-07
U-233	1.59E+05	Y	Y		3.33E+00
U-234	2.45E+05	Y	Y		8.18E+01
U-235	7.04E+08	Y	Y		5.01E-01
U-236	2.34E+07	Y	Y		1.12E+00
U-238	4.47E+09	Y	Y		2.81E-02
Np-237	2.14E+06	Y	Y		6.915E+02
Pu-238	87.74	Y	Y		4.23E+04
Pu-239	2.41E+04	Y	Y		3.57E+03
Pu-240	6537	Y	Y		1.52E+03
Pu-242	3.76E+05	Y	Y		7.45E-01
Am-241	432.2	Y	Y		1.383E+03
Am-243	7380	Y	Y		2.704E+01
TOTAL					2.045E+07

Table A.2.1.11 Design Radionuclide Inventory by Spent Fuel and Waste Type
for Graphite Fuels

Radionuclide	Half Life (yrs)	40CFR191 Controls	> 20 yr Half Life (w/ alpha)	> 20 yr Half Life (other)	Total Curies
C-14	5730	Y		Y	1.682E+02
Cl-36	3.01E+05			Y	1.889E+00
Ni-59	7.50E+04			Y	1.318E+01
Ni-63	96			Y	2.847E+02
Se-79	6.50E+04			Y	2.047E+01
Rb-87	4.70E+10			Y	1.219E-03
Sr-90	29.12	Y		Y	1.332E+06
Zr-93	1.53E+06			Y	8.109E+02
Tc-99	2.10E+05	Y		Y	5.125E+02
Pd-107	6.50E+06			Y	6.012E-01
Cd-113	9.00E+05			Y	2.782E-12
Sn-126	1.00E+05	Y		Y	8.925E+00
I-129	1.57E+07	Y		Y	1.533E+00
Cs-135	2.30E+06	Y		Y	9.752E+00
Cs-137	30	Y		Y	1.460E+06
Ce-142	5.00E+16		Y		6.734E-05
Nd-144	2.00E+15		Y		1.370E-03
Sm-147	1.06E+11		Y		4.415E+00
Sm-151	90			Y	3.937E+04
Th-232	1.41E+10	Y	Y		3.543E+00
U-232	72		Y		1.494E+03
U-233	1.59E+05	Y	Y		3.002E+03
U-234	2.45E+05	Y	Y		3.833E+02
U-235	7.04E+08	Y	Y		1.784E+00
U-236	2.34E+07	Y	Y		1.478E+01
U-238	4.47E+09	Y	Y		4.112E-02
Np-237	2.14E+06	Y	Y		1.064E+01
Pu-238	87.74	Y	Y		4.813E+04
Pu-239	2.41E+04	Y	Y		1.491E+02
Pu-240	6537	Y	Y		2.683E+02
Am-241	432.2	Y	Y		2.440E+03
Am-242m	152		Y		3.071E-01
Am-243	7380	Y	Y		1.888E+01
Cm-242	4.471E-01				5.884E+01
Cm-243	28.5		Y		1.076E+00
Cm-244	1.812E+01				3.603E+02
Cm-245	8500		Y		1.682E-01
Cm-246	4730		Y		4.867E-02
Cm-247	1.56E+07		Y		1.443E-07

TOTAL

2.890E+06

a) The design radionuclide inventory is equivalent to 35.1 MTHM vs 27 MTHM anticipated.

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Table A.2.1.12 Design Radionuclide Inventory by Spent Fuels and Waste Type
for Special Fuels

Radionuclide	Half Life (yrs)	40CFR191 Controls	> 20 yr Half Life (w/ alpha)	> 20 yr Half Life (other)	Total Curies
C-14	.730E+03	Y		Y	3.855E+02
Cl-36	3.010E+05			Y	2.855E+00
Ni-59	7.600E+04			Y	1.298E+03
Ni-63	1.000E+02			Y	1.321E+05
Se-79	6.500E+04			Y	1.344E+02
Rb-87	4.800E+10			Y	2.158E-03
Sr-90	2.910E+01	Y		Y	1.260E+07
Y-90	7.310E-03			Y	1.260E+07
Zr-93	1.500E+06			Y	6.169E+02
Zr-95	1.753E-01			Y	1.052E-26
Nb-93m	1.610E+01				4.656E+02
Nb-94	2.000E+04			Y	1.787E-03
Nb-95	9.574E-02				2.336E-26
Tc-99	2.130E+05	Y		Y	4.350E+03
Ru-106	1.020E+00				6.017E+00
Rh-106	9.475E-07			Y	6.017E+00
Pd-107	6.500E+06			Y	2.930E+01
Cd-113m	1.410E+01				2.490E+02
Sn-116	1.000E+05	Y		Y	2.231E+02
Sb-125	2.758E+00				3.654E+03
Sb-126m	3.613E-05				2.231E+02
Sb-126	3.395E-02				3.251E+01
Te-125m	1.588E-01				8.916E+02
I-129	1.570E+07	Y		Y	9.683E+00
Cs-135	2.300E+06	Y		Y	9.466E+01
Cs-137	3.017E+01	Y		Y	1.691E+07
Ba-137m	4.852E-06				1.600E+07
Ce-141	8.898E-02				5.713E-60
Ce-144	7.792E-01				1.552E+00
Pr-144m	1.369E-05				1.862E-02
Pr-144	3.285E-05				1.552E+00
Pm-147	2.623E+00				5.753E+04
Sm-147	1.060E+11		Y		6.070E-04
Sm-151	9.000E+01			Y	9.437E+04
Ho-166m	1.200E+03			Y	2.333E-03
Ti-207	9.069E-06				5.362E-03
Ti-208	5.805E-06				5.591E-02
Ti-209	4.183E-06				1.757E-07
Pb-209	3.708E-04				4.534E-05
Pb-210	2.230E+01			Y	1.577E-04
Pb-211	6.864E-05				5.377E-03
Pb-212	1.214E-03				1.556E-01
Pb-214	5.134E-05				6.016E-04
Bi-210	1.372E-02				1.577E-04
Bi-211	4.069E-06				5.377E-03
Bi-212	1.151E-04				1.556E-01
Bi-213	8.670E-05				4.534E-05
Bi-214	3.784E-05				6.016E-04
Po-210	3.789E-01				1.577E-04
Po-211	1.635E-08				6.909E-06
Po-212	9.443E-15				9.973E-02
Po-213	1.268E-13				4.436E-05
Po-214	5.187E-12				6.013E-04
Po-215	5.641E-11				5.377E-03

Table A.2.1.12 Design Radionuclide Inventory by Spent Fuels and Waste Type
for Special Fuels (Continued)

Radionuclide	Half Life (yrs)	40CFR191 Controls	> 20 yr Half Life (w/ alpha)	> 20 yr Half Life (other)	Total Curies
Po-216	4.595E-09				1.556E-01
Po-218	5.894E-06				6.016E-04
At-217	1.014E-09				4.534E-05
Rn-219	1.255E-07				5.377E-03
Rn-220	1.762E-06				1.556E-01
Rn-222	1.047E-02				6.016E-04
Fr-221	9.126E-06				4.534E-05
Fr-223	4.145E-05				3.399E-05
Ra-223	3.131E-02				5.377E-03
Ra-224	1.002E-02				1.556E-01
Ra-225	4.079E-02				4.534E-05
Ra-226	1.600E+03	Y	Y		6.016E-04
Ac-225	2.738E-02				4.534E-05
Ac-227	2.177E+01		Y		5.371E-03
Th-227	5.125E-02				5.303E-03
Th-228	1.913E+00				1.555E-01
Th-229	7.300E+03		Y		4.534E-05
Th-230	7.540E+04	Y	Y		8.880E-02
Th-231	2.910E-03				1.322E+01
Th-234	6.598E-02				7.954E+01
Pa-231	3.280E+04		Y		1.201E-02
Pa-233	7.392E-02				1.049E+02
Pa-234m	2.225E-06				7.954E+01
Pa-234	7.632E-04				2.129E-04
U-232	7.000E+01		Y		1.524E-01
U-233	1.592E+05	Y	Y		1.807E-02
U-234	2.460E+05	Y	Y		3.309E+02
U-235	7.040E+08	Y	Y		1.322E+01
U-236	2.342E+07	Y	Y		1.009E+02
U-237	1.848E-02				1.379E-02
U-238	4.470E+09	Y	Y		7.954E+01
U-240	1.609E-03				1.823E-07
Np-237	2.140E+06	Y	Y		1.050E+02
Np-239	6.448E-03				4.259E+03
Np-240m	1.373E-05				1.823E-07
Pu-238	8.770E+01	Y	Y		5.121E+05
Pu-239	2.410E+04	Y	Y		7.940E+04
Pu-240	6.560E+03	Y	Y		1.986E+03
Pu-241	1.410E+01				7.491E+06
Pu-242	3.750E+05	Y	Y		4.405E+02
Pu-244	8.000E+07		Y		1.826E-07
Am-241	4.327E+02	Y	Y		8.092E+05
Am-242m	1.410E+02		Y		1.104E+00
Am-242	1.828E-03				1.098E+00
Am-243	7.370E+03	Y	Y		4.259E+03
Cm-242	4.457E-01				6.774E+02
Cm-243	2.910E+01		Y		3.446E-01
Cm-244	1.810E+01				1.211E+05
Cm-245	8.500E+03		Y		8.706E-03
Cm-246	4.760E+03		Y		5.987E-04

TOTAL

6.743E+07

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Table A.2.1.13 Design Radionuclide Inventory by Spent Fuels and Waste Type
for Calcine Waste

Radionuclide	Half Life (yrs)	40CFR191 Controls	> 20 yr Half Life (w/ alpha)	> 20 yr Half Life (other)	Total Curies
Sr-90	29.12	Y		Y	5.888E+06
Y-90	7.31E-03				5.888E+06
Ru-106	1.01				1.650E-15
Cs-134	2.06				5.350E-07
Cs-137	30	Y		Y	6.863E+06
Ba-137m	--				6.488E+06
Ce-144	7.79E-01				2.650E-18
Pm-147	2.63				9.163E-05
Eu-154	8.61				2.425E+03
Th-230	70.700E+04	Y		Y	8.644E-02
Th-232	1.405E+10	Y	Y		2.593E-07
U-233	1.59E+05	Y	Y		4.16E+00
U-234	2.45E+05	Y	Y		1.02E+02
U-235	7.04E+08	Y	Y		6.26E-01
U-236	2.34E+07	Y	Y		1.40E+00
U-238	4.47E+09	Y	Y		3.51E-02
Np-237	2.14E+06	Y	Y		8.644E+02
Pu-238	87.74	Y	Y		5.28E+04
Pu-239	2.41E+04	Y	Y		4.46E+03
Pu-240	6537	Y	Y		1.90E+03
Pu-242	3.76E+05	Y	Y		9.31E-01
Am-241	432.2	Y	Y		1.729E+03
Am-243	7380	Y	Y		3.380E+01
TOTAL					2.557E+07

A.2.2 Radionuclide Chains and Half-Lives

The decay chains for the ICPP spent fuels and HLW are shown in Figures A.2.2.1, A.2.2.2, A.2.2.3, and A.2.2.4. The half-lives for each radionuclide, as listed in the literature by International Commission on Radiological Protection Publication 38 (ICRP, Pub 38, 1983), are shown on Figures A.2.2.1, A.2.2.2, A.2.2.3, and A.2.2.4.

A.2.3 Release Limits and Waste Unit Factor

A.2.3.1 40 CFR 191 Release Limits

40 CFR 191 Appendix A Table 1 provide the release limits per 1,000 metric tons of heavy metal (MTHM) or other unit of waste. This Table and Notes have been reproduced below.

40 CFR 191 Appendix A-Table for Subpart B

Table 1 Release Limits For Containment Requirements

[Cumulative releases to the accessible environment for 10,000 years after disposal]

Radionuclide	Release limits Per 1,000 MTHM or other units of waste(see note) (Ci)
Americium (Am) -241 or -243.....	100
Carbon (C) -14.....	100
Cesium (Cs) -135 or -137.....	1000
Iodine (I) -129.....	100
Neptunium (Np) -237.....	100
Plutonium (Pu) -238, -239, -240, or -242.....	100
Radium (Ra) -226.....	100
Strontium (Sr) -90.....	1000
Technetium (Tc) -99.....	10000
Thorium (Th) -230 or -232.....	10
Tin (Sn) -126.....	1000
Uranium (U) -233, -234, -235, -236, or -238.....	100
Any other α -emitting radionuclide with $t_{1/2} > 20$ years.....	100
Any other non α -emitting radionuclide with $t_{1/2} > 20$ years.....	1000

Application of 40 CFR 191 Appendix A-Subpart B Table 1

Note 1: Units of Waste. The release limits in Table 1 apply to the amount of wastes in any one of the following:

- (a) An amount of spent nuclear fuel containing 1,000 MTHM exposed to a burnup between 25,000 megawatt-days per metric ton of heavy metal (MWd/MTHM) and 40,000 Mwd/MTHM;
- (b) The high-level radioactive wastes generated from reprocessing

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each 1,000 MTHM exposed to a burnup between 25,000 megawatt-days per metric ton of heavy metal (Mwd/MTHM) and 40,000 Mwd/MTHM;

- (c) Each 100,000,000 curies of gamma or beta-emitting radionuclides with half-lives greater than 20 years but less than 100 years (for use as discussed in Note 5 or with materials that are identified by the commission as high-level radioactive waste in accordance with part B of the definition of the high-level waste in the NWPB);
- (d) Each 1,000,000 curies of other radionuclides such as gamma or beta-emitters with half lives greater than 100 years or any alpha-emitters with half-lives greater than 20 years (for use as discussed in Note 5 or with material that are identified by the commission as high-level radioactive waste in accordance with part B of the definition of the high-level waste in the NWPB); or
- (e) An amount of transuranic (TRU) wastes containing one million curies of alpha-emitting transuranic radionuclides with half lives greater than 20 years.

Note 2: Release Limits for Specific Disposal Systems. To develop Release Limits for a particular disposal system, the quantities in Table 1 shall be adjusted for the amount of waste included in the disposal system compared to the various units of waste defined in Note 1.

Note 3: Adjustments for Reactor Fuels with Different Burnup. For disposal systems containing reactor fuels (or high-level wastes from reactor fuels) exposed to an average burnup of less than 25,000 Mwd/MTHM or greater than 40,000 Mwd/MTHM, the units of waste defined in (a) and (b) of Note 1 shall be adjusted. The unit shall be multiplied by the ratio of 30,000 Mwd/MTHM divided by the fuel's actual average burnup, except that a value of 5,000 Mwd/MTHM may be used when the average fuel burnup is below 5,000 Mwd/MTHM and a value of 100,000 Mwd/MTHM shall be used when the average fuel burnup is above 100,000 Mwd/MTHM. This adjusted unit of waste shall then be used in determining the Release Limits for the disposal system.

Note 4: Treatment of Fractional High-Level Wastes. In some cases, a high-level waste stream from reprocessing spent nuclear fuel may have been (or will be) separated into two or more high-level waste components destined for different disposal systems. In such cases, the implementing agency may allocate the Release Limit multiplier (based upon the original MTHM and the average fuel burnup of the high-level waste stream) among the various disposal systems as it chooses, provided that the total Release Limit multiplier used for that waste stream at all of its disposal systems may not exceed the Release Limit multiplier that would be used if the entire waste stream were disposed of in one disposal system.

Note 5: Treatment of Wastes with Poorly Known Burnup or Original MTHM. In some cases, the records associated with particular high-level waste streams may not be adequate to accurately determine the original metric tons of heavy metal in the reactor fuel that created the waste, or to determine the average burnup that the fuel was exposed to. If the uncertainties are such that the original amount of heavy metal or the average fuel burnup for particular high-level waste streams cannot be quantified, the units of waste derived from (a) and (b) of Note 1 shall no longer be used.

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Instead, the units of waste defined in (c) and (d) of Note 1 shall be used for such high-level waste streams. If the uncertainties in such information allow a range of values to be associated with the original amount of heavy metal or the average fuel burnup, then the calculations described in previous Notes will be conducted using the values that result in the smallest Release Limits, except that the Release Limits need not be the smaller than those that would be calculated using the units of waste defined in (c) and (d) of Note 1.

Note 6: Use of Release Limits to Determine Compliance with 191.13. Once release limits for a particular disposal system have been determined in accordance with Notes 1 through 5, these release limits shall be used to determine compliance with the requirements of 191.13 as follows. In cases where a mixture of radionuclides is projected to be released to the accessible environment, the limiting values shall be determined as follows: For each radionuclide in the mixture, determine the ratio between the cumulative release quantity projected over 10,000 years and the limit for that radionuclide as determined from Table 1 and Notes 1 through 5. The sum of such ratios for all the radionuclides in the mixture may not exceed one with regard to 191.13(a)(1) and may not exceed ten with regard to 191.13(a)(2)

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FIGURE A.2.2.2 RADIONUCLIDE CHAINS

DECAY2.GCD 8-11-93

FIGURE A.2.2.3 RADIONUCLIDE CHAINS

Isotopes Considered in Transport Calculations

Considered in Calculations Evaluating the Release from Cuttings and Evaluating Concentrations within the Repository

DECAYS.GCD 11-24-93

FIGURE A.2.2.4 RADIONUCLIDE CHAINS

 Isotopes Considered in Transport Calculations

 Considered in Calculations Evaluating the Release from Cuttings and Evaluating Concentrations within the Repository

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A.2.3.2 Waste Unit Factor

The waste unit factor (f_w) is the sum of the following: (1) the amount of the INEL spent fuels, as defined in 40CFR191 Appendix A Notes 1(a), divided by 1,000, and (2) the curies of gamma and beta-emitting radionuclides in the HLW with half-lives greater than 20 years but less than 100 years, as defined by 40CFR191 Appendix A Note 1(c), divided by 100,000,000, or the curies of other radionuclides of gamma or beta emitters in the HLW with half-lives greater than 100 years, or any alpha emitters with half-lives greater 20 years, as defined by 40CFR191 Appendix A Note 1 (d), divided by 1,000,000. The waste unit factor (f_w) is used to calculate the release limits for a specific disposal system allowed by 40 CFR 191 Appendix A Note 2. Table A.2.3.2.1 below shows the waste unit contribution from the various DOE complexes, based on 40CFR191 Note 1(a), Note 1(c), and Note 1(d).

As indicated in item 2 above, 40 CFR 191 Appendix A Notes 1 and 5 allow two different methods of calculating the units of waste for the HLW with poorly known MTHM or burn-up. This draft PA will be based on the units of waste calculated using the 40CFR191 Note 1(d) method.

The waste unit factor (f_w), based on the table below, is 9.934 (0.327 + 9.607).

Table A.2.3.2.1 Units of Waste Based on 40CFR191 Appendix A

DOE Complexes	Beta & Gamma Nuclides > 20 < 100 yrs (Curies)	Alpha Nuclides > 20 yrs Beta & Gamma Nuclides > 100 yrs (Curies)	Units of Waste Based on Note 1(a)	Units of Waste Based on Note 1(c)	Units of Waste Based on Note 1(d)
INEL					
Graphite Fuels	N/A	N/A	0.027	N/A	N/A
Special Fuels	N/A	N/A	0.3	N/A	N/A
Calcine	2.56E+07	3.39E+05	N/A	0.256	0.339
WVDP	1.51E+07	7.15E+04	N/A	0.1512	0.0715
SRS	4.85E+08	8.04E+06	N/A	4.854	8.036
HANF	1.84E+08	1.16E+06	N/A	1.836	1.160
Total			0.327	6.841	9.607

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A.2.3.3 Release Limits for a Specific Disposal System

The Disposal System Release Limits (DSRL) for the hypothetical repository are calculated by multiplying the 40 CFR 191 Appendix A Table 1 release limits by the waste unit factor (f_w). For the hypothetical repository in this PA evaluation, the Disposal System Release Limits for evaluating compliance with 40 CFR 191 § 13 are provided in Table A.2.3.3 below.

Table A.2.3.3 Cumulative Release Limits (L_i) to the Accessible Environment 10,000 Years after Disposal for Evaluating Compliance with Containment Requirements (40 CFR 191, Appendix A, Table 1)

Radionuclide	Release limits (L_i) per 1×10^6 Ci radionuclide with $t_{1/2} > 20$ years * (Ci)	1992 Disposal System Release Limits (L_i)(f_w) (Ci)
Americium (Am) -241 or -243.....	100	993.4
Carbon (C) -14.....	100	993.4
Cesium (Cs) -135 or -137.....	1000	9934
Iodine (I) -129.....	100	993.4
Neptunium (Np) -237.....	100	993.4
Plutonium (Pu) -238, -239, -240, or -242.....	100	993.4
Radium (Ra) -226.....	100	993.4
Strontium (Sr) -90.....	1000	9934
Technetium (Tc) -99.....	10000	99340
Thorium (Th) -230 or -232.....	10	99.34
Tin (Sn) -126.....	1000	9934
Uranium (U) -233, -234, -235, -236, or -238.....	100	993.4
Any other α -emitting radionuclide with $t_{1/2} > 20$ years.....	100	993.4
Any other non α -emitting radionuclide with $t_{1/2} > 20$ years..	1000	9934

* Other units of waste described in 40 CFR 191, appendix A

A.2.4 Estimated Critical Masses

The estimated critical masses of the fissile radioisotopes (presented in the spent fuels, calcine, HLW, and Na-waste) are indicated on Table A.2.4.1.

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Table A.2.4.1 Estimated Critical Mass for Fissile Radioisotopes

Radioisotopes	Un-reflected kg(c)	Water Reflected kg	Steel Reflected kg	References
U-233(metal)	16.5	7.3	--	TID-26286
U-233(water mixture)	1.2	0.58	--	LA-10860-MS
U-235(metal)(a)	49.0	22.8	--	TID-26286
U-235(water mixture)	1.5	0.82	--	LA-10860-MS
Np-237(metal)	56	51	33	ANS-8.15 Appendix B
NpO ₂ (kg Np)	288	272	187	ANS-8.15 Appendix B
Pu-238(metal)	7.1	6.1	4.2	ANS-8.15 Appendix B
PuO ₂ (kg Pu)	21	16.6	10.4	ANS-8.15 Appendix B
Pu-239(metal)	10	5.42	--	TID-26286
Pu-239(water mixture)	0.95	0.53	--	LA-10860-MS
Pu-240(metal)(b)	33	29	19	ANS-8.15 Appendix B
PuO ₂ (kg Pu)	143	131	84	ANS-8.15 Appendix B
Pu-241(metal)	--	6.0	--	TID-26286
Pu-242(metal)	82	--	47	ANS-8.15 Appendix B
Am-241(metal)	58	51	34	ANS-8.15 Appendix B
Am-243(metal)	54	50	36	ANS-8.15 Appendix B
Am ₂ O ₃ (kg Am)	83	78	59	ANS-8.15 Appendix B
AmO ₂ (kg Am)	99	93	71	ANS-8.15 Appendix B
Cm-244(metal)	9	7.5	5.4	ANS-8.15 Appendix B
Cm ₂ O ₃ (kg Cm)	11.8	10.3	7.3	ANS-8.15 Appendix B
CmO ₂ (kg Cm)	11.5	10.1	7.3	ANS-8.15 Appendix B

- a) U-235 enrichment of 93.2 percent
b) Pu-240 content of at least 3.1 percent
c) All metal and oxides are non-moderated

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A.2.5 Molecular Diffusion Coefficient

Table A.2.5.1 provides estimated values of the free liquid diffusion coefficient of important radioisotopes in the spent fuels, calcine, and HLW.

Table A.2.5.1 Estimated Diffusion Coefficient for Radionuclide

Parameter	Median	Range		Units	Distribution Type	Notes(See Below)
		Lower Range	Upper Range			
Am	1.765E-10	5.300E-11	3.000E-10	m ² /s		a
Ca ⁺²	3.955E-10	1.978E-10	7.910E-10	m ² /s		d
Cd ⁺²	3.590E-10	1.795E-10	7.180E-10	m ² /s		d
Ce ⁺³	3.100E-10	1.550E-10	6.200E-10	m ² /s		b
Cl ⁻	1.015E-09	5.075E-10	2.030E-09	m ² /s		d
Cm	1.765E-10	5.300E-11	3.000E-10	m ² /s		a
Cs ⁺	1.030E-09	5.150E-10	2.060E-09	m ² /s		d
Gd ⁺³	2.985E-10	1.493E-10	5.970E-10	m ² /s		d
I ⁻	1.000E-09	5.000E-10	2.000E-09	m ² /s		b
Nd ⁺³	3.075E-10	1.538E-10	6.150E-10	m ² /s		d
Ni ⁺²	3.325E-10	1.663E-10	6.650E-10	m ² /s		d
Np	1.760E-10	5.200E-11	3.000E-10	m ² /s		a
Pb	4.000E-10	2.000E-10	8.000E-10	m ² /s		a
Pu	1.740E-10	4.800E-11	3.000E-10	m ² /s		a
Ra	3.750E-10	1.875E-10	7.500E-10	m ² /s		a
Rb ⁺	1.035E-09	5.175E-10	2.070E-09	m ² /s		d
Sm ⁺³	3.035E-10	1.518E-10	6.070E-10	m ² /s		d
Sr ⁺²	3.970E-10	1.985E-10	7.940E-10	m ² /s		c
TcO ₄ ⁻	7.500E-10	3.750E-10	1.500E-09	m ² /s		b
Th	1.000E-10	5.000E-11	1.500E-10	m ² /s		a
U	20.700E-10	1.100E-10	4.300E-10	m ² /s		a

Notes

- Sandia Report, "Preliminary Comparison with 40 CFR Part 191, Subpart B for the Waste Isolation Pilot Plant, December 1991", Vol. 3: Reference Data, Sand91-0893/3.UC-721.
- Miller, D.G. "Estimation of Tracer Diffusion Coefficients of Ions in Aqueous Solution" September 7, 1982, UCRL-53319.
- Li, Yuan-Hui, and Gregory, S. "Diffusion of Ions in Sea Water and in Deep-Sea Sediments", rev. March 1973, Geochimica et Cosmochimica Acta 1974, Vol. 38, pp. 703 to 714. Pergamon Press.
- Calculated through the use of the Nernst equation as stated in ref. b and c using ionic conductivities at 25°C and infinite dilution. The conductivities were found in CRC Handbook of Chemistry and Physics, 70th ed., 1989-1990.

The values obtained from references b, c, and d were taken at 25°C and infinite dilution. These values were used as the upper range. The median for references 2, 3, and 4 was found by decreasing the upper range by a factor of 2 and the lower range was found by decreasing the median by a factor of 2. (See Note (a.) pg 3-70.)

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A.2.6 Chemical Reactions

A.2.6.1 Gas Production

A.2.6.1.1 Spent Fuels

A.2.6.1.1.1 Graphite Fuels

Since the primary compounds that make up the fuel kernel of the graphite fuels are water-reactive carbides, most of the reactions with water will produce acetylene, or a more-saturated hydrocarbon plus hydrogen gas. The metal will form the hydroxide. An equilibrium reaction analysis showed that most of the fission products are also present as water-reactive carbides. Some of the fission products are present as the metal and will react to produce hydrogen and the metal hydroxide. Some typical reactions are as follows:

Without the entry of water, air, or some other reactant into the SiC-protected fuel kernel, the reactions occurring will be very slow, so that the changes caused by radioactive decay will be at chemical equilibrium. The reactivity to oxygen in the air will also be very low at ambient temperature, at about the same order of magnitude as the reaction rate with carbon. If the temperature were elevated, the rate of reaction with air would increase. The ignition point is over 1000 °C. Table A.2.6.1.1.1 shows the expected actinide and fission product compounds present. If the fuels are treated by burning to react the carbides to oxides, then the only reactions would be the dissolution of the soluble oxides.

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TABLE A.2.6.1.1.1
Expected Chemical Compounds in Graphite Fuel Kernels

ELEMENTS	PRINCIPAL COMPOUND(S)
Actinides	MC ₂
Lanthanides	MC ₂
Zr	MC
Rb, Cs	M, MI, MCl
Ba, Sr, Y	MC ₂ , MSe, MTe
Sb, Sn, Pd, Ru, Tc, Mo	M
Se, Te	CM ₂ , SrM
Ni, Co, Fe	M ₃ C
Cl, I	CsM

The calculated maximum gas generation rate for the total graphite fuel mass (base form 1) is 75.8 l/hr. Assumptions include 50 percent flooded and 0.5 percent of the cracked open particles. Reaction is controlled by a cyclic thermal suction of water into the cracks of the damaged particles. The minimum rate is assumed to be 10 percent of reasonable. If the reasonable rate is used and all of the available cracked particles are eventually flooded, it would take about 17 years to use up the available carbides.

A.2.6.1.1.2 Special Fuels

Two potential gas production mechanisms are available; the first is the reaction of active metals with water. The predominant active metal being sodium. There is an average of 5 grams of metallic sodium per canister and a high of 140 grams. Sodium will react with water to generate hydrogen gas. The sodium metal is dispersed uniformly along the inside of the fuel cladding and will only react when the cladding is breached; it will not all react at once, but only as water reaches it. If it all did react at once, 0.2 grams of hydrogen per canister would be generated. The expected reaction rate is less than 1 gram hydrogen per year.

The second gas generation mechanism is the corrosion of the metal such as the cladding, the uranium, and the canister.

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Because all of these materials are corrosion resistant, the rate will be very low and dependent on the surface area of the metal. This means that the hydrogen gas generated by the corrosion of the storage canister will be the major factor. This gas generation reaction will apply to all canisters.

Because of the age of the fuel that will be dispositioned and the expected radiation levels, the hydrolysis of water to produce hydrogen and oxygen was considered to be less than the other two mechanisms.

A.2.6.1.2 Calcines

Due to the high temperature (>400 °C) processing of the calcine wastes and the stable oxide species created during the initial formation and subsequent post-treatment, the presence of organics or other materials that might contribute to gas forming chemical reactions is considered negligible.

Measured percentages of residual nitrates retained in the various calcines have been analyzed as follows: 0.43 percent in zirconium; 1.21 percent in alumina, and 5.54 percent in zirconium-sodium blends [Ref.6].

A.2.6.2 Other Chemical Reactions

A.2.6.2.1 Graphite Fuels

No other chemical reaction is expected for Graphite Fuels.

A.2.6.2.2 Special Fuels

About 1 percent of the total mass of the fuel contains uranium zirconium hydride fuel meat. These hydrides are known to be chemically stable and are not expected to produce any rapid chemical reactions.

A.2.6.2.3 Calcines

No other chemical reactions are expected for the calcines.

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A.2.7 Solubility

The solubility of specific radionuclides was extracted from the references indicated. Further development work will be conducted in the areas of spent fuels and glass-ceramic. When new information becomes available, it will be incorporated into future Performance Assessment analyses. Tables A.2.7.1 and A.2.7.2 depict the estimated distributions of solubility for species in a typical salt and hard rock repository.

Table A.2.7.1 Solubilities in WIPP Salado Brine(a)					
	Solid species Max and Min	Most likely chemical species	Median (M)	Lower Range (M)	Upper Range (M)
Am ³⁺	Am(OH) ₃ AmOHCO ₃	(AmCl ₂) ⁺	9.0E-10	5.0E-14	1.4
Cm ³⁺	Cm(OH) ₃ CmO ₂	Cm ^{III}	9.0E-10	5.0E-14	1.4
Np ⁵⁺	NpO ₂ (OH) (amorphous) NaNpO ₂ CO ₃ •3.5H ₂ O	(NpO ₂ CO ₃) ⁻	7.0E-09	3.0E-11	1.2E-2
Np ⁶⁺	Np(OH) ₄ NpO ₂	(Np(OH) ₅) ⁻	8.0E-07	3.0E-16	2.0E-5
Pb ²⁺	PbCO ₃ PbCl ₂	PbCl ₄ ²⁻	7.0E-07 1.5	1.0E-9 carbonate present 0.01 carbonate absent	8.0E-2 carbonate present 10.0 Carbonate absent
Pu ⁴⁺	Pu(OH) ₄ PuO ₂	(Pu(OH) ₅) ⁻	5.0E-10	2.0E-16	4.0E-6
Pu ⁵⁺	Pu(OH) ₄ PuO ₂	(PuO ₂) ⁺	5.0E-10	2.5E-17	5.5E-4
Ra ²⁺	RaSO ₄ and (Ra/Ca)SO ₄ RaCO ₃ and (Ra/Ca)CO ₃ RaCl ₂ •2H ₂ O	Ra ²⁺	1.0E-08 1.5E-06 17	1.0E-11 Sulfate present 1.6E-9 Carbonate present 2.0 Carbonate and Sulfate Absent	1.0E-6 1.0 18.0
Th ⁴⁺	Th(OH) ₄ ThO ₂	Th(OH) ₄ ⁰	2.0E-10	5.5E-16	2.2E-6
U ⁴⁺	UO ₂ (amorphous) U ₃ O ₈	U(OH) ₄ ⁰	1.0E-04	1.0E-15	5.0E-2
U ⁶⁺	UO ₃ •2H ₂ O UO ₂	UO ₂ (CO ₃) ₂ ²⁻	2.0E-03	1.0E-7	1.0

Notes:

- (a) SAND91-0893/3 Preliminary Comparison with 40 CFR Part 191, Subpart B for the Waste Isolation Pilot Plant, December 1991-Volume 3 pages 3-62 thru 3-66.

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Table A.2.7.2 Solubilities in Yucca Mountain Ground Water				
	Lower Range (M)* p = 0.05*	Median (M)* p = 0.90*	Upper Range (M)* p = 0.05*	Solubility (M)*
Am	6.2E-13	3.9E-07	3.9E-06	1.0E-08
C	8.3E-05	1.2E-04	1.2E-02	large (1.0E10)
Cm	9.7E-15	3.9E-07	3.9E-06	
Cs	9.0E-06**	2.9E-03**	6.1 ^(d)	large (1.0E10)
I	7.9E-06	3.1E-03	6.1 ^(d)	
Np	1.7E-09	1.5E-03	3.0E-03	3.0E-03
Pu	2.5E-10	3.9E-09	1.8E-06	1.8E-06
Ra	4.4E-11	1.8E-09	4.4E-07	1.0E-07
Se	1.0E-02	1.0E-01	7.0	
Sn	1.1E-09	2.7E-08	1.8E-07	1.0E-09
Sr				9.4E-04
Tc	3.6E-07	1.0E-03	1.0E+01	large (1.0E10)
U	2.1E-06	1.0E-05	2.1E-04	2.1E-04

Notes:

- a. Demonstration of a Risk-Based Approach to High-Level Waste Repository Evaluation: Phase 2, Electric Power Research Institute, EPRI TR-100384, May 1992
- These estimated solubility values were derived from a variety of reported sources on radionuclide solubilities for a repository in tuff as well as for a generic "granitic" oxidizing groundwater. The "low" values correspond to the lowest solubility values cited from reports on a repository located in tuff. The exceptions were the "low" solubility for Se, taken from the "generic" oxidizing groundwater, and the "low" solubility values of C and I (for which there were no reliable data reported other than "high") that were set approximately equal to 1.0 gm/m³. All three of these radioelements are expected to be controlled by UO₂ reaction rate rather than their own solubilities, so this approximation is not deemed to introduce any significant effect on calculated results. The "high" solubility values for all radioelements were taken from the highest values reported among these sources. The "medium" values were then selected between these extremes.
- b. Kerrick J.F., Solubility Limits on Radionuclide Dissolution at a Yucca Mountain Repository, Los Alamos National Laboratory, LA-9995-MS, May 1984.
- These solubilities were estimated for a number of important waste elements in water that is characteristic of Yucca Mountain. Analyses from Well J-13 were used to define water composition. The water Ph was taken as 7. Oxidizing conditions (Eh of 700mV) were assumed because present indications are that water in the unsaturated zone and upper portions of the saturated zone at Yucca mountain may be oxidizing. This assumption leads to higher solubilities for a number of elements than would occur under mildly or strongly reducing conditions. Solubility calculations were done for Am, Pu, U, and Sr using EQ3/6. For Cs, Tc, and C, solubility limits were assumed to be so large that their dissolution rates would always be limited by bulk waste-form dissolution. Solubilities for Ra, Sn, and Np were estimated from reviews by Apps et al.¹ and Allard.² The 10 waste elements account for approximately 99 percent of spent fuel activity at 1000 years after discharge from the reactor.
1. J.A. Apps, C.L. Carnahan, P.C. Lichtner, M.C. Michel, D. Perry, R.J. Silva et al., "Status of Geochemical Problems Relating to the Burial of High-Level Radioactive Waste, 1982," Lawrence Berkeley Laboratory report LBL-15103 (March 1983).
2. B. Allard, "Solubilities of Actinides in Neutral or Basic Solutions," in Actinides in Perspective, N.M. Edelstein, Ed., Proc. of the Actinides - 1981 Conf., Pacific Grove, Ca., Sept. 10-15, 1981 (Pergamon Press, New York, 1982).

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- c. Apted, M.J.; Liebetrau, A.M.; Engel, D.W., The Analytical Repository Source Term (AREST) Model: Analysis of Spent Fuel as a Nuclear Waste Form, Pacific Northwest Laboratory, PNL-6347/UC-70, Feb 1989.

These solubility values were taken from Environmental Assessment: Yucca Mountain Site, Nevada Research and Development Area, Nevada DOE/RW-0073, vol 2 of 3, 1986.

In an effort to be conservative and report all literature values, the upper ranges used for CS and I were taken from Note c because they exceeded the values given in Note a.

* p = probability

** These Cs solubilities are suspect. Cesium should have a higher solubility than these numbers indicate.
(R.J.Kirkham)

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A.3 Base Repository Form

The base repository form will consider the spent fuels and wastes in the following manner: (1) For spent fuels, minimal conditioning will be performed and the fuels will be placed into canisters, and disposed of in the repository; (2) Liquid waste HLW and Na-waste will be assumed to be converted into calcine form and packaged in canisters for repository disposal, and, like the spent fuels, calcine wastes will be packaged in canisters for repository disposal without additional conditioning; and (3) The immobilized HLW from the other DOE complexes will be placed into canisters for repository disposal. The volume, mass, and number of canisters are indicated in Section A.1 and Table A.2.1.5 through A.2.1.7. The immobilized HLW from the other DOE complexes will be assumed to be similar to the INEL HLW borosilicate glass and glass-ceramic forms. The glass from the other DOE complexes will be the same in all of the repository forms considered in this Performance Assessment. Therefore, their description will only be given in the base form. [Ref. Attachment 1 Page 1]

Attachments 3 ("ICCP Spent Fuel and Waste Technology Development Program Waste Form, Canister Sizes and Material of Construction For 0.700 kg's in a Salt and Granite Repository") and 3a ("ICCP Spent Fuel and Waste Technology Development Program Waste Form, Canister Sizes and Material of Construction For 10 kg's in a Salt Repository Only") summarize the canister sizes, and Attachments 4 ["Waste Form Data Summary (Salt & Granite Repository—0.700 kg U Limit)"] and 4a ["Waste Form Data Summary (Salt Repository Only—10 kg U Limit)"] provide container masses and other pertinent waste form data. In all cases, the canisters will be overpacked with a corrosion-resistant material (TiCode 12 for the salt repository, and Inconel 625 for the igneous rock repository). Overpack canister sizes are also summarized in Attachment 3 & 3a. Packaging for materials other than spent fuel will be in the standard ≈ 61 by 300 cm canister and corresponding ≈ 66 by 325 cm overpack canister.

Note: Subsequent information, prior to issuing the first draft, has shown that the above combination of canister materials would appear to inhibit the computer simulations of the waste form behavior in the repository. Therefore, 304L Stainless Steel (SS) was selected to package all waste forms; overpacks were omitted for this study. In addition, Attachments 3 and 4 are summaries of canister sizes, volume, mass, number of canisters, and heat generation for 0.700 kg fissile limit in a granite repository. Attachments 3a and 4a have the same information as Attachments 3 & 4 except that information pertains to a salt repository only with a 10 kg fissile limit for waste forms 1(graphite only) and 3 (both graphite and SNF).

Graphite Fuel Base Form Description

The fuel contains a homogeneous mixture of two types of particles, fissile and fertile. Fissile particles contain thorium and 93.5 percent enriched uranium; fertile particles contain only thorium. The Fort St. Vrain and Peach Bottom Core 2 fuel kernels are coated via a fluidized bed, vapor-phase deposition process; with three fission-product-retaining layers. The inner and outer layers are of isotropic carbon, and the middle layer is of silicon carbide (hence the name TRISO-coated). There is a fourth layer, called the "buffer", of porous carbon next to the kernel of fissile or fertile material to provide a volume for accumulation of fission product gases without excessive pressure buildup. The SiC layer is highly resistant to both oxidation and moisture, even at extremely high temperature. The Peach Bottom core 1 particles do not contain the protective SiC layer and, as a result, many of

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the fuel kernels failed. The failure rate for the Peach Bottom Core 2 fuel kernels were similar to the FSV elements.

The fissile and fertile particles are blended and then molded into 1.27-cm diameter x 4.93-cm long fuel rods (compacts) in the FSVR design. In the Peach Bottom, the compacts are 6.86-cm OD, 4.25-cm ID x 7.57-cm long hollow cylinders. A carbonaceous binder is used to form the compacts that are later carbonized by firing at high temperature before insertion into the graphite blocks. It has been reported that 0.3 percent to 0.5 percent [Ref. 6] of the SiC coatings from the first three fueling segments may have failed. The next three segments are expected to be similar to the first three. A different coating method was used on the fuel that replaced the first three segments during refueling, and the expected failure rate of the SiC coatings from that fuel should be well under 0.1 percent.

The base repository form for graphite fuels will consist of cut pieces of Fort St. Vrain fuel (equivalent to two thirds of a block) or two Peach Bottom fuel elements placed into canisters. No packing or other filler material is contemplated. In a special case study for the base form, Fort St. Vrain and Peach Bottom fuels were analyzed on the basis of a 10 kg U loading in canisters in a salt repository. The 0.700 kg limit still applied to the SNF and graphite fuels in granite and SNF in salt. Cutting the Fort St. Vrain fuel will be necessitated in all cases to meet the U-235 equivalent limit to prevent criticality. The radionuclide inventory for the graphite fuels is given in Table A.2.1.1.

Fuel Burnup

The maximum burnup actually achieved in any core segment at FSVR is 52,000 MWd/MTHM, and the average value is in the range of 30,000 to 35,000 MWd/MTHM. Peach Bottom Core 1 had a burnup of 30,795 MWd/MTHM, and Core 2 had a burnup of 72,717 MWd/MTHM. Because of the relatively low amount of heavy metal in the Peach Bottom fuel, the average value range for all the graphite fuel is the same as the FSVR fuel.

Graphite Fuel Thermal Loading

The average "Thermal Output" of the fuel as is about 26 W/m³ (See Attachment 4 Salt & Granite Repository for 0.700 kg U limit) and decayed to 2030. The average "Thermal Output" for the 10 kg U limit in a salt repository is about 27 W/m³ (See Attachment 4a Salt Repository only - 10 kg U limit).

Special Fuel Base Form Description

The base repository form for special fuels consists of bare or canned fuel placed in a canister. The fuel will be dried to as low a moisture content as possible before packaging. A low-melting metal will fill the voids in the fuel and the space between the fuel and the fuel canister to the maximum extent possible. The fuel canisters will be placed into overpack canisters, and a low-melting alloy will be cast so as to fill the void between fuel canister and overpack canister. For the salt repository, the overpack will be made of titanium alloy TiCode 12. For the igneous rock repository, the overpack material will be Inconel 625. The intent of the fill material is to provide structural strength to the waste packages to withstand the lithostatic pressure. As currently stored, the fuel may be bare or canned, depending on the fuel type and the integrity of its cladding. In a few cases,

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therefore, it may not be possible to encapsulate the fuel with metal.

The total amount of fuel to be loaded into a canister is limited by criticality considerations. Initially, for highly enriched fuel, no more than 0.700 kg of U-235 is permitted for the salt & granite repository computer simulations. The fissile material limit for the salt repository was determined to be 10 kg, although only the graphite fuels were studied at this loading. Each U-235 concentration between these limits has a corresponding critical mass limit [Ref. 7]. Fuel cutting will be avoided, except for those highly enriched fuels that exceed 0.700 g U-235 per fuel element for the salt and granite repository and 10 kg's for the salt repository only test case. The total radionuclide inventory of the special fuel will be that shown in Table A.2.1.2.

Most of the special fuel consists of small, short fuel elements (typically 10 by 140 cm); however, some the fuel is full-size commercial BWR and PWR spent fuel, which are approximately 28 by 410 cm in size. Accordingly, the special fuel will be packaged in two sizes of canister to accommodate these extremes.

Special Fuel Burnup

The special fuel burnup ranges from none to a maximum of 50,000 MWd/MTHM.

Special Fuel Thermal Loading

The average special fuel thermal power density is 234 W/m³, decayed to 2030. Per canister, the average thermal power is 27.1 W (See Attachment 4).

Calcine Description

There are two major types of calcines at the ICPP, aluminum and zirconium calcines.

The aluminum wastes were among the first wastes to be generated at ICPP, as a result of processing the fuels from Test Reactor Area (TRA). The first alumina (Al₂O₃) was generated in the Waste Calcining Facility in December, 1963. This process used a fluidized bed with indirect heating to denitrate the acidic, aqueous waste and convert it into a dry, granular solid that was stored in underground bins. Typically, the wastes experienced about a 8-10:1 volume reduction in the calcination process. The dry solids consist of both a granular material and a powder (~3-18 percent of the granular in wt percent) [Ref. 5], both of which are pneumatically retrievable and transportable. In terms of short-lived radionuclide concentrations, the aluminum fuels and the resulting wastes provide the highest watt densities (Kw/m³) of any of the wastes processed in the waste calcination facilities at ICPP.

The zirconium waste originated from the processing of Navy nuclear fuels with their zirconium cladding. The Navy fuel dissolution is a more aggressive effort, using aqueous hydrogen fluoride (HF) to chemically dissolve the fuel elements. Aluminum nitrate, first as a cold feed stock, then later as a coprocessing effort, was blended with the zirconium dissolver product to complex the free fluorides prior to extraction in the stainless steel process vessels, and then storage in the tank farm. The calcination of zirconium wastes requires the addition of calcium nitrate to form calcium fluoride; all the other components of the waste are converted to their oxide form. Because of the cold chemical addition to the

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fluoride wastes, liquid to solid volume reduction factors are closer to 6-7:1. The waste densities are more moderate, compared to the aluminum calcines.

The granular material ranges in size from 0.2 to 0.5 mm. It is usually expressed as mmpd (mass median particle diameter) where 1/2 the mass of particles in a given sample pass through a hypothetical screen having openings of that particular size. So for any given mmpd, there is actually a distributed ranges of diameters. The granular calcine has a bulk density of 2.4 to 2.7 g/cm³; however, when combined with the fines, the bulk density ranges between 1.0 and 1.7 g/cm³, with 1.4 g/cm³ the most representative value.

Borosilicate Glass From Other DOE Complexes

Borosilicate glass, to be disposed of in a repository, is being produced from neutralized high-level waste in three other DOE complexes. These facilities are the West Valley Demonstration Project (WVDP), the Hanford Waste Vitrification Plant (HWVP), and the Savannah River Site (SRS). These borosilicate glass products will be very similar to the glass produced at the ICPP. The average radionuclide contents per canister from WVDP, HWVP and the SRS are 109,600 curies, 298,000 curies, and 234,000 curies, respectively. The respective canisters contain 1,900 kg, 1,650 kg, and 1,682 kg of glass.

Canister sizes for the borosilicate glass from these facilities are all $\approx$ 61 cm in diameter by 300 cm long. However, the WVDP canister has a canister neck opening of 42 cm as compared to a 12.7 cm diameter neck for the HWVP and the SRS canisters (**See Attachment 3 canister sizes and materials for 0.700 kg U limit for Salt & Granite Repository**). The canister dimensions are shown as Attachment 3 to this Appendix. The final material of construction for the canisters has not yet been determined; performance in the geological environment will dictate final material selection.

A.3.1 Volume of Base Form

A.3.1.1 Spent Fuels

Graphite Fuels

The total "Volume As Is" (**See Attachment 4**) of the spent graphite fuels from the Fort St. Vrain and the Peach Bottom Reactors is 259.8 m³. The Base Waste Form is to load the Fort St. Vrain blocks into 45.75-cm diameter by 98.6-cm long containers, the overpack container will be 50.8-cm in diameter by 124.0-cm in length. One block would be loaded into each container if the effective U-235 mass is less than 0.700 kg. If the effective U-235 is more than 0.700 kg, the blocks will need to be cut longitudinally because of criticality concerns. About 320 blocks will need sectioning. One third will be cut off each of the fissile heavy blocks, then the block section remaining will be loaded into a canister. Two of the cut-off sections will be loaded per extra canister. This makes a total of 2368 containers for the 2208 FSV blocks. Two of the Peach Bottom element containers will be loaded into a 27.31-cm diameter by 400.9-cm long container and the overpack dimensions will be 32.39-cm x 426.3-cm. This will require 800 containers for the 1599 Peach Bottom elements. The combined "Packaged Volume" would be, excluding overpack material and containers (**See Attachment 4** for further details), 541.1 m³ in 3168

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canisters. [Values calculated from information in Ref. 6,16]

The "Volume As Is" for the 10 kg simulation is the same as the 0.700 kg study, but the "Packaged Volume" will be 532.6 m³. However, the waste forms for the 10 kg case will be put into 60.96-cm diameter by 411.48-cm long container, and the overpack dimensions (for non 304L SS material) would be 66.04-cm in diameter by 426.3-cm in length. The containers will have about 5 FSV blocks and 3-4 Peach Bottom elements, which will require about 442 canisters (See Attachment 3a and 4a).

Special Fuels

The total "Volume As Is" of the special fuels alone is 89 m³. After canning, the "Packaged Volume" of the overpacked canisters is 752 m³ (See Attachment 4).

A.3.1.2 Calcine

The total volume of the calcines (including conversion of all the liquid HLW to the calcine form) is 3,750 m³. The process selected to treat the remaining sodium wastes should generate an additional 500 m³ of calcine. Six thousand nine hundred and forty-five $\approx$ 61-cm diameter x 300-cm long canisters will be required to contain the calcines alone. The overpack size will be based on the next larger, standard diameter pipe size with a 0.95-cm wall thickness. Attachment 3 (based on 0.700 kg U limit) has all the dimensions for the overpacks & waste containers for all five waste forms.

A.3.1.3 Borosilicate Glass From Other Complexes

The "Packaged Volume" of the borosilicate glass from WVDP, HWVP, and SRS is expected to be 172 m³, 1,444 m³, and 3,891 m³ (See Attachment 4), respectively. The "Volume As Is" are 193 m³, 1227 m³, and 3307 m³ (See Attachment 4) for WVDP, HWVP, and SRS, respectively. Using $\approx$ 61-cm diameter x 300-cm long (See Attachment 3) canister at 0.626 (filling the canister to 85 percent) cubic meters each, the total number of canisters required is expected to be 276 for WVDP, 1,960 for HWVP and 5,283 for SRS (See Attachment 4).

A.3.2 Mass of Base Form

A.3.2.1 Spent Fuels

Graphite Fuels

The "Mass As Is" of the Fort St. Vrain and Peach Bottom fuel totals 342 metric tons (27 MTHM). ICPP has 158 metric tons, (10 MTHM) and the balance resides at Fort St. Vrain. The "Packaged Mass" of contained Base Waste Form would be about 959 metric tons for the 0.700 kg loading limit for the granite repository (See Attachment 4). This includes the mass of the Inconel containers, but excludes overpack material and containers as shown in Attachment 4. The "Packaged Mass" of the contained Base Waste Form for the 10 kg case study for the salt repository would be about 650 metric tons (See Attachment 4a).

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Special Fuels

The "Mass As Is" of the special fuels alone is 474 metric tons (300 MTHM). The "Packaged Mass" (fuel and packaging) is 7314 metric tons.

A.3.2.2 Calcine

The "Mass As Is" of the calcines (including sodium waste) totals 5950 metric tons. The masses for the overpack containers and overpack material are included in Attachment 4.

A.3.2.3 Borosilicate Glass From Other Complexes

The treated waste form "Mass As Is" from WVDP, HWVP and SRS will be 522.5 MT, 3,234 metric tons and 8,884 MT, respectively (**See Attachment 4**). This mass does not include the weight of canister materials used to contain the waste. Uncertainties relative to material type for corrosion protection, strength requirements, and physical size will ultimately dictate the mass of material used in the canisters; for 304L canisters, the canister will be approximately 252 kg and 500 kg, respectively, for the WVDP and the HWVP or SRS canister.

A.3.2.4 Other Material Masses

The Inconel containers, the cast metal fill, and the TiCode or Inconel overpack canister make up the other materials in the base waste form. Their masses are summarized in Attachment 4 & 4a.

A.3.3 Surface Dose Rates

(Note: This section is only important for transportation and repository design. This information is not required for the PA.)

A.3.3.1 Spent Fuel Containers

Attachments 3 & 3a are summaries of the canister and overpack dimensions, and Attachments 4 & 4a give canister, cast metal fill, and overpack canister masses. The fuel canisters have a 0.9525 cm wall thickness. All other canisters and overpack canisters have a 0.635 cm wall thickness.

A.3.3.1.1 Graphite Fuel Containers

The graphite fuel containers are described under the description of waste form volumes.

A.3.3.1.2 Special Fuel Containers

The special fuel containers are described in Attachments 3 & 3a. In all, there are 6500 fuel canisters (**See Attachment 4**), of which 5613 are small canisters and 887

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are large canisters.

A.3.3.2 Calcine Containers

Radioisotopes of concern in canister shielding will be the equilibrium Yttrium-90 daughter (0.2025 Mev gamma) product for the Strontium-90 decay (beta decay only), and the delayed 0.66164 Mev gamma from the Cesium-137 decay to Barium-137m. Combined, these two isotopes and their equilibrium daughter products constitute 97-98 percent of the activity found in the aged calcine wastes. Maximum concentration of strontium and cesium in the hottest calcine is expected to range from 5-11 Ci/kg [Ref. 5] or 16,600-28,800 Ci/m³ based on glass-ceramic densities to as high as 3.2 gm/cm³. Concentrations of any radionuclide species can be apportioned as a percentage of the inventory listed in Table A.2.1.4.

Calculated surface dose rates will be affected by the thickness of the canister and the overall diameter of the calcine monolith in the canister (self-shielding considerations).

Calculated dose rates should be based on a monolithic glass-ceramic form $\approx$ 61.0 cm in diameter by 300 cm long. For estimation purposes, the canister will use an Inconel material composition with a .95 cm wall thickness.

A.3.4 Reaction Potential

A.3.4.1 Gas Generation Potential from Reactions

Calculated surface dose rates will in effect determine the extent of gas generation by radiolysis. The behavior of water in the geological strata surrounding the buried canister will in turn determine the quantity of gas generation from radiolysis.

A.3.4.1.1 Graphite Fuels

The potential for chemical reaction and the generation of flammable gas is always a possibility with the graphite fuels waste forms containing carbides (For reactions see Section A.2.6.).

A.3.4.1.2 Special Fuels

Reactions listed in section A.2.6 are applicable to the Special Fuels base waste form.

A.3.4.1.3 Calcine Material

The residual nitrates noted in Section A.2.6 will be reduced below 1.0 percent based on proposed post-treatment packaging of the calcines in a glass-ceramic matrix.

A.3.4.1.4 Borosilicate Glass From other DOE Complexes

Due to the high temperature processing during vitrification, the presence of organics

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or other materials that might contribute to gas forming chemical reactions is considered negligible. The properties of the other DOE complexes' glass should be very similar to the glass or glass-ceramic at the ICPP.

A.3.4.2 Other Important Chemical Reactions

Other chemical reactions are the same as Section A.2.6.

A.3.5 Drilling Erosion

(This section will be provided by SNL)

A.3.6 Leachability

A.3.6.1 Graphite Fuels

The graphite fuel Waste Form leachability has been roughly estimated. The values vary widely and are dependent on exposed area and containing matrix assumptions. The base leach rates used to derive the leach rate values in the Table came from data obtained on unclad Light Water Reactor (LWR) fuel pellets. The report, "Status Report on LWR Spent Fuel IAEA Leach Tests", PNL-3173, reports values near 10 g/m²-day for the most leachable constituents in the oxide matrix. The leach rates from a glass matrix are about 200 times lower. The leach rates for the actinides in glass are lower still at about .0005 g/m²-day. The actual leachability of the graphite fuels depends on the integrity of the SiC layer and the containing matrix to inhibit water from contacting the fuel kernel. The assumptions made for the scoping values are as follows:

Leach Rate 1 Assumption--Maximum

Half of the fuel particles are exposed to water and .5 percent have the protective SiC layer cracked open. Area is based on particle surface area, and the maximum leach rate applies. The probability, based upon these assumptions, is estimated to be 0.2 percent.

Leach Rate 2 Assumption--Middle

Same as for 1 above except that the area is based on the exterior of the waste form and the leach rate used was for the containing matrix. The probability, based upon these assumptions, is estimated to be 30 percent.

Leach Rate 3 Assumption--Minimum

One tenth of the fuel is water exposed, and only 0.1 percent of the fuel particles are cracked open. Leach rates are based on the exterior area of the fuel and the containing matrix. The probability, based upon these assumptions, is estimated to be 99 percent.

For the total mass of the base waste form, the overall leach rates for any nuclide are about 5,164 g/day for leach rate 1; 308 g/day for leach rate 2; and 12.31 g/day for leach rate 3.

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A.3.6.2 Special Fuels

Leachabilities for the special fuels were approximated by using the results of a leach test performed by Pacific Northwest Laboratory on light-water-reactor (LWR) spent fuel. The test was conducted at 25°C using a mixture of unclad spent LWR fuel fragments with an average burnup of 28,000 MWd/MTU, and a batch International Atomic Energy Association (IAEA) leach test procedure. Data was collected for a period of 467 days.

The values in the following table are actually 3/4 of the values in Reference (a), listed below the table, for leachabilities of spent fuel in hard rock and salt repository conditions. The reason for this reduction in value is that the leach test was performed on fuel with a higher burnup than the fuels at ICPP. In order to get an initial approximation for the leach rate of all special fuels at ICPP, a 3/4 value was chosen. (Per Discussion With D.L. Fillmore)

Leachabilities for Special Fuels					
	In WIPP Salado Brine		In Yucca Mountain ground water*		
	Lower Range (g/cm ² day)	Upper Range (g/cm ² day)	Lower Range (g/cm ² day)	Upper Range (g/cm ² day)	Note (See Below)
Ce ¹⁴⁴	4.2E-7	1.1E-4	7.5E-8	4.7E-5	a
Cm ²⁴⁴	8.2E-7	9.0E-5	1.1E-7	1.3E-4	a
Cs ¹³⁷	2.8E-6	9.0E-4	1.7E-6	1.0E-3	a
Eu ¹⁵⁴	7.1E-7	1.0E-4	1.5E-7	6.5E-5	a
Pu ^{239 + 240}	6.9E-7	1.8E-5	9.0E-8	5.3E-5	a
Ru ¹⁰⁶	2.8E-7	2.3E-5	9.7E-8	6.5E-5	a
Sb ¹²⁵	2.2E-6	8.2E-5	2.0E-6	3.4E-4	a
Sr ⁹⁰ + Y ⁹⁰	9.0E-7	1.1E-4	1.1E-7	1.3E-4	a
U	1.5E-7	3.0E-5	2.2E-7	7.5E-5	a

Note

- a. Katayama, Y.B., Bradley D.J., Harvey, C.O., Status Report on LWR Spent Fuel IAEA Leach Tests, March 1980, PNL-3173

* Yucca Mountain ground water is approximated by using the values for the 0.03M NaHCO₃ solution leach rate.

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A.3.6.3 Calcine

Calcine, as a loose material in a combination granular and powdered form, is essentially a homogenous mix of inert materials (predominantly Al_2O_3 , ZrO_2 , and/or CaF_2) and a small quantity of oxides of heavy metals, actinides, and radionuclides. Leach tests of the various calcine forms produced over the years at ICPP indicated near quantitative removal of the alkaline earth metals [Ref. 9,10,11]. The oxides of the actinide species appear more leach-resistant to typical ground water. During a modified MCC-1 leach test of radioactive calcine, analytical results showed leach rates for Sr-90, total Cs, and Tc-99 of 93 percent, 84 percent, and 54 percent, respectively, at the end of a 28 day period. Gross alpha leach rates experienced a $6.1\text{E-}02$ percent removal at the end of 28 days [Ref. 12].

A.3.6.4 Borosilicate Glass

In the case of other DOE complexes' borosilicate glass, the leach rates should be similar to the ICPP glass-ceramic as indicated in waste form 2 below.

A.3.7 Heat Generation Potential

Attachments 4 & 4a contain summaries of volumetric and per-canister thermal powers for each of the five waste forms.

A.3.7.1 Graphite Fuels

The important heat generation rate is the amount per linear distance on the container or the total amount per container. For the base waste form, the "Thermal Output" is 26 W/m^3 . The per-canister thermal power "Watt Density" is 4 W (See Attachment 4 for the 0.700 kg granite repository data). The "Thermal Output" and the "Watt Density" for the 10 kg U limit for the salt repository are 26 W/m^3 and 33 W/canister (See Attachment 4a).

Thermal properties associated with a glass matrix surrounding the graphite fuels includes an estimated mean specific heat of 20 cal/kg-K and a thermal conductivity of 0.96 w/m-K . The silicon carbide has a mean specific heat of $6.42 \text{ cal/g-mole-K}$. The graphite in the blocks has thermal conductivities and specific heat capacities of 93 w/m-K and 1.3 kJ/kg-K , respectively.

A.3.7.2 Special Fuels

The average special fuel "Thermal Output" is 234 W/m^3 , decayed to 2030. Per canister, the average thermal power is 24.8 W (See Attachment 4).

A.3.7.3 Calcine Materials

Typically, the aluminum wastes are expected to range from $40\text{-}80 \text{ W/m}^3$ of final waste volume [Ref. 5]. Zirconium and zirconium-sodium blends will range from $30\text{-}50 \text{ W/m}^3$. Sodium wastes by themselves will range as low as $0.3\text{-}15 \text{ W/m}^3$, depending on the final method selected to solidify the liquid wastes. These heat generation potentials are 1992

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values. Year 2030 heat generation potentials will be approximately 45 percent of the values listed.

Thermal conductivity of the various calcine materials ranges from 0.17-0.29 W/m-°K [Ref. 13]. Heat capacity (C_p) for commercial alumina ranges from 0.20 cal/gm/°C @ 100 °C to 0.274 cal/gm/°C @ 1500 °C [Ref. 14]; alumina calcine is expected to be similar to those values. Zirconium calcines are also expected to fall in a similar range.

Bulk densities for the various calcines range from 0.89-1.12 gm/cm³ for alumina to 1.56-1.71 gm/cm³ for zirconium calcine [Ref. 1].

A.3.7.4 Borosilicate Glass From other DOE Complexes

The borosilicate glass from the other DOE complexes varies in heat generation rates depending on the type of waste immobilized. The heat generation rates were calculated from the basis of curie inventories indicated on the tables as of November 1992.

The borosilicate glass from WVDP, HWVP and SRS has a "Watt Density" of 147, 391 and 328 W/canister, respectively. These numbers are based on a decay to 31 October, 2030 (See Attachment 4).

A.3.8 Criticality Issues

This problem applies to the spent fuels only. Initial assumptions focused on the possibility of a fully flooded (water moderated) condition for both salt and hard rock repositories. The criticality safety evaluation will consider the repository under fully flooded conditions for the hard rock repository only. Subsequent information revealed credit could be taken for water exclusion in a post-closure salt repository, thereby allowing a greater loading (10kg) per canister.

To avoid any possibility of criticality, all canisters containing spent fuel will be packaged to contain less than the subcritical mass limit based on worst-case assumptions. The subcritical mass limit depends on enrichment ranging from a 0.7 kg U-235 equivalent for high-enriched uranium to infinite mass for uranium containing less than 0.7 weight percent U-235 [Ref. 7].

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A.4 Waste Form 2

Waste form 2 packages the special fuels and the glass-ceramic into the same canisters. The generic salt and granite repositories, with a 0.700 kg U limit per canister, will be evaluated as disposal sites for waste forms 2, 4, and 5. The fuel canisters described in Waste Form 1 will be surrounded by annular rings of glass-ceramic which will, in turn, be surrounded by a canister. The canister will be overpacked, and the space between canister and overpack will be filled with a cast metal. The graphite fuel compacts will be mechanically removed and the calcines will be converted into a glass-ceramic form, for disposal. The paragraphs below describe the characteristics of Waste Form 2 [Ref. Attachment 1 Page 2].

A.4.1 Volume of Waste Form 2

A.4.1.1 Spent Fuel

Graphite fuels

Waste Form 2 includes the fuel compacts (separated from the graphite blocks) loaded into a 20.32-cm diameter x 400.9-cm long canister, and the overpack container dimensions will be 27.31-cm in diameter x 426.3-cm in length. The compacts are packaged with a glass that fills the voids and increases the bulk volume by 100 percent. The expected total or "Packaged Volume", excluding overpack material and containers (**See Attachment 4** for further details), of this waste form for all graphite fuels is 269 cubic meters in 1801 containers. Because of initial criticality concerns no more than 0.700 kg of fissile U-235 equivalent material is loaded per container. [Calculations based on information from Ref. 6,15,16,17]

Special Fuels

The total "Volume As Is" of the special fuels only is 89 m³. As "Packaged Volume", and including the glass-ceramic, the total volume will be 5353 m³ (**See Attachment 4**).

A.4.1.2 Glass-Ceramic From Calcine

Glass-ceramic waste formulations use various glass-forming additives (frit) and reactants (Ti and Al) to form the final product. The final treatment process employs temperatures up to 1050 °C and pressures ranging from 6.9×10^7 to 1.4×10^8 Pa in a Hot Isostatic Press (HIP) process [Ref 16]. The advantage of the HIP process over borosilicate glass is the higher waste loading (fewer inerts/improved volume reduction); leach rate testing has demonstrated comparable results between the two forms.

The HIP process is limited in canister size to approximately 60 cm in diameter by 20 cm thick. The thin-walled containers used in the HIP package would require a larger diameter overpack, with several HIP packages inserted in each overpack canister.

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Existing calcine plus conversion of existing and anticipated liquid wastes to a glass-ceramic solid form is expected to produce a "Volume As Is" and "Packaged Volume" are 2975 m³ and 5353 m³ respectively. (See Attachment 4).

A.4.2 Mass of Waste Form 2

A.4.2.1 Spent Fuel

Graphite Fuels

The "Packaged Mass" of graphite fuels Waste Form 2, including container mass, is 721 metric tons (See Attachment 4) (27 MTHM). This excludes overpack material & containers which are shown in Attachment 4.

Special Fuels

The mass of the special fuel in waste form 2 is 474 metric tons (300 MTHM. As packaged, the "Packaged Mass" is 16734 metric tons, including 9994 metric tons of glass-ceramic (See Attachment 4).

A.4.2.2 Glass-Ceramic

The treated waste form (not including the special fuels) is 9,994 metric tons for a glass-ceramic. This mass does not include the weight of canister materials used to contain the waste. (See Attachment 4 for details)

A.4.3 Containers for Waste Form 2

A.4.3.1 Spent Fuel Containers

Graphite Fuel Containers

The containers for the glass matrix fuel/block separated waste form are discussed in the volume section above.

Special Fuel Containers

As with Waste Form 1, two canister sizes are needed to accommodate the different sizes of spent fuel (See Attachment 3). In all, there are 8551 canisters, of which 7664 are small canisters and 887 are large canisters (See Attachment 4).

A.4.3.2 Calcine Containers

The glass-ceramic waste form will be added to the fuel canisters as annular pre-cast rings with fuel centered inside. The fill efficiency is reduced to 80 percent to accommodate unavoidable volume loss between the overpack and the HIP packages.

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A.4.4 Reactions of Waste Form 2

All reactions will be much the same as for the base waste form.

A.4.5 Drilling Erosion

Drilling erosion data will be provided by Sandia National Laboratories.

A.4.6 Leachability of Waste Form 2

Graphite Fuels

Using the same assumptions as for the base waste form, the following overall leach rates were derived: rate 1) 5164 g/day; rate 2) 308 g/day; and rate 3) 12.31 g/day.

Special Fuels

The leach rates for the special fuel in Waste Form 2 will be the same as the base form. The leach rates of the glass-ceramic that encapsulates the special fuels will be as listed below.

Glass-Ceramic

In the case of glass-ceramic forms tested, 14-day MCC-1 analytical data show cesium and strontium species leaching into the MCC-1 media at a maximum rate of 3.43 and 2.10 g/m²-d, depending on the glass-ceramic composition. For the 18 samples tested for strontium leaching, the average leach rate was 0.95 g/m²-d and the median for the samples was 0.56. Similar results for the cesium species yielded an average 0.83 g/m²-d and a median of 0.58 [Ref. 12]. The surface area of the waste form may increase due to cracking during storage. A slight surface area increase is expected, but as a bounding condition, it is assumed that the surface area could increase by 100-fold if exposed to underground water in a leaching situation.

A.4.7 Heat Generation of Waste Form 2

Graphite Fuels

The fuel is more concentrated than in the base waste form while the container diameter is smaller. The heat generation rates or "Thermal Output" are 53 W/m³. The "Watt Density" rates per canister are 8 W/m (See Attachment 4)

Special Fuels

The heat generation potential of the special fuel waste form 2 is the sum of the base form and the glass-ceramic. The average "Thermal Output" is 41.5 W/m³, and the per-canister heat generation rate "Watt Density" is 26 W (See Attachment 4).

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Glass-Ceramic

Calcine materials converted to a glass-ceramic form will vary in heat generation rates depending on both the type of waste and the final form selected. Heat generation rates were calculated on the basis of curie inventories as of November 1992. Densification during post-treatment explains the higher heat generation rate when compared with the calcine.

Typically, the glass-ceramic of aluminum wastes are expected to range from 60 to 120 W/m³ of final waste volume. Glass-ceramic of zirconium and zirconium-sodium blends will range from 45 to 75 W/m³. Glass-ceramic of sodium wastes by themselves will range as low as 0.45 to 23 W/m³ depending on the final method selected to solidify the liquid wastes. Year 2030 heat generation potentials will be approximately 45 percent of the values listed.

Glass-ceramic forms have indicated thermal conductivities of 0.6 to 0.8 W/m-°K. Canister loadings will be controlled so that a maximum centerline temperature of the canister after glass-ceramic formation will not exceed 400 °C. This limitation is intended to prevent glass phase changes that might occur above 500 °C.

A.4.8 Criticality Issues

Graphite Fuels

Restricting the loaded fissile mass per container to less than 0.700 kg of effective U-235 brings the fissile loading below single parameter limits, at which criticality is impossible regardless of other conditions.

Special Fuels

The criticality issue for the special fuel will be the same as for the base waste form.

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A.5 Waste Form 3

Waste form 3 will contain burned graphite fuels encapsulated within borosilicate glass material. The generic salt repository will be evaluated as a disposal site for waste form 3, with a 10 kg U limit per canister for the graphite and special fuels. The generic granite repository will have a 0.700 kg U limit per canister for the graphite and special fuels. Special fuel will be encapsulated in borosilicate glass produced from calcine. Excess glass not needed to encapsulate the graphite and special fuels will be packaged separately in standard ≈ 61 by 300 cm canisters. The paragraphs below describe the characteristics of waste form 3 [Ref. Attachment 1 Page 3].

A.5.1 Volume of Waste Form 3

A.5.1.1 Spent Fuel

Graphite fuels

Waste form 3 will contain burned graphite fuels packaged in 10.16cm diameter (4 inch Sch 80 pipe) by 233.1-cm long Inconel containers, and the overpack container dimensions will be 14.13-cm in diameter by 258.5-cm in length (See Attachment 3). The size reduction is necessary to bring the container U-235 equivalent loading under 0.700 kg. The 10 kg U-235 loading will be packaged in a 60.96cm diameter by 299.7cm long container, and the overpack container dimensions will be 66.04cm by 325.1cm length (See Attachment 3a). For both the FSVR and the Peach Bottom core 2, it was assumed that the bulk graphite was burned to the SiC layer. The Peach Bottom core 1 elements are assumed to be burned to oxide ash. The "Packaged Volume" of this waste form is, excluding overpack material and containers, 42.3 m³ in 1801 containers for the 0.700 kg U-235 loading limit (See Attachment 4). The "Packaged Volume" for the 10 kg loading limit is 33.9 m³ in 152 containers (See Attachment 4a). [Values are from calculations based on information contained in Ref. 6, 15, 16, 17]

Special Fuels

The "Volume As Is" of the special fuels is 89 m³. "Packaged Volume", including the excess glass, is 12637m³ for the 0.700 kg U-235 loading limit case (See Attachment 4). The "Volume As Is" for the 10 kg loading limit case doesn't change, but the "Packaged Volume" increases to 1431 m³ (fuel w/glass) and 9946 m³ of glass only (See Attachment 4a).

A.5.1.2 Borosilicate Glass From Calcine

Existing calcine plus conversion of existing and anticipated liquid wastes to a borosilicate glass solid form is expected to produce a "Volume As Is" of 10625 m³; the low calcine loading relative to the quantity of additives needed to make the borosilicate glass results in a significant volume increase (See Attachment 4). All of the glass is accounted for in the special fuels discussion above.

Borosilicate glass has a thermal conductivity range of 0.86-1.2 W/m-OK. As with the glass-ceramic form, canister maximum temperature is limited to 400 °C.

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A.5.2 Mass of Waste Form 3

A.5.2.1 Mass of Spent Fuel

Graphite Fuels

The "Mass As Is" and the "Packaged Mass" of the graphite fuel is 32.7 metric tons (27 MTHM) and 195 metric tons, respectively, for Waste Form 3 (for 0.700 kg study of granite repository), including the Inconel containers. This excludes overpack material and containers (**See Attachment 4**). The 10 kg study has a "Mass As is" and "Packaged Mass" of 32.7 metric tons and 74 metric tons, respectively (**See Attachment 4a**).

Special Fuels

The "Mass Is" of the special fuel alone in Waste Form 3 is 474 metric tons (300 MTHM). The total "Packaged Mass" of the packaged fuel and glass is 41,676 metric tons for the 0.700 kg loading limit (**See Attachment 4**). The "Mass As Is" for the 10 kg loading limit doesn't change, but the "Packaged Mass" changes to 6317 metric tons (**See Attachment 4a**).

A.5.2.2 Borosilicate Glass

Borosilicate glass, similar to what is produced at the SRS and West Valley waste facilities, is a possible alternative to treatment of the calcine at ICPP; the waste loading per canister is not as high as for the glass-ceramic form. Considerations such as acceptance of borosilicate glass as BDAT, availability of proven technology, or Na waste treatment being more practical in a glass melter may dictate adoption of this waste form at ICPP.

The treated waste form will have a "Mass As Is" of 27,625 metric tons for the borosilicate glass form (due to lower waste loading relative to inerts needed to make the glass). This mass does not include the weight of canister materials used to contain the waste.

A.5.3 Containers for Waste Form 3

The containers for the glass matrix fuel/block separated waste form are discussed in Waste Form 2. There are 6500 canisters that contain special fuel-5613 small canisters and 887 large canisters. There will be 13,687 canisters containing glass only, for a total of 20,187 canisters for the 0.700 kg limit for the granite repository (**See Attachment 4**). The salt repository with a 10 kg loading limit will have 929 (633 HEU & 296 LEU) canisters containing special fuel and 15,888 canisters containing borosilicate glass (**See Attachment 4a**).

A.5.4 Reactions of Waste Form 3

All reactions will be much the same as for the base waste form.

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A.5.5 Drilling Erosion

This section will be provided by Sandia National Laboratories.

A.5.6 Leachability of Waste Form 3

Graphite Fuels

Using the same assumptions as for the base waste form, the following overall leach rates were derived: rate 1) 5164 g/d; rate 2) fission product-0.1570 g/d, actinides-0.006 g/d; and rate 3) fission product-0.114 g/d, and actinides-0.0011 g/d. It should be noted that the uranium leach rates should lie close to the fission product rates because of uranium carbonate solubility.

Special Fuels

The leach rates for the special fuel waste form 3 will be the same as the base form. The leach rates of the borosilicate glass that encapsulates the special fuel will be the same as listed below.

Borosilicate Glass

In the case of borosilicate glass, the leach rates should be very similar to the glass-ceramic as indicated in waste form 2. The surface area of the waste form may increase due to cracking during storage. A slight surface area increase is expected, but as a bounding condition, it is assumed that the surface area could increase by 1000-fold if exposed to underground water in a leaching situation.

A.5.7 Heat Generation of Waste Form 3

Graphite Fuels

The fuel is more concentrated than in the base waste form while the container diameter is smaller. The volume containing the same heat sources is much smaller than Waste Form 2. The heat generation rates for the 0.700 kg study are: 335 W/m³ (See Attachment 4: "Thermal Output") and 8 W/canister (See Attachment 4: "Watt Density"). The heat generation rates for the 10 kg study are: 428 W/m³ (See Attachment 4a: "Thermal Output") and 96 W/canister (See Attachment 4a: "Watt Density").

Special Fuels

The heat generation potential of the special fuel waste form 3 is the sum of the base form and borosilicate glass. The average thermal power density for the 0.700 kg study are: 17.6 W/m³ (See Attachment 4: "Thermal Output"), and the per-canister thermal power is 11.0 W (See Attachment 4: "Watt Density"). The average thermal power density for the 10 kg study are: 123 W/m³ (See Attachment 4: "Thermal Output"), and 396.3 W/canister (See Attachment 4a: "Watt Density").

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Borosilicate Glass

Calcine materials converted to a borosilicate glass form will vary in heat generation rates depending on both the type of waste and the final form selected. Heat generation rates were calculated on the basis of curie inventories as of the year 2030.

Typically, the borosilicate glass from aluminum wastes is expected to range from 20 to 50 W/m³ of final waste volume. Borosilicate glass from zirconium and zirconium-sodium blends will range from 18 to 35 W/m³. Borosilicate glass from sodium wastes by themselves will range as low as 0.4 to 19 W/m³, depending on the final method selected to solidify the liquid wastes. These heat generation potentials are year 2030 values.

A.5.8 Criticality Issues

Graphite Fuels

Same as Waste form 2.

Special Fuels

The criticality issue for the special fuel will be the same as the base waste form.

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A.6 Waste Form 4

Waste form 4 will consist of calcined waste immobilized in glass-ceramic. High-level waste from processing special and graphite fuels will be mixed with the glass-ceramic; assumed use of advanced, non-aqueous processing is expected to generate minimal HLW volumes. Part of the fuel separation process includes removal of the fissile actinide species to eliminate the potential for a criticality in the repository [Reference Attachment 1, page 4].

A.6.1 Volume of Waste Form 4

A.6.1.1 Spent Fuel

Graphite fuels

The spent material after uranium and thorium removal, (other actinides may also be removed) will be packaged in a glass waste form with glass filling the voids plus 100 percent of the original volume. The SiC hulls will also be in the waste form. About 98 percent of the thorium, 99.5 percent of the uranium, and 95 percent of the particle carbon will have been removed from the burned material. The waste will be packaged in the standard 60.96cm diameter by 299.7cm long container, and the overpack container dimensions will be 66.04cm in diameter by 325.1-cm in length. The "Volume As Is" will occupy 14.1m³ (See Attachment 4). The "Packaged Volume", excluding overpack material & containers (See Attachment 4 for further details), will occupy 33.3 m³ contained in only 38 canisters. [Values calculated from information from Ref. 6, 15, 16, 19, 20]

Special Fuels

The special fuels will be processed so as to remove actinides from the fuel material, which leaves fission products and structural material as high-level waste. The high-level waste generated will be converted to the glass-ceramic waste form used to immobilize calcine. The additional "Volume As Is" of glass-ceramic due to the special fuels is 5.7 m³ (See Attachment 4). The "Packaged Volume" will occupy 5638 m³ (See Attachment 4).

A.6.1.2 Glass-Ceramic

Existing calcine plus conversion of existing and anticipated liquid wastes to a glass-ceramic solid form is expected to produce a "Volume As Is" of 2975 m³ (See Attachment 4).

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A.6.2 Mass of Waste Form 4

A.6.2.1 Mass of Spent Fuel

Graphite Fuels

The "Packaged Mass" of graphite fuels Waste Form 4 is 80 metric tons, including the mass of the containers. This excludes overpack material and containers which are shown in Attachment 4.

Special Fuels

The "Mass As Is" of the processed special fuel in Waste Form 4 will amount to 22 MT, which will be added to 9994 metric tons glass-ceramic. The total "Packaged Mass" will be 26,574 metric tons (See Attachment 4).

A.6.2.2 Glass-Ceramic

Information on the glass-ceramic is included in the information for the special fuels above, and the characteristics of glass ceramic were discussed in Waste Form 2. The "Mass As Is" of glass-ceramic is 9,994 metric tons (See Attachment 4).

A.6.3 Containers for Waste Form 4

The containers for the glass-ceramic material are the standard $\approx$ 61 by 300 cm canister (See Attachment 3). The number of canisters will be 5061 (See Attachment 4).

A.6.4 Reactions of Waste Form 4

All reactions will not be the same as for the base waste form. The graphite fuels waste form 4 will no longer have the flammable gas generation of the previous three waste forms. The oxides will leach but, will generate limited amounts of gas, primarily the same as the corrosion on special fuels.

A.6.5 Drilling Erosion

This section will be provided by Sandia National Laboratories.

A.6.6 Leachability of Waste Form 4

Graphite Fuels

Using the same assumptions as for the base waste form, with the exception of glass leaching on rate 1, the following overall leach rates were derived: rate 1) 258 g/d; rate 2) fission product-0.030 g/d, actinides-nil g/d; and rate 3) fission product-0.0063 g/d, actinides-nil g/d. It should be noted that the uranium and other actinides will be mostly removed.

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Special Fuels

The leach rates for the special fuel Waste Form 4 will be the same as for the glass-ceramic waste form.

Glass-Ceramic

The final form will be either the borosilicate glass (form 3) or glass-ceramic (form 2) depending on cost and form performance in the repository.

A.6.7 Heat Generation of Waste Form 4

Graphite Fuels

The heat generation rates are 426 W/m³ and 372W/canister (See Attachment 4).

Special Fuels

The special fuel Waste Form 4 will contain the same fission product inventory as the base form special fuel and calcine added together. The "Thermal Output" of the glass-ceramic waste is 70.1 W/m³, and the per-canister "Watt Density" is 43.9 W/canister (See Attachment 4).

Glass-Ceramic

The characteristics of glass-ceramic are the same as for Waste Form 2. The glass ceramic heat generation has been added to that of the special fuel.

A.6.8 Criticality Issues

Graphite Fuels

There is no potential for criticality with the fissile material removed.

Special Fuels

There is no potential for criticality with the fissile material removed.

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A.7 Waste Form 5

Waste form 5 will consist of calcine waste in a borosilicate glass. Special & graphite fuels would require a conversion of the enriched fuels and depleted uranium to a common form e.g. uranyl nitrate or uranium hexafluoride, for intimate chemical mixing and sampling prior to conversion back to a stable oxide for encapsulation; the blend ratio will 'denature' the HEU to less than 0.70 percent enrichment for purposes of preventing criticality in the repository [Reference Attachment 1, page 5].

A.7.1 Volume of Waste Form 5

A.7.1.1 Spent Fuel

Graphite fuels

Waste Form 5 is the burned material converted to oxides then diluted with depleted U_3O_8 to a U^{235} equivalent concentration of less than 0.7 percent. This material would then be packaged in a similar manner to Waste Form 4. The resulting "Packaged Volume", excluding overpack material and containers (See Attachment 4 for further details), would be 177.1 M^3 in 202 standard containers. The "Volume As Is" is 75.4 m^3 , respectively (See Attachment 4). [Values from calculations based on information from Ref. 6, 15, 16, 18, 19]

Special Fuels

The special fuels will be converted to the oxide form and blended with depleted uranium oxide to an equivalent U-235 concentration of less than 0.7 weight percent. The oxide will be placed into standard ≈ 61 by 300 cm canisters (See Attachment 3). The "Volume As Is" and "Packaged Volume" will be 1569 m^3 and 881 m^3 , respectively (See Attachment 4).

A.7.1.2 Glass-Ceramic

Existing calcine plus conversion of existing and anticipated liquid wastes to a borosilicate glass solid form is expected to produce a "Volume As Is" of 2975 m^3 (See Attachment 4). Using ≈ 61 -cm diameter x 300-cm long canister at 0.626 (See Attachment 3) cubic meters each, the total "Package Volume" and number of canisters required is expected to be 5353 m^3 and 7301 (See Attachment 4).

Borosilicate glass has a thermal conductivity range of 0.86 to 1.2 W/m- $^{\circ}$ K. As with the glass-ceramic form, canister maximum temperature is limited to 400 $^{\circ}$ C.

A.7.2 Mass of Waste Form 5

A.7.2.1 Mass of Spent Fuel

Graphite Fuels

The "Packaged Mass" of graphite fuels Waste Form 5 is 676 metric tons,, (158

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MTHM) including the container mass. This excludes overpack material and containers (**See Attachment 4**).

Special Fuels

The "Mass As Is" of the special fuel in Waste Form 5 after blending with depleted uranium, will be 3656 metric tons (782 MTHM). The "Packaged Mass" will have a mass of 7558 metric tons (**See Attachment 4**).

A.7.2.2 Glass-Ceramic

The characteristics of glass-ceramic are described in Waste Form 2. The "Mass As Is" will be 9994 metric tons (**See Attachment 4**).

A.7.3 Containers for Waste Form 5

The containers for Waste Form 5 are the standard canisters and are discussed in the volume section above. There are 202 canisters for graphite fuel, 1408 canisters for special fuel, and 7301 canisters for glass-ceramic (**See Attachment 4**).

A.7.4 Reactions of Waste Form 5

Reactions will be similar to the special fuels.

A.7.5 Drilling Erosion

This section will be provided by Sandia National Laboratories.

A.7.6 Leachability of Waste Form 5

Graphite Fuels

Using the same assumptions as for the base waste form, with the exception of glass leaching on rate 1, the following overall leach rates were derived: rate 1) 258 g/d; rate 2) fission product-0.16 g/d, actinides-0.0016 g/d; and rate 3) fission product-0.0064 g/d, actinides-0.00006 g/d. It should be noted that the uranium leach rates should lie close to the fission product rates because of uranium carbonate solubility.

Special Fuels

The leach rates for the special fuel Waste Form 5 will be substantial, due the fact that many of the oxides that will be formed are soluble. Additional work will be required to determine leach rates if this alternative is pursued further.

Borosilicate glass

Borosilicate glass form will be as described in Waste Form 3.

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A.7.7 Heat Generation of Waste Form 5

Graphite Fuels

The fuel is more concentrated than in the base waste form while the container diameter is smaller. The volume containing the same heat sources is much smaller than Waste Form 2. The heat generation rates are 82 W/m³ and 72 W/canister (See Attachment 4: "Thermal Output & Watt Density").

Special Fuels

The special fuel Waste Form 5 will contain the same fission product inventory as the base form. The "Thermal Output" of the oxide waste form will be 151.2 W/m³, and the per-canister "Watt Density" will be 168.4 W/canister (See Attachment 4).

Glass-Ceramic

The characteristics of glass-ceramic are approximately the same as for Waste Form 2. Heat generation rates are 48.6 W/m³ and 30.4 W/canister (See Attachment 4: "Thermal Output" and "Watt Density").

A.7.8 Criticality Issues

Graphite Fuels

The fact that the waste form is now more dilute in fissile material than natural uranium will prevent criticality.

Special Fuels

For fissile material that has less than 0.7 wt percent U-235 equivalent, the fissile material is dilute enough to present no criticality concern, regardless of the mass or geometry. However, the effectiveness of the uranium blending is a key development issue. Since plutonium is chemically different from uranium, possible concentration mechanisms will be investigated in future PAs.

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ATTACHMENT 1 PAGE 1

INEL SPENT FUEL AND HLW REPOSITORY BASE FORM

Note: Drawing Not To Scale

BASEFMINW.GCD
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ATTACHMENT 1 PAGE 2
INEL SPENT FUEL AND HLW REPOSITORY FORM NO. 2

Note: Drawing Not To Scale

ATTACHMENT 1 PAGE 3
INEL SPENT FUEL AND HLW REPOSITORY FORM NO. 3

Note: Drawing Not To Scale

FORM 3.GCD
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ATTACHMENT 1 PAGE 4
INEL SPENT FUEL AND HLW REPOSITORY FORM NO. 4

Note: Drawing Not To Scale

FORM 4.GCD
REV 1-25-93

ATTACHMENT 1 PAGE 5
INEL SPENT FUEL AND HLW REPOSITORY FORM NO. 5

Note: Drawing Not To Scale

FORMS.GCD
REV 1-28-93

MATERIAL : TYPE 304L STAINLESS STEEL

NOTE: DIMENSIONS ARE IN
CENTIMETERS

PS8503-7

Hanford HLW canister. Source: Mitchell and Nelson 1988.

Attachment 2
Page 3

Savannah River Site HLW canister. Source: Kelker 1986.

Attachment 3

ICPP Spent Fuel and Waste Technology Development Program Waste Form, Canister Sizes and Material of Construction

Canister Sizes				
Waste Forms	Special Fuel	Graphite Fuel	Calcine/Glass Glass-Ceramic	Other DOE Complex Glass
Base Form	16.83cm x 160.0cm 32.39cm x 434.3cm	45.75cm x 98.6cm 27.31cm x 400.9cm	60.96cm x 299.7cm	60.96cm x 299.7cm
Overpack size	21.91cm x 177.8cm 35.56cm x 452.1cm	50.8cm x 124.0cm 32.39cm x 426.3cm	66.04cm x 325.1cm	66.04cm x 325.1cm
Form #2	60.96cm x 177.8cm 60.96cm x 452.1cm	20.32cm x 400.9cm	Part of special fuel	60.96cm x 299.7cm
Overpack size	66.04cm x 203.0cm 66.04cm x 477.4cm	27.31cm x 426.3cm		66.04cm x 325.1cm
Form #3	60.96cm x 177.8cm 60.96cm x 452.1cm 60.96cm x 299.7cm	10.16cm x 233.1cm	60.96cm x 299.7cm	60.96cm x 299.7cm
Overpack size	66.04cm x 203.0cm 66.04cm x 477.4cm 66.04cm x 325.1cm	14.13cm x 258.5cm	66.04cm x 325.1cm	66.04cm x 325.1cm
Form #4	Part of glass-ceramic	Part of glass-ceramic	60.96cm x 299.7cm	60.96cm x 299.7cm
Overpack size			66.04cm x 325.1cm	66.04cm x 325.1cm
Form #5	60.96cm x 299.7cm	60.96cm x 299.7cm	60.96cm x 299.7cm	60.96cm x 299.7cm
Overpack size	66.04cm x 325.1cm	66.04cm x 325.1cm	66.04cm x 325.1cm	66.04cm x 325.1cm

Evaluated Material of construction for the canister 304L Stainless Steel (SS)

Proposed Material of construction for the canister Inconel 625

Proposed Material of construction for the overpack TiCode 12 in salt repository; Inconel 625 in igneous rock repository

Wall thickness for base form fuel canisters 0.9525 cm

Wall thickness for other canisters and all overpacks 0.635 cm

For Waste Form 1, special fuel, graphite fuel, and calcine will be packaged in separate canisters. For Waste Forms 2 and 3, special fuel will be packaged in fuel canisters the same size as for Waste Form 1. The fuel canisters, along with glass-ceramic or glass, will be packaged in the 60.96 x 177.8 cm and 60.96 x 452.1 cm canisters. For Waste Form 3, the extra glass will be packaged in the 60.96 x 299.7 cm canisters. (There is no extra glass-ceramic for Waste Form 2.) The fill between canister and overpack and between fuel and fuel canister has not yet been determined. The principal requirement is that the metal melt at a lower temperature than aluminum. Lead or a lead alloy would have a number of good characteristics in this application, and its density is higher than other candidate metals, so the canister masses were calculated on the basis of lead. However, lead is a characteristic waste and may be banned. In any case the choice of a metal filler will have to receive further consideration.

Attachment 3a

ICPP Spent Fuel And Waste Technology Development Program Waste Form, Canister Sizes And Material of Construction For 10 kg's In A Salt Repository Only

Canister Sizes				
Waste Forms	Special Fuels	Graphite Fuels	Calcine/Glass Glass-Ceramic	Other DOE Complex Glass
Base Form Overpack Size	N/A	60.96 cm x 411.48 cm 66.04 cm x 426.30 cm	N/A	N/A
Form #2 Overpack Size	N/A	N/A	N/A	N/A
Form #3 Overpack Size	(HE) 60.96 cm x 401.20 cm (LE) 60.96 cm x 477.40 cm (HE) 66.04 cm x 426.50 cm (LE) 71.12 cm x 503.00 cm	10.16 cm x 233.10 cm 60.96 cm x 299.7 cm 14.13 cm x 258.5 cm 66.04 cm x 325.1 cm	N/A	N/A
Form #4 Overpack Size	N/A	N/A	N/A	N/A
Form #5 Overpack Size	N/A	N/A	N/A	N/A

Evaluated Material of construction for the canister 304L Stainless Steel (SS)

Proposed Material of construction for the canister Inconel 625

Proposed Material of construction for the overpack TiCode 12 in salt repository; Inconel 625 in igneous rock repository

Wall thickness for base form fuel canisters 0.9525 cm

Wall thickness for other canisters and all overpacks 0.635 cm

For Waste Form 1 graphite fuel will be packaged in separate canisters. For Waste Forms 3, special fuel will be packaged in fuel canisters the same size as for Waste Form 1. For Waste Form 3, the extra glass will be packaged in the 60.96 x 299.7 cm canisters. The fill between canister and overpack and between fuel and fuel canister has not yet been determined. The principal requirement is that the metal melt at a lower temperature than aluminum. Lead or a lead alloy would have a number of good characteristics in this application, and its density is higher than other candidate metals, so the canister masses were calculated on the basis of lead. However, lead is a characteristic waste and may be banned. In any case the choice of a metal filler will have to receive further consideration.

Attachment 4

WASTE FORM DATA SUMMARY
 (Salt or Granite Repository -
 0.700 kg U limit)

	Waste Form 1	Waste Form 2	Waste Form 3	Waste Form 4	Waste Form 5
<u>VOLUME AS IS, cubic meters</u>					
graphite	259.8	91.0	14.1	14.1	75.4
special fuels	89	89	89	5.7	881
INEL calcine/glass-ceramic	4250	2975	10625	2975	2975
Hanford glass	1227	1227	1227	1227	1227
SRP glass	3307	3307	3307	3307	3307
West Valley glass	193	193	193	193	193
TOTAL	9326	7882	15455	7722	8658
<u>PACKAGED VOLUME, cubic meters</u>					
graphite	541.1	269.0	42.3	33.3	177.1
special fuels	752	* incl'd.	* incl'd.	* incl'd.	1569
INEL calcine/glass-ceramic	4250	5353	12637	5638	5353
Hanford glass	1444	1444	1444	1444	1444
SRP glass	3891	3891	3891	3891	3891
West Valley glass	172	172	172	172	172
TOTAL	11050	11129	18186	11178	12606
<u>MASS AS IS, metric tons</u>					
graphite	342.0	74.3	32.7	12.7	318.9
special fuels	474	474	474	22	3656
INEL calcine/glass-ceramic	5950	9994	27625	9994	9994
Hanford glass	3234.0	3234.0	3234.0	3234.0	3234.0
SRP glass	8884.0	8884.0	8884.0	8884.0	8884.0
West Valley glass	522.5	522.5	522.5	522.5	522.5
TOTAL	19407	23183	40772	22669	26609

* incl'd accounts for combined masses, volumes, etc where waste components are mixed

Attachment 4

WASTE FORM DATA SUMMARY
 (Salt or Granite Repository -
 0.700 kg U limit)

	Waste Form 1	Waste Form 2	Waste Form 3	Waste Form 4	Waste Form 5
<u>CONTAINER MASS, metric tons</u>					
graphite	* incl'd.	* incl'd.	* incl'd.	* incl'd.	* incl'd.
special fuels	6840	incl'd.	incl'd.	* incl'd.	3902
-short fuel	216.7	216.7	216.7		
-long fuel	279.4	279.4	279.4		
-short		1404.4	1404.4		
-long		564.5	564.5		
-standard			794.0	2530.5	704.0
INEL calcine/glass-ceramic	3395	4276	10318	14027	20652
Hanford glass	980	980	980	980	980
SRP glass	2641	2641	2641	2641	2641
West Valley glass	69	69	69	69	69
TOTAL	14422	10431	17268	20248	28949
<u>PACKAGED MASS, metric tons</u>					
graphite	959	721	195	80	676
special fuels	7314	* incl'd.	* incl'd.	* incl'd.	7558
INEL calcine/glass-ceramic	9345	16734	41676	26574	30646
Hanford glass	4214	4214	4214	4214	4214
SRP glass	11525	11525	11525	11525	11525
West Valley glass	592	592	592	592	592
TOTAL	33949	33787	58203	42985	55211

* incl'd accounts for combined masses, volumes, etc where waste components are mixed

Attachment 4

WASTE FORM DATA SUMMARY
 (Salt or Granite Repository -
 0.700 kg U limit)

	Waste Form 1	Waste Form 2	Waste Form 3	Waste Form 4	Waste Form 5
NUMBER OF CONTAINERS					
graphite	3168	1801	1801	38	202
special fuels	6500	* incl'd.	* incl'd.	* incl'd.	1408
INEL calcine/glass-ceramic	6789	8551	20187	5061	7301
Hanford glass	1960	1960	1960	1960	1960
SRP glass	5283	5283	5283	5283	5283
West Valley glass	276	276	276	276	276
TOTAL	23976	17871	29506	12617	16430
MASS OF FILL, Pb/Cnstr (MT)					
graphite	1865.0	1334.5	646.6	152.1	808.6
special fuels					
-short fuel	1470.61	1470.61	1470.61		
-long fuel	2496.02	2496.02	2496.02		
-short	1396.0	8722.6	8722.6		
-long	510.3	2243.2	2243.2		
-standard			3152.2	1985.0	1985.0
INEL calcine/glass-ceramic	13659.7	17204.8	40616.0	10182.7	14689.6
SUBTOTAL	21397.6	33471.8	59347.2	12319.8	17483.2
MASS of OVERPACK/CONTAINER (MT)					
graphite	330.9	381.8	158.5	12.0	63.8
special fuels					
-short	50.4	1788.0	1788.0		
-long	211.9	420.5	420.5		
-standard			286.5	286.5	286.5
INEL calcine/glass-ceramic	1962.1	2471.3	5834.0	1462.6	2110.0
SUBTOTAL	2555	5062	8488	1761	2460

* incl'd accounts for combined masses, volumes, etc where waste components are mixed

Attachment 4

WASTE FORM DATA SUMMARY
(Salt or Granite Repository -
0.700 kg U limit)

	Waste Form 1	Waste Form 2	Waste Form 3	Waste Form 4	Waste Form 5
<u>TOTAL PACKAGE MASS (MT)</u>					
graphite	3154.6	2439.9	999.7	1524.1	8352.2
special fuels	13449.1	* incl'd.	* incl'd.	* incl'd.	9829.5
INEL calcine/glass-ceramic	24966	36411	88126	38219	47446
TOTAL	41570	38851	89126	39743	65627
<u>DENSITY of PKG'D VOLUME, MT/M3</u>					
graphite	1.77	2.68	4.61	2.40	3.81
special fuels	9.73	* incl'd.	* incl'd.	* incl'd.	4.82
INEL calcine/glass-ceramic	1.84	5.50	5.14	4.26	5.81
Hanford glass	2.91	2.91	2.91	2.91	2.91
SRP glass	2.96	2.96	2.96	2.96	2.96
West Valley glass	2.61	2.61	2.61	2.61	2.61
<u>DENSITY AS IS, MT/M3</u>					
graphite	1.32	0.82	2.32	0.90	4.23
special fuels	5.33	5.33	5.33	* incl'd.	4.15
INEL calcine/glass-ceramic	1.40	3.36	3.32	3.36	3.36
Hanford glass	2.63	2.63	2.63	2.63	2.63
SRP glass	2.73	2.73	2.73	2.73	2.73
West Valley glass	2.70	2.70	2.70	2.70	2.70

* incl'd accounts for combined masses, volumes, etc where waste components are mixed

Attachment 4

WASTE FORM DATA SUMMARY
 (Salt or Granite Repository -
 0.700 kg U limit)

	Waste Form 1	Waste Form 2	Waste Form 3	Waste Form 4	Waste Form 5
<u>FISSILE MATERIAL DENSITY</u>					
kg/M3 10/31/2030					
graphite	2.01	4.03	25.69	0.33	6.13
special fuels	13.80	* incl'd.	* incl'd.	0.00	6.59
INEL calcine/glass-ceramic	low	1.93	1.45	low	low
Hanford glass	na	na	na	na	na
SRP glass	na	na	na	na	na
West Valley glass	na	na	na	na	na
<u>FISSILE MATERIAL, mean kg/can</u>					
graphite	0.343	0.603	0.603	0.287	5.373
special fuels	1.59	* incl'd.	* incl'd.	0	7.35
INEL calcine/glass-ceramic	low	1.591	1.280	low	low
<u>TOTAL</u>					
<u>WATT DENSITY, watts /canister</u>					
[decayed to 10/31/2030]					
graphite	4	8	8	372	72
special fuels	24.8	* incl'd.	* incl'd.	incl'd	168.4
INEL calcine/glass-ceramic	9.0	26.0	11.0	43.9	30.4
Hanford glass	391	391	391	391	391
SRP glass	328	328	328	328	328
West Valley glass	147	147	147	147	147

* incl'd accounts for combined masses, volumes, etc where waste components are mixed

Attachment 4

WASTE FORM DATA SUMMARY
 (Salt or Granite Repository -
 0.700 kg U limit)

	Waste Form 1	Waste Form 2	Waste Form 3	Waste Form 4	Waste Form 5
Thermal Output, watts/M3 in 2030					
graphite	26	53	335	426	82
special fuels	234.0	* incl'd.	* incl'd.	* incl'd.	151.2
INEL calcine/glass-ceramic	14.4	41.5	17.6	70.1	48.6
Hanford glass	625	625	625	625	625
SRP glass	524	524	524	524	524
West Valley glass	210	210	210	210	210
Radionuclide concentration, Ci/M3					
graphite	4102	8252	52482	66666	12535
special fuels	71700	* incl'd.	* incl'd.	* incl'd.	34385
INEL calcine/glass-ceramic	4812	6874	1925	13205	7500
Hanford glass	181994	181994	181994	181994	181994
SRP glass	143151	143151	143151	143151	143151
West Valley glass	78488	78488	78488	78488	78488
Radionuclide inventory, Ci					
graphite	2.22E+06	2.22E+06	2.22E+06	2.22E+06	2.22E+06
special fuels	5.40E+07	5.40E+07	5.40E+07	5.40E+07	5.40E+07
INEL calcine/glass-ceramic	2.05E+07	2.05E+07	2.05E+07	2.05E+07	2.05E+07
Hanford glass	2.63E+08	2.63E+08	2.63E+08	2.63E+08	2.63E+08
SRP glass	5.57E+08	5.57E+08	5.57E+08	5.57E+08	5.57E+08
West Valley glass	1.35E+07	1.35E+07	1.35E+07	1.35E+07	1.35E+07
TOTAL	9.10E+08	9.10E+08	9.10E+08	9.10E+08	9.10E+08

* incl'd accounts for combined masses, volumes, etc where waste components are mixed

Attachment 4a

WASTE FORM DATA SUMMARY
 (Salt Repository [only] -
 10 kg U Limit)

	Waste Form 1	Waste Form 2	Waste Form 3	Waste Form 4	Waste Form 5
<u>VOLUME AS IS, cubic meters</u>					
graphite	259.8	91.0	14.1	14.1	75.4
special fuels	89	89	89	5.7	881
INEL calcine/glass-ceramic	4250	2975	10625	2975	2975
Hanford glass	1227	1227	1227	1227	1227
SRP glass	3307	3307	3307	3307	3307
West Valley glass	193	193	193	193	193
TOTAL	9326	7882	15455	7722	8658
<u>PACKAGED VOLUME, cubic meters</u>					
graphite	532.6	269.0	33.9	33.3	177.1
special fuels	752	* incl'd.	1431 *	* incl'd.	1569
INEL calcine/glass-ceramic	4250	5353	9546 **	5638	5353
Hanford glass	1444	1444	1444	1444	1444
SRP glass	3891	3891	3891	3891	3891
West Valley glass	172	172	172	172	172
TOTAL	11042	11129	16918	11178	12606
			*fuel w/glass		
			**glass only		
<u>MASS AS IS, metric tons</u>					
graphite	342.0	74.3	32.7	12.7	318.9
special fuels	474	474	474	22	3656
INEL calcine/glass-ceramic	5950	9994	27625	9994	9994
Hanford glass	3234.0	3234.0	3234.0	3234.0	3234.0
SRP glass	8884.0	8884.0	8884.0	8884.0	8884.0
West Valley glass	522.5	522.5	522.5	522.5	522.5
TOTAL	19407	23183	40772	22669	26609

*incl'd accounts for combined masses, volumes, etc. where waste components are mixed

Attachment 4a

WASTE FORM DATA SUMMARY
(Salt Repository [only] -
10 kg U Limit)

	Waste Form 1	Waste Form 2	Waste Form 3	Waste Form 4	Waste Form 5
<u>CONTAINER MASS, metric tons</u>					
graphite	* incl'd.	* incl'd.	* incl'd.	* incl'd.	* incl'd.
special fuels	6840	* incl'd.	* incl'd.	* incl'd.	3902
-short fuel	216.7	216.7	163.8		
-long fuel	279.4	279.4	268.1		
-short		1404.4	334.9		
-long		564.5	204.3		
-standard				2530.5	704.0
INEL calcine/glass-ceramic	3395	4276	7944	14027	20652
Hanford glass	980	980	980	980	980
SRP glass	2641	2641	2641	2641	2641
West Valley glass	69	69	69	69	69
TOTAL	14422	10431	12606	20248	28949
<u>PACKAGED MASS, metric tons</u>					
graphite	650	721	74	80	676
special fuels	7314	* incl'd.	6317	* incl'd.	7558
INEL calcine/glass-ceramic	9345	16734	37014	26574	30646
Hanford glass	4214	4214	4214	4214	4214
SRP glass	11525	11525	11525	11525	11525
West Valley glass	592	592	592	592	592
TOTAL	33640	33787	59736	42985	55211

*incl'd accounts for combined masses, volumes, etc. where waste components are mixed

Attachment 4a

WASTE FORM DATA SUMMARY
 (Salt Repository [only] -
 10 kg U Limit)

	Waste Form 1	Waste Form 2	Waste Form 3	Waste Form 4	Waste Form 5
NUMBER OF CONTAINERS					
graphite	442	1801	152	38	202
special fuels	6500	* incl'd.	929 *	* incl'd.	1408
INEL calcine/glass-ceramic	6789	8551	15888	5061	7301
Hanford glass	1960	1960	1960	1960	1960
SRP glass	5283	5283	5283	5283	5283
West Valley glass	276	276	276	276	276
TOTAL	21250	17871	24488	12617	16430
			*63% HEU & 296 LEU		
MASS OF FILL, Total Pb (MT)					
graphite	865.4	1334.5	5733.4	152.1	808.6
special fuels					
-short fuel	1470.61	1470.61	1382.00		
-long fuel	2496.02	2496.02	1690.00		
-short	1396.0	8722.6	649.0		
-long	510.3	2243.2	793.0		
-standard				1985.0	1985.0
INEL calcine/glass-ceramic	13659.7	17204.8	10397.0	10182.7	14689.6
SUBTOTAL	20398.0	33471.8	20644.4	12319.8	17483.2
MASS of OVERPACK/CONTAINER (MT)					
graphite	19.0	381.8	26.4	12.0	63.8
special fuels					
-short	50.4	1788.0	223.6		
-long	211.9	420.5	134.4		
-standard				286.5	286.5
INEL calcine/glass-ceramic	1962.1	2471.3	1501.0	1462.6	2110.0
SUBTOTAL	2243	5062	1885	1761	2460

*incl'd accounts for combined masses, volumes, etc. where waste components are mixed

Attachment 4a

WASTE FORM DATA SUMMARY
 (Salt Repository [only] -
 10 kg U Limit)

	Waste Form 1	Waste Form 2	Waste Form 3	Waste Form 4	Waste Form 5
<u>TOTAL PACKAGE MASS (MT)</u>					
graphite	2384.8	2439.9	137.9	1524.1	8352.2
special fuels	13449.1	* incl'd.	5843	* incl'd.	9829.5
INEL calcine/glass-ceramic	24966	36411	14517	38219	47446
TOTAL	40800	38851	20498	39743	65627
<u>DENSITY of PKG'D VOLUME, MT/M3</u>					
graphite	1.22	2.68	2.17	2.40	3.81
special fuels	9.73	* incl'd.	4.41	* incl'd.	4.82
INEL calcine/glass-ceramic	1.84	5.50	4.67	4.26	5.81
Hanford glass	2.91	2.91	2.91	2.91	2.91
SRP glass	2.96	2.96	2.96	2.96	2.96
West Valley glass	2.61	2.61	2.61	2.61	2.61
<u>DENSITY AS IS, MT/M3</u>					
graphite	1.32	0.82	2.32	0.90	4.23
special fuels	5.33	5.33	5.33	* incl'd.	4.15
INEL calcine/glass-ceramic	1.40	3.36	3.32	3.36	3.36
Hanford glass	2.63	2.63	2.63	2.63	2.63
SRP glass	2.73	2.73	2.73	2.73	2.73
West Valley glass	2.70	2.70	2.70	2.70	2.70

*incl'd accounts for combined masses, volumes, etc. where waste components are mixed

Attachment 4a

WASTE FORM DATA SUMMARY
 (Salt Repository [only] -
 10 kg U Limit)

	Waste Form 1	Waste Form 2	Waste Form 3	Waste Form 4	Waste Form 5
<u>FISSILE MATERIAL DENSITY</u>					
kg/M3 10/31/2030					
graphite	2.04	4.03	32.00	0.33	6.13
special fuels	13.80	* incl'd.	7.23	0.00	6.59
INEL calcine/glass-ceramic	low	1.93	low	low	low
Hanford glass	na	na	na	na	na
SRP glass	na	na	na	na	na
West Valley glass	na	na	na	na	na
<u>FISSILE MATERIAL, mean kg/can</u>					
graphite	2.456	0.603	7.141	0.287	5.373
special fuels	1.59	* incl'd.	11.1	0	7.35
INEL calcine/glass-ceramic	low	1.591	low	low	low
<u>TOTAL</u>					
<u>WATT DENSITY, watts /canister</u>					
[decayed to 10/31/2030]					
graphite	33	8	96	372	72
special fuels	24.8	* incl'd.	396.3	* incl'd.	168.4
INEL calcine/glass-ceramic	9.0	26.0	14.0	43.9	30.4
Hanford glass	391	391	391	391	391
SRP glass	328	328	328	328	328
West Valley glass	147	147	147	147	147

*incl'd accounts for combined masses, volumes, etc. where waste components are mixed

Attachment 4a

WASTE FORM DATA SUMMARY

(Salt Repository [only] -

10 kg U Limit)

	Waste Form 1	Waste Form 2	Waste Form 3	Waste Form 4	Waste Form 5
<u>Thermal Output, watts/M3 in 2030</u>					
graphite	27	53	428	426	82
special fuels	234.0	* incl'd.	123.0	* incl'd.	151.2
INEL calcine/glass-ceramic	14.4	41.5	22.3	70.1	48.6
Hanford glass	625	625	625	625	625
SRP glass	524	524	524	524	524
West Valley glass	210	210	210	210	210
<u>Radionuclide concentration, Ci/M3</u>					
graphite	4168	8252	65486	66666	12535
special fuels	71700	* incl'd.	37701	* incl'd.	34385
INEL calcine/glass-ceramic	4812	6874	3513	13205	7500
Hanford glass	181994	181994	181994	181994	181994
SRP glass	143151	143151	143151	143151	143151
West Valley glass	78488	78488	78488	78488	78488
<u>Radionuclide inventory, Ci</u>					
graphite	2.22E+06	2.22E+06	2.22E+06	2.22E+06	2.22E+06
special fuels	5.40E+07	5.40E+07	5.40E+07	5.40E+07	5.40E+07
INEL calcine/glass-ceramic	2.05E+07	2.05E+07	2.05E+07	2.05E+07	2.05E+07
Hanford glass	2.63E+08	2.63E+08	2.63E+08	2.63E+08	2.63E+08
SRP glass	5.57E+08	5.57E+08	5.57E+08	5.57E+08	5.57E+08
West Valley glass	1.35E+07	1.35E+07	1.35E+07	1.35E+07	1.35E+07
TOTAL	9.10E+08	9.10E+08	9.10E+08	9.10E+08	9.10E+08

*incl'd accounts for combined masses, volumes, etc. where waste components are mixed

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Appendix B: Development of EPA 40 CFR 191

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Appendix B: Development of EPA 40 CFR 191

M. M. Gruebel

Although international organizations such as the International Commission on Radiological Protection (ICRP) and the Nuclear Energy Agency (NEA) provided considerable guidance for writing radiological standards (NEA, 1984; ICRP, 1985; IAEA, 1989), promulgation of 40 CFR 191 was mandated by Congress in the Nuclear Waste Policy Act (NWPA) of 1982 before the guidance. The standards in 40 CFR 191 represented the first legislated attempt to formulate a regulation for disposal of high-level waste (HLW), spent nuclear fuel (SNF), and transuranic (TRU) radioactive waste. Prior to 40 CFR 191, United States agencies had little experience and no proven methodologies or procedures to draw on. As a result, 40 CFR 191 contains some new and innovative methods of regulating radioactive-waste disposal. Although the final regulation is relatively short and simple, it underwent as many as 25 revisions before its promulgation in September 1985. The text of the 1985 version of 40 CFR 191 is an attachment to this appendix.

Unless otherwise noted, the information in this discussion was derived from the environmental impact statement (EPA, 1982) published at the same time that a draft of the original 40 CFR 191 was produced in 1982; the 1985 version of 40 CFR 191 (EPA, 1985a); the background information document (EPA, 1985b) issued with 40 CFR 191; and a report by a group contracted by the U.S. Environmental Protection Agency (EPA) (EPA, 1985c) that provided many of the background calculations on which the regulation was based.

This discussion is not intended to trace the history of the regulation from its original draft form to the anticipated repromulgation in late 1993. Instead, this discussion uses the original documentation published by the EPA in 1982 and 1985 to present what the EPA stated at the time was its philosophy and methodology concerning the content of the 1985 version of 40 CFR 191. Because of the lengthy history of this regulation and its remand in 1987, the EPA has published a considerable amount of additional material, especially since early 1990 when the agency began to produce a series of working drafts in advance of repromulgation. Only the Individual Protection and Groundwater Protection Requirements of Subpart B were specifically cited by the court, but review panels, advisory boards, and individual investigators have recommended numerous modifications to the EPA radioactive-waste standards as a whole. Many of the reviewers recommended substantial changes in regulation philosophy, format, and stringency. However, much of the information has not been published. Several in-depth discussions of the history and implications of various parts of 40 CFR 191 can be found in published reports (NAS, 1983; SAB, 1984; Klett, 1991; Klett and Gruebel, 1992).

The EPA considered five general questions when it set the standards in 40 CFR 191 (EPA, 1982, p. 37):

1. How should risks to future generations be treated?
2. How should the capability of disposal systems to reduce long-term risks be assessed?
3. How should the uncertainties in predicting the future be considered?
4. How should the standards be expressed to best reflect the desired environmental protection objectives?
5. How should the regulatory time period be selected?

The ways in which the EPA incorporated its views about these questions into 40 CFR 191 are discussed in the following sections.

B.1 Risks to Future Generations

The EPA stated that it did not attempt to resolve the issue of intergenerational risk by making risk-versus-benefit judgments (EPA, 1982, pp. 38-48). The EPA believed that foreseeable disposal technologies were not likely to

eliminate risks to future generations; thus, risks to future generations should be small—at the very least, future generations should not bear any radiation risks from disposal that the current generation was not willing to assume. The EPA's analysis of the risks associated with undisturbed uranium ore bodies reinforced their decision about the reasonableness of the residual risks permitted by 40 CFR 191. According to the EPA (EPA, 1982, pp. 38-48; EPA, 1985a, p. 38083), the analysis indicated that leaving equivalent amounts of uranium ore unmined compared to the amount of ore needed to produce high-level wastes that would fill a repository presented at least as great a risk to future generations as disposal of the wastes covered by the regulation. Additionally, the residual risk represented in 40 CFR 191 was compared to variations in natural background and to natural radionuclide concentrations in groundwater (EPA, 1985a, p. 38083).

Fundamental criteria are the only standards that explicitly define the radiological safety requirements of repositories (Klett, 1991, pp. 5-14; Klett and Gruebel, 1992, pp. 9-12). The fundamental criterion for spent fuel, high-level and transuranic waste in 40 CFR 191 allows no more than 1000 premature cancer deaths over the first 10,000 years from disposal of the wastes produced from 100,000 metric tons of heavy metals (MTHM) (average of 10^{-6} health effects/MTHM-yr) (EPA, 1985a, p. 38082). The EPA's rationale for using this criterion was that it satisfied the following two objectives (Klett, 1991, p. 5):

The risk from the high-level waste disposal segment of the fuel cycle would be at least a factor of 100 lower than predicted for an equivalent amount of unmined uranium from three actual ore bodies and a factor of 100 higher than predicted for a hypothetical minimum release ore body (EPA, 1982, pp. 38-48; EPA, 1985a, p. 38083; EPA, 1985b, p. 8-49 to 8-50).

The requirements appear to be achievable by a variety of combinations of repository sites and designs without significant effects on the cost of disposing of these wastes (EPA, 1985a, p. 38084).

B.2 Assessment and Reduction of Risks

In developing the standards for disposal of HLW, SNF, and TRU wastes, the EPA concluded that mined geologic repositories had good capability for isolating wastes from the environment (EPA, 1985a, p. 38067). The EPA also indicated that the various provisions of Subpart B were intended to be met through a combination of steps involving disposal system site selection, design, and operational techniques (i.e., engineered barriers). The standards were intended to be taken into consideration in devising the proper selection of controls (EPA, 1985a, p. 38070).

In developing the Containment Requirements in 40 CFR 191, the EPA assessed the characteristics and performance of a number of model geologic repositories similar to those systems being considered by the DOE. The EPA stated that "potential radionuclide releases over 10,000 years were evaluated, and very general models of environmental transport and a linear, non-threshold dose-effect relationship were used to relate these releases to the incidence of premature cancer deaths they might cause" (EPA, 1985a, p. 38071).

The EPA chose for the Containment Requirements a single derived standard limiting time-integrated radionuclide releases from repository boundaries that applies to all HLW, SNF, and TRU-waste repositories and all release modes. Derived standards are computed from dose criteria (Table B-1) using nuclide-transport models. The purpose of derived standards is to facilitate regulations, control, and monitoring. They do not replace the fundamental or dose criteria and are a measure of risk only as they relate to the fundamental criteria (ICRP, 1984). The uses and limitations of derived standards are discussed in other reports (Klett, 1991; Klett and Gruebel, 1992).

The generic models used by the EPA considered four general modes of radionuclide releases to the environment: direct releases to a river, an ocean, and a land surface, and releases due to a volcanic or meteoritic interaction with a waste repository. The purpose of the EPA analyses was to compute the number of premature cancer deaths per curie (risk factors) of each radionuclide released to the biosphere via the various release modes. The analyses assumed that all radionuclides leaving any surface of any repository boundary were instantaneously deposited in the river or ocean, or on the land surface (EPA, 1985b, p. 7-13).

Table B-1. Criteria Traceability Using Examples of Design Requirements for the Proposed Yucca Mountain Repository in Nevada (Klett and Gruebel, 1992, p. 10)

Criteria Level	Criteria Category	Criteria Example
1	Fundamental	Level of allowable risk to the populace (40 CFR 191 [EPA, 1985a])
2	Dose	Level of allowable dose (ICRP, 1990; Working Draft 4 of 40 CFR 191 [EPA, 1992])
3	Derived	Time-integrated radionuclide flux across repository boundaries (40 CFR 191 [EPA, 1985a])
4	Prescriptive	Retrieval, containment, and groundwater travel time (10 CFR 60 [NRC, 1983])
5	Functional Requirements	Dry canister, no spalling, no phase change, limit on fuel rod failure, limit on surface condition change, limit on rock thermal fracturing, and maintain human access to drifts
6	Design Requirements	Temperature limits on canisters, boreholes, drifts, and the formation
7	Secondary Design Requirements	Areal power density, repository layout, inventory, ventilation cooling, and ventilation drying

In modeling modes of release, the EPA stated that it used very general models of environmental pathways and assumed population sizes and characteristics similar to present-day conditions (EPA, 1985c, p. 5-1). The EPA specifically mentioned that it had avoided detailed analytical techniques that would be more appropriate for near-term environmental assessments of specific facilities. Total health effects were calculated for the entire population exposed to the releases from a repository rather than terminated at an arbitrary distance from the repository (EPA, 1985b, p. 7-2). Total fatal cancers per curie were calculated for each of the four release modes. The most stringent of the sets of values—that for surface water—was used in calculating the release limits for the Containment Requirements (EPA, 1985b, p. 7-13).

B.3 Dealing with Uncertainties

The EPA recognized the potential for uncertainties in expectations about the future in its consideration of the possible harm that might be incurred by future generations as a result of releases from a disposal system. In its development of 40 CFR 191, the EPA used conservative values in its assessments and included qualitative assurance requirements. The EPA divided uncertainties into two categories: technical uncertainties and uncertainties in human behavior (EPA, 1982, pp. 54-62).

Technical uncertainties were considered to be those pertaining to the effectiveness of the barriers between the waste and the human environment (EPA, 1982, pp. 54-57). The EPA made their estimates by considering the performance of each of the barriers, using conservative values, and combining them into an assessment of the performance of the entire system. Uncertainties in the estimates were taken into account by selecting values for the effectiveness of each barrier that were believed to be well within the capabilities of engineering and site characterization. Redundant, independent barriers, as required by the Assurance Requirements, were considered a means of qualitatively dealing with technical uncertainties. An additional Assurance Requirement requires a repository design that allows recovery of most of the wastes for a reasonable period after disposal in the event that a disposal system becomes excessively hazardous. The recovery requirement was included to provide for the possibility of currently unknown processes and phenomena that could produce much greater risks than predicted by analysts at the time of disposal (EPA, 1982, p. 56).

Inability to predict human behavior in the future adds to uncertainty in assessments of repository performance. The EPA listed two ways in which future human behavior might affect the risks from disposal facilities (EPA, 1982, p. 57):

Human actions might change the ability of the system to isolate the wastes (e.g., if a person drilled through a geologic repository while exploring for mineral resources, a pathway might be created through which radionuclides could reach humans).

Human actions might affect the likelihood of human exposure to radionuclides that have escaped from isolation (e.g., people are exposed to radionuclides in contaminated groundwater only as a result of using the water).

To compensate for uncertainties in future human behavior, EPA's Assurance Requirements require institutional controls intended to assure to the extent possible, the integrity of a repository and address the possibility of human damage to isolation capability (EPA, 1982, pp. 57-61). Reliance on "active" institutional controls, such as guarding a disposal site and performing maintenance operations, is limited to 100 years. The EPA expected "passive" institutional controls, such as permanent markers placed at a disposal site, public records or archives, Federal ownership or control of land use, and other methods of preserving knowledge about the disposal site, to reduce but not eliminate the chance of inadvertent human intrusion well beyond 100 years. It also expected such measures to be effective in deterring systematic or persistent exploitation of a disposal site. An additional Assurance Requirement is that wastes should not be placed in an area where there is a reasonable expectation of future mining or exploration unless the isolating capabilities of the site compensate for the increased probability of intrusion (EPA, 1992, pp. 61-62). This requirement was aimed at sites containing a substantial concentration of any material that is not widely available to allow for uncertainty about what materials may be considered valuable in the future.

B.4 Format for Disposal Standards

The EPA considered a number of possible ways to limit risks to individuals or population groups (EPA, 1982, pp. 62-64). The form selected for 40 CFR 191 expresses the requirements in terms of total integrated radionuclide releases (curies) to the accessible environment over a long period of time. The main reason for choosing this form was ease of implementation—it requires estimation of the probable quantities of radionuclides leaving the disposal site and their physical transport to a location accessible to humans. Other alternatives (such as standards expressed in numbers of health effects over some period of time, in population dose equivalents over some period of time, in dose equivalent to individuals over a lifetime, in dose equivalent rates to individuals, in radionuclide release rates to the accessible environment, or concentration of radionuclides in environmental media) required assumptions and knowledge that the EPA thought would be difficult or impossible to justify or obtain, might not assure adequate protection, or would be difficult to implement.

B.5 Regulatory Time Periods

The EPA's goal for the disposal standards was to provide protection as long as the wastes presented an unacceptable risk (EPA, 1982, pp. 64-68). The regulation requires a reasonable expectation that releases over a specified time will be less than the specified limits. The EPA indicated that stating a period of time for regulation did not imply that radioactive wastes would be released after that time; rather, providing a time period established the period over which the releases must be assessed.

The EPA selected 10,000 years as the time period for regulation for the Containment Requirements. The agency believed that within 10,000 years "many of the radionuclides that [could] pass through groundwater transport paths [would] have reached surface water and the accessible environment" and that "the radiological hazard of the wastes would be substantially reduced through the decay of most of the significant fission products and many of the actinides" (EPA, 1982, p. 66). The EPA also stated that, because 10,000 years is a short period of time geologically, changes in geological conditions were expected to be small (EPA, 1982, p. 67). Time periods of 100 and 1000 years

1 were rejected because few health effects were predicted for that time span—many nuclides would be in transit toward
2 the accessible environment, the probability of disruptive events would be low, and many of the important nuclides
3 would not have undergone significant radioactive decay. Longer time periods than 10,000 years were rejected
4 because making quantitative assessments of the potential environmental impacts of waste disposal over longer peri-
5 ods of time would be difficult and predictions of geological changes over longer time periods would not be reliable.
6 An additional alternative, to limit the risks so that no future generation would incur more than a fraction of the risks
7 that would be acceptable to the current generation, was considered too difficult to implement because the risks to each
8 generation for a long period of time would have to be examined to determine if they were below the appropriate level
9 (EPA, 1982, pp. 67-68).
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13 The time period for the Individual Protection and Groundwater Protection Requirements is 1000 years of undis-
14 turbed performance. The EPA believed that this time period was long enough to ensure that particularly good engi-
15 neered barriers would be used at any potential site where some groundwater would be expected to flow through the
16 repository (EPA, 1985a, p. 38073). The agency felt that demonstrating compliance with individual exposure limits
17 longer than 1000 years would be difficult because of the analytical uncertainties involved. The EPA believed that the
18 only certain way at most potential sites to comply with individual- and groundwater-protection requirements related
19 to a time period of 10,000 years would be to use very expensive engineered barriers that would rule out any potential
20 releases over most of the period. The EPA stated that “while such barriers could provide longer-term protection for
21 individuals, they would not provide substantial benefits to populations because the containment and assurance
22 requirements already reduce populations risks to very small levels” (EPA, 1985a, p. 38073).
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Attachment A:

**40 CFR 191, Environmental Standards
for the Management and Disposal of
Spent Nuclear Fuel, High-Level
and Transuranic Radioactive Wastes;
Final Rule**

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Thursday
September 19, 1985

Part II

Environmental Protection Agency

40 CFR Part 191

Environmental Standards for the
Management and Disposal of Spent
Nuclear Fuel, High-Level and Transuranic
Radioactive Wastes; Final Rule

ENVIRONMENTAL PROTECTION AGENCY

40 CFR Part 191

(AH-FRL-2870-3)

Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes

AGENCY: Environmental Protection Agency.

ACTION: Final rule.

SUMMARY: The Environmental Protection Agency (EPA) is promulgating generally applicable environmental standards for the management and disposal of spent nuclear fuel and high-level and transuranic radioactive wastes. The standards apply to management and disposal of such materials generated by activities regulated by the Nuclear Regulatory Commission (NRC) and to disposal of similar materials generated by atomic energy defense activities under the jurisdiction of the Department of Energy (DOE). These standards have been developed pursuant to the Agency's authorities and responsibilities under the Atomic Energy Act of 1954, as amended; Reorganization Plan No. 3 of 1970; and the Nuclear Waste Policy Act of 1982.

Subpart A of these standards limits the radiation exposure of members of the public from the management and storage of spent fuel or high-level or transuranic wastes prior to disposal at waste management and disposal facilities regulated by the NRC. Subpart A also limits the radiation exposures to members of the public from waste emplacement and storage operations at DOE disposal facilities that are not regulated by the NRC.

Subpart B establishes several different types of requirements for disposal of these materials. The primary standards for disposal are long-term containment requirements that limit projected releases of radioactivity to the accessible environment for 10,000 years after disposal. These release limits should insure that risks to future generations from disposal of these wastes will be no greater than the risks that would have existed if the uranium ore used to create the wastes had not been mined to begin with. A set of six qualitative assurance requirements is an equally important element of Subpart B designed to provide adequate confidence that the containment requirements will be met. The third set of requirements are limitations on exposures to individual members of the public for 1,000 years after disposal.

Finally, a set of ground water protection requirements limits radionuclide concentrations for 1,000 years after disposal in water withdrawn from most Class I ground waters to the concentrations allowed by the Agency's interim drinking water standards (unless concentrations in the Class I ground waters already exceed the limits in 40 CFR Part 141, in which case this set of requirements would limit the increases in the radionuclide concentrations to those specified in 40 CFR Part 141). Subpart B also contains informational guidance for implementation of the disposal standards to clarify the Agency's intended application of these standards, which address a time frame without precedent in environmental regulations. Although disposal of these materials in mined geologic repositories has received the most attention, the disposal standards apply to disposal by any method, except disposal directly into the oceans or ocean sediments.

This notice describes the final rule that the Agency developed after considering the public comments received on the proposed rule published on December 29, 1982, and the recommendations of a technical review conducted by the Agency's Science Advisory Board (SAB). The major comments received on the proposed standards are summarized together with the Agency's responses to them. Detailed responses to all the comments received are discussed in the Response to Comments Document prepared for this final rule.

DATE: These standards shall be promulgated for purposes of judicial review at 1:00 p.m. eastern time on October 3, 1985. These standards shall become effective on November 18, 1985.

ADDRESSES: *Background Information*—The technical information considered in developing this rule, including risk assessments of disposal of these wastes in mined geologic repositories, is summarized in the Background Information Document (BID) for 40 CFR Part 191, EPA 520/1-85-023. Single copies of both the BID and the Response to Comments Document, as available, may be obtained from the Program Management Office (ANR-458), Office of Radiation Programs, Environmental Protection Agency, Washington, DC 20460; telephone number (703) 557-9351.

Docket—Docket Number R-82-3 contains the rulemaking record for 40 CFR Part 191. The docket is available for inspection between 8 a.m. and 4 p.m. on weekdays in the West Tower Lobby, Gallery 1, Central Docket Section, 401 M Street, SW., Washington, DC. A

reasonable fee may be charged for copying.

FOR FURTHER INFORMATION CONTACT: Dan Egan or Ray Clark, Criteria and Standards Division (ANR-460), Office of Radiation Programs, Environmental Protection Agency, Washington, DC 20460; telephone number (703) 557-8610.

SUPPLEMENTARY INFORMATION:

Fissioning of nuclear fuel in nuclear reactors creates a small quantity of highly radioactive materials. Virtually all of these materials are retained in the "spent" fuel elements when they are removed from the reactor. If the fuel is then reprocessed to recover unfissioned uranium and plutonium, most of the radioactivity goes into acidic liquid wastes that will later be converted into various types of solid materials. These highly radioactive liquid or solid wastes from reprocessing spent nuclear fuel have traditionally been called "high-level wastes." If it is not to be reprocessed, the spent fuel itself becomes a waste. The nuclear reactors operated by the nation's electrical utilities currently generate about 2,000 metric tons of spent fuel per year. The relatively small physical quantity of these wastes is apparent when compared to the chemically hazardous wastes regulated under the Resource Conservation and Recovery Act, which are produced at a rate of about 150,000,000 metric tons per year.

Although they are produced in small quantities, proper management and disposal of high-level wastes and spent nuclear fuel are essential because of the inherent hazard of the large amounts of radioactivity they contain. Spent fuel from commercial nuclear power reactors contains about 1.6 billion curies of radionuclides with half-lives greater than 20 years. Over the next decade, this inventory is projected to grow at a rate of about 300 million curies per year from reactors currently licensed to operate. Most of this spent fuel is currently stored at reactor sites. Reprocessing reactor fuel used for national defense activities has produced about 700 million curies of radionuclides with half-lives greater than 20 years. Most of these wastes are stored in various liquid and solid forms on three Federal reservations in Idaho, Washington, and South Carolina.

In addition, a wide variety of wastes contaminated with man-made radionuclides heavier than uranium have been created by various processes, mostly from the atomic energy defense activities conducted by the DOE and its predecessor agencies (the Atomic Energy Commission and the Energy

Research and Development Administration). These wastes are usually called "transuranic" wastes. Most of them are stored at Federal reservations in Idaho, Washington, New Mexico, and South Carolina.

National Programs for Disposal of These Wastes

Since the inception of the nuclear age in the 1940's, the Federal government has assumed ultimate responsibility for the care and disposal of these wastes regardless of whether they are produced by commercial or national defense activities. In October 1976, President Ford ordered a major expansion of the Federal program to demonstrate a permanent disposal method for high-level wastes. The Agency was directed to develop generally applicable environmental standards to govern the management and disposal of these wastes as part of this initiative. Among EPA's first activities in response to this directive were a series of public workshops conducted in 1977 and 1978 to better understand the various public concerns and technical issues associated with radioactive waste disposal.

In 1981, the DOE, after completing a comprehensive programmatic environmental impact statement, decided to focus the national program on disposal in mined geologic repositories (46 FR 26677). In 1982, Congress passed the Nuclear Waste Policy Act (henceforth designated "NWPA"), which President Reagan signed into law on January 7, 1983. The NWPA contains several provisions that are relevant to this rulemaking. First, it affirmed the DOE's 1981 decision that mined repositories should receive primary emphasis in the national program, although research on some other technologies would be continued. Second, it established formal procedures regarding the evaluation and selection of sites for geologic repositories, including steps for the interaction of affected States and Indian tribes with the Federal Government regarding site selection decisions. Third, the NWPA levied a fee on utilities that generate electrical power with nuclear reactors in order to pay for Federal management and disposal of their spent fuel or high-level wastes. Fourth, the NWPA reiterated the existing responsibilities of the Federal agencies involved in the national program to develop mined geologic repositories, and it assigned some additional tasks regarding site evaluation. Finally, the Act provided a timetable for several key milestones that the Federal agencies were to meet in carrying out the program.

Section 121 of the NWPA reiterated the Agency's responsibility for developing the overall framework of requirements needed to assure protection of public health and the environment, in accordance with the Agency's authorities under the Atomic Energy Act of 1954 and Reorganization Plan Number 3 of 1970. Section 121 also called for the Agency to promulgate these standards by January 7, 1984. The Agency did not meet this deadline. On February 8, 1985, the Natural Resources Defense Council and four other environmental interest groups filed suit to bring about compliance with the NWPA mandate. This litigation was settled by the Agency and the plaintiffs agreeing to a consent order requiring promulgation not later than August 15, 1985. The generally applicable environmental standards promulgated by this notice satisfy the terms of this consent order. However, they also represent the culmination of an effort that began almost nine years ago and that has included frequent interactions with the public to help formulate standards responsive to the concerns about disposal of these dangerous materials.

Objective and Implementation of the Standards

In developing the standards for disposal of spent nuclear fuel and high-level and transuranic radioactive wastes, the Agency has carefully evaluated the capabilities of mined geologic repositories to isolate the wastes from the environment. Because such repositories are capable of performing so well, it has been possible to choose containment requirements that will provide exceptionally good protection to current and future populations for at least 10,000 years after disposal. In fact, EPA's analyses indicate that the small residual risks allowed by the disposal standards would be comparable to the risks that future populations would have been exposed to if the uranium ore used to produce the high-level wastes had not been mined to begin with.¹ The Agency

¹ Specifically, the Agency estimates that compliance with the disposal standards would allow no more than 1,000 premature deaths from cancer in the first 10,000 years after disposal of the high-level wastes from 100,000 metric tons of reactor fuel, an average of no more than one premature death every ten years. As this residual risk level is referred to in the following discussion, it should be remembered that it is a speculative calculation that is primarily intended as a tool for comparing risk levels; it should not be considered a reliable projection of the "real" number of health effects resulting from compliance with the disposal standards.

believes that achieving this protection should not significantly increase the cost or difficulty of carrying out the national program for disposing of the wastes from commercial nuclear power plants. In addition, the containment requirements in the final rule are complemented by six qualitative assurance requirements designed to provide confidence that the containment requirements will be met, given the substantial uncertainties inherent in predictions of systems performance over 10,000 years. Because of this comprehensive framework, the Agency is confident that the national program to dispose of these wastes will be carried out with exceptional protection of public health and the environment.

The Nuclear Regulatory Commission (NRC) and the DOE are responsible for implementing these standards. The NRC has already promulgated procedural and technical requirements in 10 CFR Part 60 for disposal of high-level wastes in mined geologic repositories (46 FR 13971, 48 FR 28194). The NRC will obtain compliance with 40 CFR Part 191 for disposal of all high-level wastes by issuing licenses to the DOE, in accordance with 10 CFR Part 60, at various steps in the construction and operation of a repository. The NWPA directs the DOE to select a number of potential sites for geologic repositories, successively reducing this set of alternatives from five to three to one, in consultation with affected States and Indian Tribes and with participation by the public in key steps in the selection process. The DOE will accomplish this through use of site selection guidelines (10 CFR Part 960) that it has developed in accordance with section 112 of the NWPA. Both NRC's 10 CFR Part 60 and DOE's 10 CFR Part 960 incorporate the standards the Agency is promulgating today as the overall performance requirements for a geologic repository. Both of these other rules were designed in concert with EPA's ongoing development of 40 CFR Part 191. However, both the NRC and DOE must now review these regulations to determine what specific changes will be needed to properly implement the final version of 40 CFR Part 191.

Review of the Proposed Standards

On December 29, 1982, shortly before the NWPA was enacted, the Agency published 40 CFR Part 191 for public review (47 FR 58196) and asked that comments be received by May 2, 1983. Eighty-three substantive replies were received from a broad spectrum of private citizens, public interest groups, members of the scientific community,

representatives of industry, and State and Federal agencies. These responses contained information and recommendations regarding seven issues on which the Agency sought further public comment (48 FR 21666). Questions concerning these issues were directed to all of the witnesses at two public hearings held during May 1983 in Washington, D.C. and in Denver (48 FR 13444). Copies of these questions were also sent to all those who responded to the initial request for comment, and the availability of these questions was announced in the *Federal Register* (48 FR 21666). The comment period was then held open until June 20, 1983, to receive responses to these additional questions. Responses to major comments—including all those specifically highlighted for public review—are summarized below. Detailed responses to the full range of comments received is described in the Response to Comments Document prepared for the final rule.

Review of the Technical Basis of the Standards

In parallel with this public review and comment, the Agency conducted an independent scientific review of the technical basis for the proposed 40 CFR Part 191 through a special Subcommittee of the Agency's Science Advisory Board (SAB) (48 FR 509). This Subcommittee held nine public meetings from January 18, 1983, through September 21, 1983, and prepared a final report that was transmitted on February 17, 1984. While finding that the Agency had generally prepared comprehensive and scientifically competent technical analyses to support the proposed standards, the SAB review developed 46 findings and recommendations regarding specific improvements in the technical analyses and in the standards themselves. Since many of the SAB recommendations were to be considered in developing the final rule, the Agency sought public comment on the information and recommendations presented in the final SAB report (49 FR 19604).

Most of the SAB recommendations involve specific details of the technical assessments and judgments the Agency made in developing these standards. After evaluating the public comments received on the SAB report, the Agency agrees with almost all of the SAB's technical recommendations and has made corresponding changes in the technical basis of the final rule. A few of the Subcommittee's recommendations have implications that involve broader policy judgments. These recommendations have been treated as

part of the public comment record and are described below as the major comments on the proposed 40 CFR Part 191 are discussed. A complete itemization of the Agency's responses to each of the findings and recommendations of the SAB is contained in the Response to Comments Document, together with a synopsis of the public comments on the SAB report.

Summary of the Final Rule

The rule being promulgated today establishes generally applicable environmental standards for the management and disposal of spent nuclear fuel, high-level radioactive wastes, and transuranic radioactive wastes. The final rule differs in a number of ways from the proposed rule because of changes the Agency has made in response to public comments and in response to the recommendations of the technical review by the Agency's Science Advisory Board. This section provides an overview of the major provisions of the final rule, and changes from the proposed rule are noted. More detail on many of these provisions is provided later as part of the discussion of the comments considered in development of 40 CFR Part 191. The final rule:

(1) Applies to management and disposal of spent nuclear fuel, high-level radioactive wastes as defined by the NWSA, and transuranic wastes containing more than 100 nanocuries per gram of alpha-emitting transuranic isotopes, except for wastes that either the NRC or the Administrator determines do not need the degree of isolation required by this rule. (The proposed rule applied to spent nuclear fuel, high-level wastes exceeding a specific set of concentration limits, and to transuranic wastes containing more than 100 nanocuries per gram.)

(2) Through Subpart A, "Standards for Management and Storage," establishes limits on annual doses to members of the public of 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ from exposures associated with management, storage, and preparation for disposal of any of these materials at facilities regulated by the NRC. These limits apply to the combined exposures from all NRC-licensed facilities covered by this Part or 40 CFR Part 190, the Agency's standards for the commercial uranium fuel cycle. Subpart A also limits annual doses to members of the public from management and storage operations at DOE disposal facilities that are not regulated by the NRC to 25 millirems to the whole body and 75 millirems to any other organ. (The

proposed rule applied to the combined exposures from operations regulated by 40 CFR Part 190, waste management and storage operations regulated by the NRC or Agreement States, and waste management and storage operations conducted at all DOE facilities.) Subpart A also contains a provision that allows the Administrator to issue alternative standards for waste management and storage operations at DOE disposal facilities that are not regulated by the NRC. (The proposed rule contained a provision to allow the implementing agency, either the NRC or the DOE, to grant variances for unusual operating conditions.)

(3) Establishes several sets of requirements for disposal of these wastes through Subpart B, "Standards for Disposal." The primary standards are *containment requirements* that limit projected releases of radioactivity to the accessible environment for 10,000 years after disposal. Equally important is a set of six *assurance requirements* chosen to provide adequate confidence that the containment requirements will be met. In addition, Subpart B of the final rule includes *individual protection requirements* that limit annual exposures from the disposal facility to members of the public in the accessible environment to 25 millirems to the whole body and 75 millirems to any organ for 1,000 years after disposal. The Subpart also contains *ground water protection requirements* that limit radioactivity concentrations in water withdrawn from most Class I ground waters near a disposal system (as defined in conjunction with the Agency's Ground Water Protection Strategy published in August 1984) for 1,000 years after disposal. Finally, Subpart B provides *guidance for implementation* that indicates how the Agency intends the various numerical standards to be applied. (The proposed rule contained only containment requirements, assurance requirements and procedural requirements; this last category provided some of the basis for the "guidance for implementation" in the final rule.) Major provisions of each of these sets of requirements include the following:

(a) The containment requirements (Section 191.13) limit the total projected release of specific radionuclides over the entire 10,000-year period after disposal. Releases from all expected and accidental causes are included, except for releases from conceivable events that are judged to have an incredibly small likelihood of occurrence. Quantitative terms are used to identify the probabilities of the releases to which

the containment requirements apply; however, the final rule acknowledges that determination of compliance will have to tolerate much larger uncertainties than would be appropriate for short-term estimates and that judgments may have to be substituted for quantitative predictions in certain situations. Disposal in compliance with the containment requirements is projected to cause no more than 1,000 premature cancer deaths over the entire 10,000-year period from disposal of all existing high-level wastes and most of the wastes yet to be produced by currently operating reactors—an average of 0.1 fatality per year. This level of residual risk to future generations would be comparable to the risks that those generations would have faced from the uranium ore used to create the wastes if the ore had never been mined. Actual risks will probably be significantly less because of the conservative approach called for by the other parts of Subpart B. (The quantitative probabilities in the proposed rule were an order of magnitude smaller than those incorporated into the final rule. The release limits in the final rule are different than those in the proposed rule due to changes in EPA's technical analyses that were recommended by the SAB Subcommittee; however, the level of residual risk is the same as for the proposed rule.)

(b) The assurance requirements (Section 191.14) call for cautious steps to be taken in disposing of these wastes because of the inherent uncertainties in selecting and designing disposal systems that must be very effective for more than 10,000 years. The assurance requirements incorporate the following principles:

(i) Although active institutional controls, such as guarding and maintaining a disposal site, should be encouraged, they cannot be relied upon to isolate these wastes from the environment for more than 100 years after disposal. (The proposed rule limited reliance to "a few hundred years" after disposal.)

(ii) Disposal systems must be monitored to detect substantial changes from their expected performance until the implementing agency determines that there are no significant concerns to be addressed by further monitoring. (This requirement was not included in the proposed rule.)

(iii) The sites where disposal systems are located must be identified by permanent markers, widespread records, and other passive institutional controls to warn future generations of the dangers and location of the wastes.

(iv) Disposal systems must use several different types of barriers, including both engineered and natural ones, to isolate the wastes from the environment to help guard against unexpectedly poor performance from one type of barrier.

(v) Sites for disposal systems should be selected to avoid places where resources have previously been mined, where there is a reasonable expectation of exploration for scarce or easily accessible resources, or where there is a significant concentration of any material which is not otherwise available. (The wording in the proposed rule would have ruled out sites with a significant possibility of being considered for resource exploration in the future. The final rule revises this requirement to allow use of sites with some resource potential if they have other significant advantages compared to potential alternative sites.)

(vi) Recovery of most of the wastes must not be precluded for a reasonable period after disposal if unforeseen events require this in the future.

(c) The individual protection requirements (Section 191.15) limit annual exposures to members of the public in the accessible environment from the disposal system to 25 millirems to the whole body and 75 millirems to any organ. These requirements apply to undisturbed performance of the disposal system for 1,000 years after disposal. All potential pathways of radiation exposure from the disposal system to people must be considered, including the assumption that individuals consume all of their drinking water (2 liters per day) from any "significant source of ground water" located outside the "controlled area" established around a disposal system. A "significant source" is identified by several parameters intended to describe an aquifer sufficient to meet the needs of a "community water system" as defined in the Agency's National Interim Primary Drinking Water Regulations (40 CFR Part 141). (No explicit individual protection requirements were included in the proposed rule.)

(d) The ground water protection requirements (Section 191.16) limit the concentrations of radioactivity (or the increases in concentrations, if preexisting concentrations already exceed these limits) in waters withdrawn from most Class I sources of ground water near a disposal system to no more than 15 picocuries per liter of alpha-emitting radionuclides (including no more than 5 picocuries per liter of radium-226 and radium-228 but excluding radon) and to no more than the combined concentrations of radionuclides that emit either beta or

gamma radiation that would produce an annual dose equivalent to the total body or any internal organ greater than 4 millirems if individuals consumed all of their drinking water from that source of ground water. These concentration limits are similar to those set in 40 CFR Part 141 for community water systems. Like the individual protection requirements, the ground water protection requirements apply to undisturbed behavior of the disposal system for a period of 1,000 years after disposal. (No explicit ground water protection requirements were included in the proposed rule.)

(e) Section 191.17 of the final rule establishes minimum procedural requirements that the Administrator must follow if additional information considered in the future indicates that it would be appropriate to modify any portion of the disposal standards through further rulemaking. (No similar provision was included in the proposed rule.)

(f) The "guidance for implementation" included as Appendix B to the final rule describes certain analytical approaches and assumptions through which the Agency intends the various long-term numerical standards of Subpart B to be applied. This guidance is particularly important because there are no precedents for the implementation of such long-term environmental standards, which will require consideration of extensive analytical projections of disposal system performance. (The proposed rule contained a corresponding, but less extensive, section entitled "procedural requirements.")

Overall Approach of the Final Rule

In general, the Agency developed the various elements of this rule by balancing several perspectives. One set of considerations was the expected capabilities of the waste management and disposal technologies to reduce both short- and long-term risks to public health and the environment. These capabilities were examined through a number of performance assessments of the waste management, storage, and disposal facilities planned for the wastes generated by commercial nuclear power plants. Since detailed plans have not yet been determined for disposition of the wastes generated by atomic energy defense activities, similar assessments were generally not performed for these materials. A second consideration, where applicable, was consistency with related environmental standards for radiation exposure. A third factor was evaluation of various

benchmarks to assess the acceptability of the residual risks that might be allowed by the rule. This was particularly important for the disposal standards, where there were few precedents to guide the Agency's judgments. Finally, the Agency placed considerable emphasis on the public concerns expressed during the various phases of this rulemaking, particularly where these concerns involved addressing the substantial uncertainties inherent in the unprecedented time periods of interest.

The final rule reflects a combination of all these perspectives—no single factor predominated. For instance, no portion of this rule is based solely on projections of the "best" protection that technology might provide. If this had been the case, the rule would have been significantly different. On the other hand, the rule cannot be interpreted as setting precedents for "acceptable risk" levels to future generations that should not be exceeded regardless of the circumstances. Instead, because of a number of unique circumstances, the Agency has been able to develop standards for the management and disposal of these wastes that are both reasonably achievable—with little, if any, effort beyond that already planned for commercial wastes—and that limit risks to levels that the Agency believes are clearly acceptably small. The following paragraphs describe how these various perspectives were used in developing the final rule.

Standards for Management and Storage (Subpart A)

Upon surveying the expected performance of the technologies planned for the management, storage, and preparation of these wastes for disposal, the Agency found that the likely exposures to members of the public would generally be very small. Therefore, compatibility with related radiation protection standards became a more important perspective for Subpart A.

For waste management and storage operations to be regulated by the NRC, the most relevant existing standards are those provisions of 40 CFR Part 190 that limit annual exposures of members of the public to 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ from uranium fuel cycle facilities. Accordingly, the Agency has decided to extend this coverage to include such waste management and storage operations so that the combined exposure from all of the NRC-licensed facilities covered under Part 190 and Subpart A of Part 191 shall not exceed

these limits. This will include all operations prior to final closure at high-level waste disposal facilities, since these are to be regulated by the NRC.

For waste management and storage operations conducted at atomic energy defense facilities operated for the Department of Energy (which are not regulated by the NRC), the most relevant existing standards are the 40 CFR Part 61 limitations on air emissions of radionuclides that were recently promulgated under the Agency's Clean Air Act authorities (50 FR 5190). These standards limit annual exposures to members of the public to 25 millirems to the whole body and 75 millirems to any organ, with less stringent alternative standards available if it can be shown that no member of the public will receive a continuous exposure of more than 100 millirems per year or an infrequent exposure of more than 500 millirems per year from all sources (excluding natural background and medical exposures.) These Clean Air Act standards are applicable to those facilities not covered by 40 CFR Parts 190, 191 or 192. For DOE waste disposal facilities covered by this rule but not regulated by NRC (i.e., those for defense transuranic wastes), the Agency has included standards in Subpart A similar to those included in the Clean Air Act rule.

For other DOE waste management and storage operations, which are usually conducted on large facilities with many other potential sources of radionuclide emissions, the Agency believes that continued regulation under the broader scope of 40 CFR Part 61 is the most effective and practical approach. Otherwise, similar types of emissions from adjoining operations would have to be assessed and regulated through separate rules developed under different authorities; this would cause complex implementation practices without providing any additional protection.

Standards for Disposal (Subpart B)

Developing the standards for disposal of spent fuel and high-level and transuranic wastes involved much more unusual circumstances than those for waste management and storage. Because these materials are dangerous for so long, very long time frames are of interest. Standards must be implemented in the design phase for these disposal systems because active surveillance cannot be relied upon over such periods. At the same time, the standards must accommodate large uncertainties, including uncertainties in our current knowledge about disposal system behavior and the inherent

uncertainties regarding the distant future. Subpart B addresses these issues by combining several different types of standards. The primary objective of these standards is to isolate most of the wastes from man's environment by limiting long-term releases and the associated risks to populations. In addition, Subpart B limits risks to individuals in ways compatible with this primary objective.

Although developed primarily through consideration of mined geologic repositories, these disposal standards apply to disposal of spent fuel and high-level and transuranic radioactive wastes by any method—with one exception. The standards do not apply to ocean disposal or disposal in ocean sediments because such disposal of high-level waste is prohibited by the Marine Protection, Research, and Sanctuaries Act of 1972. If this law is ever changed to allow such disposal (DOE continues to study the feasibility of this technology, consistent with the NWPAL), the Agency will develop appropriate regulations in accordance with the different authorities that would apply.

Also, these disposal standards do not apply to wastes that have already been disposed of. The various provisions of Subpart B are intended to be met through a combination of steps involving disposal system site selection, design, and operational techniques (i.e., engineered barriers). Therefore, the Agency believes it appropriate that these disposal standards only apply to disposal occurring after the standards have been promulgated—so that they can be taken into consideration in devising the proper selection of controls. Some transuranic wastes produced in support of national defense programs were disposed of before the current DOE procedures for transuranic waste management were adopted in 1970. The exclusion of wastes already disposed of applies to these transuranic wastes, for which selection of disposal system sites, designs, and operational techniques are no longer options.

Containment Requirements (Section 191.13)

To develop the containment requirements, the Agency assumed that some aspects of the future can be predicted well enough to guide the selection and development of disposal systems for these wastes. A period of 10,000 years was considered because that appears to be long enough to distinguish geologic repositories with relatively good capabilities to isolate wastes from those with relatively poor capabilities. On the other hand, this

period is short enough so that major geologic changes are unlikely and repository performance might be reasonably projected.

The Agency assessed the performance of a number of model geologic repositories similar to those systems now being considered by DOE. Potential radionuclide releases over 10,000 years were evaluated, and very general models of environmental transport and a linear, non-threshold dose-effect relationship were used to relate these releases to the incidence of premature cancer deaths they might cause. For the various repository types, these assessments indicate that disposal of the wastes from 100,000 metric tons of reactor fuel would cause a population risk ranging from no more than about ten to a little more than one hundred premature deaths over the entire 10,000-year period, assuming that the existing provisions of 10 CFR Part 60 regarding engineered barriers are met.

The Agency also evaluated the health risks that future generations would be exposed to from the amount of uranium ore needed to produce 100,000 metric tons of reactor fuel, if this ore had not been mined to begin with. Population risks ranging between 10 and 100,000 premature cancer deaths over 10,000 years were associated with this much unmined uranium ore, depending upon the analytical assumptions made.

These analyses, which have been updated from those prepared for the proposed standards, reinforce the Agency's conclusion that limiting radionuclide releases to levels associated with no more than 1,000 premature cancer deaths over 10,000 years from disposal of the wastes from 100,000 metric tons of reactor fuel satisfies two important objectives. First, it provides a level of protection that appears reasonably achievable by the various options being considered within the national program for commercial wastes. Second, the Agency believes that such a limitation would clearly keep risks to future populations at acceptably small levels, particularly because it appears to limit risks to no more than the midpoint of the range of estimated risks that future generations would have been exposed to if the uranium ore used to create the wastes had never been mined. Thus, because mined geologic repositories appear capable of providing such good protection, the Agency has decided to establish containment requirements that meet these two objectives.

The specific release limits for different radionuclides in Table 1 of the final rule were developed by estimating how many curies of each radionuclide would

cause 1,000 premature deaths over 10,000 years if released to the environment. The limits were then stated in terms of the allowable release from 1,000 metric tons of reactor fuel (so that the actual curie values in Table 1 correspond to a risk level of 10 premature deaths over 10,000 years). All of these limits have been rounded to the nearest order of magnitude because of the approximate nature of these calculations. For particular disposal systems, release limits based upon the amount of waste in the system will be developed and will be used in a formula that insures that the desired risk level will not be exceeded if releases of more than one type of radionuclide are predicted. For some of the wastes covered by this rule, 1,000 metric tons of reactor fuel is not an appropriate unit of waste. In these situations, the various Notes to Table 1 provide instructions on how to calculate the proper release limits. In particular, the final rule includes provisions for high-level wastes from reactor fuels that have received substantially different uses in national defense applications (and contain much different amounts of radioactivity) than is typical of most reactor fuel used to generate electricity. The proposed rule would have allowed releases for these different types of fuels to occur in much different proportions to their total radioactivity than the Agency intended.

The release limits apply to radionuclides that are projected to move into the "accessible environment" during the first 10,000 years after disposal. The accessible environment includes all of the atmosphere, land surface, surface waters, and oceans. However, it does not include the lithosphere (and the ground water within it) that is below the "controlled area" surrounding a disposal system. The standards are formulated this way because the properties of the geologic media around a mined repository are expected to provide much of the disposal system's capability to isolate these wastes over these long time periods. Thus, a certain area of the natural environment is envisioned to be dedicated to keeping these dangerous materials away from future generations and may not be suitable for certain other uses. In the final rule, this "controlled area" is not to exceed 100 square kilometers and is not to extend more than five kilometers in any direction from the original emplacement of the wastes in the disposal system. The implementing agencies may choose a smaller area whenever appropriate.

The containment requirements apply to accidental disruptions of a disposal system as well as to any expected

releases. Accordingly, they are stated in terms of the probability of releases occurring. This is done in two steps.

First, the release limits calculated in accordance with Notes 1 through 5 to Table 1 apply to those release levels that are projected to occur with a cumulative probability greater than 0.1 for the entire 10,000-year period over which these disposal standards apply. This includes the total releases from those processes that are expected to occur as well as relatively likely disruptions (which the Agency assumes will primarily include predictions of inadvertent human intrusion).

Second, these release limits multiplied by ten apply to all of the releases projected to occur with a cumulative probability greater than 0.001 over the 10,000-year period. The Agency expects that this will include releases that might occur from the more likely natural disruptive events, such as fault movement and breccia pipe formation (near soluble media such as salt formations). This range of probabilities was selected to include the anticipated uncertainties in predicting the likelihood of these natural phenomena. Greater releases are allowed for these circumstances because they are so unlikely to occur.

Finally, the containment requirements place no limits on releases projected to occur with a cumulative probability of less than 0.001 over 10,000 years. Probabilities this small would tend to be limited to phenomena such as the appearance of new volcanos outside of known areas of volcanic activity, and the Agency believes there is no benefit to public health or the environment from trying to regulate the consequences of such very unlikely events.

The containment requirements call for a "reasonable expectation" that their various quantitative tests be met. This phrase reflects the fact that unequivocal numerical proof of compliance is neither necessary nor likely to be obtained. A similar qualitative test, that of "reasonable assurance," has been used with NRC regulations for many years. Although the Agency's intent is similar, the NRC phrase has not been used in 40 CFR Part 191 because "reasonable assurance" has come to be associated with a level of confidence that may not be appropriate for the very long-term analytical projections that are called for by 191.13. The use of a different test of judgment is meant to acknowledge the unique considerations likely to be encountered upon implementation of these disposal standards.

Assurance Requirements (Section 191.14)

In contrast to the containment requirements, the assurance requirements were developed from that point of view that there may be major uncertainties and gaps in our knowledge of the expected behavior of disposal systems over many thousands of years. Therefore, no matter how promising the analytical projections of disposal system performance appear to be, these materials should be disposed in a cautious manner that reduces the likelihood of unanticipated types of releases. Because of the inherent uncertainties associated with these long time periods, the Agency believes that the principles embodied in the assurance requirements are important complements to the containment requirements that should insure that the level of protection desired is likely to be achieved.

Each of the assurance requirements was chosen to reduce the potential harm from some aspect of our uncertainty about the future. Designing disposal systems with limited reliance on active institutional controls reduces the risks if future generations do not maintain surveillance of disposal sites. On the other hand, planning for long-term monitoring helps reduce the chances that unexpectedly poor performance of a disposal system would go unnoticed. Using extensive markers and records and avoiding resources when selecting disposal sites both serve to reduce the chances that people may inadvertently disrupt a disposal system because of incomplete understanding of its location, design, or hazards. Designing disposal systems to include multiple types of barriers, both engineered and natural, reduces the risks if one type of barrier performs more poorly than current knowledge indicates. Finally, designing disposal systems so that it is feasible for the wastes to be located and recovered gives future generations an opportunity to rectify the situation if new discoveries indicate compelling reasons (which would not be foreseeable now) to change the way these wastes are disposed of.

The proposed standards contained two other assurance requirements intended to reduce the risks of uncertainty. One of them called for these wastes to be disposed of promptly to reduce the uncertainties associated with storing these materials for indefinitely long times with methods that require active human involvement. However—after this rule was published for public comment—the NWPAA was enacted, setting up mandates and

procedures intended to insure development of the necessary disposal systems for spent fuel and high-level wastes. Furthermore, the Department has made substantial progress towards developing a repository for disposal of the transuranic wastes from atomic energy defense activities. Because of these steps, the Agency decided that the call for prompt disposal was no longer needed, and this assurance requirement has not been included in the final rule.

The other proposed assurance requirement deleted from the final rule is the provision that called for releases to be kept as small as reasonably achievable even when the numerical containment requirements have been complied with. This would have increased the confidence of achieving the desired level of protection even if there were major uncertainties in analytical projections of long-term isolation. However, the Agency does not believe that it is necessary to retain this assurance requirement in the final standards because of two aspects of the related rules subsequently promulgated by the NRC and DOE for disposal of spent fuel and high-level wastes.

First, NRC's 10 CFR Part 60 implemented the multiple barrier principle by requiring very good performance from two types of engineered components: A 300 to 1,000-year lifetime for waste packages during which there would be essentially no expected release of waste, and a subsequent long-term release rate from the waste form of no more than one part in 100,000 per year. The Agency fully endorses this approach and believes that it represents the best performance reasonably achievable for currently foreseeable engineered components. Second, the DOE has included a provision in its site selection guidelines (10 CFR Part 960) that calls for significant emphasis to be placed on selecting sites that demonstrate the lowest releases over 100,000 years compared to the other alternatives available. Particularly because of the longer time frame involved in this comparison, the Agency believes that this provides adequate encouragement to choose sites that provide the best isolation capabilities available. Therefore, the concept of keeping long-term releases as small as reasonably achievable has been embodied by other agencies' regulations for both the engineered and natural components of disposal systems.

The final rule incorporates the five remaining assurance requirements plus the requirement for long-term monitoring, but it makes them

applicable only to disposal facilities that are not regulated by the NRC. In its comments on the proposed rule, the NRC objected to inclusion of the assurance requirements, asserting that they were not properly part of the Agency's authorities assigned by Reorganization Plan No. 3 of 1970. The Agency continues to believe that provisions such as the assurance requirements are an appropriate part of generally applicable standards where they are necessary to establish the regulatory context for numerical standards—as they are in these circumstances because of the major uncertainties involved. However, the two agencies have agreed to resolve this issue by having the Commission modify 10 CFR Part 60 where necessary to incorporate the intent of the assurance requirements, rather than have them included in 40 CFR Part 191 for NRC-licensed disposal facilities. Thus, 10 CFR Part 60 will establish the context needed for appropriate implementation of 40 CFR Part 191.

The NRC staff is preparing the appropriate revisions to Part 60 and has told the Agency that they will be published in the *Federal Register* for public review and comment within approximately 120 days of today's promulgation of 40 CFR Part 191. EPA has provided NRC with all of the comments received on the assurance requirements during the 40 CFR Part 191 rulemaking, and the Agency will participate in the NRC rulemaking to facilitate our objective of having the intent of all of the assurance requirements embodied in Federal regulation. Finally, the Agency will review the record and outcome of the Part 60 rulemaking to determine if any subsequent modifications to 40 CFR Part 191 are needed.

Individual and Ground Water Protection Requirements (Sections 191.15 and 191.16)

While the primary objective of both the proposed and final disposal standards has been to limit potential long-term releases from disposal systems (and the population risks associated with such releases), these two sections have been added to the final rule to provide protection for those individuals in the vicinity of a disposal system. There are a number of difficult issues involved in formulating standards for individual protection in this situation, as discussed later in the "Release Limits vs. Individual Dose Limits" section. However, after evaluating the various comments received on this topic, the Agency

believes that there are also important advantages in providing for individual protection in ways compatible with the containment and assurance requirements. In discussing this issue, the SAB Subcommittee stated that: "We support the use of a population risk criteria. We believe it is impractical to provide absolute protection to every individual for all postulated events or for very long periods. On the other hand, in our view it is important that, for the first several hundred years, residents of the region immediately outside the accessible environment have very great assurance that they will suffer no, or negligible, ill effects from the repository."

The individual protection requirements in the final rule limit the annual exposure from the disposal system to a member of the public in the accessible environment, for the first 1,000 years after disposal, to no more than 25 millirems to the whole body or 75 millirems to any organs. These limitations apply to the predicted behavior of the disposal system, including consideration of the uncertainties in predicted behavior, assuming that the disposal system is not disrupted by human intrusion or the occurrence of unlikely natural events. The Agency chose the limits of 25 millirem/year to the whole body and 75 millirem/year to any organ because it believes that they represent a sufficiently stringent level of protection for situations where no more than a few individuals are likely to receive this exposure. If such an individual were exposed to this level over a lifetime (which seems particularly unlikely given the localized pathways through which waste might escape from a geologic repository), the Agency estimates this would cause a 5×10^{-4} chance of incurring a premature fatal cancer.

In choosing a time period for these requirements to protect individuals nearby disposal systems, the Agency took into account concerns such as those expressed by the SAB by examining the effects of choosing different time frames. As 10,000 years was chosen for the containment requirements because it is long enough to encourage use of disposal sites with natural characteristics that enhance long-term isolation, 1,000 years was chosen for the individual protection provisions because the Agency's assessments indicate it is long enough to insure that particularly good engineered barriers would need to be used at potential sites where some ground water would be expected to flow through a mined geologic repository. Use of a time

much shorter than 1,000 years would not call for substantial engineered barriers even at disposal sites with a lot of ground water flow.

On the other hand, demonstrating compliance with individual exposure limits for times much longer than 1,000 years appears to be quite difficult because of the analytical uncertainties involved. It would require predicting radionuclide concentrations—even from releases of tiny portions of the waste—in all the possible ground water pathways flowing in all directions from the disposal system, at all depths down to 2,500 feet, as a function of time over many thousands of years. At some of the sites being considered (and possibly all of them, depending upon what is discovered during site characterization) the only certain way to comply with such requirements for periods on the order of 10,000 years appears to be to use very expensive engineered barriers that would rule out any potential releases over most of this period. While such barriers could provide longer-term protection for individuals, they would not provide substantial benefits to populations because the containment and assurance requirements already reduce population risks to very small levels.

Based on all of these considerations, the Agency has decided that a 1,000-year duration is adequate for quantitative limits on individual exposures after disposal. For longer time periods, several of the qualitative assurance requirements should help to reduce the chances that individuals will receive serious radiation exposures. In addition, 40 CFR Part 191 in no way limits the future applicability of the Agency's drinking water standards (40 CFR Part 141)—which protect community water supply systems through institutional controls—or of similar standards that future generations may choose to adopt.

In assessing the performance of a disposal system with regard to individual exposures, all pathways of radioactive material or radiation from the disposal system to people shall be considered. In particular, the assessments must assume that individuals consume all of their drinking water (2 liters per day) from any portion of a "significant source of ground water" anywhere outside of the "controlled area" surrounding the disposal system. Significant sources of ground water are defined to include underground formations that are likely to be able to provide enough water for a community water system as defined in 40 CFR Part 141. (More information regarding this

definition is provided later in the "Release Limits vs. Individual Dose Limits" discussion.) Formations that could only provide smaller amounts of potable water have not been included because the Agency wants to avoid discriminating against the use of low-productivity geologic formations that might provide very good long-term isolation as disposal sites. The Agency believes this is reasonable for these standards because of the very small number of such disposal facilities that are contemplated (no more than three or four over the next 100 years.) However, the Agency has no plans to use this classification for other ground water related standards, which usually affect a far greater number of situations.

The Agency has not required these individual protection provisions to assume ground water use within the controlled area because geologic media within the controlled area are an integral part of the disposal system's capability to provide long-term isolation. (But if the implementing agency plans to allow individuals to use ground water within the controlled area, such planned use would have to be considered within the pathways evaluated to determine compliance with § 191.15.) The potential loss of ground water resources is very small because of the small number of such disposal facilities contemplated. Nevertheless, the Agency has also added ground water protection requirements to the final rule (Section 191.16) that protect certain sources of ground water even within the controlled area. These ground water protection requirements are similar to the individual protection requirements because they apply to undisturbed performance for 1,000 years after disposal. However, the ground water protection requirements apply only to those Class I ground waters, as they are identified in accordance with the Agency's Ground-Water Protection Strategy published in August 1984, that meet the following three conditions: (1) They are within the controlled area or near (less than five kilometers beyond) the controlled area; (2) they are supplying drinking water for thousands of persons as of the date that the Department selects the site for extensive exploration as a potential location of a disposal system; and (3) they are irreplaceable in that no reasonable alternative source of drinking water is available to that population.

For such Class I ground waters, § 191.16 limits the radionuclide concentrations in water withdrawn from any portion of them to no more than concentration limits similar to those

established for the output of community water systems in 40 CFR Part 141. However, if the preexisting concentrations of radioactivity in the Class I aquifer already exceed any of these limits at a particular site, § 191.16 then limits any increases in the preexisting concentrations to these same concentration limits. The Agency believes these provisions are necessary and adequate to avoid any significant degradation of the important drinking water resources provided by these Class I ground waters.

Alternative Provisions for Disposal (Section 191.17)

In developing the disposal standards, the Agency has had to make many assumptions about the characteristics of disposal systems that have not been built, about plans for disposal that are only now being formulated, and about the probable adequacy of technical information that will not be collected for many years. Thus, although the Agency believes that the disposal standards being promulgated today are appropriate based upon current knowledge, we cannot rule out the possibility that future information may indicate needs to modify the standards.

In recognition of this possibility, § 191.17 of the final rule sets forth procedures under which the Administrator may develop modifications to Subpart B, should the need arise. Any such changes would have to proceed through the usual notice-and-comment rulemaking process, and § 191.17 stipulates that such a rulemaking would require a public comment period of at least 90 days, to include public hearings in affected areas of the country. Although such procedures are common practice in rulemakings of this type, they are not required by the statutes relevant to this rule (Administrative Procedures Act mandates can be satisfied by a comment period as short as 14 days). Thus, § 191.17 insures an opportunity for significant public interaction regarding any proposed changes to the disposal standards.

There are several areas of uncertainty the Agency is aware of that might cause suggested modifications of the standards in the future. One of these concerns implementation of the containment requirements for mined geologic repositories. This will require collection of a great deal of data during site characterization, resolution of the inevitable uncertainties in such information, and adaptation of this information into probabilistic risk assessments. Although the Agency is currently confident that this will be

successfully accomplished, such projections over thousands of years to determine compliance with an environmental regulation are unprecedented. If—after substantial experience with these analyses is acquired—disposal systems that clearly provide good isolation cannot reasonably be shown to comply with the containment requirements, the Agency would consider whether modifications to Subpart B were appropriate.

Another situation that might lead to suggested revisions would be if additional information were developed regarding the disposal of certain wastes that appeared to make it inappropriate to retain generally applicable standards addressing all of the wastes covered by this rule. For example, the DOE is considering disposal of some defense wastes by stabilizing them in their current storage tanks, rather than relocating them to a mined repository. The Agency has not assessed the ramifications of such disposal yet, and it is certainly possible that it could be carried out in compliance with all the provisions of Subpart B being promulgated today. However, it is also possible that there may be benefits associated with such disposal that would warrant changes in Subpart B for these types of waste. If so, § 191.17 would govern the consideration of any such revisions.

Other examples of developments that might offer reasons to consider alternative provisions in the future include: The use of reactor fuel cycles or utilizations substantially different than today's; new models of the environmental transport and biological effects of radionuclides that indicate major changes (i.e., approaching an order of magnitude) in the relative risks associated with different radionuclides and the level of protection sought by the disposal standards; or information that indicates that particular assurance requirements might not be needed in certain situations to insure adequate confidence of long-term environmental protection.

Guidance for Implementation (Appendix B)

This supplement to the final rule is based upon some of the analytical assumptions that the Agency made in developing the technical basis used for formulating the numerical disposal standards. These analytical assumptions incorporate information assembled as part of the technical basis used to develop the proposed rule. In particular, Appendix B discusses: (1) The consideration of all barriers of a disposal system in performance

assessments; (2) reasonable limitations on the scope of performance assessments; (3) the use of average or "mean" values in expressing the results of performance assessments; (4) the types of assumptions regarding the effectiveness of institutional controls; and (5) limiting assumptions regarding the frequency and severity of inadvertent human intrusion into geologic repositories.

The implementing agencies are responsible for selecting the specific information to be used in these and other aspects of performance assessments to determine compliance with 40 CFR Part 191. However, the Agency believes it is important that the assumptions used by the implementing agencies are compatible with those used by EPA in developing this rule. Otherwise, implementation of the disposal standards may have effects quite different than those anticipated by EPA. The final rule to be published in the Code of Federal Regulations will include this informational appendix as guidance to the implementing agencies. Although the other agencies are not bound to follow this guidance, EPA recommends that it be carefully considered in planning for the application of 40 CFR Part 191. The Agency will monitor implementation of the disposal standards as it develops over the next several years to determine whether any changes to the rule are called for to meet the Agency's objectives for these standards.

Comments on Issues Highlighted for Public Review

The Agency particularly requested public comment on six issues associated with the proposed rule (47 FR 58196). After these comments were received, additional comments and information were requested on seven issues raised by the initial comments (48 FR 21666). Two of these seven issues (the definition of high-level waste and the use of individual dose limitations in the disposal standards) had been included among the first six issues that were highlighted. Thus, a total of eleven questions received particular attention during the public review and comment process. The following paragraphs summarize the comments received on each of these issues and the Agency's responses to them, including descriptions of any resulting changes made in the final rule.

Definition of "High-Level Waste"

Traditionally, the term "high-level waste" has meant the highly radioactive liquid wastes remaining from the

recovery or uranium and plutonium in a nuclear fuel reprocessing plant, and other liquid or solid forms into which such liquid wastes are converted to facilitate managing them. This traditional use of the term has not included radioactive materials from other sources, no matter how radioactive they are. However, somewhat different definitions of high-level waste have appeared in certain laws and regulations affecting specific aspects of radioactive waste management. Most notably, some of these definitions have included unreprocessed spent fuel as the prospects for a commercial fuel reprocessing industry became more uncertain.

In the proposed rule, high-level waste was defined in the traditional sense, including spent fuel if disposed of without reprocessing. But the proposed definition also included minimum radioactivity concentrations below which such materials would not be subject to the stringent isolation requirements of 40 CFR Part 191. To identify these minimum concentrations, the maximum concentrations that the NRC determined that it would generally accept in near-surface disposal facilities under 10 CFR Part 61 (47 FR 57446) were adapted. Since this represented a modification of the traditional meaning of high-level waste, the Agency particularly sought comment on this aspect of the proposed rule.

Shortly after 40 CFR Part 191 was published for public review, the NWPA was enacted. The NWPA distinguished between spent nuclear fuel and high-level waste, and it defined high-level waste to include both: "(A) The highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and (B) other highly radioactive material that the Commission, consistent with existing law, determines by rule requires permanent isolation." This definition allow for inclusion of highly radioactive material not related to reprocessing of spent nuclear fuel, and it reflects the concept that some derivatives of nuclear fuel reprocessing may not contain sufficient radioactivity to warrant exceptional isolation.

Many of the comments regarding the proposed definition suggested that EPA adopt the definition in the NWPA, although in response to the specific questions distributed in conjunction with the Agency's public hearings, many

responders thought that the Agency should define the phrase "sufficient concentrations" contained in part A of the NWPA definition. However, several commenters argued that the proposed lower limits for high-level waste concentrations had been improperly taken out of the context of 10 CFR Part 61 and could require expensive disposal of wastes with relatively small hazards.

After considering these comments and other information currently available, the Agency decided to incorporate the NWPA definition of high-level waste in the final 40 CFR Part 191 without further elaboration of the phrase "sufficient concentrations." The Agency recognizes that this introduces some uncertainty regarding the applicability of this rule. However, the Commission is now beginning a rulemaking that should assemble the technical information needed to develop a more specific definition of high-level wastes. Since the NRC definition would not necessarily apply to all the situations covered by 40 CFR Part 191 (e.g., management and storage of defense high-level wastes prior to disposal is not regulated by NRC), the Agency will follow the Commission's rulemaking to determine what appropriate elaborations of the NWPA definition should be incorporated into 40 CFR Part 191. Upon completion of the NRC rulemaking, the Agency will initiate steps to appropriately modify this rule. In addition, EPA will address disposal of any radioactive wastes that are not covered by 40 CFR Part 191 or 40 CFR Part 192 (the Agency's standards for disposal of uranium mill tailings) as it considers standards for disposal of low-level radioactive wastes (48 FR 39563).

Finally, incorporating the NWPA definition of high-level waste also includes the phrase "consistent with existing law" when describing the NRC's responsibilities to identify materials as high-level waste. Promulgation of 40 CFR Part 191 with this definition does not signify Agency acceptance or endorsement of any particular interpretation of the phrase "consistent with existing law." The Agency presumes that the Commission will specify the applicability of its existing authorities as it conducts the relevant rulemaking efforts.

The Level of Protection

In the proposed rule, the containment requirements for disposal systems limited the residual risks to no more than an estimated 1,000 premature cancer deaths over the first 10,000 years after disposal of the wastes from 100,000 metric tons of heavy metal (MTHM) used as fuel in a nuclear reactor. The

Agency pointed out that a variety of mined repository designs using different combinations of geologic media and engineered controls were expected to meet these requirements. It was also estimated that the residual risks to future generations appeared to be no greater than if the uranium ore used to create the wastes had not been mined. EPA particularly asked for comment on whether it had taken an appropriate and reasonable approach in choosing this level of protection based upon these considerations.

Most of the public comments found this approach satisfactory. However, some commenters argued that the risks from unmined uranium ore did not necessarily define an acceptably low level of residual risks. They pointed out that such risks may vary from place to place (and a high-level waste repository could "redistribute" them) and that society sometimes does take measures to clean-up naturally-occurring radioactivity, implying that such natural risks are not always "acceptable."

On the other hand, some commenters felt that the level of protection sought in the proposed rule was far too stringent when compared to risks allowed and accepted by society from other activities. For example, the SAB Subcommittee recommended that the desired level of protection be relaxed by at least a factor of ten for this reason, coupled with the Subcommittee's concern that the uncertainties in analytical projections over thousands of years could make it difficult to demonstrate compliance with the proposed containment requirements.

After evaluating the public comments and updated performance assessments of geologic repositories, the Agency has retained the proposed level of protection as the basis for the long-term containment requirements in the final rule—even though it is true that long-term assessments of repository performance will encounter substantial uncertainties, as the SAB Subcommittee pointed out. Three reasons support this decision.

First, revising the performance assessments in accordance with many of the technical recommendations of the SAB has reinforced the Agency's conclusion that the proposed level of protection can reasonably be achieved by a variety of combinations of repository sites and designs—and EPA's regulatory impact analyses indicate that this level of protection can be achieved without significant effects on the cost of disposing of these wastes.

Second, comparing this level of protection with the comparable risks

from equivalent amounts of unmined uranium ore continues to reinforce the Agency's belief that this is an acceptably small residual risk for future generations. Therefore, the Agency believes that this level of protection represents a reasonable basis for these disposal standards.

Third, rather than relax the level of protection, the Agency has chosen to address the uncertainties that concerned the SAB Subcommittee by adding § 191.13(b) and by providing a more detailed "Guidance for Implementation" section to replace the proposed "Procedural Requirements." For example, this guidance points out that the entire range of possible projections of releases need not meet the containment requirements. Rather, compliance should be based upon the projections that the implementing agencies believe are more realistic. Furthermore, these revisions acknowledge that the quantitative calculations needed may have to be supplemented by reasonable qualitative judgments in order to appropriately determine compliance with the disposal standards.

In retaining the proposed level of protection, the Agency emphasizes that it is making a decision applicable only to the circumstances involving disposal of spent nuclear fuel and high-level and transuranic wastes. This rule cannot be used to establish precedents such as "no incremental risk to future generations" for extrapolation to other disposal problems. For other situations, evaluations of technological feasibility and cost-effectiveness must be considered for the particular set of circumstances. If mined geologic repositories were not capable of providing such good protection, the Agency might have chosen considerably different standards.

Time Period for Containment Requirements

Many commenters addressed the 10,000-year period used for the proposed containment requirements. A few argued that this period was too long and that EPA should only be concerned with a few hundred to a thousand years. A number of commenters supported the focus on 10,000 years. However, many commenters felt that it was inappropriate for the standards to ignore the period after 10,000 years. Some suggested that the containment requirements should address periods ranging from 50,000 to 500,000 years.

In the proposed rule, the Agency indicated that 10,000 years was chosen, in part, because compliance with quantitative standards for a

substantially longer period would have entailed considerably more uncertain calculations. There was no intention to indicate that times beyond 10,000 years were unimportant, but the Agency felt that a disposal system capable of meeting the proposed containment requirements for 10,000 years would continue to protect people and the environment well beyond 10,000 years. The SAB Subcommittee reviewed and supported these technical arguments for limiting the containment requirements to a 10,000-year period. Those commenters who argued for longer periods did not suggest effective ways that might compensate for the substantially greater uncertainties inherent in longer projections of disposal system performance.

However, many of the commenters and the SAB Subcommittee suggested that more qualitative or comparative assessments beyond 10,000 years might be appropriate. The Agency agreed with these comments and worked with the DOE to formulate comparative assessment provisions that have been incorporated into the final version of the Department's site selection guidelines (10 CFR Part 960). These provisions call for comparisons of the projected releases from undisturbed performance of alternative repository sites over 100,000 years to be a significant consideration in site selection. Since natural barriers are expected to provide the primary protection for such long time frames, this provision should allow for appropriate consideration of longer time periods without requiring the absolute values of these very uncertain calculations to meet a specific quantitative test. With the inclusion of this comparative test in 10 CFR Part 960, the Agency believes that no modification is needed in 40 CFR Part 191.

Use of Quantitative Probabilities in the Containment Requirements

The containment requirements in the proposed rule applied to two categories of potential releases ("reasonably foreseeable" and "very unlikely") based upon their projected probabilities of occurrence over the first 10,000 years after disposal. In its comments on the proposed rule, the NRC objected to the proposed quantitative definitions of these probabilities on the basis that calculation of such probabilities could be so uncertain that it would be impractical to determine whether the standards had been complied with. Instead, the NRC suggested substitution of qualitative terms to identify the two categories of potential releases. The wording proposed by the NRC was

formulated in terms of releases that might be caused by geologic processes and events.

In the second round of comment, the Agency sought information on whether to adopt the NRC's recommended wording or to retain definitions based on quantitative probabilities. Although a number of commenters agreed with the NRC position, the preponderance of comments supported retention of the quantitative probabilities. The SAB Subcommittee strongly supported retention of the probabilistic structure, but with substantially less restrictive probabilities and with the proviso that the Agency be sure that such conditions would be "... practical to meet and [would] not lead to serious impediments, legal or otherwise, to the licensing of high-level waste repositories." After considering all of this information, the Agency has revised the structure of the containment requirements in several ways that will retain quantitative objectives for long-term containment while allowing the implementing agencies enough flexibility to make qualitative judgments when necessary.

First, the final rule does not use the terms "reasonably foreseeable" and "very unlikely" releases. Instead, the permissible probabilities for two different levels of cumulative releases (over 10,000 years after disposal) are now incorporated directly into the containment requirements.

Second, the numerical probabilities associated with the two release categories have been increased by an order of magnitude to reflect further assessments of the uncertainties associated with projecting the probabilities of geologic events such as fault movement.

Third, the final rule clearly indicates that comprehensive performance assessments, including estimates of the probabilities of various potential releases whenever meaningful estimates are practicable, are needed to determine compliance with the containment requirements.

Fourth, a paragraph has been added to the final containment requirements (Section 191.13) to emphasize that unequivocal proof of compliance is neither expected nor required because of the substantial uncertainties inherent in such long-term projections. Instead, the appropriate test is a reasonable expectation of compliance based upon practically obtainable information and analysis. This paragraph was patterned after a paragraph that considered similar issues in NRC's 10 CFR Part 60.

Finally, the "Guidance for Implementation" section has been

added (Appendix B). This part of the rule describes the Agency's assumptions regarding performance assessments and uncertainties and should discourage overly restrictive or inappropriate implementation of the containment requirements.

The Agency believes that these revisions to the proposed rule preserve an objective framework for application of the containment requirements that requires very stringent isolation while allowing the implementing agencies adequate flexibility to handle specific uncertainties that may be encountered.

Within this framework, the possibility of inadvertent human intrusion into or nearby a repository requires special attention. Such intrusion can significantly disrupt the containment afforded by a geologic repository (as well as being dangerous for the intruders), and repositories should be selected and designed to reduce the risks from such potential disruptions. However, assessing the ways and the reasons that people might explore underground in the future—and evaluating the effectiveness of passive controls to deter such exploration near a repository—will entail informed judgment and speculation. It will not be possible to develop a "correct" estimate of the probability of such intrusion. The Agency believes that performance assessments should consider the possibilities of such intrusion, but that limits should be placed on the severity of the assumptions used to make the assessments. Appendix B to the final rule describes a set of parameters about the likelihood and consequences of inadvertent intrusion that the Agency assumed were the most pessimistic that would be reasonable in making performance assessments. The implementing agencies may adopt these assumptions or develop similar ones of their own. However, as indicated under the discussion of institutional controls, the Agency does not believe that institutional controls can be relied upon to completely eliminate the possibility of inadvertent intrusion.

Definition of "Accessible Environment"

The containment requirements limit releases to the "accessible environment" for 10,000 years after disposal. In the proposed rule, ground water within 10 kilometers of a disposal system was excluded from the definition of accessible environment. This definition was intended to reflect the concept that the geologic media surrounding a mined repository are part of the long-term containment system, with disposal sites being selected so that the surrounding media prevent or

retard transport of radionuclides through ground water. Such surrounding media would be dedicated for this purpose, with the intention to prohibit incompatible activities (either those that might disrupt the disposal system or those that could cause significant radiation exposures) in perpetuity. Applying standards to the ground water contained within these geologic media surrounding a repository would ignore the role of this natural barrier, and it could reduce the incentive to search for sites with characteristics that would enhance long-term containment of these wastes. (At the same time, the Agency recognized that the institutional controls designed to reserve this area around a disposal system cannot be considered infallible, and other provisions of the rule are designed to reduce the consequences of potential failures.)

Many commenters objected to the definition of accessible environment incorporated in the proposed rule. Some recommended that all ground water, or all "potable" ground water, should be included. Others agreed that it was appropriate to exclude some ground water in the immediate vicinity of a repository, but argued that the proposed 10-kilometer distance was too long—particularly for ground water sources that were likely to be used in the future. A few commenters thought that the proposed definition was too restrictive by including all ground water beyond 10 kilometers; they suggested that poor quality ground water sources unlikely to be used in the future should not be part of the accessible environment at all.

After considering these comments, the Agency has decided to make several changes in the definition of the "accessible environment." First, the concept of a "controlled area" has been adopted from NRC's 10 CFR Part 60. This establishes an area around a disposal system that is to be identified by markers, records, and other passive institutional controls intended to prohibit incompatible activities from the area. Consistent with the proposed 40 CFR Part 191, the current NRC definition of "controlled area" limits its distance from the edge of a repository to no more than 10 kilometers. The final 40 CFR Part 191 defines "accessible environment" to include: (1) The atmosphere, land surfaces, surface waters, and the oceans, wherever they are located; and (2) portions of the lithosphere—and the ground water within it—that are beyond the controlled area.

Second, the Agency has made the definition of the "controlled area" more restrictive than that currently

incorporated in 10 CFR Part 60. This revised definition limits the controlled area to a distance no greater than five kilometers from the original emplacement of wastes in a disposal system, rather than 10 kilometers. Furthermore, the revised definition limits the area encompassed by the controlled area to no more than 100 square kilometers, which is approximately the area that would be encompassed by a controlled area at a distance of three kilometers from all sides of a typical repository configuration. (A distance of five kilometers from all sides of a typical repository would correspond to an area of about 200 square kilometers, whereas a distance of ten kilometers from all sides corresponds to an area of almost 500 square kilometers.) This revised definition substantially reduces the area of the lithosphere that would have been removed from the "accessible environment" defined in the proposed rule, and it somewhat reduces the distance used in the proposed rule. The five-kilometer distance was chosen to retain reasonable compatibility with the NRC's requirement for a preemplacement ground water travel time of 1,000 years to the accessible environment (one of the 10 CFR Part 60 requirements developed in concert with the proposed rule), while still providing for greater isolation than called for by the proposed rule. This definition of the accessible environment will allow a controlled area to be established asymmetrically around a repository based upon the particular characteristics of a site.

Release Limits vs. Individual Dose Limits

The Agency believes that the containment requirements in § 191.13 will insure that the overall population risks to future generations from disposal of these wastes will be acceptably small. However, the situation with regard to potential individual doses is more complicated. Even with good engineering controls, some waste may eventually (i.e., several hundreds or thousands of years after disposal) be released into any ground water that might be in the immediate vicinity of a geologic repository. Since ground water generally provides relatively little dilution, anyone using such contaminated ground water in the future may receive a substantial radiation exposure (e.g., several rems per year or more). This possibility is inherent in collecting a very large amount of radioactivity in a small area.

The proposed rule did not contain any numerical restrictions on such potential individual doses after disposal. Rather, the proposal relied on several of the qualitative assurance requirements to greatly reduce the likelihood of such exposures. In particular, the assurance requirement calling for extensive permanent markers and records was intended to perpetuate information to future generations about the dangers of intruding into the vicinity of a repository. The assurance requirement to avoid sites with significant resources was intended to reduce the incentive to explore around a repository even if the information passed on was ignored or misunderstood. And the assurance requirements to use multiple barriers, both engineered and natural, and to keep releases as small as reasonably achievable were intended to encourage reduction of releases to ground water beyond that needed to meet the containment requirements—further reducing the potential for harmful individual exposures.

This approach to potential individual exposures was highlighted for comment when 40 CFR Part 191 was proposed. After receiving many recommendations to incorporate a limitation on individual doses after disposal, the Agency sought comment on further details of such a limitation in the second round of comments. For example, EPA asked whether such a limitation should apply to ground water use, whether it should apply only for ground water at some distance from a geologic repository or for any ground water source, and whether reliance on existing individual dose limitations (such as 40 CFR Part 141 or 10 CFR Part 20) for protection regarding ground water would be adequate.

The responses resulting from these questions offered a wide range of suggestions. A number of commenters opposed inclusion of an individual dose limitation for disposal on the grounds that calculations to judge compliance with such a standard would be highly speculative and not an appropriate basis upon which to judge the adequacy of a disposal system. In contrast, some other commenters argued that an individual dose standard in the 5 to 25 millirems per year range should apply to use of ground water in the accessible environment for an indefinitely long period into the future. Another group of commenters supported inclusion of some limitation on individual exposure, but only to the extent that it would not compromise the primary intent of long-term isolation and containment of the wastes.

These comments did not offer information that changed the Agency's perception of some of the problems associated with individual dose limitations for disposal. First, relying *only* upon an individual dose standard for disposal could encourage disposal methods that would enhance dilution of any wastes released. Thus, disposal sites near bodies of surface water or large sources of ground water might be preferred—which the Agency believes is an inappropriate policy that would usually increase overall population exposures.

This concern could be met by *adding* an individual dose limitation to the proposed containment requirements, rather than replacing them. However, the Agency's performance assessments of geologic repositories indicate that doses from using ground water close to a repository can become substantial (e.g., several rems per year) after a few hundred or thousand years, because the geological and geochemical characteristics of appropriate sites tend to concentrate eventual releases of wastes in any ground water that is close to the site. A study published by the National Academy of Sciences in April 1983 confirms this potential for large individual doses if flowing ground water can contact the wastes after the waste canisters are presumed to start leaking. Although it might be possible to find certain geologic settings that avoid this problem, such restrictive siting prerequisites could substantially delay development of disposal systems without providing significantly more protection to populations. Furthermore, even if reasonable limitations on individual exposure might be met at certain sites for very long times, demonstrating compliance with such limitations could be very difficult because of the additional complexities involved in estimating individual exposures rather than amounts of radioactivity released. The SAB Subcommittee report generally agreed with the technical aspects of these conclusions.

On the other hand, analyses of repository systems with good engineering controls show that they should be able to prevent significant doses from ground water use for at least a thousand years after disposal. Such protection would be compatible with both the proposed containment and assurance requirements. Accordingly, the SAB Subcommittee recommended that the Agency include a requirement limiting individual doses for the first 500 years after disposal, and one of the States that commented on the proposed

rule suggested an individual dose limit for 1,000 years after disposal.

After considering all of this information, the Agency has decided to include two new sections in the final rule. The first (Section 191.15) limits exposures to members of the public after disposal, while the second (Section 191.16) limits concentrations in water withdrawn from certain important sources of ground water after disposal.

The individual protection requirements in § 191.15 limit exposures from a disposal system to individuals in the accessible environment to 25 millirems per year to the whole body and 75 millirems per year to any organ. These limits apply only to undisturbed performance of the disposal system (i.e., without any consideration of human intrusion or disruption by unlikely natural events), and they apply for the first 1,000 years after disposal. All potential pathways of radiation or radioactive material from the disposal system to people (associated with undisturbed performance) shall be considered, including the assumption that an individual drinks two liters per day of water from any "significant source of ground water" outside of the "controlled area" surrounding a disposal system. If the implementing agency plans to allow individuals to use ground water within the controlled area, such planned use would also have to be considered within the pathways evaluated to determine compliance with § 191.15.

"Significant sources of ground water" are defined to include any aquifer currently providing the primary source of water for a community water system or any aquifer that satisfies all of the following five conditions: (1) It is saturated with water containing less than 10,000 milligrams per liter of total dissolved solids; (2) it is within 2,500 feet of the land surface; (3) it has a transmissivity of at least 200 gallons per day per foot, provided that (4) each of the underground formations or parts of underground formations included within the aquifer must have an individual hydraulic conductivity greater than 2 gallons per day per square foot; and (5) it must be capable of providing a sustained yield of 10,000 gallons per day of water to a pumped or flowing well.

Although such quantitative distinctions are inevitably somewhat arbitrary, the Agency believes that they provide reasonable demarcations to identify underground formations that could meet the needs of community water systems in the future. The selected transmissivity of 200 gallons per day per foot and the sustained yield

of 10,000 gallons per day roughly correspond to the size of a ground water source required to support the needs of about 20 households; this is similar to the size of the community water system considered in 40 CFR Part 141. The water quality criterion of 10,000 milligrams per liter of total dissolved solids has been used in several previous Agency regulations and is based upon congressional guidance in the legislative history of the Safe Drinking Water Act. The maximum depth criterion of 2,500 feet was chosen because almost all of the wells used to provide water to significant numbers of people do not extend below this depth. The minimum hydraulic conductivity criterion of 2 gallons per day per square foot was chosen to insure that only reasonably permeable formations are considered, rather than including unproductive formations that might be in the vicinity of a "significant source of ground water."

The ground water protection requirements in § 191.16(a) limit the concentrations in water withdrawn from any "special source of ground water" in the vicinity of a disposal system to concentrations similar to those established for the output of community water systems by 40 CFR Part 141: (1) 5 picocuries per liter of radium-226 and radium-228; (2) 15 picocuries per liter of alpha-emitting radionuclides (including radium-226 and radium-228 but excluding radon); or (3) the combined concentrations of radionuclides that emit either beta or gamma radiation that would produce an annual dose equivalent to the total body or any internal organ greater than 4 millirems per year if an individual continuously consumed 2 liters per day of drinking water from that source of water. However, if the preexisting radionuclide concentrations in the special source of ground water already exceed any of these limits, then § 191.16(b) limits any increases in the preexisting concentrations to the concentration limits set in § 191.16(a). Like the individual protection requirements, the ground water protection requirements apply only for undisturbed performance of the disposal system and apply for the first 1,000 years after disposal. Unlike the individual protection requirements, the ground water requirements would apply to a "special source" if it was within the controlled area.

"Special sources" are defined to include only those Class I ground waters—to be identified in accordance with the Agency's Ground-Water Protection Strategy published in August 1984—that meet the following three

conditions: (1) They are within the controlled area or near (less than five kilometers beyond) the controlled area; (2) they are supplying drinking water for thousands of persons as of the date that the Department selects the site for extensive exploration as a potential location of a disposal system; and (3) they are irreplaceable in that no reasonable alternative source of drinking water is available to that population.

Need for the Assurance Requirements

The preceding issues dealt with the quantitative requirements of the disposal standards. While numerical standards are important to bring about appropriate selection and design of disposal systems, the Agency has long recognized that the numerical standards chosen for Subpart B, by themselves, do not provide either an adequate context for environmental protection or a sufficient basis to foster public confidence in the national program. There are too many uncertainties in projecting the behavior of natural and engineered components for many thousands of years—and too many opportunities for mistakes or poor judgments in such calculations—for the numerical requirements on overall system performance in Subpart B to be the sole basis to determine the acceptability of disposal systems for these very hazardous wastes. These uncertainties and potential errors in quantitative analysis could ultimately prevent the degree of protection sought by the Agency from being achieved. (Theoretically, it might be possible to develop adequate confidence in achieving this level of protection by choosing much more stringent numerical standards, but this could lead to substantial difficulties in implementation.) Therefore, the proposed standards also included qualitative assurance requirements chosen to ensure that cautious steps are taken to reduce the problems caused by these uncertainties. The proposed rule emphasized that the assurance requirements were an essential complement to the quantitative containment requirements that were selected.

In its comments on the proposed rule, the NRC argued that the assurance requirements were not properly part of the Agency's generally applicable standards. The Commission agreed that the overall numerical performance standards were not sufficient, but suggested that its regulations and procedures were the appropriate vehicle to provide the necessary confidence that the inherent uncertainties would not

compromise environmental protection. The Agency believes that it does have the authority to give regulatory expression to the context within which it has chosen to establish one set of numerical standards rather than another. However, because it might not be appropriate to exercise this authority, the Agency sought public comment on the need for the assurance requirements in the second round of comments.

The preponderance of comments received on this question strongly supported retention of the assurance requirements in 40 CFR Part 191. In particular, virtually all of the various State governments that commented on the rule described the assurance requirements as an essential part of the regulations governing disposal of these wastes. Subsequently, two of these States, Nevada and Minnesota, petitioned the Commission to incorporate the assurance requirements proposed as part of 40 CFR Part 191 into its own rules (50 FR 18267).

Based upon these comments, the Agency and the NRC have reached an agreement that should accomplish the desired regulatory goals while avoiding the jurisdictional issue. EPA has included the assurance requirements in the final rule, modified as appropriate in response to other comments. However, these requirements will not be applicable to disposal facilities to be licensed by the Commission. Instead, as discussed previously, the NRC staff plans to propose modifications to 10 CFR Part 60, developed in consultation with EPA, for public review and comment within approximately 120 days to insure that the objectives of all of the assurance requirements in 40 CFR Part 191 will be accomplished through compliance with 10 CFR Part 60. The Agency has provided the Commission with all of the comments received by EPA regarding the assurance requirements, so that the NRC can use them in its rulemaking. In addition, the Agency will participate in the NRC rulemaking to facilitate incorporation of the principles of all of the assurance requirements in Federal regulation. Finally, the Agency will review the record and outcome of the Part 60 rulemaking to determine if any subsequent modifications to 40 CFR Part 191 are needed.

Approach Toward Institutional Controls

The Agency particularly sought comment on its proposed approach to reliance on institutional controls. The proposed rule limited reliance on "active institutional controls" (such as controlling access to a disposal site,

performing maintenance operations, or cleaning up releases) to a reasonable period of time after disposal, described as on the order of a "few hundred years." On the other hand, "passive institutional controls" (such as permanent markers, records, archives, and other methods of preserving knowledge) were considered to be at least partially effective for a longer period of time.

Few commenters argued with the distinction between active and passive institutional controls, or with the amount of reliance the proposed rule envisioned for passive controls. However, many commenters felt that "a few hundred years" was too long a period to count on active controls. Accordingly, the final rule limits reliance on active institutional controls to no more than 100 years after disposal. This was the time period the Agency considered in criteria for radioactive waste disposal that were proposed for public comment in 1978 (43 FR 53262), a period that was generally supported by the commenters on that proposal. After this time, no contribution from any of the active institutional controls can be projected to prevent or limit potential releases of waste from a disposal system.

The concept of passive institutional controls has now been incorporated into the definition of "controlled area" that is used to establish one of the boundaries for applicability of the containment requirements and the individual protection requirements in the final rule. Because the assumptions made about the effectiveness of passive institutional controls can strongly affect implementation of the containment requirements, the Agency's intent has been elaborated in the "guidance for implementation" section. The Federal Government is committed to retaining control over disposal sites for these wastes as long as possible. Accordingly (and in compliance with one of the assurance requirements), an extensive system of explanatory markers and records will be instituted to warn future generations about the location and dangers of these wastes. These passive controls have not been assumed to prevent all possibilities of inadvertent human intrusion, because there will always be a realistic chance that some individuals will overlook or misunderstand the markers and records. (For example, exploratory drilling operations occasionally intrude into areas that clearly would have been avoided if existing information had been obtained and properly evaluated.) However, the Agency assumed that

society in general will retain knowledge about these wastes and that future societies should be able to deter systematic or persistent exploitation of a disposal site.

The Agency also assumed that passive institutional controls should reduce the chance of inadvertent intrusion compared to the likelihood if no markers and records were in place. Specific judgments about the chances and consequences of intrusion should be made by the implementing agencies when more information about particular disposal sites and passive control systems is available. The parameters described in the "guidance for implementation" represent the most severe assumptions that the Agency believed were reasonable to use in its analyses to evaluate the feasibility of compliance with this rule (analyses that are summarized in the BID). The implementing agencies are free to use other assumption if they develop information considered adequate to support those judgments.

The role envisioned for institutional controls in this rulemaking has been adapted from the general approach the Agency has followed in its activities involving disposal of radioactive wastes since the initial public workshops conducted in 1977 and 1978. The Agency's overall objective has been to protect public health and the environment from disposal of radioactive wastes without relying upon institutional controls for extended periods of time—because such controls do not appear to be reliable enough over the very long periods that these wastes remain dangerous. Instead, the Agency has pursued standards that call for isolation of the wastes through the physical characteristics of disposal system siting and design, rather than through continuing maintenance and surveillance. This principle was enunciated in the general criteria published for public comment in 1978 (43 FR 53262), and it has been incorporated into the Agency's standards for disposal of uranium mill tailings (48 FR 590, 48 FR 45926).

This approach has been tailored to fit two circumstances associated with mined geologic repositories. First, 40 CFR Part 191 places containment requirements on a broad range of potential unplanned releases as well as the expected behavior of the disposal system. Therefore, determining compliance with the standards involves performance assessments that consider the probabilities and consequences of a variety of disruptive events, including potential human intrusion. Not allowing

passive institutional controls to be taken into account to some degree when estimating the consequences of inadvertent human intrusion could lead to less protective geologic media being selected for repository sites. The Agency's analyses indicate that repositories in salt formations have particularly good capabilities to isolate the wastes from flowing ground water and, hence, the accessible environment. However, salt formations are also relatively easy to mine and are often associated with other types of resources. If performance assessments had to assume that future societies will have no way to ever recognize and limit the consequences of inadvertent intrusion (from solution mining of salt, for example), the scenarios that would have to be studied would be more likely to eliminate salt media from consideration than other rock types. Yet, this could rule out repositories that may provide the best isolation, compared to other alternatives, if less pessimistic assumptions about survival of knowledge were made.

The second circumstance that the Agency considered in evaluating the approach towards institutional controls taken in this rule is the fact that the mined geologic repositories planned for disposal of the materials covered by 40 CFR Part 191 are different from the disposal systems envisioned for any other types of waste. The types of inadvertent human activities that could lead to significant radiation exposures or releases of material from geologic repositories appear to call for much more intensive and organized effort than those which could cause problems at, for example, an unattended surface disposal site. It appears reasonable to assume that information regarding the disposal system is more likely to reach (and presumably deter) people undertaking such organized efforts than it is to inform individuals involved in mundane activities.

These considerations led the Agency to conclude that a limited role for passive institutional controls would be appropriate when projecting the long-term performance of mined geologic repositories to judge compliance with these standards. However, such assumptions would not necessarily be applicable to other Agency actions where different issues are involved.

Avoiding Sites With Natural Resources

The proposed rule contained an assurance requirement that would have prohibited use of sites where there is a reasonable expectation that future exploration for scarce or easily

accessible resources might occur. The comments received on this issue generally agreed that sites with resources should be avoided. However, some commenters suggested that the requirement should be more restrictive, to include "potentially accessible" resources. Other commenters argued that the Agency should be less restrictive regarding sites with possible resource potential—discouraging but not prohibiting their use—because other attributes of the site might overcome the relative disadvantages presented by resource potential.

After considering these comments, the Agency agreed with the latter viewpoint. This judgment was reinforced by the belief that disposal sites should be chosen after comparative evaluation of a variety of alternatives, and the proposed assurance requirement could have inhibited this process. Therefore, this assurance requirement has been revised in the final rule to identify resource potential as a disincentive but not as an outright prohibition for site selection. Instead, the revised assurance requirement states that places with resource potential shall not be used "unless the favorable characteristics of such places compensate for their greater likelihood of being disturbed in the future."

This wording implies a qualitative comparison, because the Agency is not aware of quantitative formulas comprehensive enough to provide adequate comparisons to govern site selection. However, the Agency does not intend that sites with resource potential can be used merely upon identification of a few features that might be more favorable than at a site without significant resources. Rather, sites with resources should only be used if it is reasonably certain that they would provide better *overall* protection than the practical alternatives that are available.

The following example illustrates the effect of the change in this assurance requirement. When discussing the proposed assurance requirement, the Agency implied that disposal in salt domes might not be acceptable because such formations seemed more likely than others to attract exploration in the future. The modification of this assurance requirement in the final rule means that salt domes should not be peremptorily removed from consideration, but should be compared against all of the characteristics of alternative sites in terms of the overall environmental protection expected.

Long-Term Monitoring

The proposed rule addressed active institutional controls over a disposal site only in a negative sense—to prohibit reliance upon them for more than a few hundred years after disposal. The Agency's intent was to be sure that long-term protection of the environment did not depend upon positive actions by future generations. Almost all commenters agreed with this intent, although many suggested a shorter period of reliance was appropriate (see the preceding discussion under "Approach Towards Institutional Controls").

However, several commenters (including most of the States) also urged addition of a requirement for long-term monitoring of a repository after disposal. This view did not deny the need to select and design disposal systems without depending upon active controls in the future. However, it broadened this perspective by arguing that a disposal system so designed should still be monitored for a long time after disposal to guard against unexpected failures.

The Agency had not considered this viewpoint in developing the proposed rule. Accordingly, further information on this idea was sought during the "second round" of public comment, and the Agency surveyed the capabilities and expectations of long-term monitoring approaches. Evaluating this information led the Agency to several conclusions:

(1) Perhaps most importantly, the techniques used for monitoring after disposal must not jeopardize the long-term isolation capabilities of the disposal system. Furthermore, plans to conduct monitoring after disposal should never become an excuse to relax the care with which systems to isolate these wastes must be selected, designed, constructed, and operated.

(2) Monitoring for radionuclide releases to the accessible environment is not likely to be productive. Even a poorly performing geologic repository is very unlikely to allow measurable releases to the accessible environment for several hundreds of years of more, particularly in view of the engineered controls needed to comply with 10 CFR Part 60. A monitoring system based only on detecting radionuclide releases—a system which would almost certainly not be detecting anything for several times the history of the United States—is not likely to be maintained for long enough to be of much use.

(3) Within the above constraints, however, there are likely to be monitoring approaches which may, in a relatively short time, significantly improve confidence that a repository is

performing as intended. Two examples are of particular interest. One involves the concept of monitoring ground water sources at a variety of distances for benign tracers intentionally released to the ground water in the repository; this approach can evaluate the delay involved in ground water movement from the repository to the environment and can serve to validate expectations of the performance expected from the system's natural barriers. Another concept involves monitoring the small uplift of the land surface over the repository in order to validate predictions of the system's thermal behavior. Both of these approaches can be carried out without enhancing pathways for the wastes to escape from the repository.

Based on these conclusions and the public comments on this question, the Agency has included a provision for long-term monitoring after disposal in the assurance requirements of the final rule: "Disposal systems shall be monitored after disposal to detect substantial and detrimental deviations from expected performance. This monitoring shall be done with techniques that do not jeopardize the isolation of the wastes and shall be conducted until there are no significant concerns to be addressed by further monitoring." This new provision is consistent with the overall intent of the assurance requirements: To take prudent and cautious steps necessary to minimize the risks posed by the inherent uncertainties in expectations of the future. Beyond this broad mandate, however, the Agency has not specified the details of a monitoring program. That is properly left to the implementing agencies. Furthermore, the precise objectives of an appropriate monitoring program probably should not be spelled out until much more information is gathered about the characteristics and expected behavior of specific sites and designs.

Ability To Recover Wastes After Disposal

The proposed rule included an assurance requirement that recovery of these wastes be feasible for "a reasonable period of time" after disposal. The Agency specifically sought comment on whether this was a desirable provision, since it would rule out certain disposal concepts, such as deep-well injection of liquid wastes. The comments received were split about evenly between those who thought the provision should be retained and those who thought it was detrimental to the overall rule. Many of those who opposed

the requirement argued that it would encourage designing a geologic repository to make retrieving waste relatively easy—which might compromise the isolation capabilities of the repository or which might encourage recovery of the waste to make use of some intrinsic value it might retain (the potential energy content of spent nuclear fuel, for example).

The intent of this provision was not to make recovery of waste easy or cheap, but merely possible in case some future discovery or insight made it clear that the wastes needed to be relocated. EPA reiterates the statement in the preamble to the proposal that *any* current concept for a mined geologic repository meets this requirement *without* any additional procedures or design features. For example, there is no intent to require that a repository shaft be kept open to allow future recovery. To meet this assurance requirement, it only need be technologically feasible (assuming current technology levels) to be able to mine the sealed repository and recover the waste—albeit at substantial cost and occupational risk. The Commission's requirements for multiple-engineered barriers within a repository (10 CFR Part 60) adequately address any concerns about the feasibility of recovering wastes from a repository.

Therefore, this provision should not have any effect upon plans for mined geologic repositories. Rather, it is intended to call into question any other disposal concept that might not be so reversible—because the Agency believes that future generations should have options to correct any mistakes that this generation might unintentionally make. Almost all of the commenters agreed with the validity of this objective. Accordingly, the Agency has decided to retain this assurance requirement in the final rule as proposed.

Health Impacts of 40 CFR Part 191

Waste Management and Storage.

Waste management and storage activities conducted in accordance with Subpart A would limit the maximum risk to a member of the public in the general environment to a 5×10^{-4} chance of incurring a premature fatal cancer over a lifetime. Of course, a risk this large would exist only for an individual continuously exposed to the full amount of the dose limits over his or her lifetime. Because the Agency believes that such continuous exposure is very unlikely, the actual risks to individuals are expected to be much lower. It is theoretically possible under the final rule that an individual could be exposed to 25 millirems per year (to the whole

body) from both an NRC-licensed facility and a DOE facility not licensed by NRC, for a total of 50 millirem/year. However, the Agency believes that this is particularly improbable and does not foresee a significant public health impact from this possibility.

Waste Disposal. A disposal system complying with Subpart B would confine almost all of the radioactive wastes to the immediate vicinity of the repository for a very long time. Because the wastes would be so well isolated from the environment, the Agency is confident that any risks to future populations would be very small. Similarly, risks to most future individuals would also be very small (and effectively zero in almost all cases)—except for the possibility that an individual in the distant future might use ground water from the vicinity of a repository. In this case, there is a chance that such an individual might receive a substantial exposure. The following paragraphs describe the possible health impacts of the residual risks from a disposal system that would be in compliance with 40 CFR Part 191.

Population Risks: With regard to exposure of populations, the Agency has estimated the potential long-term health risks to future generations from various types of mined geologic repositories using very general models of environmental transport and a linear, nonthreshold dose-effect relationship between radiation exposures and premature deaths from cancer. Food chains, ways of life, and the size and geographical distributions of populations will undoubtedly change over a 10,000-year period. Unlike geological processes, factors such as these cannot be usefully predicted over such long periods of time. Thus, in making these health effects projections, the Agency found it necessary to depend upon very general models of environmental pathways and to assume current population distributions and death rates. The SAB Subcommittee evaluated these models carefully, and, although a number of specific changes were recommended for particular parameters, the Subcommittee endorsed the general approach. As a consequence of using these generalized models, EPA's projections are intended to be used primarily as a tool for comparing the performance of one waste disposal system to another and for comparison of the risks of waste disposal with those of undisturbed ore bodies. The results of these analyses should not be considered a reliable projection of the "real" or absolute number of health effects

resulting from compliance with the disposal standards.

These health risk models were used to assess the long-term health risks from several different model repositories containing the wastes from 100,000 MTHM—which could include all existing wastes and the future wastes from all currently operating reactors. The Agency estimates that this quantity of waste, when disposed of in accordance with the proposed standards, would cause no more than 1,000 premature deaths from cancer in the first 10,000 years after disposal: an average of no more than one premature death every 10 years. Most of the model repositories considered had projected population risks at least a factor of ten below this, or about 100 deaths over 10,000 years. The projections for the actual repositories that are constructed are expected to be closer to this lower figure. Any such increase in the number of cancer deaths would be very small compared to today's incidence of cancer, which kills about 350,000 people per year in the United States. Similarly, any such increase would be much less than the approximately 6,000 premature cancer deaths per year that the same linear, non-threshold dose-effect relationship predicts for the nation due to natural background radiation.

Individual Risks: With regard to exposures of individuals, the Agency examined the potential doses to persons who might use ground water from the immediate vicinity of a repository at various times in the future. For these analyses, only the expected undisturbed performance of a repository was considered (e.g. there was no evaluation of exposures that might occur if a repository was disrupted by movement of a fault). In most of the cases studied, no exposures occurred for more than one thousand years after disposal. After that, these analyses predict that significant exposures (on the order of a few rems per year in the vicinity of the repository over the next several thousands of years) may appear for some of the geologic media considered. These projections are similar to those contained in the April 1983 report published by the National Academy of Sciences. The BID contains more detailed descriptions of the Agency's individual dose calculations.

Intergenerational Risk: As described earlier, the Agency has chosen to rely on provisions that limit risks to populations as the primary standards for the long-term performance of disposal systems. Although the projections of the residual population risk are clearly very small, the discontinuity between when the

wastes are generated and when the projected health effects manifest themselves made it difficult to determine what level of residual risk should be allowed by these disposal standards. The difficulty arose because most of the benefits derived in the process of waste production fall upon the current generation, while most of the risks fall upon future generations. Thus, a potential problem of intergenerational equity with respect to the distribution of risks and benefits became apparent. This problem is sometimes referred to as the intergenerational risk issue, and it is not unique to the disposal of high-level radioactive wastes. If the Agency tried to insure that these standards fully satisfied a criterion of intergenerational equity with respect to the distribution of risks and benefits, it might appear that no risk should be passed on to future generations. This is a condition which the Agency believes cannot be met by disposal technologies foreseeable within this century. However, there is one particular factor which has reinforced EPA's decision about the reasonableness of the risks permitted under the disposal standards. This is the following evaluation of the risks associated with undisturbed uranium ore bodies. Additionally, for the purpose of comparing the risks permitted under the standards to other radiation risks which people are currently exposed to, a brief discussion of the risks from other natural sources of radiation is also included.

Uranium Ore: Most uranium ore in the United States occurs in permeable geologic strata containing flowing ground water. Radionuclides in the ore, particularly uranium and radium, continuously enter this ground water. EPA estimated the potential risks from these undisturbed ore bodies using the same generalized environmental models that were used for releases from a waste repository. The effects associated with the amount of ore needed to produce the high-level wastes that would fill the model geologic repository can vary considerably. Part of this variation corresponds to actual differences from one ore body to another; part can be attributed to uncertainties in the assessment. After revising the population risk models in accordance with the recommendations of the SAB Subcommittee, these estimates of the risks from unmined ore bodies ranged from about 10 to more than 100,000 excess cancer deaths over 10,000 years. Thus, leaving the ore unmined appears to present a risk to future generations comparable to the risks from disposal of wastes covered by these standards.

Variations in Natural Background: Radionuclides occur naturally in the earth in very large amounts, and are produced in the atmosphere by cosmic radiation. Everyone is exposed to natural background radiation from these natural radionuclides and from direct exposure to cosmic radiation. Individual exposures average about 100 millirems per year, with a range of about 60 to 200 millirem/year. These natural background radiation levels have remained relatively constant for a very long time. According to the same linear, nonthreshold dose effect relationship used in EPA's other analyses, an increase of one millirem per year (about one percent) in natural background in the United States would result in about 60 additional deaths per year, or 600,000 over a 10,000-year period.

Natural Radionuclide Concentrations in Ground Water: One source of this exposure to natural background radiation comes from naturally occurring radionuclides found in ground water. Radium is the most important of the naturally occurring radioactive materials likely to occur in public water supply systems, but uranium is also found in ground waters due to its natural occurrence. Surveys of radionuclides in ground water systems indicate: a United States range of 0.1 to 50 picocuries (pCi) per liter for radium-226 (with isolated sources exceeding 100 pCi/liter); up to 74 pCi/liter for all alpha-emitting radionuclides other than uranium (although most of the alpha-emitting concentrations are below 3 pCi/liter); and up to 650 pCi/liter for total uranium concentrations. Elevated radium-226 concentrations are found along the Atlantic coastal region and the Midwest; low levels are usually found in the treated water supplies in the western States. Elevated uranium and alpha-emitting radionuclide concentrations are generally limited to the Rocky Mountain region and Maine and Pennsylvania in the east.

The Agency's primary drinking water regulations (40 CFR Part 141) limit the contamination levels for radium-226 and radium-228 to 5 pCi/liter and the levels for total alpha-emitting contamination (excluding radon and uranium) to 15 pCi/liter. Elevated concentrations of radium in drinking water are generally a problem associated with smaller community water systems, with an estimated 500 systems exceeding 5 pCi/liter. The Agency's risk assessments indicate that continuous consumption of water containing the maximum amount of radium allowed may cause between 0.7 and 3 cancers per year per million exposed persons.

Environmental Impacts

A Draft Environmental Impact Statement (EIS) was prepared for the proposed rule, in accordance with the Agency's procedures for the voluntary preparation of EIS's (30 FR 37419). However, section 121(c) of the NHPA subsequently exempted this action from preparation of an EIS under section 102(2)(C) of the National Environmental Policy Act of 1969 (NEPA) and from any environmental review under subparagraph (E) or (F) of section 102(2) of the NEPA. Accordingly, a Final EIS has not been prepared for promulgation of this rule. The potential health impacts of this action are summarized above, and much of the information that would have been contained in a Final EIS is documented in the Background Information Document that accompanies this final version of 40 CFR Part 191.

Regulatory Impacts

This rule was submitted to the Office of Management and Budget (OMB) for review as required by Executive Order 12291. The final rule has not been classified as a "major rule" in accordance with the guidelines provided by the Executive Order. Any comments received from OMB and EPA's responses to those comments are available for public inspection in the docket cited above under the heading "ADDRESSES."

The Agency has had to take an unusual approach in considering the regulatory impacts of this proposed action—as required by Executive Order 12291. In most cases, a regulation concerns an ongoing activity and may be considered a burden whose costs should be judged against the regulatory benefits. Here, it was not possible to quantify the costs and benefits of this action compared to the consequences of no regulation because there is no specific "baseline" program to consider. The appropriate regulations must be established before the regulated activity can even begin. Thus, the typical perspectives on costs and benefits are altered. Instead, the Agency evaluated how the costs of commercial waste management and disposal might change in response to different levels of protection from the containment requirements. Similar evaluations were not performed for the wastes from atomic energy defense activities because sufficient information was not available.

To evaluate the effects of different levels of protection, EPA considered the performance of different repository designs in several different geologic

media. The costs of the various engineering controls that might be needed to meet different levels of protection were estimated. In addition, allowances were made for the increased research and development costs that might be needed to demonstrate compliance with the standards if projected performance for a particular disposal system indicated releases less than an order of magnitude below the long-term radionuclide release limits in § 191.13.

Since the regulatory impact analyses that supported the proposed rule were performed, the NRC has promulgated minimum requirements for the engineered barriers of a disposal system (in 10 CFR Part 60), more data concerning disposal sites being considered by the Department have become available, and the Agency has reviewed its performance assessments to reduce overestimates of long-term risks in accordance with the SAB review. After evaluating all of this new information, the Agency believes that there need not be any significant additional costs to the national program for disposal of commercial wastes caused by retaining the proposed level of protection in the final rule, compared to the costs of choosing levels considerably less stringent. In other words, all of the disposal sites being evaluated by the Department, assuming compliance with the existing requirements of 10 CFR Part 60, are expected to be able to meet these disposal standards without additional precautions beyond those already planned.

List of Subjects in 40 CFR Part 191

Environmental protection, Nuclear energy, Radiation protection, Uranium, Waste treatment and disposal.

Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Administrator hereby certifies that this rule will not have any significant impact on small businesses or other entities, and that a Regulatory Flexibility Analysis is not required. This rule will affect only a small number of facilities, most of which are or will be operated by the United States Government.

Dated: August 15, 1985.

Lee M. Thomas,
Administrator.

A new Part 191 is hereby added to Title 40, Code of Federal Regulations, as follows:

SUBCHAPTER F—RADIATION PROTECTION PROGRAMS

PART 191—ENVIRONMENTAL RADIATION PROTECTION STANDARDS FOR MANAGEMENT AND DISPOSAL OF SPENT NUCLEAR FUEL, HIGH-LEVEL AND TRANSURANIC RADIOACTIVE WASTES

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Appendix A Table for Subpart B

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Authority: The Atomic Energy Act of 1954, as amended; Reorganization Plan No. 3 of 1970; and the Nuclear Waste Policy Act of 1982.

Subpart A—Environmental Standards for Management and Storage

§ 191.01 Applicability.

This Subpart applies to:

(a) Radiation doses received by members of the public as a result of the management (except for transportation) and storage of spent nuclear fuel or high-level or transuranic radioactive wastes at any facility regulated by the Nuclear Regulatory Commission or by Agreement States, to the extent that such management and storage operations are not subject to the provisions of Part 190 of title 40; and

(b) Radiation doses received by members of the public as a result of the management and storage of spent nuclear fuel or high-level or transuranic wastes at any disposal facility that is operated by the Department of Energy and that is not regulated by the Commission or by Agreement States.

§ 191.02 Definitions.

Unless otherwise indicated in this Subpart, all terms shall have the same meaning as in Subpart A of Part 190.

(a) "Agency" means the Environmental Protection Agency.

(b) "Administrator" means the Administrator of the Environmental Protection Agency.

(c) "Commission" means the Nuclear Regulatory Commission.

(d) "Department" means the Department of Energy.

(e) "NWP" means the Nuclear Waste Policy Act of 1982 (Pub. L. 97-425).

(f) "Agreement State" means any State with which the Commission or the Atomic Energy Commission has entered into an effective agreement under subsection 274b of the Atomic Energy Act of 1954, as amended (68 Stat. 919).

(g) "Spent nuclear fuel" means fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing.

(h) "High-level radioactive waste," as used in this Part, means high-level radioactive waste as defined in the Nuclear Waste Policy Act of 1982 (Pub. L. 97-425).

(i) "Transuranic radioactive waste," as used in this Part, means waste containing more than 100 nanocuries of alpha-emitting transuranic isotopes, with half-lives greater than twenty years, per gram of waste, except for: (1) High-level radioactive wastes; (2) wastes that the Department has determined, with the concurrence of the Administrator, do not need the degree of isolation required by this Part; or (3) wastes that the Commission has approved for disposal on a case-by-case basis in accordance with 10 CFR Part 61.

(j) "Radioactive waste," as used in this Part, means the high-level and transuranic radioactive waste covered by this Part.

(k) "Storage" means retention of spent nuclear fuel or radioactive wastes with the intent and capability to readily retrieve such fuel or waste for subsequent use, processing, or disposal.

(l) "Disposal" means permanent isolation of spent nuclear fuel or radioactive waste from the accessible environment with no intent of recovery, whether or not such isolation permits the recovery of such fuel or waste. For example, disposal of waste in a mined geologic repository occurs when all of the shafts to the repository are backfilled and sealed.

(m) "Management" means any activity, operation, or process (except for transportation) conducted to prepare spent nuclear fuel or radioactive waste for storage or disposal, or the activities associated with placing such fuel or waste in a disposal system.

(n) "Site" means an area contained within the boundary of a location under the effective control of persons possessing or using spent nuclear fuel or radioactive waste that are involved in

any activity, operation, or process covered by this Subpart.

(o) "General environment" means the total terrestrial, atmospheric, and aquatic environments outside sites within which any activity, operation, or process associated with the management and storage of spent nuclear fuel or radioactive waste is conducted.

(p) "Member of the public" means any individual except during the time when that individual is a worker engaged in any activity, operation, or process that is covered by the Atomic Energy Act of 1954, as amended.

(q) "Critical organ" means the most exposed human organ or tissue exclusive of the integumentary system (skin) and the cornea.

§ 191.03 Standards.

(a) Management and storage of spent nuclear fuel or high-level or transuranic radioactive wastes at all facilities regulated by the Commission or by Agreement States shall be conducted in such a manner as to provide reasonable assurance that the combined annual dose equivalent to any member of the public in the general environment resulting from: (1) Discharges of radioactive material and direct radiation from such management and storage and (2) all operations covered by Part 190; shall not exceed 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other critical organ.

(b) Management and storage of spent nuclear fuel or high-level or transuranic radioactive wastes at all facilities for the disposal of such fuel or waste that are operated by the Department and that are not regulated by the Commission or Agreement States shall be conducted in such a manner as to provide reasonable assurance that the combined annual dose equivalent to any member of the public in the general environment resulting from discharges of radioactive material and direct radiation from such management and storage shall not exceed 25 millirems to the whole body and 75 millirems to any critical organ.

§ 191.04 Alternative standards.

(a) The Administrator may issue alternative standards from those standards established in 191.03(b) for waste management and storage activities at facilities that are not regulated by the Commission or Agreement States if, upon review of an application for such alternative standards:

(1) The Administrator determines that such alternative standards will prevent

any member of the public from receiving a continuous exposure of more than 100 millirems per year dose equivalent and an infrequent exposure of more than 500 millirems dose equivalent in a year from all sources, excluding natural background and medical procedures; and

(2) The Administrator promptly makes a matter of public record the degree to which continued operation of the facility is expected to result in levels in excess of the standards specified in 191.03(b).

(b) An application for alternative standards shall be submitted as soon as possible after the Department determines that continued operation of a facility will exceed the levels specified in 191.03(b) and shall include all information necessary for the Administrator to make the determinations called for in 191.04(a).

(c) Requests for alternative standards shall be submitted to the Administrator, U.S. Environmental Protection Agency, 401 M Street, SW., Washington, DC 20460.

§ 191.05 Effective date.

The standards in this Subpart shall be effective on November 18, 1985.

Subpart B—Environmental Standards for Disposal

§ 191.11 Applicability.

(a) This Subpart applies to:

(1) Radioactive materials released into the accessible environment as a result of the disposal of spent nuclear fuel or high-level or transuranic radioactive wastes;

(2) Radiation doses received by members of the public as a result of such disposal; and

(3) Radioactive contamination of certain sources of ground water in the vicinity of disposal systems for such fuel or wastes.

(b) However, this Subpart does not apply to disposal directly into the oceans or ocean sediments. This Subpart also does not apply to wastes disposed of before the effective date of this rule.

§ 191.12 Definitions.

Unless otherwise indicated in this Subpart, all terms shall have the same meaning as in Subpart A of this Part.

(a) "Disposal system" means any combination of engineered and natural barriers that isolate spent nuclear fuel or radioactive waste after disposal.

(b) "Waste," as used in this Subpart, means any spent nuclear fuel or radioactive waste isolated in a disposal system.

(c) "Waste form" means the materials comprising the radioactive components of waste and any encapsulating or stabilizing matrix.

(d) "Barrier" means any material or structure that prevents or substantially delays movement of water or radionuclides toward the accessible environment. For example, a barrier may be a geologic structure, a canister, a waste form with physical and chemical characteristics that significantly decrease the mobility of radionuclides, or a material placed over and around waste, provided that the material or structure substantially delays movement of water or radionuclides.

(e) "Passive institutional control" means: (1) Permanent markers placed at a disposal site, (2) public records and archives, (3) government ownership and regulations regarding land or resource use, and (4) other methods of preserving knowledge about the location, design, and contents of a disposal system.

(f) "Active institutional control" means: (1) Controlling access to a disposal site by any means other than passive institutional controls; (2) performing maintenance operations or remedial actions at a site, (3) controlling or cleaning up releases from a site, or (4) monitoring parameters related to disposal system performance.

(g) "Controlled area" means: (1) A surface location, to be identified by passive institutional controls, that encompasses no more than 100 square kilometers and extends horizontally no more than five kilometers in any direction from the outer boundary of the original location of the radioactive wastes in a disposal system; and (2) the subsurface underlying such a surface location.

(h) "Ground water" means water below the land surface in a zone of saturation.

(i) "Aquifer" means an underground geological formation, group of formations, or part of a formation that is capable of yielding a significant amount of water to a well or spring.

(j) "Lithosphere" means the solid part of the Earth below the surface, including any ground water contained within it.

(k) "Accessible environment" means: (1) The atmosphere; (2) land surfaces; (3) surface waters; (4) oceans; and (5) all of the lithosphere that is beyond the controlled area.

(l) "Transmissivity" means the hydraulic conductivity integrated over the saturated thickness of an underground formation. The transmissivity of a series of formations is the sum of the individual

transmissivities of each formation comprising the series.

(m) "Community water system" means a system for the provision to the public of piped water for human consumption, if such system has at least 15 service connections used by year-round residents or regularly serves at least 25 year-round residents.

(n) "Significant source of ground water," as used in this Part, means: (1) An aquifer that: (i) Is saturated with water having less than 10,000 milligrams per liter of total dissolved solids; (ii) is within 2,500 feet of the land surface; (iii) has a transmissivity greater than 200 gallons per day per foot, provided that any formation or part of a formation included within the source of ground water has a hydraulic conductivity greater than 2 gallons per day per square foot; and (iv) is capable of continuously yielding at least 10,000 gallons per day to a pumped or flowing well for a period of at least a year; or (2) an aquifer that provides the primary source of water for a community water system as of the effective date of this Subpart.

(o) "Special source of ground water," as used in this Part, means those Class I ground waters identified in accordance with the Agency's Ground-Water Protection Strategy published in August 1984 that: (1) Are within the controlled area encompassing a disposal system or are less than five kilometers beyond the controlled area; (2) are supplying drinking water for thousands of persons as of the date that the Department chooses a location within that area for detailed characterization as a potential site for a disposal system (e.g., in accordance with Section 112(b)(1)(B) of the NHPA); and (3) are irreplaceable in that no reasonable alternative source of drinking water is available to that population.

(p) "Undisturbed performance" means the predicted behavior of a disposal system, including consideration of the uncertainties in predicted behavior, if the disposal system is not disrupted by human intrusion or the occurrence of unlikely natural events.

(q) "Performance assessment" means an analysis that: (1) Identifies the processes and events that might affect the disposal system; (2) examines the effects of these processes and events on the performance of the disposal system; and (3) estimates the cumulative releases of radionuclides, considering the associated uncertainties, caused by all significant processes and events. These estimates shall be incorporated into an overall probability distribution of cumulative release to the extent practicable.

(r) "Heavy metal" means all uranium, plutonium, or thorium placed into a nuclear reactor.

(s) "Implementing agency," as used in this Subpart, means the Commission for spent nuclear fuel or high-level or transuranic wastes to be disposed of in facilities licensed by the Commission in accordance with the Energy Reorganization Act of 1974 and the Nuclear Waste Policy Act of 1982, and it means the Department for all other radioactive wastes covered by this Part.

§ 191.13 Containment requirements.

(a) Disposal systems for spent nuclear fuel or high-level or transuranic radioactive wastes shall be designed to provide a reasonable expectation, based upon performance assessments, that the cumulative releases of radionuclides to the accessible environment for 10,000 years after disposal from all significant processes and events that may affect the disposal system shall:

(1) Have a likelihood of less than one chance in 10 of exceeding the quantities calculated according to Table 1 (Appendix A); and

(2) Have a likelihood of less than one chance in 1,000 of exceeding ten times the quantities calculated according to Table 1 (Appendix A).

(b) Performance assessments need not provide complete assurance that the requirements of 191.13(a) will be met. Because of the long time period involved and the nature of the events and processes of interest, there will inevitably be substantial uncertainties in projecting disposal system performance. Proof of the future performance of a disposal system is not to be had in the ordinary sense of the word in situations that deal with much shorter time frames. Instead, what is required is a reasonable expectation, on the basis of the record before the implementing agency, that compliance with 191.13 (a) will be achieved.

§ 191.14 Assurance requirements.

To provide the confidence needed for long-term compliance with the requirements of 191.13, disposal of spent nuclear fuel or high-level or transuranic wastes shall be conducted in accordance with the following provisions, except that these provisions do not apply to facilities regulated by the Commission (see 10 CFR Part 60 for comparable provisions applicable to facilities regulated by the Commission):

(a) Active institutional controls over disposal sites should be maintained for as long a period of time as is practicable after disposal; however, performance assessments that assess isolation of the wastes from the accessible environment

shall not consider any contributions from active institutional controls for more than 100 years after disposal.

(b) Disposal systems shall be monitored after disposal to detect substantial and detrimental deviations from expected performance. This monitoring shall be done with techniques that do not jeopardize the isolation of the wastes and shall be conducted until there are no significant concerns to be addressed by further monitoring.

(c) Disposal sites shall be designated by the most permanent markers, records, and other passive institutional controls practicable to indicate the dangers of the wastes and their location.

(d) Disposal systems shall use different types of barriers to isolate the wastes from the accessible environment. Both engineered and natural barriers shall be included.

(e) Places where there has been mining for resources, or where there is a reasonable expectation of exploration for scarce or easily accessible resources, or where there is a significant concentration of any material that is not widely available from other sources, should be avoided in selecting disposal sites. Resources to be considered shall include minerals, petroleum or natural gas, valuable geologic formations, and ground waters that are either irreplaceable because there is no reasonable alternative source of drinking water available for substantial populations or that are vital to the preservation of unique and sensitive ecosystems. Such places shall not be used for disposal of the wastes covered by this Part unless the favorable characteristics of such places compensate for their greater likelihood of being disturbed in the future.

(f) Disposal systems shall be selected so that removal of most of the wastes is not precluded for a reasonable period of time after disposal.

§ 191.15 Individual protection requirements.

Disposal systems for spent nuclear fuel or high-level or transuranic radioactive wastes shall be designed to provide a reasonable expectation that, for 10,000 years after disposal, undisturbed performance of the disposal system shall not cause the annual dose equivalent from the disposal system to any member of the public in the accessible environment to exceed 25 millirems to the whole body or 75 millirems to any critical organ. All potential pathways (associated with undisturbed performance) from the disposal system to people shall be

considered, including the assumption that individuals consume 2 liters per day of drinking water from any significant source of ground water outside of the controlled area.

§ 191.16 Ground water protection requirements.

(a) Disposal systems for spent nuclear fuel or high-level or transuranic radioactive wastes shall be designed to provide a reasonable expectation that, for 1,000 years after disposal, undisturbed performance of the disposal system shall not cause the radionuclide concentrations averaged over any year in water withdrawn from any portion of a special source of ground water to exceed:

(1) 5 picocuries per liter of radium-226 and radium-228;

(2) 15 picocuries per liter of alpha-emitting radionuclides (including radium-226 and radium-228 but excluding radon); or

(3) The combined concentrations of radionuclides that emit either beta or gamma radiation that would produce an annual dose equivalent to the total body or any internal organ greater than 4 millirems per year if an individual consumed 2 liters per day of drinking water from such a source of ground water.

(b) If any of the average annual radionuclide concentrations existing in a special source of ground water before construction of the disposal system already exceed the limits in 191.16(a), the disposal system shall be designed to provide a reasonable expectation that, for 1,000 years after disposal, undisturbed performance of the disposal system shall not increase the existing average annual radionuclide concentrations in water withdrawn from that special source of ground water by more than the limits established in 191.16(a).

§ 191.17 Alternative provisions for disposal.

The Administrator may, by rule, substitute for any of the provisions of Subpart B alternative provisions chosen after:

(a) The alternative provisions have been proposed for public comment in the Federal Register together with information describing the costs, risks, and benefits of disposal in accordance with the alternative provisions and the reasons why compliance with the existing provisions of Subpart B appears inappropriate;

(b) A public comment period of at least 90 days has been completed, during which an opportunity for public hearings in affected areas of the country has been provided; and

(c) The public comments received have been fully considered in developing the final version of such alternative provisions.

§ 191.18 Effective date.

The standards in this Subpart shall be effective on September 19, 1985.

Appendix A—Table for Subpart B

TABLE 1.—RELEASE LIMITS FOR CONTAINMENT REQUIREMENTS

(Cumulative releases to the accessible environment for 10,000 years after disposal)

Radionuclide	Release limit per 1,000 MTHM or other unit of waste (see notes) (curies)
Americium-241 or -243	100
Carbon-14	100
Cesium-135 or -137	1,000
Iodine-129	100
Neptunium-237	100
Plutonium-238, -239, -240, or -242	100
Radium-226	100
Strontium-90	1,000
Technetium-99	10,000
Thorium-230 or -232	10
Tin-126	1,000
Uranium-233, -234, -235, -236, or -238	100
Any other alpha-emitting radionuclide with a half-life greater than 20 years	100
Any other radionuclide with a half-life greater than 20 years that does not emit alpha particles	1,000

Application of Table 1

Note 1: Units of Waste. The Release Limits in Table 1 apply to the amount of wastes in any one of the following:

(a) An amount of spent nuclear fuel containing 1,000 metric tons of heavy metal (MTHM) exposed to a burnup between 25,000 megawatt-days per metric ton of heavy metal (MWd/MTHM) and 40,000 MWd/MTHM;

(b) The high-level radioactive wastes generated from reprocessing each 1,000 MTHM exposed to a burnup between 25,000 MWd/MTHM and 40,000 MWd/MTHM;

(c) Each 100,000,000 curies of gamma or beta-emitting radionuclides with half-lives greater than 20 years but less than 100 years (for use as discussed in Note 5 or with materials that are identified by the Commission as high-level radioactive waste in accordance with part B of the definition of high-level waste in the NWSA);

(d) Each 1,000,000 curies of other radionuclides (i.e., gamma or beta-emitters with half-lives greater than 100 years or any alpha-emitters with half-lives greater than 20 years) (for use as discussed in Note 5 or with materials that are identified by the

Commission as high-level radioactive waste in accordance with part B of the definition of high-level waste in the NWSA); or

(e) An amount of transuranic (TRU) wastes containing one million curies of alpha-emitting transuranic radionuclides with half-lives greater than 20 years.

Note 2: Release Limits for Specific Disposal Systems. To develop Release Limits for a particular disposal system, the quantities in Table 1 shall be adjusted for the amount of waste included in the disposal system compared to the various units of waste defined in Note 1. For example:

(a) If a particular disposal system contained the high-level wastes from 50,000 MTHM, the Release Limits for that system would be the quantities in Table 1 multiplied by 50 (50,000 MTHM divided by 1,000 MTHM).

(b) If a particular disposal system contained three million curies of alpha-emitting transuranic wastes, the Release Limits for that system would be the quantities in Table 1 multiplied by three (three million curies divided by one million curies).

(c) If a particular disposal system contained both the high-level wastes from 50,000 MTHM and 5 million curies of alpha-emitting transuranic wastes, the Release Limits for that system would be the quantities in Table 1 multiplied by 55:

$$\frac{50,000 \text{ MTHM}}{1,000 \text{ MTHM}} + \frac{5,000,000 \text{ curies TRU}}{1,000,000 \text{ curies TRU}} = 55$$

Note 3: Adjustments for Reactor Fuels with Different Burnup. For disposal systems containing reactor fuels (or the high-level wastes from reactor fuels) exposed to an average burnup of less than 25,000 MWd/MTHM or greater than 40,000 MWd/MTHM, the units of waste defined in (a) and (b) of Note 1 shall be adjusted. The unit shall be multiplied by the ratio of 30,000 MWd/MTHM divided by the fuel's actual average burnup, except that a value of 5,000 MWd/MTHM may be used when the average fuel burnup is below 5,000 MWd/MTHM and a value of 100,000 MWd/MTHM shall be used when the average fuel burnup is above 100,000 MWd/MTHM. This adjusted unit of waste shall then be used in determining the Release Limits for the disposal system.

For example, if a particular disposal system contained only high-level wastes with an average burnup of 3,000 MWd/MTHM, the unit of waste for that disposal system would be:

$$1,000 \text{ MTHM} \times \frac{(30,000)}{(5,000)} = 6,000 \text{ MTHM}$$

If that disposal system contained the high-level wastes from 60,000 MTHM (with an average burnup of 3,000 MWd/MTHM), then

the Release Limits for that system would be the quantities in Table 1 multiplied by ten:

$$\frac{60,000 \text{ MTHM}}{6,000 \text{ MTHM}} = 10$$

which is the same as:

$$\frac{60,000 \text{ MTHM}}{1,000 \text{ MTHM}} \times \frac{(5,000 \text{ MWd/MTHM})}{(30,000 \text{ MWd/MTHM})} = 10$$

Note 4: Treatment of Fractionated High-Level Wastes. In some cases, a high-level waste stream from reprocessing spent nuclear fuel may have been (or will be) separated into two or more high-level waste components destined for different disposal systems. In such cases, the implementing agency may allocate the Release Limit multiplier (based upon the original MTHM and the average fuel burnup of the high-level waste stream) among the various disposal systems as it chooses, provided that the total Release Limit multiplier used for that waste stream at all of its disposal systems may not exceed the Release Limit multiplier that would be used if the entire waste stream were disposed of in one disposal system.

Note 5: Treatment of Wastes with Poorly Known Burnups or Original MTHM. In some cases, the records associated with particular high-level waste streams may not be adequate to accurately determine the original metric tons of heavy metal in the reactor fuel that created the waste, or to determine the average burnup that the fuel was exposed to. If the uncertainties are such that the original amount of heavy metal or the average fuel burnup for particular high-level waste streams cannot be quantified, the units of waste derived from (a) and (b) of Note 1 shall no longer be used. Instead, the units of waste defined in (c) and (d) of Note 1 shall be used for such high-level waste streams. If the uncertainties in such information allow a range of values to be associated with the original amount of heavy metal or the average fuel burnup, then the calculations described in previous Notes will be conducted using the values that result in the smallest Release Limits, except that the Release Limits need not be smaller than those that would be calculated using the units of waste defined in (c) and (d) of Note 1.

Note 6: Uses of Release Limits to Determine Compliance with 191.13 Once release limits for a particular disposal system have been determined in accordance with Notes 1 through 5, these release limits shall be used to determine compliance with the requirements of 191.13 as follows. In cases where a mixture of radionuclides is projected to be released to the accessible environment, the limiting values shall be determined as follows: For each radionuclide in the mixture, determine the ratio between the cumulative release quantity projected over 10,000 years and the limit for that radionuclide as determined from Table 1 and Notes 1 through 5. The sum of such ratios for all the radionuclides in the mixture may not exceed one with regard to 191.13(a)(1) and may not exceed ten with regard to 191.13(a)(2).

For example, if radionuclides A, B, and C are projected to be released in amounts Q_A , Q_B , and Q_C , and if the applicable Release Limits are RL_A , RL_B , and RL_C , then the cumulative releases over 10,000 years shall be limited so that the following relationship exists:

$$\frac{Q_A}{RL_A} + \frac{Q_B}{RL_B} + \frac{Q_C}{RL_C} < 1$$

Appendix B—Guidance for Implementation of Subpart B

[Note: The supplemental information in this appendix is not an integral part of 40 CFR Part 191. Therefore, the implementing agencies are not bound to follow this guidance. However, it is included because it describes the Agency's assumptions regarding the implementation of Subpart B. This appendix will appear in the Code of Federal Regulations.]

The Agency believes that the implementing agencies must determine compliance with §§ 191.13, 191.15, and 191.16 of Subpart B by evaluating long-term predictions of disposal system performance. Determining compliance with § 191.13 will also involve predicting the likelihood of events and processes that may disturb the disposal system. In making these various predictions, it will be appropriate for the implementing agencies to make use of rather complex computational models, analytical theories, and prevalent expert judgment relevant to the numerical predictions. Substantial uncertainties are likely to be encountered in making these predictions. In fact, sole reliance on these numerical predictions to determine compliance may not be appropriate; the implementing agencies may choose to supplement such predictions with qualitative judgments as well. Because the procedures for determining compliance with Subpart B have not been formulated and tested yet, this appendix to the rule indicates the Agency's assumptions regarding certain issues that may arise when implementing §§ 191.13, 191.15, and 191.16. Most of this guidance applies to any type of disposal system for the wastes covered by this rule. However, several sections apply only to disposal in mined geologic repositories and would be inappropriate for other types of disposal systems.

Consideration of Total Disposal System. When predicting disposal system performance, the Agency assumes that reasonable projections of the protection expected from all of the engineered and natural barriers of a disposal system will be considered. Portions of the disposal system should not be disregarded, even if projected performance is uncertain, except for portions of the system that make negligible contributions to the overall isolation provided by the disposal system.

Scope of Performance Assessments. Section 191.13 requires the implementing agencies to evaluate compliance through performance assessments as defined in § 191.12(q). The Agency assumes that such performance assessments need not consider

categories of events or processes that are estimated to have less than one chance in 10,000 of occurring over 10,000 years. Furthermore, the performance assessments need not evaluate in detail the releases from all events and processes estimated to have a greater likelihood of occurrence. Some of these events and processes may be omitted from the performance assessments if there is a reasonable expectation that the remaining probability distribution of cumulative releases would not be significantly changed by such omissions.

Compliance with Section 191.13. The Agency assumes that, whenever practicable, the implementing agency will assemble all of the results of the performance assessments to determine compliance with § 191.13 into a "complementary cumulative distribution function" that indicates the probability of exceeding various levels of cumulative release. When the uncertainties in parameters are considered in a performance assessment, the effects of the uncertainties considered can be incorporated into a single such distribution function for each disposal system considered. The Agency assumes that a disposal system can be considered to be in compliance with § 191.13 if this single distribution function meets the requirements of § 191.13(a).

Compliance with Sections 191.15 and 191.16. When the uncertainties in undisturbed performance of a disposal system are considered, the implementing agencies need not require that a very large percentage of the range of estimated radiation exposures or radionuclide concentrations fall below limits established in §§ 191.15 and 191.16, respectively. The Agency assumes that compliance can be determined based upon "best estimate" predictions (e.g., the mean or the median of the appropriate distribution, whichever is higher).

Institutional Controls. To comply with § 191.14(a), the implementing agency will assume that none of the active institutional controls prevent or reduce radionuclide releases for more than 100 years after disposal. However, the Federal Government is committed to retaining ownership of all disposal sites for spent nuclear fuel and high-level and transuranic radioactive wastes and will establish appropriate markers and records, consistent with § 191.14(c). The Agency assumes that, as long as such passive institutional controls endure and are understood, they: (1) can be effective in deterring systematic or persistent exploitation of these disposal sites; and (2) can reduce the likelihood of inadvertent, intermittent human intrusion to a degree to be determined by the implementing agency. However, the Agency believes that passive institutional controls can never be assumed to eliminate the chance of inadvertent and intermittent human intrusion into these disposal sites.

Consideration of Inadvertent Human Intrusion into Geologic Repositories. The most speculative potential disruptions of a mined geologic repository are those associated with inadvertent human intrusion. Some types of intrusion would have virtually no effect on a repository's containment of

waste. On the other hand, it is possible to conceive of intrusions (involving widespread societal loss of knowledge regarding radioactive wastes) that could result in major disruptions that no reasonable repository selection or design precautions could alleviate. The Agency believes that the most productive consideration of inadvertent intrusion concerns those realistic possibilities that may be usefully mitigated by repository design, site selection, or use of passive controls (although passive institutional controls should not be assumed to completely rule out the possibility of intrusion). Therefore, inadvertent and intermittent intrusion by exploratory drilling for resources (other than any provided by the disposal system itself) can be the most severe intrusion scenario assumed by the implementing agencies. Furthermore, the implementing agencies can assume that

passive institutional controls or the intruders' own exploratory procedures are adequate for the intruders to soon detect, or be warned of, the incompatibility of the area with their activities.

Frequency and Severity of Inadvertent Human Intrusion into Geologic Repositories. The implementing agencies should consider the effects of each particular disposal system's site, design, and passive institutional controls in judging the likelihood and consequences of such inadvertent exploratory drilling. However, the Agency assumes that the likelihood of such inadvertent and intermittent drilling need not be taken to be greater than 30 boreholes per square kilometer of repository area per 10,000 years for geologic repositories in proximity to sedimentary rock formations, or more than 3 boreholes per square kilometer per 10,000 years for repositories in other geologic

formations. Furthermore, the Agency assumes that the consequences of such inadvertent drilling need not be assumed to be more severe than: (1) Direct release to the land surface of all the ground water in the repository horizon that would promptly flow through the newly created borehole to the surface due to natural lithostatic pressure—or (if pumping would be required to raise water to the surface) release of 200 cubic meters of ground water pumped to the surface if that much water is readily available to be pumped; and (2) creation of a ground water flow path with a permeability typical of a borehole filled by the soil or gravel that would normally settle into an open hole over time—not the permeability of a carefully sealed borehole.

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Appendix C: Scoping Calculations

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Appendix C: Scoping Calculations

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This appendix contains details of scoping calculations that were performed before the Complex and Simple performance assessments (PAs) to provide information about how to model the disposal system. For the remainder of this appendix, the word "formal" is used to mean both the Complex and Simple PAs, which are described in the main text. Note that the data used in the scoping calculations may not always match the data used in the Complex and Simple PAs. This situation arises from the nature of the scoping calculations themselves, i.e., the calculations may have been simplified to focus attention on a particular aspect quickly and/or the results of the calculations may point to a need to use different data in the actual assessment. The details on the scoping calculations are provided here so that the reader can better understand how decisions about the conceptual models and calculational design for the Complex and Simple PAs were made. Most conclusions reached as a result of the scoping calculations are presented in Chapter 8. An exception is a set of scoping-type calculations that was used to design some aspects of the repository. The conclusions from these design calculations are discussed in Chapter 6, "Repository Design."

C.1 Scoping Performance Assessment in Salt Repository

Scoping calculations were run in January 1993 to evaluate the whole block disposal option for graphite spent fuel in the salt repository and the option's potential compliance with EPA's standard, 40 CFR 191. The study looked at graphite spent fuel from the Fort St. Vrain, CO, reactor. As described below, the scoping evaluation uncovered no phenomena or graphite spent fuel characteristic that would preclude this disposal option in either bedded salt or granite with regard to EPA's standard.

The scoping performance assessment for the whole block option in bedded salt modeled the whole system; conditions that differ from the final conceptual model are detailed below.

C.1.1 Disposal System Description

The disposal system consists of the engineered barrier (waste parcel and repository) and the geologic barrier.

Engineered Barrier. The engineered barrier for these calculations consisted of the waste parcel, backfill, and the subsurface facilities (repository) for handling and permanent disposal of the waste parcels. (The host rock is not included in the engineered barrier.)

The total inventory at the potential year of emplacement (2030) was about 2.2 million curies. The calculated radionuclide releases for the scenario in which cuttings are brought to the surface from drilling were based on 34 radionuclides from the graphite spent fuel and decay products in the inventory. Ten long-lived radionuclides with EPA normalized releases greater than 1 for at least 1000 yr were considered in transport calculations (^{241}Am , ^{14}C , ^{237}Np , ^{237}Pu , ^{240}Pu , ^{226}Ra , ^{229}Th , ^{230}Th , ^{233}Th , ^{234}U).

For disposal, one fuel block was emplaced in a container if the effective ^{235}U mass in a fuel block was less than 0.7 kg (and cut into sections, if the fuel block was more than 0.7 kg). This option required about 2370 containers for the Fort St. Vrain reactor graphite spent fuel (which filled only half a panel of an actual repository). The total number of containers of graphite spent fuel, including those from the Peach Bottom reactor, was 3168. However, because only Fort St. Vrain reactor waste was considered in the scoping calculations, the 2370 containers were scaled up to a total of 4406 containers per panel.

The assumed conditions for the engineered barrier that differ or were added to those mentioned in Chapters 4 or 6 are as follows:

General

- No other DOE SNF/HLW in repository.
- Fort St. Vrain graphite spent fuel scaled up to 374 metric tons of heavy metal to fill 10-panel repository.
- Graphite spent fuel has characteristics of only Fort St. Vrain reactor.
- Average radionuclide inventory of a container given by total graphite spent fuel divided by the number of containers required (volume average).

Waste Parcel

- Container is assumed to consist of a single, thick, moderately corrosion-resistant metal.
- Less than 0.7 kg ^{235}U is placed in a container to prevent critical conditions for ^{235}U . (See Section C.3, "Bounding Calculations of Critical Conditions in a Container.")
- Container breach determined by time to failure model with user-supplied parameters.

Repository

- Effects of brine inflow from generating gas by corrosion are bounded by single phase flow results.
- Brine flow to intrusion boreholes in Sodium Springs Formation is modeled as axisymmetric flow in a confined aquifer.

Geologic Barrier. Formations that were assumed to be important to the site were the following: Tower, Sodium Springs Formation, and Cattle River Formation. See Chapter 5 for a detailed description of these formations.

C.1.2 Scenario Development

The event selected for modeling was an inadvertent human intrusion drilling event, which was divided into three summary scenarios: a drilling event into one panel that hits a hypothetical underlying brine reservoir (called E1); a drilling event into one panel that misses a hypothetical underlying brine reservoir (called E2); and a combination of the two (two drilling events into one panel, with one missing the brine reservoir and the other hitting it; called E1E2). The releases from the E1 scenario were assumed to be bound by those of the E2 scenario. Permutations of these three summary scenarios were then used to arrive at other drilling situations (e.g., three hits into the repository: one into a panel missing a brine reservoir, and the other two in another panel).

C.1.3 Consequence Modeling

This section describes the phenomena that were included, and the manner in which they were included, when the selected summary scenario events described above were evaluated.

Release when Drilling into Repository (CUTTINGS). For these scoping calculations in salt, five intrusion times were assumed over the 10,000-yr regulatory period (1000, 3000, 5000, 7000, and 9000 yr). At these times, the contents (i.e., decayed activity) of one entire container was assumed to be transported to the ground surface. The inventory was assumed to be divided equally among 3168 containers to obtain the inventory per container; 4406 of these containers were then assumed in each panel of the salt repository.

The probability of intrusions within the time intervals centered about 1000, 3000, 5000, 7000, and 9000 yr were evaluated by the probability model based on a Poisson distribution (see Section C.2.4).

No conditions in addition to those mentioned in Chapter 10, "Modeling Drilling into Repository," were assumed.

Radionuclide Source Term. For the source-term calculations, IPANEL, a modified version of PANEL (Rechard, Ed., 1992), was used. The code was modified so that instead of assuming that all the inventory of each isotope is immediately available for dissolution, only the inventory within particles with failed SiC coatings, and within breached containers was available for dissolution. The amount of each isotope in solution was limited only by solubility (or by available inventory if the entire inventory could be dissolved). The volume of fluid used in the solubility

calculations is the entire panel pore fluid volume. In effect, the diffusivity of isotopes out of the containers and throughout the repository pore fluid was assumed to be infinite.

Containers were assumed to exist in one of two states: intact or breached (failed). All inventory in breached containers was assumed to be available for transport in the repository fluid, except for inventory within intact SiC-coated particles, and graphite-block C-14 inventory, as discussed below.

The ^{14}C inventory was divided into two categories: the ^{14}C located in the fuel itself and the ^{14}C in the graphite blocks. The ^{14}C in the failed particles, like all the other isotopes, was conservatively assumed to be available immediately upon container breach. The ^{14}C in the graphite of breached containers was made available for release only as fast as the graphite could oxidize to CO_2 . This assumption is consistent with Lotts et al. (1992); however it may result in an underprediction of the release of ^{14}C from the graphite blocks.

A simple distribution of container failure times was adopted, characterized by four parameters: P_0 , the probability that a container has a defect at time zero; r_0 , a lower limit for the corrosion rate of the container material; r_1 , an upper limit to the corrosion rate of the container material; and T , the container material thickness (refer to the following section on Probability Models).

For each time step, (1) the number of breached containers is calculated; (2) radionuclides from particles with failed SiC coatings within the containers breached since the last time step are added to the mass available for dissolution; (3) ^{14}C from oxidation of graphite in breached containers is added to the mass available for dissolution; (4) the inventories are adjusted for decay and build-up since the last time step; (5) the concentration of each isotope dissolved within the panel is calculated based on total volume of water within the pore space of the entire panel, solubility of each element, fraction of isotopes for each element during time step, and amount of each isotope available for dissolution, and (6) the amount of each isotope within the panel was decreased by the amount removed by flow during the time step.

The following conditions for the source term were assumed:

- The repository is filled with brine at the start of the calculations.
- Non-failed SiC coatings never fail. This assumption is based on the observation by Lotts et al. (1992) that SiC particles withstand high temperatures and pressures, and resist oxidation and aqueous corrosion, and that there are no known mechanisms that would cause failure in a repository. The upper limit of the initial failure fraction distribution was set to 100% (with small probability) to represent the operation of unknown mechanisms.
- The entire inventory within the particles with failed SiC coatings is available for dissolution by the liquid within the entire panel upon breach of each container. This assumes that dissolution and transport through any surrounding graphite and failed container, and throughout the pore space of the panel is instantaneous. Given the porosity and permeability of the backfill within the panel, this assumption neglects a very significant delay in the availability of the isotopes for transport out of the panel. (Delay was considered in part in the formal performance assessment by modeling transport from waste-emplacement package holes through the rooms and drifts [see Chapters 13 to 15]; however, transport through a failed waste emplacement package was not modeled [see Chapter 11].)
- Retardation of isotopes by graphite and waste parcel corrosion products is neglected.
- ^{14}C in voids of the waste parcel is immediately available for transport after the container fails.
- ^{14}C present in the graphite is released as the graphite dissolves.
- ^{14}C is released as dissolved CO_2 with essentially infinite solubility since gas flow is not modeled.
- Release from the repository, calculated by IPANEL, is instantaneously injected into the overlying Snake Dolomite brine aquifer.
- All the brine in pores of a panel is available for dissolving radionuclides from breached containers (panel-sized mixing cell).

Fluid Flow. Two fluid flow models were used: a repository-flow model to estimate flow rates through the repository and an aquifer-flow model to estimate flow through the overlying lateral transport pathway in the event of intrusion. The flow rates from each model were uncoupled. Flow rates from the repository-flow model influence corrosion rates and radionuclide concentrations in the repository and therefore the calculations were done directly in the source-term model, IPANEL.

The following conditions for the repository flow were assumed:

- Flow is single phase (no gas generation).
- In the absence of gas accumulation in the repository, flow to an intrusion borehole is limited by its ability to drain pore fluid from the repository backfill. This effect is approximated in the model by representing the backfill as an infinite confined aquifer with an inner radius equal to the intrusion borehole radius.
- The panel and borehole are assumed to rapidly respond to changes in flow rates, such that hydraulic behavior of these components is completely characterized by an effective conductance, which is a function of the component geometry and hydraulic conductivity.
- Fluid flow from the lateral portion of the host rock salt connected to the panel is a linear function of the panel depressurization (e.g., an infinite Darcian median).
- The brine reservoir is finite and has infinite hydraulic conductivity. The reservoir is therefore completely characterized by a bulk storativity (volume of fluid released per unit depressurization) and an initial pressure.
- All other elements of the disposal system (e.g., panel seals, host rock above and below repository) are impermeable.
- Flow rates from the repository model are uncoupled from aquifer flow model.

Radionuclide Transport in Snake Brine Aquifer. The transport behavior of the system is represented by a dual-porosity model. Based on the following simplifications, an analytical 2-D model for a uniform flow field was used to calculate radionuclide releases due to groundwater transport:

- The flow field is assumed to be constant over the 10,000-yr regulatory period. Under the assumption of uniform flow, the Snake Dolomite flow field is completely characterized by pathline distance and travel time, which were derived from the Culebra Dolomite permeability used in the 1992 WIPP performance assessment. Travel distance and time were assumed to be independent.
- Radionuclides move as passive tracers and do not chemically interact with the dolomite matrix or fracture filling material.
- The release due to transport through spatially variable flow fields is approximated by the release due to transport through a uniform flow field representing a stream tube around the actual (tortuous) pathway.
- Diffusion of tracer into the dolomite matrix retards tracer transport by increasing the apparent porosity of the fracture network. The extent of this increase is a function of time, and depends on the tracer residence time, fracture spacing, nuclide diffusion coefficient, and other factors. The potentially complex dynamics of fracture/matrix interaction are ignored. Some portion of the matrix pore space is assumed to immediately come to equilibrium with the fracture concentration. The fraction of available matrix pore space is described by a single uncertainty parameter, f_m .

In addition, the following conditions for transport were assumed:

- Diffusion is assumed to occur instantaneously into some portion of the matrix pore space ($n = n_f + f_m \cdot n_m$, where n_f is fracture porosity, f_m is fraction of available matrix pore space, and n_m is matrix porosity).
- Velocity (U) is constant.

Dagan (1982), citing Carslaw and Jaeger (1959), presents a solution for concentration due to introduction into a uniform flow field of a mass Δm at the origin at time 0:

$$\Delta c_s(\mathbf{x}, t) = \frac{\Delta m}{n} \frac{1}{4\pi(D_I D_{II})^{1/2} t} \exp \left[-\frac{(x_1 - Ut)^2}{4D_I t} - \frac{x_2^2}{4D_{II} t} \right] \quad (C-1)$$

where:

Δc_s	=	concentration (mass/volume) of a stable isotope
$\mathbf{x}=(x_1, x_2)$	=	position
t	=	time since introduction of Δm
n	=	medium porosity
D_x	=	dispersion coefficients
U	=	fluid velocity.

Coordinate x_1 is assumed to be aligned along the flow direction. If the tracer undergoes radioactive decay, and the daughter products are transported in the same way as the parent, then the concentrations of daughter products at any point are given by the solution of the Bateman equations:

$$\Delta c_i(\mathbf{x}, t) = \lambda_1 \lambda_2 \dots \lambda_{i-1} \Delta c_s(\mathbf{x}, t) \sum_{j=1}^i \frac{e^{-\lambda_j t}}{\prod_{k \neq j} (\lambda_k - \lambda_j)} \quad (\text{C-2})$$

The IPANEL discharge, which forms the source-term for the lateral transport model, is given as a series of integrated releases from the borehole at discrete times. The total mass released through the lateral discharge boundary during an interval can be calculated in the following way:

- The mass introduced over each interval is assumed to be introduced in a pulse at the center of the interval.
- Equations C-2 and C-1 are used to derive the integrated release associated with the interval mass.
- The total integrated release is the sum of the release for all source intervals.

This method relies (among other things) on the assumption that the interval size is small relative to the half-life of the tracer, and that the travel time is large relative to interval size. These assumptions seem reasonable given the time step typically used in IPANEL, but have not yet been properly evaluated. Other methods of release calculation can be predicated on Equation C-1 if either of these conditions does not hold.

Assuming that the dispersive flux is negligible for the purpose of calculating integrated release, the mass flux density of nuclide i is:

$$\Delta q_1(\mathbf{x}, t) = nU \Delta c_i(\mathbf{x}, t) \quad (\text{C-3})$$

While the integrated discharge rate across an infinite boundary perpendicular to flow at $x_1 = X_B$ is given by:

$$\begin{aligned} \Delta Q_i(t) &= \int_{-\infty}^{\infty} \Delta q_i(\mathbf{x}, t) dx_2 \Big|_{x_1 = X_B} = nU \int_{-\infty}^{\infty} \Delta c_i(\mathbf{x}, t) dx_2 \Big|_{x_1 = X_B} \\ &= \lambda_1 \lambda_2 \dots \lambda_{i-1} \Delta Q_s(\mathbf{x}, t) \sum_{j=1}^i \frac{e^{-\lambda_j t}}{\prod_{k \neq j} (\lambda_k - \lambda_j)} \end{aligned} \quad (\text{C-4})$$

where,

$$\Delta Q_s(t) = \int_{-\infty}^{\infty} \Delta c_s(\mathbf{x}, t) dx_2 \Big|_{x_1 = X_B} \frac{\Delta m U}{2(\pi D_I t)^{1/2}} \exp \left[-\frac{(X_B - Ut)^2}{4 D_I t} \right]$$

The total amount of each nuclide released due to Δm is given by:

$$\Delta R_i(t) = \int_0^t \Delta Q_i(\tau) d\tau$$

Taking note of the time dependencies in Equation C-4, $\Delta R_i(t)$ is evidently a linear combination of the form,

$$\Delta R_i(\mathbf{x}, t) = \lambda_1 \lambda_2 \dots \lambda_{i-1} \sum_{j=1}^i \frac{\Delta D(\lambda_i, f)}{\prod_{k \neq j} (\lambda_k - \lambda_j)}$$

where,

$$\begin{aligned} \Delta D(\lambda, t) &= \int_0^t e^{-\lambda \tau} \Delta Q_s(\tau) d\tau \\ &= \frac{\Delta m U}{2} \int_0^t \frac{e^{-\lambda \tau}}{(\pi D_I \tau)^{1/2}} \exp \left[-\frac{(X_B U \tau)^2}{4 D_I \tau} \right] \\ &= \frac{\Delta m}{2\beta} \left(e^{\frac{X_D}{2}(1-\beta)} \operatorname{erfc} \left[\frac{1}{2} \sqrt{X_D X_A} \left(1 - \frac{\beta}{X_A} \right) \right] \right. \\ &\quad \left. - e^{\frac{X_D}{2}(1-\beta)} \operatorname{erfc} \left[\frac{1}{2} \sqrt{X_D X_A} \left(1 + \frac{\beta}{X_A} \right) \right] \right) \end{aligned} \quad (C-5)$$

with,

$$\begin{aligned} X_D &= \frac{X_B}{\alpha_I} \\ X_A &= \frac{X_B}{U t} \\ \beta &= \sqrt{1 + 4 \frac{\lambda \alpha_I}{U}} \end{aligned} \quad (C-6)$$

C.1.4 Probability Modeling

Besides the subjective probability models for varied parameters, two probability models were included in these scoping calculations: the estimation of the event of human intrusion into the bedded salt repository and the time to failure of containers.

Model Parameters Selected for Sampling. Twenty-three model parameters were varied (and sampled 100 times). The parameters represent characteristics of the graphite spent fuel, geologic barrier, and engineered barrier and agents acting on the disposal system (Table C-1). The other model parameters and dimensions were held constant (Table C-2).

Probability of Human Intrusion. The estimation of human intrusion was the same as reported for the overall calculations and was based on a Poisson distribution; however, the drilling rate was varied:

- Drilling rate was assumed to vary uniformly between 0 and $\lambda_{\max}^*$ (see Chapter 9 "Probability Modeling").

Table C-1. Distributions of Parameters Varied in Scoping PA for Salt Repository*

Parameter**	Median	Range		Units	Distribution Type*
Engineered Barrier					
Waste Parcel					
Whole Block Graphite Waste Form					
SNF form 1 (base case)					
Fraction of fuel coatings failed (f_{graphite})	1.6×10^{-3}	1×10^{-3}	1.0	none	Constructed
Radionuclide Solubility					
• Am (log S_{AM})	-9.00	-13.3	0.0	log (molar)	Constructed
• Pu (log (S_{Pu}))	-9.22	-16.5	-3.24	log (molar)	Constructed
• Th (log (S_{Th}))	-10.0	-15.2	-5.6	log (molar)	Constructed
• U (Log (S_{U}))	-3.27	-15.0	0.0	log (molar)	Constructed
3-Layer Container					
• Fraction of initially defective container (f_{defect})	0.1	0	0.3	none	Constructed
• Average container corrosion rate (@90°C) ($\bar{r}_{\text{can}}$)	2×10^{-6}	1×10^{-8}	1×10^{-5}	m/yr	Constructed
• Log of variability of container corrosion rate	3×10^{-1}	1×10^{-1}	1	none	Uniform
Repository					
Drift Backfill					
• Permeability (k)	1.93×10^{-21}	1×10^{-22}	1×10^{-15}	m ²	Constructed
Geologic Barrier					
Halite within Sodium Spring Formation					
• Log of permeability (log k), undisturbed	-21.2	-24.0	-19.0	log (m ²)	Constructed
• Pore pressure (p)	13.0	8.0	18.0	MPa	Uniform
Tower Formation Brine Reservoir					
• Initial pressure (p _i)	12.6	11.0	21.0	MPa	Constructed
• Storativity, bulk (S_{bulk})	0.2	0.02	2.0	m ³ /Pa	Lognormal
Snake Dolomite Member in Cattle River Formation					
• Fracture spacing (2(B+b)***)	0.4	0.06	8.0	m	Constructed
• Fracture porosity (ϕ_f)	1×10^{-3}	1×10^{-4}	1×10^{-2}	none	Lognormal
• Matrix porosity (ϕ_m)	0.139	0.096	0.208	none	Constructed
• Index for Snake Dolomite transmissivity fields [†]	0.5	0	1	none	Uniform
• Pathline distance to discharge boundary (X_B)	5662	5260	7398	m	Constructed
• Travel time to discharge boundary at 16% porosity [†] (Δt_x)	8.42×10^{11}	4.52×10^{11}	3.032×10^{12}	s	Constructed
• Fraction of matrix pore space available*** (f_ϕ)	0.1	0.01	1	none	Loguniform

* For the scoping calculations, we did not attempt to defend the validity of the distribution types or values. Instead, this defense is provided in the formal performance assessment, when similar distribution types and values were used.

** Flow parameters and boundary conditions influencing transmissivity fields not listed (see Chapter 13, instead).

*** These two parameters are -100% correlated

[†] These two parameters are 100% correlated

Table C-1. Distributions of Parameters Varied in Scoping PA for Salt Repository* (Continued)

Parameter**	Median	Range	Units	Distribution Type	
Agents Acting on Disposal System					
Human Intrusion Probability Model					
• Drilling rate (λ)	1.5×10^{-9}	0	3.0×10^{-9}	$1/\text{m}^2 \text{ yr}$	Uniform
• Fraction of overlap of brine reservoir (f_{Area})	0.45	0.25	0.62	none	Constructed
• Permeability of borehole from human intrusion (k_{bore})	3.16×10^{-12}	1×10^{-14}	1×10^{-11}	m^2	Lognormal

* For the scoping calculations, we did not attempt to defend the validity of the distribution types or values. Instead, this defense is provided in the formal performance assessment, when similar distribution types and values were used.

** Flow parameters and boundary conditions influencing transmissivity fields not listed.

Table C-2. Model Parameters Held Constant in Scoping PA for Salt Repository*

Parameter	Symbol	Median	Units
Waste Parcel Characteristics			
• Oxidation rate of graphite block	r_{O_2}	1.7×10^{-10}	m/yr
• Fraction ^{14}C located in block	f_{C}	0.93	
• Solubility of C	S_{C}	100^2 (infinite)	molar
• Solubility of Np	S_{Np}	1.0×10^{-7}	molar
• Solubility of Ra	S_{Ra}	11	molar
Geologic Barrier			
• Dispersivity of Snake Aquifer	α	50	m
Engineered Barrier			
• Backfill porosity	ϕ_{drill}	6×10^{-2}	
• Containers in a panel	N_{panel}	4406	
• Containers of Graphite SNF	N	3168	
• Container thickness		1.9×10^{-2}	m
• Final drift height		2	m
• Panel area	A	2.592×10^4	m^2
• Specific storage	S	1×10^{-6}	m^{-1}
Global Parameters			
Water			
• Viscosity	μ_{w}	1.6×10^{-3}	Pa/s
• Density	ρ_{w}	1.01×10^3	kg/m^3
• Molecular weight	M_{w}	1.8×10^{-2}	kg/gmole
Repository			
• Pore volume		1.82×10^5	m^3
• Temperature	T	300	K

* For the scoping calculations, we did not attempt to defend the validity of the distribution types or values. Instead, this defense is provided in the formal performance assessment, when similar distribution types and values were used.

Time to Failure of Containers. This model assumes N containers in a panel. Each container has probability p_0 of being defective at closure time ($t=0$) and each corrodes at a constant rate R (a random variable). A randomly selected container is assumed failed at time > 0 if and only if

- (a) Container is defective at closure, or
- (b) Container is not defective at closure and $Rt \geq T$, where T is the initial thickness of the container wall.

For these scoping calculations:

- The expected value of n ($=NP$) is an estimate of the number of containers that are defective at time $t > 0$; N , P_0 , r_1 , r_0 are imprecisely known parameters, and T is precisely known.
- The corrosion rate is constant.
- The corrosion rate is the rate of an unnamed but moderately corrosion-resistant metal.

C.1.5 Results

The results of the scoping calculations are presented as a composite CCDF and as conditional CCDFs. The conditional CCDFs include release from the repository due to groundwater flow for single instances of the two intrusion scenarios E2 and E1E2. The conditional CCDFs were used to assess the variability of release with the intrusion scenario. Integrated releases were calculated both at the accessible environment boundary and at the repository.

Composite CCDF. The composite CCDF incorporates the likelihood of occurrence of all combinations of the basic scenarios, using the repository geometry and drilling frequency. Figure C-1 shows the composite mean CCDF of total integrated release to the accessible environment due to all mechanisms and scenarios considered. The composite CCDF includes the effects of multiple instances of the fundamental intrusion scenarios E2 and E1E2 occurring in a 10,000-yr period as well as the undisturbed (no intrusion) scenario. The probability associated with a given number of intrusions is a function of future drilling frequency, while the consequence is simply the sum of the releases for the individual events.

Figure C-1. Composite mean CCDF of total integrated release to the accessible environment due to all mechanisms and all basic intrusion scenarios considered.

The mean CCDF curve lies to the left of the curve representing the EPA containment requirement. At all levels of probability, the predicted release is at least an order of magnitude lower than the containment requirement release.

Parameter importance for the composite CCDF is identified as follows: for a given release level, the probability of release exceeding that level is identified for each vector; partial correlation coefficients are then calculated between the exceedance probability levels and all sampled parameters; parameters with large correlations to exceedance probability have the most influence on the composite CCDF. Parameters influencing the CCDF are listed in Table C-3 in order of importance.

Table C-3. Three Most Important Modeling Parameters Influencing the EPA Summed Normalized Releases

Parameter	Rank*
Agents Acting on Disposal System	
Drilling frequency	1
Geologic Barrier	
Fraction of Snake Dolomite matrix pore space available for radionuclide diffusion	2
Waste Parcel	
Fraction of graphite spent fuel particles that are initially failed	3
* 1 is most important	

Conditional CCDFs. The conditional CCDFs describe aquifer transport release, contribution of drill cuttings to release, and repository release conditional on the fact that an intrusion has occurred. (Consequently, the drilling frequency parameter is removed from consideration.)

Aquifer Transport. Figure C-2 shows the CCDFs without multiple intrusions release for the E2 and E1E2 scenarios conditional on the fact that intrusions do not penetrate containers. The major contributing nuclides in both scenarios are ^{233}U , ^{229}Th , and ^{234}U . The maximum integrated release is approximately 0.5, while the median for both scenarios is less than 10^{-7} .

The higher integrated releases (Figure C-2) are approximately the same as the releases directly from the repository, as discussed below, indicating relatively little attenuation between the intrusion borehole and the accessible environment boundary. This result is due to the small sampled value for the fraction of matrix pore volume, which approximates dual-porosity transport. High groundwater releases correspond to small available matrix pore volumes. For larger sampled fractions of available matrix pore space, releases are greatly attenuated by lateral transport. If all matrix pore space is available for diffusion during transport, nearly all radionuclides released at the intrusion borehole remain within the regulatory boundary for the entire regulatory lifetime of the repository, resulting in a very small integrated release.

The stepwise regression identified six important parameters that influenced the predicted release (Table C-4).

The fourth, fifth, and sixth parameters in the E2 scenario (Table C-5) are important for total release because they tend to control repository release. The first, second, and third collectively determine the mean travel time to the accessible environment boundary. Travel times exceeding 10,000 yr preclude significant release, while travel times much smaller than 10,000 yr result in discharge of nearly all nuclides introduced through the intrusion borehole.

Release from Drill Cuttings/Cavings. For the median borehole diameter, approximately 10% of drilling intrusions into the repository would intersect a waste container. In constructing the composite CCDF, 10% of drilling intrusions were assumed to release the entire contents of a container to the accessible environment, in addition to any releases due to groundwater flow. The integrated release per breached container is 3×10^{-2} ; the principal contributing

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Figure C-2. CCDFs without multiple intrusions release for the E2 and E1E2 scenarios conditional on the fact that intrusions do not penetrate containers. (See Section C.1.2 for discussion of scenarios for salt disposal system.)

Table C-4. Six Most Important Modeling Parameters Considering only EPA Summed Normalized Releases from Groundwater Transport in E2 and E1E2 Scenarios (No Drilling Cuttings/Cavings)

Parameter	Importance Rank* of Parameter Scenario	
	E2	E1E2**
Agents Acting on Disposal System		
Permeability of intrusion borehole	5	--
Geologic Barrier		
Fraction of Snake Dolomite matrix pore space available for radionuclide diffusion	1	1
Travel time to 5-km boundary	2	2
Matrix porosity	3	3
Far-field pore pressure of salt formation	6	--
Waste Parcel		
Fraction of graphite spent fuel particles with initially failed SiC coatings	4	4

* 1 is the most important.

** First six parameters are listed unless fewer than six parameters entered into the regression model from STEPWISE (see Appendix G).

nuclides are ^{233}U , ^{241}Am , and ^{234}U . Figure C-3 shows the CCDF of release conditional on the fact that the intrusions penetrate a container. Direct removal has a significant impact on minimum release, increasing the minimum value from nearly 0 to approximately 3×10^{-2} .

Release at Boundary of Panel. Figure C-4 shows the conditional CCDF of fluid-borne release directly from a panel for the E2 and E1E2 scenarios. Release from cuttings is not included. The maximum normalized release is approximately 0.5, indicating that, for an unfavorable combination of parameters, a significant fraction of the inventory of a panel is released. The median release is approximately 4×10^{-4} and 6×10^{-4} for the E2 and E1E2 scenarios, respectively. Integrated release tends to be dominated by ^{233}U , ^{229}Th , and ^{234}U . ^{14}C did not contribute to integrated release.

The similarity of releases for the two scenarios is in striking contrast with results from the performance assessment of the WIPP site. Preliminary performance assessment calculations for the WIPP show much larger releases in the E1E2 scenario than in the E2 scenario (WIPP PA Department, 1992). The permeability of the backfill in the salt repository is much lower than the expected permeability of WIPP TRU waste placed in the drifts, resulting in much lower flow rates between boreholes in the same panel for the spent fuel and high-level waste repository.

The stepwise regression of the rank transformation of the integrated release against the sampled parameter identified six important parameters whose uncertainty controls the uncertainty in predicted fluid-borne release from the repository (Table C-5).

The fraction of particles with failed SiC coatings, the fraction of initially failed containers, and the corrosion rate determine the concentration of radionuclides in the repository pore fluid. The backfill permeability, far-field salt pressure, and borehole permeability control the amount of fluid that flows through the repository in the event of intrusion.

The fraction of particles with failed SiC coatings is either the first or second most important uncertainty parameter for the source term. The ten vectors with a fraction of failed coatings greater than 0.005 have total releases significantly above the mean of 4×10^{-4} . If the Technology Development Program can demonstrate with a high degree of certainty that there are no mechanisms (other than the mechanical grinding of the particles as might occur during

Figure C-3. CCDF of total release (groundwater and cuttings) conditional on the fact that the intrusions penetrate a container.

Figure C-4. Conditional CCDF of release directly from a panel for the E2 and E1E2 scenarios without cuttings.

Table C-5. Six Most Important Modeling Parameters for Determining Radionuclides Leaving Panel in E2 and E1E2 Scenarios without Cuttings

Parameter	Importance Rank* at Parameter Scenario	
	E2	E1E2
Agents Acting on Disposal System		
Permeability of intrusion borehole	4	4
Geologic Barrier		
Far-field pore pressure of salt formation	3	3
Repository		
Consolidated permeability of crushed backfill	2	1
Waste Parcel		
Fraction of particles with failed SiC coatings	1	2
Fraction of initially defective waste parcel	5	6
Average corrosion rate of waste parcel	6	5
* 1 is the most important		

drilling through a graphite block) for the breach of the SiC coatings, then the repository could be redesigned to reduce cost. In particular, the criticality considerations are quite different when the mode of concentration the ^{235}U is not dissolution and reprecipitation, but mobilization and concentration of intact thorium/uranium carbide particles. A calculation of the minimum sphere of intact fissile particles that is needed for criticality might show that more than one graphite block could be packaged in a single container. Container size, waste-parcel spacing, and repository size could then be optimized to reduce cost while still complying with 40 CFR 191.

Another important uncertain parameter is the fraction of containers that have failed prior to closure of the repository. If good nondestructive leak testing methods are developed and used, the fraction of defective containers might be reduced to less than 1%. Reduction in the number of early failures is more important than the reduction of failure at later times because it reduces the amount of radionuclides that have a long time to travel out of the repository within the 10,000 yr.

We re-ran the calculations with 100% initially failed containers to judge the possibility of compliance without any canister. The resulting CCDFs of the summed normalized releases from the panel are compared with those previously calculated in Figure C-5. While there is an increase in the release, it remains well within the regulatory limits. Full analysis with releases due to cuttings also shows that compliance is maintained. Thus while a container is necessary for the transport and handling of the graphite spent fuel, it is not necessary for compliance with 40 CFR 191 within the modeled salt repository.

C.2 Scoping Performance Assessment for the Granite Repository

In April 1993, a scoping performance assessment evaluated disposal of the graphite blocks from the Fort St. Vrain reactor in a granite repository. The modeling system was modified to include two-phase flow (liquid and gas). Other differences from the scoping performance assessment done for the salt repository are highlighted below.

Figure C-5. Comparison of conditional CCDFs for E2 and E1E2 scenarios with and without containers.

C.2.1 Disposal System Description

The granite disposal system's engineered barrier (waste form [whole block graphite], container [moderately corrosion-resistant material], and repository design [drift and panel]) were similar to the engineered barrier for the salt repository. The geologic barrier for the granite repository is described below.

The granite repository was assumed to be located entirely within an intact block of granite. Localized zones of high fracture density above and below the repository provided the only lateral pathways to the accessible environment (see Figure 5-2, Chapter 5). Repository access shaft(s) were assumed to pass through an overlying fracture zone.

Based on a cursory examination of granite site characterization data from Canada, permeability ranges were set for the fracture zones above and below the granite repository. The assigned permeability range resulted in very slow travel times to the accessible environment (5 km from the repository) (Figure C-6). Travel times through the granite fracture zone were calculated from the lumped parameters used to describe the fracture zones:

$$T_t = \frac{L\phi}{KI}$$

where

- $L=2.4$ km = the distance to the accessible environment boundary
- ϕ = effective porosity of the fracture zone
- K = hydraulic conductivity of the fracture zone
- I = lateral hydraulic gradient in the fracture zone.

The porosity, hydraulic conductivity, and hydraulic gradient were sampled from the distributions described in Section C.2.4 (see Table C-6).

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Figure C-6. Comparison of cumulative travel times distribution in Snake Dolomite aquifer above salt repository and fracture zone above granite repository.

Travel times through the Snake Dolomite were calculated from an assumed distribution of travel times (T_{t0}) at a constant porosity (ϕ_0), the total matrix porosity (ϕ_m), and the fraction of the matrix available to tracers carried in the fracture (f_m):

$$T_t = T_{t0} \frac{\phi_m f_m}{\phi_0}$$

Distributions for these parameters are given in Table C-1 (see also Sandia WIPP Project, 1992, Figure 2.6-8). Travel time from the repository to the fracture zone was assumed to be zero.

The Monte Carlo simulation of the slow travel times ensured that most of the graphite spent fuel would have no significant release after 10,000 yr. Thus, graphite spent fuel would meet the containment requirements of 40 CFR 191. Although this information proved reassuring, the intent of the assessment was to discover the margin of safety provided in the existing waste forms. Hence, the focus of the scoping PA was to model the release out of the repository.

C.2.2 Scenario Development

Because the massive gray granite was assumed impermeable, release from the repository could occur only after connection of one or more fracture zones at one or more locations. Like the salt repository, the selected basic event was human intrusion through exploratory drilling. Other fracture connections were also considered plausible, e.g., fracturing of the host granite around the repository from mechanical or thermal stresses (see Figure 7-6, Chapter 7), or failure of a shaft seal.

Combinations of these events were grouped together to describe particular model situations (scenarios) of interest. These release scenarios were divided into two groups: a single connection between the repository and a fracture zone and multiple connections between the repository and fracture zone(s). Continuous circulation of water through the repository was possible with multiple connections, while a single connection had to act both as a conduit to fill the repository with water, and later to discharge contaminated water or generated gas, if release was to occur.

C.2.3 Consequence Modeling

This section describes the phenomena that were included, and the manner in which they were included, when evaluating the selected summary scenario events. Like the consequence modeling for the scoping PA of the salt repository, the consequence modeling for the granite repository divided the modeling system into two categories: (1) repository and fluid flow, and (2) contaminant transport. The major change from the scoping PA in the salt was the mathematical model of fluid flow (e.g., the same mixing cell and source-term model, IPANEL, was used). This model, which is briefly discussed here, served as the basis for the flow models in the formal Simple PA and thus is more thoroughly described in Chapter 15, "Simplified Consequence Modeling for the Granite Repository."

Fluid Flow. The representation of flow through the system uses a single set of lumped parameters to describe the hydrologic properties of the primary features (conceptual model) (Figure C-7). As outlined in Chapter 5, all elements of the disposal system were treated as porous media obeying Darcy's law, with uncertain lumped parameters describing flow and transport behavior (e.g., hydraulic conductivity and effective porosity). The porous medium approximation was not expected to be valid for any particular values of pore geometry. The intention was to assign ranges to these parameters that cover the range of possible releases resulting from a more realistic model. For example, the effective porosity for the porous medium was estimated by considering ranges of possible total fracture porosity, connectivity, and flow channeling within individual fractures.

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Figure C-7. Lumped parameter representation of flow through granite disposal system.

Flow rate through any component (e.g., fracture zone) of the system is in general a function of conductive properties of the component. In the fracture zones, fracture frequency, extent aperture, and connectivity, as well as the degree of flow channeling within individual fractures, combined to determine lumped hydraulic and transport properties of the component. The range in possible flow rates was large because of the underlying uncertainty in fracture parameters. A similar uncertainty exists in characterizing the flow through failed shaft seals because the mode and degree of failure were uncertain.

The flow model used in the simplified scoping calculations is the same as that used in the simplified formal PA (described in Chapter 15) with the exception of the gas generation model.

In the scoping calculations, the gas generation rate was assumed to be a function of the initial exposed metal area and a sampled generalized corrosion rate:

$$R_g = S_w A_m$$

Gas generation was included in the scoping calculations to explore the potential for repository-generated gas to displace contaminated liquid.

The repository is initially devoid of liquid. The repository resaturates through connections to the upper and lower fracture zones. The time of resaturation is primarily determined by the conductive properties of these connections.

The conductance of the connections with respect to the liquid or gas phase is assumed to be a function of the repository saturation, with respect to that phase, as discussed in Chapter 15.

Contaminant Transport. Radionuclide release from the repository is a function of the radionuclide release rate from the waste parcels in the repository and the transport characteristics of the shaft and fracture zones. Waste parcel release as estimated by the mixing cell source-term uses, in general, a function of liquid flow rate into the mixing cell (which removes dissolved species), and therefore of flow rate through the repository.

Only transport of radionuclides dissolved in the liquid phase were considered. Concentrations were assumed to be sufficiently small to allow the transport problem to be decoupled from the flow problem. Once flow rates were

estimated in the repository, integrated release due to transport into Fracture Zone 2 (Figure C-7) was estimated using the following assumptions in addition to or in place of those in the scoping calculations for the salt repository:

- Effectively infinite diffusivity of radionuclides exists in the repository pore fluid. In effect, the entire repository is assumed to be a mixing cell, with the total amount of available radionuclides being a function of the number of failed canisters and the number of failed SiC coatings.
- Transport time through the failed seal and connections is negligible.
- Transport in the upper fracture zone is approximated by a one-dimensional advection-dispersion equation using a uniform flow field. As discussed in Section 15.1.1, the porous medium approximation is not expected to be valid for any particular value of pore velocity and dispersivity. The intention is to assign ranges to these parameters that cover the range of possible releases resulting from a more realistic model. For example, the effective porosity of the porous medium may be estimated by considering ranges of possible total fracture porosity, connectivity, and flow channeling within individual fractures.
- Chemical retardation and matrix diffusion are not included in transport. (However, chemical retardation in the backfill buffer is included in CONCSNF.)

Other Modeling Systems. Modeling systems for evaluating consequences of cuttings release and radionuclide source term were the same as those used for the scoping PA for the salt repository.

C.2.4 Probability Modeling

Model Parameters Selected for Sampling. One hundred samples of 17 model parameters were generated. Ten of the model parameters were identical to those varied in the scoping calculation of the salt repository (Table C-6). The differences in number and type of model parameters sampled were primarily because of the change in geologic media. However, one was added because the transport model for the scoping calculation in granite could handle gas phase transport, and so the gas generation rate was varied (treatment of gas generation is described in Chapters 14 and 15); another one was added because borehole diameter could affect the flow rates through the repository between the two fracture zones. Parameters held constant are listed in Table C-7.

Probability of Human Intrusion. No probability modeling was done for the granite repository. The probability of a connection to the fracture zone through an intrusion borehole could have been modeled as in the scoping PA for the salt repository, except that the drilling rate would have dropped an order of magnitude from 30 boreholes/km² to 3 boreholes/km², as permitted by Appendix B of 40 CFR 191. However, because consequence models were not used for examining the two other means of forming secondary connections, (1) the formation of fractures from mechanical or thermal stresses, or (2) the failure of seals in the shafts, these phenomena would normally need to be handled by some type of probability model, possibly subjective estimates that these connections would form. Defensible probability models (or consequence models) for these later connections were not possible in the time allowed for this scoping calculation (or for the final simple or Complex PA) and thus were not attempted. Rather, CCDFs for releases at the repository were produced that were conditional on the fact that some type of connection occurred. (Recall that because of the permeabilities assumed for the two fracture zones and resulting groundwater travel times, based on data from the Canadian repository site, no release could occur at a 5-km boundary.) As shown in Figure C-8, assuming that the connections occurred, the movement of fluid through an intrusion borehole always bounds the movement of fluid through a failed shaft seal or secondary fracture.

C.2.5 Results

As noted earlier, cumulative radionuclide releases at the repository boundary from the mixing cell, source-term model, IPANEL, and at the accessible environment (5-km) boundary were calculated for 9 scenarios (see Figure C-8). The cumulative radionuclide activity from the source term model of the whole block graphite spent nuclear fuel was significant for scenarios 6, 8, and 9, with the cumulative activity from scenarios 2, 6, and 8 quite similar and scenario 9 somewhat less (Figure C-9). Because no probability modeling was done for this scoping PA in granite, comparisons with the EPA containment requirements or the complete CCDFs for the scoping PA in salt are inappropriate.

Table C-6. Distributions of Parameters Varied in Scoping PA for Granite Repository*

Parameter	Median	Range		Units	Distribution Type*
Engineered Barrier					
Waste Parcel					
Whole Block Graphite Waste Form					
SNF form 1 (base case)					
• Fraction of fuel coatings failed (f_{graphite})	1.6×10^{-3}	1×10^{-3}	5×10^{-3} (90%)	1 (100%)	Constructed
Radionuclide Solubility					
• Am (log S_{AM})	-9.00	-13.3	0.15	log (molar)	Constructed
• Pu (log (S_{Pu}))	-9.22	-16.6	-3.26	log (molar)	Constructed
• Th (log (S_{Th}))	-10.0	-15.2	-5.6	log (molar)	Constructed
• U (Log (S_{U}))	-3.27	-15.0	0.0	log (molar)	Constructed
Gas Generation					
• Rate (100%) (S_w)	-10.5	-13.0	-8.0	log (mole/m ² /s)	Uniform
• Saturation threshold	0.2	0.0	0.4	none	Uniform
Waste Container					
• Fraction of initially defective container	1.0×10^{-1}	0	3×10^{-1}	none	Constructed
• Average container corrosion rate (@90°C) (F_{can})**	2×10^{-7}	6×10^{-10}	6×10^{-5}	m/yr	Constructed
• Log of variability of container corrosion rate	3×10^{-1}	1×10^{-1}	1	none	Uniform
Geologic Barrier					
Fracture Zone					
• Transmissivity (log(Tr))	-8.5	-10.0	-7.0	log (m ² /s)	Uniform
• Porosity (ϕ)	5.71×10^{-3}	4.32×10^{-3}	7.1×10^{-3}		Uniform
• Lateral hydraulic gradient (l)	9×10^{-3}	2×10^{-3}	1.6×10^{-2}		Uniform
• Hydraulic head change over 10,000 yr (ΔH)	10	0	20	m	
Secondary Fracture					
• Flow rate maximum (q_{max})	4×10^{-9}	4×10^{-10}	2×10^{-7}	m ³ /s	Constructed
Agents Acting on Disposal System					
Human Intrusion Probability Model					
• Permeability of borehole from human intrusion (k_{bore})	3.16×10^{-12}	1×10^{-14}	1×10^{-11}	m ²	Lognormal
• Diameter of borehole	0.3556	0.2667	0.4145	m	Uniform

* For the scoping calculations, we did not attempt to defend the validity of the distribution types or values. Instead, this defense is provided in the formal performance assessment, when similar distribution types and values were used.

** 100% correlation

Even comparisons with the conditional CCDFs for the scoping PA in salt are dangerous since some important initial conditions changed. For example, the repository was assumed filled with liquid in the salt, but assumed empty in granite at the start of the calculations. Qualitatively, however, the cumulative, summed normalized activity leaving the repository is less in the salt media (compare Figures C-4 and C-9).

Because the groundwater travel time to the 5-km disposal system boundary is so long (see Figure C-6), all the potential releases for these four scenarios are far below the containment requirements in the granite (the containment requirements are not shown); furthermore, only the releases from scenarios 2 and 6 are easily plotted (Figure C-10).

Table C-7. Model Parameters Held Constant in Scoping PA for Granite Repository*

Parameter		Median	Units
Waste Parcel Characteristics			
• Oxidation rate of graphite block	R_{O_2}	1.7×10^{-10}	m/yr
• Fraction ^{14}C located in block	f_c	0.93	
• Solubility of CO_2	S_c	100^2 (infinite)	molar
• Solubility of Np	S_{Np}	1.0×10^{-7}	molar
• Solubility of Ra	S_{Ra}	11	molar
Geologic Barrier			
• Dispersivity of fracture zone	α	50	m
Engineered Barrier			
• Containers in a panel	N_{panel}	4406	
• Containers of Graphite SNF	N	3168	
• Container thickness		1.9×10^{-2}	m
• Final drift height		6	m
• Panel area	A	2.592×10^4	m^2
• Specific storage	S	1×10^{-6}	m^{-1}
• Surface area container	A_s	8.1×10^4	m^2
Global Parameters			
Water			
• Viscosity	μ_w	1.6×10^{-3}	Pa/s
• Density	ρ_w	1.01×10^3	kg/m^3
• Molecular weight	M_w	1.8×10^{-2}	kg/gmole
Gas (H_2)			
• Viscosity	μ_g	1.58×10^{-5}	Pa/s
• Molecular weight	M_g	2.0×10^{-3}	kg/mole
Repository			
• Pore volume		1.82×10^5	m^3
• Temperature	T	300	K

* For the scoping calculations, we did not attempt to defend the validity of the distribution types or values. Instead, this defense is provided in the formal performance assessment, when similar distribution types and values were used.

The radionuclides that are dominating the releases are the same as in the salt repository. For scenario 2, the principal contributing nuclides to cumulative activity leaving the repository are ^{233}U , ^{234}U , and ^{229}Th . Figure C-11 shows the contribution as summed normalized activity, while Figure C-12 shows the contribution in percent of total normalized activity.

The model parameters with variation that most influenced the variability in the summed normalized activity are presented in Table C-8. The model parameters representing agents acting on the disposal system were most important (i.e., the intrusion borehole permeability and diameter [the latter was not sampled for the scoping PA in salt]), whereas the salt repository tended to have model parameters of the waste parcel near the top (though the sensitivity analysis of the granite repository still identified model parameters of the waste parcel as important).

The rankings of parameter importance for the four scenarios are listed in Table C-8. The ranks are conditional on the fact that a connection to the fracture zone has occurred. Hence, the drilling frequency parameter and the probability of a fracture forming are removed from consideration.

Scenario	Description		Flow	Source	5 km
Number			Rates	Release	Groundwater
1	Fracture Connection to Upper and Lower Zones		Non-Zero Gas Release for 1 Vector	N/A	0
2	Borehole Connection to Upper and Lower Zones		Release	Maximum EPA Release = 30	Maximum EPA Release = 1×10^{-8}
3	Fracture Connection to Upper Zone Only		No-Fluid Release	NA	0
4	Fracture Connection to Lower Zone Only		No-Fluid Release	NA	0

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Figure C-8. Results for scenarios 1 through 9. Significant source-term releases were found for scenarios 2, 6, 8, and 9. Groundwater release was significant for scenarios 2, 6, and 8.

Scenario Number	Description		Flow Rates	Source Release	5 km Groundwater Release
5	Fracture Connection to Both Zones; Preferential Flow of Liquid Through Connections		No-Fluid Release	N/A	0
6	Borehole Connection to Both Zones; Preferential Flow of Liquid Through Connections		Release	Maximum EPA Release = 50	Maximum EPA Release = 2×10^{-4}
7	Fracture Connection to Upper Zone; Preferential Flow of Liquid Through Connections		No-Fluid Release	N/A	0
8	Borehole Connection to Upper Zone; Preferential Flow of Liquid Through Connections		Release	Maximum EPA Release = 8	Maximum EPA Release = 8×10^{-20}
9	Borehole Connection to Upper Zone Only		Release	Maximum EPA Release = 6×10^{-5}	0

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Figure C-8. Results for scenarios 1 through 9. Significant source-term releases were found for scenarios 2, 6, 8, and 9. Groundwater release was significant for scenarios 2, 6, and 8 (continued).

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Figure C-9. Summed normalized activity at repository boundary (IPANEL) conditional on scenarios 2, 6, 8, and 9 occurring (other scenarios had zero releases). (Direct comparison of the CCDFs with the EPA containment requirements is inappropriate, but it is shown for reference.) This figure is similar to Figure C-4 for the salt repository.

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Figure C-10. Releases at the 5-km boundary in Fracture Zone 2 conditional on scenarios 2 and 6 occurring (other scenarios had summed normalized releases less than 10^{-14}). This figure is similar to Figure C-2 for the salt disposal system.

Figure C-11. Contribution of nuclides to total release in granite disposal system for scenario 2.

Figure C-12. Contribution of radionuclides to percentage of total release in granite repository for scenario 2.

Table C-8. Five Most Important Modeling Parameters Influencing Cumulative Normalized Activity Leaving Repository for Scenarios 2, 6, 8, and 9 Conditional on Existence of a Connection to the Fracture Zones (No Cuttings/Cavings)

Parameter	Importance Rank* of Parameter Scenario			
	S2	S6	S8	S9
Agents Acting on Disposal System				
Permeability of intrusion borehole	1	1	1	--
Intrusion borehole diameter	2	3	3	1
Geologic Barrier				
Change in fracture zone hydraulic head over 10,000 yr	4	--	--	--
Container				
Gas generation rate when waste saturated	--	2	2	--
Average corrosion rate of container	3	--	--	--
Fraction of initially defective containers	5	4	--	--

* 1 is most important.

C.3 Bounding Calculations of Critical Conditions in a Container*

C.3.1 Objective of Criticality Analysis

Geologic disposal of spent fuel is an expensive proposition. The cost of geologic disposal increases as restraints are imposed on the methods by which it is accomplished. Imposing restraints to ensure criticality safety will, by definition, increase the cost of geologic disposal. The object of criticality analysis is, therefore, to minimize the restraints imposed to ensure adequate criticality safety and thereby minimize the cost of geologic disposal.

The objective of burying spent nuclear fuel in a geologic repository is not the safe dispositioning of fissile material; rather, it is the safe dispositioning of radionuclides. After many thousands of years, a large portion of those buried radionuclides will have decayed into stable atoms. The radiological hazard posed by radionuclides, when measured in terms of total activity, decreases as this time passes. In contrast, fissile material present in spent fuel poses a criticality hazard for a time on the order of billions of years.[†] Pending further guidance, restraints imposed on a disposal system to ensure criticality safety may legitimately be expected to last forever.

C.3.2 Minimum Spacing of Canisters

Calculations were performed to determine the amount of either salt or volcanic tuff[‡] required to isolate neighboring fissile regions. Infinite slabs of fissile material were assumed to be separated by infinite slabs of either salt or tuff. The fissile regions were assumed to be unmoderated ²³⁵U. The resulting very fast neutron energy spectrum serves to maximize the interaction between neighboring slabs. This approach is conservative and envelopes the cylindrical

* This section summarizes the report, "Criticality Safety Issues Associated with the Burial of Highly Enriched Nuclear Fuel in a Geologic Repository," by B. Palmer (Palmer, 1992).

[†] The half life of ²³⁵U is $\sim 7 \times 10^8$ years. The half-life of ²³⁹Pu is only $\sim 2.4 \times 10^4$ years; however, it decays such that ²³⁵U is produced.

[‡] Volcanic tuff was used in preparation for follow-on studies.

Figure C-13. Interaction of fissile regions as a function of the thickness of the material between them (Palmer, 1992, Figure 1).

repository configuration. TWODANT** was the code used to determine reactivity. TWODANT is a one- or two-dimensional, multigroup, discrete ordinates transport code. Revision 1.6 dated February 1, 1990, was used for this study. TWODANT is well suited for clearly defining the interaction between fissile regions. Figure C-13 shows the results. As seen in the figure, approximately 2 meters of separation between fissile regions is required to guarantee an effect on reactivity that is less than 1% in the multiplication factor, k_{eff} .

C.3.3 Demonstrating Criticality Safety

The methodology used to demonstrate criticality safety for this performance assessment is illustrated in Figure C-14. This methodology begins with a set of extremely conservative assumptions, i.e., to avoid altogether a critical condition.†† These assumptions will yield a low fissile material loading. Economics drives a desire for higher fissile loadings and therefore a technical basis for less conservative assumptions must be provided to guarantee adequate safety.

For the hypothetical repositories considered in this performance assessment, the following assumptions were made:

- A criticality incident in a geologic repository is unacceptable.

** The codes described in this section are not listed in Appendix G.

†† A critical condition is one in which fission (splitting of nuclei) is self sustaining by producing on average one slow neutron for every one absorbed in the fission (subcritical conditions produce less than one; supercritical conditions produce more). Though an individual fission releases a lot of energy (about 80% of which is heat), a critical condition in a repository does not imply release of massive amounts of heat on a very short time scale (e.g., explosion). For instance, the critical condition may be self controlling (or shut down) in that uranium density decreases as the material heats up (or the moderating material disappears [e.g., water vaporizes]), causing the reaction to go subcritical such that low, relatively constant power output is achieved (or causing the reaction to stop).

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Figure C-14. Methodology for demonstrating criticality safety.

- Within the time frame of interest, the fuel and disposal system will completely disintegrate.
- Fissile material preferentially migrates to the same spatial region.
- Optimum flooding.
- Disposal positions are isolated, i.e., neutronically isolated as well as no migration of fissile material between disposal positions.
- Fissile isotope is 100% enriched.

These assumptions lead to very low fissile loading per canister. Technical justification to deviate from this conservative set of assumptions was then sought. The deviations were justified by assuming that one criterion used in selecting a repository site would be geologic stability. The question then arose: Are there features inherent in the repository site that would tend to reduce the reactivity of a completely disintegrated disposal system?

Granite Repository. With the exception of the presence of large quantities of granite surrounding disposal containers, there are no features inherent in a granite repository that affect the reactivity of a fuel disposal system. A limit of 0.70 kg of ^{235}U per disposal position was selected from Figure C-15 as the single parameter subcritical mass limit for aqueous solutions containing ^{235}U .

Salt Repository. There are features inherent in a salt repository that would aid in the justification of a higher ^{235}U mass limit than that cited for the granite repository. These features are the following:

- The formation entombs the disposal container.
- Because of rapid salt creep, the maximum water volume fraction is limited to less than 5%.
- All water in a salt repository is in the form of a saturated brine solution. This solution contains effective neutron absorbers, i.e., chlorine.
- Reflecting material outside of the disposal container is salt.

In the absence of large quantities of other moderating materials, the limit of water volume fraction^{††} yields a greatly increased ^{235}U mass limit. Paxton and Pruvost (1987) indicate that a water reflected sphere of ^{235}U with a water volume fraction of 5% requires >20 kg of ^{235}U to achieve critical. To allow for reasonable quantities of carbon, such as that found in waste form 1, a limit of 10 kg ^{235}U was selected.

^{††} A volume limit may be inappropriate in a material where the waste can react with the solid to form hydrate oxides since the data shown in Figure C-15 ultimately are based on the ration of hydrogen atoms to ^{235}U atoms.

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Figure C-15. Minimum spherical critical masses as functions of ^{235}U enrichment in homogeneous and heterogeneous hydrogen-moderated systems (Paxton and Pruvost, 1987).

The limits mentioned about—0.70 kg and 10 kg—are better understood when the data in Figure C-16 is considered. In a granite repository, there is no limit on the water volume fraction and therefore no limit on the density of ^{235}U . Figure C-16 shows a minimum spherical critical mass of approximately 0.8 kg corresponding to a density of 50 kg/m^3 ^{235}U . In a salt repository, where water volume fraction is limited, reactivity is maximized when the density ^{235}U is highest. The minimum spherical critical mass is slightly greater than 20 kg for a ^{235}U density near that of uranium metal (Figure C-16).

In addition to the analysis described above, two specific intact fuels were analyzed: Shippingport PWR-Core 2 Seed 2 and Fort St. Vrain. Analysis of these fuels was not directly used within this performance assessment, although noteworthy results were generated. Shippingport PWR is water moderated and zircaloy clad with a ^{235}U loading of ~15 kg per cluster, while Fort St. Vrain fuel is graphite moderated with a relatively low ^{235}U loading (~1 kg per element). These two fuels are representative of many fuels stored at the Chem Plant.

The CSAS25 module of the code SCALE (Palmer and Crawford, 1991a, 1991b, 1991c) was used for Shippingport PWR Core-2 Seed-2. The CSAS25 module consists of cross section preprocessing followed by KENO-Va. KENO-Va is a Monte Carlo-based neutron transport code. Results of these calculations indicated that a 2×2 array stacked to an infinite height with full water reflection had a k_{eff} (an effective multiplication factor) ~0.60 as long as the fuel was dry. If the array is flooded with water, this configuration yields a reactivity well in excess of prompt critical. Flooding of this array with a saturated brine solution instead of water yields a k_{eff} ~0.97.***

Fort St. Vrain fuel elements were analyzed using the MCNP-4.2 code (Crawford and Palmer, 1992a, 1992b, 1992c, 1992d). MCNP is also a Monte Carlo-based transport computer code. MCNP offers more powerful geometry

*** These calculations led to the conclusion that the SCALE27-Group cross sections are inadequate for systems containing large amounts of chlorine.

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Figure C-16. Critical masses of homogeneous spheres of uranium moderated by water (Paxton and Provost, 1987, Figure 10).

description capabilities than SCALE and therefore is more suited to the relatively complex geometric structure of a Fort St. Vrain fuel element. Cross section data that are continuous in energy were used to perform these calculations. Even with full water reflection and infinite height assumptions, the k_{eff} of seven dry elements in a tight hexagonal array is only ~ 0.77 . Once again, allowing this system to become flooded with water drives the reactivity to well in excess of prompt critical. Flooding with a saturated brine solution, however, yields a k_{eff} of as low as ~ 0.69 .

A key result of these fuel-specific calculations is that flooding a system with a brine solution is significantly less reactive than flooding with pore water. Figure C-17 shows how the multiplication factor of a system of Shippingport PWR elements decreases with the addition of chlorine to the flooding water.

C.3.4 Summary and Conclusions

For the hypothetical repository types, granite and salt, considered in this PA evaluation, the following assumptions were made:

- A criticality incident in a geologic repository is unacceptable.
- Within the time frame of interest, the fuel will completely disintegrate (and the host rock and container do not mix with the fuel).
- Fissile material preferentially migrates to the same spatial region in the now disintegrated waste parcel.
- Fluid floods the pore spaces in the waste parcel and host rock.
- Waste parcels are isolated neutronically.
- No migration of fissile material occurs between waste parcels.
- Fissile material is 100% ^{235}U .

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Figure C-17. Effect of chlorine on the reactivity of flooded Shippingport PWR Core-2 Seed-2. The chlorine exists in the water used to flood the cluster but not in the reflecting water. An atom density of 30×10^{-4} atoms/barn-cm corresponds to ~150 ppt of chlorine (Palmer, 1992).

Given these assumptions, an equivalent ^{235}U limit was assigned to each repository type. The limits are generic in that they apply to nearly all foreseeable waste forms. The limits are

Salt Repository $\Rightarrow$ 10 kg equivalent ^{235}U per disposal position

Granite Repository $\Rightarrow$ 0.7 kg equivalent ^{235}U per disposal position.

The difference in the two mass limits is due to features inherent in a salt repository that would tend to lower the reactivity of systems, e.g., limited availability of pore space for water and neutron absorption of chlorine (Figure C-17).

Criticality analysis was performed for some specific fuels. These calculations showed that much higher position loadings are possible using the assumption that the fuel remains intact. If some technical basis for this assumption is available, the demonstration of criticality safety in a geologic repository would be greatly simplified.

Deviation from the existing very conservative approach to demonstrating criticality safety in a geologic repository will require additional guidance. This guidance should define some acceptable level of risk over time. Unfortunately the traditional methods of criticality safety cannot be applied to geologic repositories. The principle of double contingency, though widely used, assumes that remedial action is possible in the event of contingency failure. This type of remedial action is not possible in a geologic repository.

C.3.5 Future Work

Future work will be to challenge the conservative set of assumptions listed above. Some of these assumptions can be challenged as follows:

- A criticality incident in a geologic repository is unacceptable.

A technical basis for deviating from this assumption can likely be developed. For instance, current American National Standards form the basis for nearly all NRC regulations and DOE orders related to criticality safety. These standards recognize that it may be reasonable to reduce criticality control requirements when shielding protects personnel. Although clarification of the applicability of criticality safety requirements needs to be pursued for the period after closure and when waste is not accessible to individuals, clearly the geologic repository provides sufficient shielding from the direct radiation that is associated with a criticality event. Currently, we think an analysis of the implications of an on-going, constant power criticality event on radionuclide migration would be feasible and not overly difficult. The result may be that even if a criticality event occurs, it would not enhance radionuclide migration or significantly increment the inventory of radionuclides and, thus, the integrated release over 10,000 yr. We expect that it may be undetectable except for direct intrusions.

- Within the time frame of interest, the fuel will completely disintegrate.

The key phrase in this assumption is "time frame of interest." There is no question that all fuel and disposal systems will eventually disintegrate. A demonstration of the physical integrity of a key component of a system becomes possible only after the time frame is defined. The acceptable level of risk mentioned above must be a function of time.

- Fissile material preferentially migrates to the same region in the waste parcel.

Fissile elements have distinct chemical and physical properties. Although under extreme conditions these elements might preferentially migrate to the same region,^{†††} future work could include establishing an understanding of the situations in which these extreme conditions can happen as well as measures that could be included in a disposal system to prevent these conditions. A broad spectrum of possibilities exists for incorporating controls to provide long-term assurance of maintaining subcritical conditions. For example, Waste Disposal Group 5 provides such control by diluting the enriched uranium with depleted uranium to render reactivity of the waste form similar to natural uranium. Adding depleted uranium or chemically compatible neutron absorbers to the other waste forms is certainly a possibility to ensure that criticality is no longer a credible hazard.

- Waste parcels are isolated neutronically.
- No migration of fissile material between waste parcels.

Disposal of large quantities of fissile material will require isolation of fissile units from each other. In a geologic repository, this isolation of disposal positions will be time dependent. Providing a technical basis to ensure adequate isolation of disposal positions over the time frame of interest can be done once this time frame is defined.

- The fissile material is 100% ^{235}U .

This assumption has been shown to be untrue, but Figure C-15 shows how the minimum critical mass of ^{235}U in a flooded system depends on enrichment. Lowering the enrichment of the waste would, in general, lower the reactivity within the waste parcels even further.

^{†††} If a box filled with a mixture of cotton balls and pea-sized balls of uranium is shaken, a reasonable expectation is that nearly all the uranium will be on the bottom of the box.

C.4 Detailed Single Canister Simulations

As a prelude to the complex consequence modeling presented in Chapters 12 and 13, a detailed analysis of the fluid and heat transport around a single waste parcel was performed using BRAGFLO_T. Also modeled was the corrosion/gas generation process of the waste package container. An important limitation of these calculations, though, is that they were made using the original gas generation submodel present in BRAGFLO, the parent code of BRAGFLO_T.

The primary objective of the study was to identify modeling parameters that might impact the release of radionuclides in the performance assessment. The detailed analysis also resulted in some observations concerning the gas generation process of the waste parcel. This information was used to improve the modeling effort in the formal performance assessment.

C.4.1 Conceptual Model

Both bedded salt and granite were investigated as potential host formations. Each repository was modeled axisymmetric with the z-axis oriented along the center of the canister. In both repositories, a single waste package was disposed of vertically 1 m below the center of a room, which was 9 m in diameter, and 6 m high. The room dimensions were selected to correspond with the dimensions of the rooms and drifts in the performance assessment (9 m wide by 6 m high). After placing the waste parcel in the disposal facility, the room and the annulus around the waste parcel were assumed to be filled with a low permeability material. In the salt site, crushed rock salt was used as the backfill in both the room and the annulus. In the granite site, bentonite clay was used around the waste parcel and crushed granite was used in the room above (no bentonite clay, in contrast to the Complex PA). Diagrams depicting a logical mesh of the salt and granite repositories are shown in Figures C-18 and C-19, respectively.

In the original gas generation model, each element was assigned an initial metal concentration (kg/m^3). The corrosion of the waste parcel container was implemented by distributing the metal mass of the container in the grid blocks adjacent to the canister. These grid blocks were the only cells with metal concentrations greater than zero, and are referred to in Figures C-18 and C-19 as the gas generation zone. It should be recognized that the grid-block metal concentration was a function of both the mass of metal and the size of the cell.

One main difference between the two sites was the source of water to the waste parcel. At the salt site, water in the form of pressurized brine flowed through the salt into the repository. A disturbed rock zone (DRZ) that extended 1 m radially away from the waste parcel hole, 1 m below the drift, and 3 m above the drift aided this process. At the granite site, the host rock was assumed to be impermeable, so the only source of water was a single fracture 10 cm in radius, extending from the surface to the center of the repository. The permeability of the fracture was computed such that gravitational forces would initially inject $9.4 \times 10^{-8} \text{ m}^3/\text{s}$ (0.34 l/hr) of water into the repository through the fracture. The value, $9.4 \times 10^{-8} \text{ m}^3/\text{s}$ (0.34 l/hr), was derived by scaling fracture inflow rate observed at the Stripa mine (Neretnieks, 1986), $2.2 \times 10^{-7} \text{ m}^3/\text{s}$ (0.8 l/hr), by the ratio of the single canister granite repository surface area (296.9 m^2) to the surface area monitored at the Stripa mine (700 m^2).

C.4.2 Parameter Values

Different waste parcels with different dimensions, corrosion rates, and heat generating potential were used in the two sites (Table C-9). The reason for this is because the container was constructed of different materials in the two sites (Inconel 625 in the salt, and stainless steel Nitronic 50 in the granite), and because these parameters were updated as the study progressed. The waste parcel used in the granite site is identical to the graphite waste parcel of Waste Disposal Group 4, and the corrosion rates of Inconel 625 and stainless steel Nitronic 50 were obtained from Molecke et al. (1981). Other important base case parameters are included in Figures C-18 and C-19.

	k (m^2)	ϕ	S_{wi}	P_t (MPa)	C_p (J/kg K)	k_h (Watt/m K)	S_{wr}
Intact Halite	10^{-23}	0.01	1.0	50.800	868	7.20	0.20
Disturbed Rock Zone	10^{-21}	0.03	1.0	10.300	868	7.20	0.20
Backfill in Room	10^{-20}	0.09	1.0	4.658	868	7.20	0.20
Backfill around Waste Parcel	10^{-20}	0.09	0.3	4.658	868	7.20	0.40
Gas Generation Zone	10^{-20}	0.09	0.3	4.658	868	7.20	0.40
Waste Parcel	0	--	--	--	800	7.20	0.20

Heat Generation = 1274 Watts/Canister
= 621 Watts/ m^3
Corrosion Rate = 0.008 mm/yr
= 1×10^6 mole/ m^3 /s (Units for Original Gas
Generation Model in BRAGFLO)

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Figure C-18. Conceptual/applied model for single waste parcel in salt repository, shown on a logical grid.

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Corrosion Rate = 0.005 mm/yr
= 2.68×10^{-7} mole/m³/s

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65
66

Table C-9. Waste Parcel Parameters

	Hole Diameter (m)	Container Diameter (m)	Container Length (m)	Container Thickness (cm)	Corrosion Rate (mm/yr)	Heat Load (W/Container)
Salt	1.00	0.7228	5.00	0.6400	0.008	1274
Granite	1.26	0.6604	3.25	0.9525	0.005	407

Several of these parameters were varied one at a time (*ceteris paribus*) from the base case analysis to evaluate the sensitivity of the parameter with respect to the "consumption time" of the waste parcel container. The consumption time was defined as the time in years when the metal concentration of a single gas generating cell is completely consumed in the corrosion process.

C.4.3 Results

Scoping calculations of the single canister produced information concerning the use of large elements and other modeling issues through (1) a parametric sensitivity study and (2) a cursory comparison of the behavior of the two repositories.

Parametric Sensitivity Study in Salt. Several model parameters were varied one at a time to evaluate their influence on the consumption time of a waste parcel container in a salt repository (Table C-10). Of the parameters varied, only three had a substantial influence on the consumption time of the waste parcel container:

- (1) the corrosion rate of the metal container (runs 20-23)
- (2) decay constant in smoothing function for the reaction rate, α (runs 10-12)
- (3) the threshold pressure of the annulus (runs 2-4, and 6-9).

Parameters having little or no impact on the consumption time were:

- (1) the initial brine saturation of the backfill in the room (runs 29,30)
- (2) the initial brine saturation of the backfill in the annulus (runs 16,17)
- (3) the permeability of the backfill in the room (runs 18,19)
- (4) the permeability of the backfill in the annulus (runs 24,25)
- (5) the heat source of the waste parcel (runs 26-28).

The base case (run 1) should also be included in each of the above analyses.

The effect of the corrosion rate on the consumption time is readily apparent from runs 20-23 in Table C-10. The smoothing function decay constant, α , also influenced the consumption time of the container. The parameter α is used to control the gas generation rate as water, a principle reactant, becomes limited. Water surrounding the waste parcel will corrode the container only if it can flow to the canister wall. Therefore, if the liquid saturation in the gas generation zone falls below the residual saturation, the reaction will stop. To model this change in reaction rate, we introduced the exponential smoothing function

$$r/r_{\max} = 1 - \exp[-\alpha(S_w - S_{wr})] \quad (C-7)$$

where r is reaction rate over a given time step, $r_{\max}$ is the maximum reaction rate, S_w is the liquid (wetting phase) saturation, S_{wr} is the residual liquid saturation, and α is the smoothing function decay constant. The behavior of Equation C-7 as a function of the liquid saturation is depicted in Figure C-20a, where the different curves correspond to different values of α . Notice the reaction rate goes to zero as the liquid saturation approaches its residual value. Also, as the value of α is increased, the smoothing function begins to resemble a step function.

Table C-10. Parametric Sensitivity Study of Model Parameters Probably Influencing the Consumption Time of a Single Canister in Salt

Run Number	α	Backfill			Room			Corrosion Rate (mm/yr) ^a	Initial Heat Loads (W/canister)	Consumption Time (yr) ^b
		Threshold Pressure (MPa)	Permeability (m ²)	Initial Water Saturation	Threshold Pressure (MPa)	Permeability (m ²)	Initial Water Saturation			
1 (Base Case)	10	4.658	1x10 ⁻²⁰	0.30	4.658	1x10 ⁻²⁰	1.0	0.008	1274	880
2 ^c		0.9466								10,000 +
3		0.1924								10,000 +
4		0								10,000 +
5		0.9466	1x10 ⁻¹⁸							10,000 +
6		0.1924	1x10 ⁻¹⁶							10,000 +
7		0.9466	1x10 ⁻¹⁶							10,000 +
8		4.658	1x10 ⁻¹⁶							830
9		0	1x10 ⁻¹⁶							10,000 +
10	20									810
11	5									1080
12	3									1380
16 ^d				0.50						880
17				0.70						880
18					0.9466	1x10 ⁻¹⁸				840
19					0.1924	1x10 ⁻¹⁶				850
20								0.016		505
21								0.004		1665
22								0.002		3255
23								0.001		6435
24		0.9466	1x10 ⁻¹⁸		0.9466	1x10 ⁻¹⁸				800
25		0.1924	1x10 ⁻¹⁶		0.1924	1x10 ⁻¹⁶				790
26									2400	880
27									3600	880
28									4800	880
29							0.70			880
30							0.50			900

a Gas generation was modeled using original gas generation submodel of BRAGFLO, parent of BRAGFLO.T.

b Time when metal concentration (kg/m³) equal to zero in at least one computational element of the container.

c Only values that differ from base case value are shown in the main body of the table.

d Runs 13, 14 and 15 examined issue of volume of corrosion layer (not important for INEL gas generation)

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(a)

TRI-6342-4150-0

(b)

Figure C-20. Effect of corrosion rate on consumption time: (a) corrosion rate as a function of water saturation for different values of the decay constant in the exponential smoothing function between $r = 0$ and r_{max} , α , and (b) as the value of α is increased, the consumption time approaches its minimum possible value, $r \approx r_{max}$.

In runs 10 to 12, the value of the decay constant was varied while all others were maintained. These results are presented graphically in Figure C-20b. The vertical dotted/dashed line at 792 yr represents the consumption time if the reaction proceeded continuously at $r_{\max}$ independent of S_w . To approximate a step function and still remain numerically stable, a value of 20 was selected for α in the Complex PA.

For naturally occurring porous rock, the threshold pressure can be correlated with the permeability of the rock (Davies, 1991) using the equation (see also Chapter 5),

$$P_t = 0.56 * k^{-0.346} \quad (C-8)$$

where P_t is the threshold pressure in MPa, and k is the permeability in m^2 . In the base case (run 1), the threshold pressure was computed using Equation C-7 from a permeability of $10^{-20} m^2$. Threshold pressure values of 0.9466 and 0.1924 shown in Table C-10 were computed from permeabilities of 10^{-18} and $10^{-16} m^2$, respectively.

The threshold pressure in the annulus was varied independent of the permeability to examine the influence of capillary pressure on the corrosion process of the waste parcel container. The permeability was held constant at $10^{-20} m^2$ in runs 1 to 4, and $10^{-16} m^2$ in runs 6 to 9. As shown in Table C-9, the container failed only in the two runs where the threshold pressure in the annulus was equal to the threshold pressure in the room: runs 1 and 8. Of the runs in which the threshold pressure in the annulus was less than that of the room, the container remained intact for the entire 10,000-yr period. These results indicate the capillary pressure characteristics of the backfill material can significantly impact the longevity of the waste parcel container.

This raises an interesting question regarding the selection of the backfill materials. It appears the combination of a low capillary pressure material around the waste parcel relative to that of the room would enhance the life of the waste parcel packaging. However, based on Equation C-8, a material with a low capillary pressure will also be highly permeable. From a transport point of view, once the waste packaging has failed, it is desirable that all backfill materials have low permeabilities. Clearly, there are tradeoffs between threshold pressure and permeability.

Parametric Sensitivity Study in Granite. As was done in the salt repository, several model parameters were varied one at a time to evaluate their influence on the consumption time of a waste parcel container in a granite repository (Table C-11). Of the parameters varied, only three had a substantial influence on the consumption time:

- (1) the corrosion rate of the metal container (runs 19-21)
- (2) the permeability of the room backfill (runs 15-18)
- (3) the permeability of the vertical secondary fracture (runs 2-6).

Parameters having little or no impact on the consumption time were:

- (1) the permeability of the backfill in the annulus (runs 7-9)
- (2) the initial water saturation of the backfill in the annulus (runs 10-12)
- (3) the porosity of the backfill in the room (runs 13,14)
- (4) the heat load of the waste parcel (runs 22-24).

Just as in the salt repository, the corrosion rate of the metal significantly influenced the life of the container (Figure C-21a). At very low rates, the waste parcel container remained intact almost the entire 10,000-yr period. This parameter is clearly important to the results in both repositories. Unlike the salt repository though, the permeability of the backfill in the room was important, but only at very low values (Figure C-21b). The reason the room backfill permeability was important in the granite repository and not in the salt, is because water must flow through the room to reach the canister only in the granite repository. Finally, the fracture permeability was important, but once again, only at low values (Figure C-21c).

Comparison of the Single Canister in Salt and Granite. A comparison of the performance of the waste parcels in the salt and granite repositories is described for container failure, corrosion profile, and temperature.

Table C-11. Parametric Sensitivity Study of Model Parameters Potentially Influencing the Consumption Time of a Single Canister in Granite

Run Number	Fracture Permeability (m ²) ^a	Annulus Permeability (m ²)	Annulus Initial Water Saturation	Room Porosity	Room Permeability (m ²)	Corrosion Rate (mm/yr) ^b	Initial Heat Load (W/canister)	Consumption ^a Time (yr)
1 (Base Case)	3.5x10 ⁻¹³	1x10 ⁻²⁰	0.001	0.40	1.0x10 ⁻¹⁰	0.005	407	2185
2 ^b	7.0x10 ⁻¹³							2164
3	1.75x10 ⁻¹³							2227
4	8.75x10 ⁻¹⁴							2308
5	1.0x10 ⁻¹⁴							3555
6	1.0x10 ⁻¹⁵							7765
7		1.0x10 ⁻²¹						3760
8		1.0x10 ⁻¹⁹						2025
9		1.0x10 ⁻¹⁸						2000
10			0.40					2155
11			0.70					2000
12			1.0					1980
13				0.20				2164
14				0.60				2206
15					1.0x10 ⁻¹³			2188
16					1.0x10 ⁻¹⁵			2217
17					1.0x10 ⁻¹⁷			5090
18					1.0x10 ⁻¹⁹			10000+
19						0.0100		1190
20						0.0025		4150
21						0.00125		8600
22							890	2183
23							1334	2181
24							222	2186

a Time when metal concentration (kg/m³) equal to zero in at least one computational element of the waste parcel.

b Only values that differ from base case value are shown in the main body of the table.

c The decay of the heat source was assumed to be similar to commercial high level waste.

(a)

(b)

(c)

Figure C-21. Variation of consumption time (zero metal left in one computational element) in a granite repository with change in (a) corrosion rate of the metal container; (b) permeability of the room backfill; and (c) permeability of the secondary fracture.

Container Failure. In general, the behavior around the waste parcel in the two repositories was very similar. Corrosion of the container was controlled principally by the availability of water and the corrosion rate. For the base case in both sites, sufficient water was available to begin and maintain corrosion quite early in the simulation. Because the corrosion rate used in the salt repository was much higher than the corrosion rate used in the granite repository, the container failed much earlier in the salt site (880 vs. 2185 yr). However, due to the uncertainty of these rates, it cannot be concluded the waste parcel container will last longer in a granite repository than in a salt repository.

Corrosion Profile. A second notable difference between the salt and granite repositories is the corrosion profile of the canister container. Snapshots in time of the container metal concentration are presented in Figures C-22a and C-22b for the salt and granite repositories, respectively. The length of the canister from the base to the top is represented along the x-axis. In the salt repository, brine flows through the backfill from the surrounding salt and then into the room above. This behavior leads to a fairly uniform corrosion profile ultimately failing at the top of the waste parcel. In the granite repository, water flows into the room from the vertical fracture and then into the backfill around the canister. Water is abundantly available at the top of the waste parcel and relatively scarce near the bottom. Consequently, in many cases the top of the canister corroded completely before water ever reached the bottom of the canister.

Temperature. The most dramatic difference between the salt and granite repositories was the near-field temperature response due to similar thermal loading of the waste parcel (1274 W/container for the salt [run 1], and 1334 W/container for the granite [run 23]). In Figures C-23a and C-23b, the temperature at three points is plotted for the salt and the granite site, respectively. The three curves show the temperature of the container, the backfill next to the container, and the host rock adjacent to the backfill. The curves are very similar, except the magnitude of the temperature increase is substantially larger in the granite site (note the scale change). The reason the temperatures were much lower in the salt repository is that salt has a much higher thermal conductivity than granite, allowing heat generated by the canister to dissipate into the surrounding rock. In addition, a significant amount of heat is transported away from the canister convectively due to brine flow through the backfill around the canister.

This cooling effect can be seen in Figures C-24a and C-24b, where the simulation results of two different energy transport codes, BRAGFLO_T and SUMQ_3D, are presented (SUMQ_3D is discussed in Section C.5). Concerning heat flow, the main difference between the two codes is that SUMQ_3D does not account for convective transport but BRAGFLO_T does. In the granite repository (Figure C-24b), there was little liquid flow near the container during the period shown. Consequently, the results of the two codes are very similar. The slight differences most likely can be attributed to the fact that different thermal properties were used in the room and the annulus by the two codes. In BRAGFLO_T, spatially varying properties were used, while SUMQ_3D requires that homogenous properties are used. In the salt repository (Figure C-24a), the results of the two codes were quite different even though homogenous thermal properties were used by both codes in these simulations. The temperature rise when energy was transported convectively (BRAGFLO_T), was approximately three times less than when convective transport was not modeled (SUMQ_3D).

TRI-6342-3583-0

(a)

TRI-6342-3582-0

(b)

Figure C-22. Snapshots in time (yr) of the metal concentration along the container from bottom to top for (a) salt site, and (b) granite site.

TRI-6342-3578-0

(a)

TRI-6342-3579-0

(b)

Figure C-23. Temperature versus time along horizontal plane intersecting the midpoint of the container (note change in temperature scale): (a) salt site, 1274 W/container; (b) granite site, 1334 W/container.

TRI-6342-3581-0

(a)

TRI-6342-3580-0

(b)

Figure C-24. Convection heat transport shown by comparing the results of two codes, BRAGFLO_T and SUMQ_3D, for (a) salt repository and (b) granite repository.

C.5 Scoping Heat Conduction Calculations

Several simplified heat conduction design calculations were performed using the SUMQ_3D code (Berglund, in prep., a) for the five different processed waste forms. In addition, other scoping calculations were performed using the DOE wastes generated by the West Valley Demonstration Project, Defense Waste Processing Plant (Savannah River), and the Hanford Waste Vittrification Plant, assuming a total mass of 12,640 MT and a total volume of 4706 m³. The calculations are described below.

C.5.1 SUMQ_3D Calculations for Waste Forms

The SUMQ_3D code calculates the time-dependent temperature at any point within an arbitrary array of linear heat sources of finite length embedded in an infinite, uniform, isotropic material (Figure C-25). By assuming a waste parcel distribution within a disposal room, the temperature at any backfill-buffer/host-rock boundary can be computed and the peak temperature recorded for use in repository design. Waste parcel arrays that produce peak backfill-buffer/host-rock interface temperatures below 100°C are considered acceptable as long as criticality or constructibility constraints are not violated. A minimum hexagonal waste parcel spacing of 2.6 m center to center was assumed for ease of construction. Waste parcel arrays generating power at rates less than approximately 390 W/waste parcel (waste parcel in granite repository with 1.2-m borehole diameter) (backfill-buffer 0.3 m thick) were spaced at this minimum constructibility spacing.⁺⁺⁺ An ambient repository temperature of 30°C was assumed, and all spacing calculations were based on the assumption that waste placement occurred in the year 2030. With waste placement in 2030, the thermal decay was assumed to follow that of commercial high level waste (CHLW) as defined in Raines et al. (1981, p. 3) and shown in normalized form in Figure C-26.

SUMQ_3D is based on the superposition of many one-dimensional point-source heat-conduction solutions arrayed in a manner to approximate the waste parcel field. For heat conduction in salt, the thermal conductivity is $K = 5.0 \text{ W/(m}\cdot\text{K)}$ and thermal diffusivity is $k = 2.71 \times 10^{-6} \text{ m}^2/\text{s}$. For granite, a thermal conductivity of $K = 2.56 \text{ W/(m}\cdot\text{K)}$ was assumed as well as a thermal diffusivity of $k = 1.05 \times 10^{-6} \text{ m}^2/\text{s}$.

Salt Repository. The backfill-buffer salt boundary temperatures calculated for all cases achieved a peak value within a 10-yr period after emplacement. As an example of the temperature variation with time, results for a 4-row hexagonal array of waste parcels filled with special LEU spent fuel (WP17) with a uniform center-to-center spacing of 2.6 m is shown in Figure C-27. The example has a thermal output of 464 W/waste parcel (year 2030). This case represents a worst-case situation because WP17 has the greatest initial heat rate of all waste parcels and all parcels in the array were assumed to generate heat at an initial rate of 464 W/waste parcel.^{****} The peak temperature rise of 47°C is below the permissible 70°C limit selected to keep the backfill/intact salt boundary below 100°C (70°C + 30°C ambient). A listing of one of the SUMQ_3D calculations performed for the salt repository is provided in Section C.5.3.

Granite Repository. The backfill/hard-rock interface temperatures calculated for all cases achieved a peak value within a 10-yr period after emplacement. The temperature response with time for the granite was very similar to that for salt shown in Figure C-28. A listing of some of the SUMQ_3D calculations performed for the saturated granite repository is provided in Section C.5.3.

C.5.2 SUMQ_3D and COND_1D Calculations for High-Level Waste

To assess the importance of the temperatures experienced in a bentonite backfill-buffer material, calculations were run to determine the transient temperature distribution surrounding a waste parcel containing Hanford

⁺⁺⁺In a salt repository, the higher thermal conductivity allows the power limit to be raised to 690 W/waste parcel, assuming a 1.2-m crushed-salt backfill-buffer diameter.

^{****}A mixed array of waste parcels with different (and in most cases, lower thermal outputs) will be placed in the repository.

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TRI-6342-3569-0

Figure C-25. Computational basis for SUMQ_3D, point source representation of canister field.

TRI-6342-3504-0

Figure C-26. Normalized heat rate for Commercial High-Level Waste used in SUMQ_3D calculations.

TRI-6342-3571-0

Figure C-27. Temperature rise at canister wall for 407-W waste emplacement packages (WEPs), 4 rows hexagonal, 2.6-m center-to-center spacing, with decay from SUMQ_3D calculations.

(a) 12 m Scale

(b) 2 m Scale

TRI-6342-3572-0

Figure C-28. Temperature distribution out to (a) 12 m and (b) 2 m for a single canister at 6.5 yr using COND_1D and SUMQ_3D for Hanford high-level waste in borosilicate glass (WP8).

high-level waste (WP8). Both COND_1D (Berglund, in prep., b) and SUMQ_3D (Berglund, in prep., a) codes were used (Berglund, 1992). COND_1D is a one-dimensional thermal-conduction code that provides transient temperature data both inside and outside of a single waste parcel (i.e., both near-field [at the waste parcel interior] and far-field); however only a single waste parcel can be modeled and the waste parcel must be infinitely long. SUMQ_3D is a three-dimensional code that calculates the time-dependent temperature at any point within an arbitrary array of linear heat sources of finite length, but it is limited to providing reliable temperature data outside the waste parcel (i.e., outside the bentonite/host rock). Since the ideal was to determine the temperature of the bentonite for an array of finite-length canisters, calculations were run using both COND_1D and SUMQ_3D, thus taking advantage of the codes' respective attributes and compensating for their respective limitations.

For the calculations, the following conditions were taken into account:

- The waste parcel considered was WP8, which contains high-level waste from Hanford immobilized in glass. This waste parcel has an initial heat rate of 407 W (slightly higher than that shown in Chapter 6, Table 6-2, for WP8).
- The waste canister is about 61 cm in diameter.
- Before emplacement in the repository, the waste canister is placed in a 66-cm-diameter overpack.
- Generally, the thickness of the waste canister wall and the overpack wall is less than 1 cm each.
- The internal backfill between the waste canister and the overpack is assumed to be a poured metal, such as lead.
- Once the canister/overpack is emplaced in the repository, a 0.3-m-thick layer of bentonite is placed between the overpack and the host rock.
- The thermal output of the graphite and glass waste form in the year 2030 is assumed to be approximately 464 W/m³ (slightly higher than that shown in Chapter 6, Table 6-2, for WP 8).
- The thermal properties assumed were

	Glass	Bentonite Backfill	Granite
Thermal conductivity, w/(m•k)	1.0	0.7	2.56
Thermal diffusivity, m ² /s	4.03×10^{-7}	4.38×10^{-7}	1.06×10^{-6}

Also, for computational simplicity, the following condition was assumed:

- The combined thickness of the waste canister, internal backfill, and overpack was neglected.

The COND_1D calculation was performed for an infinitely long single waste parcel using the conditions described above. Two computations were performed with SUMQ_3D. The first considered a single, very long waste parcel (33 m) that generated heat in a manner consistent with the COND_1D computation described above. The second considered an array of true-length canisters in a 2.6-m, canister center-to-center, hexagonal spacing. All calculations were run and compared at 6.5 yr because previous array calculations using SUMQ_3D had indicated that peak temperatures occurred at 6.5 yr.

Figures C-28a and C-28b show the temperature distributions from COND_1D and SUMQ_3D at 6.5 yr after emplacement for a single canister. As shown in Figure C-28a, the temperature distributions for both calculations are very similar outside the bentonite. Because the SUMQ_3D data so closely match the COND_1D data, this result suggests that the SUMQ_3D data outside the bentonite layer can be used as representative of far-field results for a true-length canister backfilled with bentonite. Figure C-28b provides a detailed look at the same temperature distributions for both codes at the overpack/bentonite interface.

Figure C-29 combines the COND_1D and SUMQ_3D calculations for the single waste parcel with the SUMQ_3D calculations for an array of true-length waste parcels. As seen in Figure C-29, the peak temperature rise for an array of canisters at 6.5 yr at the bentonite/granite boundary is 78°C. Therefore, to approximate the temperature rise within the bentonite for an array of true-length canisters, it is only necessary to add the bentonite temperature rise that was obtained from COND_1D at 6.5 yr. Doing so results in a peak temperature change at the overpack/bentonite boundary of 95°C (Figure C-28) or a total temperature of 125°C after the ambient temperature is added. This

Figure C-29. Temperature distribution out to 2 m for a single canister and array at 6.5 yr using COND_1D and SUMQ_3D for Hanford high-level waste in borosilicate glass (WP8).

value is below the 130°C limit recommended for bentonite stability.^{††††} It is observed that the centerline temperature for the waste parcel reaches a peak of 135°C (105 + 30 ambient), well below the 500°C temperature that promotes crystal growth in glass (Tang and Saling, 1990).

C.5.3 Listing of SUMQ_3D Calculations

Calculation for Salt Repository. One of the SUMQ_3D calculations performed for canister spacing design in a salt repository is shown on the following pages.

^{††††}If crushed-salt backfill in a salt repository were assumed to have properties similar to bentonite, it is clear that backfill peak temperatures would be less than 125°C.

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5      U1: [JWBERGL.USERS]SUMQ_3D.DBG
6      .....
7      CHO INPUT TEXT FILE:
8      .....
9      PROBLEM 'GRAPHITE_4 SALT'
10     XSTAG=1
11     YSTAG=0
12     DECAY=1
13     XSPACE=2.6
14     YSPACE=2.25
15     HEAT_RATE=464.
16     DIFFUS=2.71E-6
17     CONDUCT=5.0
18     TIME_INC=20
19     TOTAL_TIME=3.15567E8
20     X_CAN=15
21     Y_CAN=4
22     *X, Y, Z LOCATION OF POINTS OF INTEREST
23     POINT 1 19.5 2.85 2.39
24     CAN_SOURCES=50
25     CAN_LENGTH=4.77
26     .....

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37     GRAPHITE_4 SALT
38     NUMBER OF CANISTERS ALONG X DIRECTION..... 15
39     CANISTER SPACING ALONG X DIRECTION..... 2.60000
40     NUMBER OF CANISTERS ALONG Y DIRECTION..... 4
41     CANISTER SPACING ALONG Y DIRECTION..... 2.25000
42     NUMBER OF HEAT SOURCES ALONG CANISTER LENGTH.. 50
43     LENGTH OF CANISTER..... 4.77000
44     DIFFUSIVITY..... 0.27100E-05
45     CONDUCTIVITY..... 5.00000
46     HEAT RATE PER CANISTER..... 0.46400E+03
47     AGE OF CANISTER WHEN PLACED IN REPOSITORY.... 0.00000E+00
48     TEMPERATURE DECAY..... T
49     STAGGER X COORDINATES..... T
50     STAGGER Y COORDINATES..... F

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CANISTER #	X COORD	Y COORD
1	0.00	0.00
2	2.60	0.00
3	5.20	0.00
4	7.80	0.00
5	10.40	0.00
6	13.00	0.00
7	15.60	0.00
8	18.20	0.00
9	20.80	0.00
10	23.40	0.00
11	26.00	0.00
12	28.60	0.00

Appendix C: Scoping Calculations

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5	13	31.20	0.00
6	14	33.80	0.00
7	15	36.40	0.00
8	16	1.30	2.25
9	17	3.90	2.25
10	18	6.50	2.25
11	19	9.10	2.25
12	20	11.70	2.25
13	21	14.30	2.25
14	22	16.90	2.25
15	23	19.50	2.25
16	24	22.10	2.25
17	25	24.70	2.25
18	26	27.30	2.25
19	27	29.90	2.25
20	28	32.50	2.25
21	29	35.10	2.25
22	30	37.70	2.25
23	31	0.00	4.50
24	32	2.60	4.50
25	33	5.20	4.50
26	34	7.80	4.50
27	35	10.40	4.50
28	36	13.00	4.50
29	37	15.60	4.50
30	38	18.20	4.50
31	39	20.80	4.50
32	40	23.40	4.50
33	41	26.00	4.50
34	42	28.60	4.50
35	43	31.20	4.50
36	44	33.80	4.50
37	45	36.40	4.50
38	46	1.30	6.75
39	47	3.90	6.75
40	48	6.50	6.75
41	49	9.10	6.75
42	50	11.70	6.75
43	51	14.30	6.75
44	52	16.90	6.75
45	53	19.50	6.75
46	54	22.10	6.75
47	55	24.70	6.75
48	56	27.30	6.75
49	57	29.90	6.75
50	58	32.50	6.75
51	59	35.10	6.75
52	60	37.70	6.75

COORDINATES OF POINT OF INTEREST			
POINT #	X COORD	Y COORD	Z COORD
1	19.50	2.85	2.39

TEMPERATURE

PT# 1 (X,Y,Z)	
(0.1950E+02	
0.2850E+01	
0.2390E+01)	
IME	
0.1578E+08	0.327538E+02
0.3156E+08	0.393267E+02
0.4734E+08	0.425555E+02
0.6311E+08	0.444361E+02

3 of 5

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2		
3	0.7889E+08	0.456097E+02
4	0.9467E+08	0.463570E+02
5	0.1104E+09	0.468231E+02
6	0.1262E+09	0.470916E+02
7	0.1420E+09	0.472153E+02
8	0.1578E+09	0.472284E+02
9	0.1736E+09	0.471952E+02
10	0.1893E+09	0.471152E+02
11	0.2051E+09	0.469874E+02
12	0.2209E+09	0.468177E+02
13	0.2367E+09	0.466120E+02
14	0.2525E+09	0.463754E+02
15	0.2682E+09	0.461117E+02
16	0.2840E+09	0.458244E+02
17	0.2998E+09	0.455161E+02
18	0.3156E+09	0.451894E+02
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Calculation for Granite Repository. Some of the calculations performed for canister spacing design in a saturated granite repository are provided on the following pages.

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4      U1: {JWBBERGL.USERS}SUMQ_3D.DBG
5      .....
6      CHO INPUT TEXT FILE:
7      .....
8      PROBLEM 'WP8 HWVP IN GRANITE'
9      XSTAG=1
10     YSTAG=0
11     DECAY=1
12     XSPACE=2.6
13     YSPACE=2.25
14     HEAT_RATE=391.
15     DIFFUS=1.05E-6
16     CONDUCT=2.56
17     TIME_INC=20
18     TOTAL_TIME=3.15567E8
19     X_CAN=15
20     Y_CAN=4
21     *X, Y, Z LOCATION OF POINTS OF INTEREST
22     POINT 1 19.5 2.85 1.625
23     CAN_SOURCES=50
24     CAN_LENGTH=3.251
25     .....

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SUMQ_3D C-1.01VV (11/20/92)

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36     WP8 HWVP IN GRANITE
37     NUMBER OF CANISTERS ALONG X DIRECTION..... 15
38     CANISTER SPACING ALONG X DIRECTION..... 2.60000
39     NUMBER OF CANISTERS ALONG Y DIRECTION..... 4
40     CANISTER SPACING ALONG Y DIRECTION..... 2.25000
41     NUMBER OF HEAT SOURCES ALONG CANISTER LENGTH.. 50
42     LENGTH OF CANISTER..... 3.25100
43     DIFFUSIVITY..... 0.10500E-05
44     CONDUCTIVITY..... 2.56000
45     HEAT RATE PER CANISTER..... 0.39100E+03
46     AGE OF CANISTER WHEN PLACED IN REPOSITORY..... 0.00000E+00
47     TEMPERATURE DECAY..... T
48     STAGGER X COORDINATES..... T
49     STAGGER Y COORDINATES..... F

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CANISTER #	X COORD	Y COORD
1	0.00	0.00
2	2.60	0.00
3	5.20	0.00
4	7.80	0.00
5	10.40	0.00
6	13.00	0.00
7	15.60	0.00
8	18.20	0.00
9	20.80	0.00
10	23.40	0.00
11	26.00	0.00
12	28.60	0.00

Appendix C: Scoping Calculations

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5	13	31.20	0.00
6	14	33.80	0.00
7	15	36.40	0.00
8	16	1.20	2.25
9	17	3.90	2.25
10	18	6.50	2.25
11	19	9.10	2.25
12	20	11.70	2.25
13	21	14.30	2.25
14	22	16.90	2.25
15	23	19.50	2.25
16	24	22.10	2.25
17	25	24.70	2.25
18	26	27.30	2.25
19	27	29.90	2.25
20	28	32.50	2.25
21	29	35.10	2.25
22	30	37.70	2.25
23	31	0.00	4.50
24	32	2.60	4.50
25	33	5.20	4.50
26	34	7.80	4.50
27	35	10.40	4.50
28	36	13.00	4.50
29	37	15.60	4.50
30	38	18.20	4.50
31	39	20.80	4.50
32	40	23.40	4.50
33	41	26.00	4.50
34	42	28.60	4.50
35	43	31.20	4.50
36	44	33.80	4.50
37	45	36.40	4.50
38	46	1.30	6.75
39	47	3.90	6.75
40	48	6.50	6.75
41	49	9.10	6.75
42	50	11.70	6.75
43	51	14.30	6.75
44	52	16.90	6.75
45	53	19.50	6.75
46	54	22.10	6.75
47	55	24.70	6.75
48	56	27.30	6.75
49	57	29.90	6.75
50	58	32.50	6.75
51	59	35.10	6.75
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COORDINATES OF POINT OF INTEREST			
POINT #	X COORD	Y COORD	Z COORD
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1	19.50	2.85	1.63

T E M P E R A T U R E		
	PT# 1 (X,Y,Z)	
	(0.1950E+02	
	0.2850E+01	
IME	0.1625E+01)	
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0.1578E+08	0.420262E+02	
0.3156E+08	0.533698E+02	
0.4734E+08	0.597221E+02	
0.6311E+08	0.638413E+02	

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4 0.7889E+08 0.666903E+02
5 0.9467E+08 0.687259E+02
6 0.1104E+09 0.702002E+02
7 0.1262E+09 0.712668E+02
8 0.1420E+09 0.720254E+02
9 0.1578E+09 0.725442E+02
10 0.1736E+09 0.729203E+02
11 0.1893E+09 0.731759E+02
12 0.2051E+09 0.733175E+02
13 0.2209E+09 0.733605E+02
14 0.2367E+09 0.733194E+02
15 0.2525E+09 0.732050E+02
16 0.2682E+09 0.730262E+02
17 0.2840E+09 0.727915E+02
18 0.2998E+09 0.725073E+02
19 0.3156E+09 0.721783E+02
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5   Input Text File:
6       U1:[JWBERGL.USERS]SUMQ_3D.INP:124
7       Written on 04/01/93 08:21:16
8   Graph Data File:
9
10  Output Diagnostics/Debug File:
11      U1:[JWBERGL.USERS]SUMQ_3D.DBG
12  .....
13  CHO INPUT TEXT FILE:
14  .....
15  PROBLEM 'GRAPHITE_4 GRANITE'
16  XSTAG=1
17  YSTAG=0
18  DECAY=1
19  XSPACE=2.6
20  YSPACE=2.25
21  HEAT_RATE=407.
22  DIFFUS=1.052E-6
23  CONDUCT=2.56
24  TIME_INC=20
25  TOTAL_TIME=3.15567E8
26  X_CAN=15
27  Y_CAN=4
28  *X, Y, Z LOCATION OF POINTS OF INTEREST
29  POINT 1 19.5 2.58 1.625
30  POINT 2 19.5 2.75 1.625
31  POINT 3 19.5 3.05 1.625
32  POINT 4 19.5 3.25 1.625
33  POINT 5 19.5 3.75 1.625
34  POINT 6 19.5 4.25 1.625
35  CAN_SOURCES=50
36  CAN_LENGTH=2.997
37  .....

```

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38  SUMQ_3D C-1.01VV (11/20/92)                                04/01/93 08:21:23
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40  GRAPHITE_4 GRANITE
41  NUMBER OF CANISTERS ALONG X DIRECTION..... 15
42  CANISTER SPACING ALONG X DIRECTION..... 2.60000
43  NUMBER OF CANISTERS ALONG Y DIRECTION..... 4
44  CANISTER SPACING ALONG Y DIRECTION..... 2.25000
45  NUMBER OF HEAT SOURCES ALONG CANISTER LENGTH.. 50
46  LENGTH OF CANISTER..... 2.99700
47  DIFFUSIVITY..... 0.10520E-05
48  CONDUCTIVITY..... 2.56000
49  HEAT RATE PER CANISTER..... 0.40700E+03
50  AGE OF CANISTER WHEN PLACED IN REPOSITORY..... 0.00000E+00
51  TEMPERATURE DECAY..... T
52  STAGGER X COORDINATES..... T
53  STAGGER Y COORDINATES..... F

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54  CANISTER #      X COORD      Y COORD
55  1              0.00          0.00
56  2              2.60          0.00
57  3              5.20          0.00
58  4              7.80          0.00
59  5             10.40          0.00
60  6             13.00          0.00

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6	7	15.60	0.00
7	8	16.20	0.00
8	9	20.80	0.00
9	10	23.40	0.00
10	11	26.00	0.00
11	12	28.60	0.00
12	13	31.20	0.00
13	14	33.80	0.00
14	15	36.40	0.00
15	16	1.30	2.25
16	17	3.90	2.25
17	18	6.50	2.25
18	19	9.10	2.25
19	20	11.70	2.25
20	21	14.30	2.25
21	22	16.90	2.25
22	23	19.50	2.25
23	24	22.10	2.25
24	25	24.70	2.25
25	26	27.30	2.25
26	27	29.90	2.25
27	28	32.50	2.25
28	29	35.10	2.25
29	30	37.70	2.25
30	31	0.00	4.50
31	32	2.60	4.50
32	33	5.20	4.50
33	34	7.80	4.50
34	35	10.40	4.50
35	36	13.00	4.50
36	37	15.60	4.50
37	38	18.20	4.50
38	39	20.80	4.50
39	40	23.40	4.50
40	41	26.00	4.50
41	42	28.60	4.50
42	43	31.20	4.50
43	44	33.80	4.50
44	45	36.40	4.50
45	46	1.30	6.75
46	47	3.90	6.75
47	48	6.50	6.75
48	49	9.10	6.75
49	50	11.70	6.75
50	51	14.30	6.75
51	52	16.90	6.75
52	53	19.50	6.75
53	54	22.10	6.75
54	55	24.70	6.75
55	56	27.30	6.75
56	57	29.90	6.75
57	58	32.50	6.75
58	59	35.10	6.75
59	60	37.70	6.75
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COORDINATES OF POINT OF INTEREST			
POINT #	X COORD	Y COORD	Z COORD
1	19.50	2.58	1.63
2	19.50	2.75	1.63
3	19.50	3.05	1.63
4	19.50	3.25	1.63
5	19.50	3.75	1.63
6	19.50	4.25	1.63

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T E M P E R A T U R E

	PT# 1 (X,Y,Z)	PT# 2 (X,Y,Z)	PT# 3 (X,Y,Z)	PT# 4 (X,Y,Z)	PT# 5 (X,Y,Z)	PT# 6 (X,Y,Z)
	(0.1950E+02	(0.1950E+02	(0.1950E+02	(0.1950E+02	(0.1950E+02	(0.1950E+02
	0.2580E+01	0.2750E+01	0.3050E+01	0.3250E+01	0.3750E+01	0.4250E+01
IME	0.1625E+01)	0.1625E+01)	0.1625E+01)	0.1625E+01)	0.1625E+01)	0.1625E+01)
-----	-----	-----	-----	-----	-----	-----
0.1578E+08	0.486909E+02	0.457238E+02	0.429087E+02	0.419229E+02	0.410195E+02	0.408744E+02
0.3156E+08	0.604194E+02	0.575138E+02	0.547633E+02	0.537990E+02	0.528915E+02	0.526762E+02
0.4734E+08	0.669662E+02	0.641036E+02	0.613966E+02	0.604469E+02	0.595447E+02	0.593061E+02
0.6311E+08	0.711930E+02	0.683682E+02	0.656983E+02	0.647615E+02	0.638672E+02	0.636185E+02
0.7889E+08	0.740994E+02	0.713104E+02	0.686749E+02	0.677504E+02	0.668650E+02	0.666117E+02
0.9467E+08	0.761600E+02	0.734058E+02	0.708034E+02	0.698905E+02	0.690147E+02	0.687595E+02
0.1104E+09	0.776374E+02	0.749168E+02	0.723473E+02	0.714456E+02	0.705795E+02	0.703242E+02
0.1262E+09	0.786907E+02	0.760038E+02	0.734661E+02	0.725757E+02	0.717197E+02	0.714650E+02
0.1420E+09	0.794237E+02	0.767703E+02	0.742642E+02	0.733850E+02	0.725391E+02	0.722859E+02
0.1578E+09	0.799075E+02	0.772871E+02	0.748128E+02	0.739445E+02	0.731088E+02	0.728573E+02
0.1736E+09	0.802516E+02	0.776592E+02	0.752111E+02	0.743523E+02	0.735254E+02	0.732759E+02
0.1893E+09	0.804702E+02	0.779056E+02	0.754839E+02	0.746339E+02	0.738152E+02	0.735680E+02
0.2051E+09	0.805702E+02	0.780336E+02	0.756383E+02	0.747978E+02	0.739881E+02	0.737429E+02
0.2209E+09	0.805671E+02	0.780588E+02	0.756905E+02	0.748596E+02	0.740591E+02	0.738158E+02
0.2367E+09	0.804775E+02	0.779969E+02	0.756545E+02	0.748328E+02	0.740408E+02	0.738001E+02
0.2525E+09	0.803111E+02	0.778584E+02	0.755424E+02	0.747297E+02	0.739462E+02	0.737080E+02
0.2682E+09	0.800774E+02	0.776526E+02	0.753632E+02	0.745599E+02	0.737859E+02	0.735499E+02
0.2840E+09	0.797858E+02	0.773891E+02	0.751260E+02	0.743320E+02	0.735666E+02	0.733331E+02
0.2998E+09	0.794429E+02	0.770742E+02	0.748372E+02	0.740524E+02	0.732956E+02	0.730643E+02
0.3156E+09	0.790536E+02	0.767123E+02	0.745022E+02	0.737266E+02	0.729787E+02	0.727502E+02

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- 6.5 yr
peak


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3
4 Input Text File:
5   U1:(JWBERGL.USERS)SUMQ_3D.INP;130
6   Written on 04/01/93 09:58:37
7 Graph Data File:
8
9 Output Diagnostics/Debug File:
10  U1:(JWBERGL.USERS)SUMQ_3D.DBG
11 .....
12 CHO INPUT TEXT FILE:
13 .....
14 PROBLEM 'GRAPHITE_4 GRANITE'
15 XSTAG=1
16 YSTAG=0
17 DECAY=1
18 XSPACE=2.6
19 YSPACE=2.25
20 HEAT_RATE=407.
21 DIFFUS=1.052E-6
22 CONDUCT=2.56
23 TIME_INC=20
24 TOTAL_TIME=3.15567E8
25 X_CAN=1
26 Y_CAN=1
27 *X, Y, Z LOCATION OF POINTS OF INTEREST
28 POINT 1 0. 0.33 1.5
29 POINT 2 0. 0.5 1.5
30 POINT 3 0. 0.8 1.5
31 POINT 4 0. 1.0 1.5
32 POINT 5 0. 1.5 1.5
33 POINT 6 0. 2.0 1.5
34 CAN_SOURCES=50
35 CAN_LENGTH=2.997
36 .....

```

SUMQ_3D C-1.01VV (11/20/92)

04/01/93 09:58:44

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38 GRAPHITE_4 GRANITE
39 NUMBER OF CANISTERS ALONG X DIRECTION..... 1
40 CANISTER SPACING ALONG X DIRECTION..... 2.60000
41 NUMBER OF CANISTERS ALONG Y DIRECTION..... 1
42 CANISTER SPACING ALONG Y DIRECTION..... 2.25000
43 NUMBER OF HEAT SOURCES ALONG CANISTER LENGTH.. 50
44 LENGTH OF CANISTER..... 2.99700
45 DIFFUSIVITY..... 0.10520E-05
46 CONDUCTIVITY..... 2.56000
47 HEAT RATE PER CANISTER..... 0.40700E+03
48 AGE OF CANISTER WHEN PLACED IN REPOSITORY..... 0.00000E+00
49 TEMPERATURE DECAY..... T
50 STAGGER X COORDINATES..... T
51 STAGGER Y COORDINATES..... F

```

CANISTER # X COORD Y COORD

1 0.00 0.00

COORDINATES OF POINT OF INTEREST

POINT # X COORD Y COORD Z COORD

1 0.00 0.33 1.50

Appendix C: Scoping Calculations

2	0.00	0.50	1.50
3	0.00	0.80	1.50
4	0.00	1.00	1.50
5	0.00	1.50	1.50
6	0.00	2.00	1.50

T E M P E R A T U R E

	PT# 1 (X,Y,Z)	PT# 2 (X,Y,Z)	PT# 3 (X,Y,Z)	PT# 4 (X,Y,Z)	PT# 5 (X,Y,Z)	PT# 6 (X,Y,Z)
	(0.0000E+00	(0.0000E+00	(0.0000E+00	(0.0000E+00	(0.0000E+00	(0.0000E+00
	0.3300E+00	0.5000E+00	0.8000E+00	0.1000E+01	0.1500E+01	0.2000E+01
IME	0.1500E+01)	0.1500E+01)	0.1500E+01)	0.1500E+01)	0.1500E+01)	0.1500E+01)
0.1578E+08	0.168062E+02	0.134623E+02	0.985835E+01	0.826270E+01	0.565733E+01	0.410053E+01
0.3156E+08	0.171193E+02	0.138125E+02	0.102473E+02	0.866787E+01	0.608546E+01	0.453709E+01
0.4734E+08	0.171470E+02	0.138776E+02	0.103527E+02	0.879098E+01	0.623665E+01	0.470381E+01
0.6311E+08	0.170818E+02	0.138501E+02	0.103657E+02	0.882188E+01	0.629658E+01	0.478065E+01
0.7889E+08	0.169729E+02	0.137787E+02	0.103349E+02	0.880908E+01	0.631312E+01	0.481456E+01
0.9467E+08	0.168392E+02	0.136825E+02	0.102793E+02	0.877149E+01	0.630498E+01	0.482399E+01
0.1104E+09	0.166897E+02	0.135706E+02	0.102080E+02	0.871821E+01	0.628120E+01	0.481787E+01
0.1262E+09	0.165297E+02	0.134481E+02	0.101261E+02	0.865425E+01	0.624674E+01	0.480113E+01
0.1420E+09	0.163619E+02	0.133179E+02	0.100364E+02	0.858261E+01	0.620462E+01	0.477675E+01
0.1578E+09	0.161885E+02	0.131820E+02	0.994104E+01	0.850523E+01	0.615677E+01	0.474664E+01
0.1736E+09	0.160372E+02	0.130624E+02	0.985574E+01	0.843535E+01	0.611221E+01	0.471738E+01
0.1893E+09	0.158855E+02	0.129425E+02	0.977013E+01	0.836474E+01	0.606637E+01	0.468683E+01
0.2051E+09	0.157314E+02	0.128204E+02	0.968242E+01	0.829229E+01	0.601864E+01	0.465372E+01
0.2209E+09	0.155751E+02	0.126960E+02	0.959249E+01	0.821765E+01	0.596910E+01	0.461910E+01
0.2367E+09	0.154170E+02	0.125697E+02	0.950067E+01	0.814113E+01	0.591765E+01	0.458276E+01
0.2525E+09	0.152572E+02	0.124419E+02	0.940726E+01	0.806301E+01	0.586461E+01	0.454482E+01
0.2682E+09	0.150961E+02	0.123127E+02	0.931249E+01	0.798353E+01	0.581020E+01	0.450550E+01
0.2840E+09	0.149338E+02	0.121823E+02	0.921654E+01	0.790288E+01	0.575461E+01	0.446500E+01
0.2998E+09	0.147705E+02	0.120509E+02	0.911958E+01	0.782120E+01	0.569798E+01	0.442347E+01
0.3156E+09	0.146063E+02	0.119186E+02	0.902172E+01	0.773862E+01	0.564046E+01	

T E M P E R A T U R E

	PT# 1 (X,Y,Z)	PT# 2 (X,Y,Z)	PT# 3 (X,Y,Z)	PT# 4 (X,Y,Z)
	(0.0000E+00	(0.0000E+00	(0.0000E+00	(0.0000E+00
	0.2600E+01	0.5200E+01	0.7800E+01	0.1040E+02
	0.1500E+01)	0.1500E+01)	0.1500E+01)	0.1500E+01)
IME				
11	0.1578E+08	0.291885E+01	0.860090E+00	0.278053E+00
12	0.3156E+08	0.335446E+01	0.122563E+01	0.534897E+00
13	0.4734E+08	0.353117E+01	0.140149E+01	0.682795E+00
14	0.6311E+08	0.362017E+01	0.150442E+01	0.777753E+00
15	0.7889E+08	0.366699E+01	0.157039E+01	0.843292E+00
16	0.9467E+08	0.368965E+01	0.161459E+01	0.890590E+00
17	0.1104E+09	0.369692E+01	0.164464E+01	0.925665E+00
18	0.1262E+09	0.369367E+01	0.166489E+01	0.952072E+00
19	0.1420E+09	0.368283E+01	0.167801E+01	0.972063E+00
20	0.1578E+09	0.366629E+01	0.168573E+01	0.987142E+00
21	0.1736E+09	0.364879E+01	0.168999E+01	0.998532E+00
22	0.1893E+09	0.362988E+01	0.169200E+01	0.100736E+01
23	0.2051E+09	0.360856E+01	0.169174E+01	0.101399E+01
24	0.2209E+09	0.358499E+01	0.168946E+01	0.101867E+01
25	0.2367E+09	0.356014E+01	0.168550E+01	0.102181E+01
26	0.2525E+09	0.353378E+01	0.167998E+01	0.102347E+01
27	0.2682E+09	0.350605E+01	0.167312E+01	0.102394E+01
28	0.2840E+09	0.347714E+01	0.166509E+01	0.102331E+01
29	0.2998E+09	0.344719E+01	0.165634E+01	0.102170E+01
30	0.3156E+09	0.341634E+01	0.164672E+01	0.101924E+01
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U1:[JWBBERGL.USERS]SUMQ_3D.INP;127
 Written on 04/01/93 09:36:51
 Graph Data File:

Output Diagnostics/Debug File:
 U1:[JWBBERGL.USERS]SUMQ_3D.DBG

ECHO INPUT TEXT FILE:

PROBLEM 'GRAPHITE_4 GRANITE'
 XSTAG=1
 YSTAG=0
 DECAY=1
 XSPACE=2.6
 YSPACE=2.25
 HEAT_RATE=4481.
 DIFFUS=1.052E-6
 CONDUCT=2.56
 TIME_INC=20
 TOTAL_TIME=3.15567E8
 X_CAN=1
 Y_CAN=1
 *X, Y, Z LOCATION OF POINTS OF INTEREST
 POINT 1 0. 0.33 16.5
 POINT 2 0. 0.5 16.5
 POINT 3 0. 0.8 16.5
 POINT 4 0. 1.0 16.5
 POINT 5 0. 1.5 16.5
 POINT 6 0. 2.0 16.5
 POINT 7 0. 2.6 16.5
 POINT 8 0. 5.2 16.5
 CAN_SOURCES=100
 CAN_LENGTH=33.

1

SUMQ_3D C-1.01VV (11/20/92)

04/01/93 09:36:59

GRAPHITE_4 GRANITE
 NUMBER OF CANISTERS ALONG X DIRECTION..... 1
 CANISTER SPACING ALONG X DIRECTION..... 2.60000
 NUMBER OF CANISTERS ALONG Y DIRECTION..... 1
 CANISTER SPACING ALONG Y DIRECTION..... 2.25000
 NUMBER OF HEAT SOURCES ALONG CANISTER LENGTH.. 100
 LENGTH OF CANISTER..... 33.00000
 DIFFUSIVITY..... 0.10520E-05
 CONDUCTIVITY..... 2.56000
 HEAT RATE PER CANISTER..... 0.44810E+04
 AGE OF CANISTER WHEN PLACED IN REPOSITORY..... 0.00000E+00
 TEMPERATURE DECAY..... T
 STAGGER X COORDINATES..... T
 STAGGER Y COORDINATES..... F

CANISTER #	X COORD	Y COORD
1	0.00	0.00

COORDINATES OF POINT OF INTEREST			
POINT #	X COORD	Y COORD	Z COORD
1	0.00	0.33	16.50
2	0.00	0.50	16.50
3	0.00	0.80	16.50
4	0.00	1.00	16.50
5	0.00	1.50	16.50
6	0.00	2.00	16.50
7	0.00	2.60	16.50

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      8      0.00      5.20      16.50

      T E M P E R A T U R E

      PT# 1 (X,Y,Z) PT# 2 (X,Y,Z) PT# 3 (X,Y,Z) PT# 4 (X,Y,Z) PT# 5 (X,Y,Z) PT# 6 (X,Y,Z) P*
(0.0000E+00) (0.0000E+00) (0.0000E+00) (0.0000E+00) (0.0000E+00) (0.0000E+00) (0*
0.3300E+00 0.5000E+00 0.8000E+00 0.1000E+01 0.1500E+01 0.2000E+01 0*
TIME 0.1650E+02) 0.1650E+02) 0.1650E+02) 0.1650E+02) 0.1650E+02) 0.1650E+02) 0*
-----
0.1578E+08 0.244189E+02 0.209442E+02 0.170441E+02 0.152033E+02 0.118941E+02 0.959750E+01 0*
0.3156E+08 0.269895E+02 0.235497E+02 0.196819E+02 0.178510E+02 0.145417E+02 0.122190E+02 0*
0.4734E+08 0.282617E+02 0.248598E+02 0.210326E+02 0.192192E+02 0.159360E+02 0.136233E+02 0*
0.6311E+08 0.289890E+02 0.256257E+02 0.218409E+02 0.200470E+02 0.167963E+02 0.145028E+02 0*
0.7889E+08 0.294195E+02 0.260949E+02 0.223534E+02 0.205796E+02 0.173640E+02 0.150931E+02 0*
0.9467E+08 0.296657E+02 0.263800E+02 0.226821E+02 0.209287E+02 0.177494E+02 0.155028E+02 0*
0.1104E+09 0.297877E+02 0.265410E+02 0.228868E+02 0.211541E+02 0.180117E+02 0.157903E+02 0*
0.1262E+09 0.298212E+02 0.266135E+02 0.230032E+02 0.212912E+02 0.181860E+02 0.159903E+02 0*
0.1420E+09 0.297887E+02 0.266200E+02 0.230536E+02 0.213624E+02 0.182947E+02 0.161252E+02 0*
0.1578E+09 0.297055E+02 0.265759E+02 0.230534E+02 0.213831E+02 0.183530E+02 0.162098E+02 0*
0.1736E+09 0.296173E+02 0.265206E+02 0.230355E+02 0.213830E+02 0.183855E+02 0.162654E+02 0*
0.1893E+09 0.295065E+02 0.264430E+02 0.229951E+02 0.213601E+02 0.183945E+02 0.162972E+02 0*
0.2051E+09 0.293727E+02 0.263424E+02 0.229318E+02 0.213145E+02 0.183808E+02 0.163057E+02 0*
0.2209E+09 0.292187E+02 0.262216E+02 0.228484E+02 0.212489E+02 0.183473E+02 0.162947E+02 0*
0.2367E+09 0.290475E+02 0.260836E+02 0.227478E+02 0.211660E+02 0.182966E+02 0.162667E+02 0*
0.2525E+09 0.288614E+02 0.259306E+02 0.226323E+02 0.210683E+02 0.182311E+02 0.162239E+02 0*
0.2682E+09 0.286622E+02 0.257647E+02 0.225038E+02 0.209575E+02 0.181526E+02 0.161683E+02 0*
0.2840E+09 0.284517E+02 0.255874E+02 0.223639E+02 0.208354E+02 0.180628E+02 0.161012E+02 0*
0.2998E+09 0.282312E+02 0.254001E+02 0.222140E+02 0.207033E+02 0.179629E+02 0.160243E+02 0*
0.3156E+09 0.280018E+02 0.252039E+02 0.220552E+02 0.205623E+02 0.178542E+02 0.159384E+02 0*
[EOB]

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T E M P E R A T U R E					
	PT# 1 (X,Y,Z)	PT# 2 (X,Y,Z)	PT# 3 (X,Y,Z)	PT# 4 (X,Y,Z)	
	(0.0000E+00	(0.0000E+00	(0.0000E+00	(0.0000E+00	
	0.2600E+01	0.5200E+01	0.7800E+01	0.1040E+02	
IME	0.1650E+02)	0.1650E+02)	0.1650E+02)	0.1650E+02)	
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0.1578E+08	0.757060E+01	0.289724E+01	0.106241E+01	0.347675E+00	
0.3156E+08	0.101348E+02	0.499352E+01	0.253458E+01	0.124698E+01	
0.4734E+08	0.115371E+02	0.627613E+01	0.359305E+01	0.204541E+01	
0.6311E+08	0.124286E+02	0.714232E+01	0.436181E+01	0.268119E+01	
0.7889E+08	0.130364E+02	0.776282E+01	0.493917E+01	0.318557E+01	
0.9467E+08	0.134662E+02	0.822403E+01	0.538489E+01	0.359052E+01	
0.1104E+09	0.137753E+02	0.857494E+01	0.573605E+01	0.391991E+01	
0.1262E+09	0.139979E+02	0.884564E+01	0.601674E+01	0.419082E+01	
0.1420E+09	0.141558E+02	0.905577E+01	0.624325E+01	0.441556E+01	
0.1578E+09	0.142639E+02	0.921882E+01	0.642704E+01	0.460316E+01	
0.1736E+09	0.143402E+02	0.934653E+01	0.657704E+01	0.476054E+01	
0.1893E+09	0.143924E+02	0.944849E+01	0.670138E+01	0.489389E+01	
0.2051E+09	0.144216E+02	0.952809E+01	0.680443E+01	0.500752E+01	
0.2209E+09	0.144307E+02	0.958851E+01	0.688920E+01	0.510436E+01	
0.2367E+09	0.144230E+02	0.963239E+01	0.695841E+01	0.518669E+01	
0.2525E+09	0.144007E+02	0.966184E+01	0.701396E+01	0.525633E+01	
0.2682E+09	0.143656E+02	0.967863E+01	0.705764E+01	0.531501E+01	
0.2840E+09	0.143193E+02	0.968429E+01	0.709084E+01	0.536384E+01	
0.2998E+09	0.142630E+02	0.968054E+01	0.711469E+01	0.540399E+01	
0.3156E+09	0.141979E+02	0.966834E+01	0.713022E+01	0.543639E+01	

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Appendix D: Waste Parcel Design and Contents

Appendix D: Waste Parcel Design and Contents

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Appendix D: Waste Parcel Design and Contents

J. W. Berglund

The waste parcel consists of the waste form, internal backfill, waste canister, and overpack. This chapter discusses the types of waste canisters that were under consideration and defines those that were modeled in the performance assessment. A total of 26 types of waste parcels were modeled; the differences among the types of waste parcels generally were the form of the waste they enclosed and their size. In most calculations, the waste canister itself was stainless steel and an overpack was not modeled; exceptions to these conditions are described below. The contents (in curies) of the waste was computed for each type of waste parcel; these values are also provided.

D.1 Waste Parcel Design

The waste parcel consists of the waste form, waste canister, internal backfill, and overpack. The waste parcel serves as a physical barrier for waste containment. It must provide sufficient mechanical strength for waste transportation and handling operations, both during disposal and retrieval, should the latter be required. It must also be able to withstand the high temperature of the waste processing (e.g., glassification inside the canister or ceramic heater processes) (Molecke et al., 1981, p. 8). After emplacement, the waste parcel must be able to sustain the hydrostatic and lithostatic pressures associated with deep geologic disposal and the chemical corrosive effects of groundwater.

The waste parcel is a component of the engineered barrier (in conjunction with the backfill-buffer, drift-backfill, and seals). Requirements established for the engineered barrier system for spent fuel and high-level waste by 10 CFR 60 are summarized below (Hill, 1986, p. 3-14):

1. Contain the waste for 300 to 1000 yr.
2. Maintain a release rate less than 10^{-5} of the radionuclide inventory present at 1000 yr.
3. Maintain a retrievability for 50 yr after emplacement of the first waste package.
4. Control criticality so as not to exceed an effective multiplication factor (K_{eff}) of 0.95 unless at least two unlikely, independent, and concurrent or sequential changes occur.
5. Prevent release of radioactive material in excess of applicable federal standards under expected loads during or after transportation, handling, emplacement, retrieval, and seismic events. Further, these loads must not compromise long-term performance.
6. Retain legible, externally labeled identification as long as retrievability is required.

Design concepts from the Yucca Mountain Project (YMP) were investigated before selecting the waste parcels to be modeled in this performance assessment. Note that the YMP concepts described below include the entire *waste package*, i.e., the waste parcel (waste form, canister, internal backfill, overpack), and backfill (buffer) material between the waste parcel and the host rock.

D.1.1 Container Options under Consideration

Several canister concepts including corrosion-allowance designs, corrosion-resistant designs, and self-shielding designs were considered for disposal of spent fuel and high-level waste in a repository. Spain, for example (Nilsson et al., 1993), is considering the use of carbon-steel corrosion-allowance canisters that utilize the extensive manufacturing, processing, and material property data base available for this material. Sweden (Vogt, 1993) and Finland (Aikas, 1993), on the other hand, are planning to utilize a composite copper and steel canister (ACP canister). This design consists of an outer oxygen-free copper corrosion shield and an inner steel liner as a load-bearing element.

The Yucca Mountain Project (YMP) has developed an advanced conceptual design waste package (Doering, 1992), which involves several waste package options. These advanced conceptual design waste packages define a multi-barrier as two or more layers of corrosion allowance and/or corrosion resistant materials. Currently, the YMP

is considering seven waste package design concepts, and one alternative waste package design. The YMP advanced conceptual design considers: drift and/or borehole emplacement, metallic and/or composite barriers, multi-barriers, complete and/or partial radiation shielding, filler materials used as buffers and/or barriers, and backfill materials as waste package design parameters. The seven YMP concepts are described below.

Concept 1: Metallic Multi-Barrier. Concept 1 is intended to provide substantially complete containment for more than 1000 yr. Each barrier will not only provide a primary containment barrier, but will also provide backup to the other materials, a defense-in-depth design approach. Filler material(s) insulate the SNF so that the thermal design goal may be exceeded, possibly degrading the performance of the fuel rod cladding material. The backfill is another barrier in the defense-in-depth design philosophy. Features of Concept 1 include:

- Drift emplacement,
- Metallic barriers,
- Multi-barrier,
- Partially shielded,
- Capable of containing filler materials as buffers and/or barriers,
- Able to take advantage of backfill materials.

Concept 2: Metallic Multi-Barrier Totally Shielded. This design is intended to contain and lower ionizing radiation at the surface of the Waste Package (WP). Ionizing radiation shielding is massive, thus restricting the number of SNF assemblies per package. The total weight of WP is a design limit. Concept 2 disposes of less SNF per WP. Features of Concept 2 include:

- Drift emplacement,
- Metallic barriers,
- Multi-barrier,
- Ionizing radiation totally shielded,
- Capable of containing filler materials as buffers and/or barriers,
- Able to take advantage of backfill performance.

Concept 3: Small Metallic Multi-Barrier Borehole. Concept 3 was developed to be emplaced in a borehole (vertical) and is limited to smaller overall WP diameters (< 1 m). Features of Concept 3 include

- Borehole emplacement,
- Metallic barriers,
- Multi-barrier,
- No attempt to minimize ionizing radiation from SNF,
- Capable of containing filler materials as buffers and/or barriers,
- Not able to take advantage of drift backfill performance.

Concept 4: Non-Metallic Multi-Barrier. In this design, any one or all of the barriers could be a ceramic or composite material. Depending on the characteristics of the ceramic or composite material, ionizing radiation shielding qualities can vary. Thus shielding characteristics are material dependent. Features of Concept 4 include

- Borehole and/or drift emplacement,
- Ceramic and/or composite barriers,
- Multi-barrier,
- Material-dependent ionizing radiation shielding,
- Capable of containing filler materials as buffers and/or barriers,
- May take advantage of backfill performance.

Concept 5: Overpack Multi-Element Sealed Canister (MESC) (or Universal Basket). This system consists of several sealed SNF canisters bundled into an overpack or as a unit. Final canisterized waste would be overpacked with a containment barrier at the Mined Geological Disposal System (MGDS) for final storage. Features of Concept 5 include

- Drift emplacement,
- Metallic and/or ceramic barriers,
- Multi-barrier,
- May attempt to minimize ionizing radiation from the SNF,
- Not capable of containing filler materials as buffers and/or barriers,
- May take advantage of backfill performance.

Concept 6: Universal Cask-Waste Package. Concept 6 provides a single unit that can be used for on-site storage by the utilities, transportation to and storage at the Monitored Retrieval Site (MRS), transportation to the MGDS and final emplacement of the waste without further handling of the SNF assemblies. This package provides the least amount of waste disposal compared to overall weight. Concept 6 would be drift emplaced due to massive size and weight. Features of Concept 6 include

- Drift emplacement,
- Metallic,
- May be multi-barrier,
- May attempt to minimize ionizing radiation from the SNF,
- Not capable of containing filler materials as buffers and/or barriers,
- May take advantage of backfill performance.

Concept 7: Site Characterization Plan-Canister Design Reference (SCP-CDR) (Single Container). This design makes use of a thin-walled, vertically emplaced WP with an air gap between the packages and the rock wall. There are several design options in which the WP contains 3 Pressure Water Reactor (PWR) SNF assemblies, or 6 Boiling Water Reactor (BWR) SNF assemblies, or 6 consolidated PWR SNF assemblies, or 18 consolidated BWR SNF assemblies, or a mix of 3 PWR and 4 BWR fuel assemblies. Features of Concept 7 include

- Borehole and/or drift emplacement,
- Ceramic and/or composite barriers,
- Single container design,
- Ionizing radiation totally shielded,
- Capable of containing filler materials as buffers and/or barriers,
- May take advantage of backfill performance.

Concept 8: High-Level Waste Glass Container. Savannah River, Hanford, and West Valley generate HLW pour canisters. These glass-pour canisters have a thermal output ranging from 200 to 800 W. The ionizing radiation is greater than 30,000 Rads, thus shielding is a significant concern for operations and waste handling prior to emplacement. One Concept 8 option is to overpack the vitrified waste form with a thin-walled overpack, which could also be a corrosion-allowance material (YMP denotes this design as their reference DHLW package). Another Concept 8 option would be consolidation of four pour canisters into one overpack design.

Extensive material studies were conducted for use in YMP canister design. Alloys including austenitic stainless steel AISI 304L, austenitic stainless steel AISI 316L, austenitic nickel-based alloy 825, high purity copper CDA 102, copper-nickel alloy CDA 715, aluminum bronze CDA 613, nickel-based alloy C-4, and titanium grade 12 were all considered as potential candidates. Using peer-reviewed selection criteria, Alloy 825, Alloy C-4, and Titanium Grade 12 were judged to be suitable for a thin-walled canister design (Stahl and Harrison, 1993).

D.1.2 Waste Canisters

A single thin-walled type canister, similar to Concept 7 of the YMP advanced conceptual design, was selected as the most appropriate type for both the salt and granite repositories for this performance assessment.

Material Construction. The standard waste canister is assumed to be a 300-series stainless steel canister, with a wall thickness of 0.0095 m, similar to the YMP reference waste canister. Typical of spent fuel and high-level waste container vessels, the standard canisters are thin-walled cylindrical canisters with domed tops. The domed or curved head is a more efficient canister design and allows less material to be subjected to higher stress levels (Trabia and Kiley, 1992, p. 1252).

In one set of calculations, an Inconel canister was modeled to study the effects of highly corrosive-resistant canisters.

Size and Geometry. Diameters and lengths of the canisters vary depending on the waste type and form (see Figure D-1 and Appendix A).

Standard Thin-Shelled Canister. YMP has a standard canister design constructed of 300-series stainless steel. The dimensions – outside diameter, 0.66 m; length 4.766 m; wall thickness, 0.0095 m – provide sufficient strength to contain the various waste configurations, as discussed in the YMP advanced conceptual design Concept 7 (see Doering, 1992).

Defense Waste Processing Plant (Savannah River) Canister. Savannah River has designed a high-level canister to contain their borosilicate-glass (vitrified type) waste form. The waste package is similar to the standard, thin-shelled canister design of YMP but has different dimensions: 0.61 m, outside diameter; 2.99 m length; and wall thickness, 0.0095 m, constructed of 304L stainless steel (U.S. DOE, 1992, p. 3.3-4).

West Valley Demonstration Project (WVDP) Canister. West Valley has reprocessed spent fuel into a borosilicate-glass seal inside a 304L stainless steel canister (U.S. DOE, 1992, p. 3.2-3 to 3.2-4). These borosilicate-glass waste canisters are 0.61 m in diameter, 2.99 in length, with a wall thickness of 0.0034 m.

Hanford Waste Vitrification Plant (HWVP) Canister. Hanford has generated a similar borosilicate-glass waste form to West Valley as a byproduct of plutonium production operations. The canister dimensions are similar to the West Valley's: 0.61 m, outside diameter; 2.99 m, length; and 0.0095-m wall thickness (U.S. DOE, 1992, p. 3.4-4).

D.1.3 Waste Overpacks

The waste overpack is a container in which the waste canister is placed. The primary function of an overpack is to provide containment during the operating period (transfer, emplacement, and potential retrieval) and for thousands of years after repository closure (decommissioning). As a primary containment barrier, the overpack should be designed for at least a 1000-yr corrosion lifetime (Tang and Saling, 1990, p. 125). In addition, the overpack should also be designed (with sufficient thickness) to resist crushing from external, lithostatic loads with sufficient allowance to prevent failure from any corrosion processes (Kircher and Bradley, 1983, p. 384).

Steel, copper, nickel-alloys (such as Hastelloy), and titanium-alloys (such as Ti-Grade 12 and Ti-Grade 2) are common overpack materials of several proposed waste parcels in various geologic-type repositories. The overpack material must be compatible with the geological environment of the repository (i.e., granite, tuff, basalt, salt). Perhaps the most important material consideration for overpack longevity is resistance to corrosion (assuming the designed canister dimensions prevent structural failure from lithostatic loads and/or thermal-induced stresses). Overpack corrosion is a result of groundwater leaching into the repository. Although a backfill-buffer material such as a bentonite clay/sand mixture can provide protection of the overpack from moisture, should the overpack come in contact with groundwater, corrosion resistance is critical to maintain isolation of the waste from the environment.

Although overpacks were not modeled in the majority of the performance assessment calculations, one calculation set did model overpacks to study the effects of highly corrosive-resistant containers. Materials investigated for these calculations are described below.

Figure D-1. Relative sizes for 10 different waste canisters considered in the performance assessment (dimensions in m).

Material Options for Overpacks. Steel-, nickel-, and titanium-based materials were examined.

Steel Overpack. Mild and cast iron steels are engineering materials that are well understood and possess desirable qualities suited to overpack designs. Because of the abundance of data and the good understanding of the behavior of steels, there remains high confidence that a steel overpack can be designed to withstand an engineered life (Nowak, 1980). The philosophy behind steel overpack designs is that of "corrosion allowance." Proposed corrosion allowance canister designs range from 0.0025-m to 0.030-m wall thickness. Overpack wall thicknesses of 0.030 m or greater also provide radiation shielding (Doering, 1992). However, such massive designs would be costly and introduce new problems involving fabrication, handling, and transportation (Nowak, 1980).

Nickel Overpack. Recently, a comprehensive corrosion study was performed on several overpack materials subjected to salt brines (Smailos and Köster, 1992, p. 1676-1680). The most promising passively corroding materials of the study were the titanium-alloys (99.8%-Ti 0.2%-Pd) and the nickel-based alloys (Hastelloy C4, Ni 19.4-Cr 13.8-Mo). Of the three brines considered, the Mg-Cl₂ rich brines had observable pitting damage on the Hastelloy C4 (Smailos and Köster, 1992, p. 1678). Other corrosion studies involving similar brine solutions also indicate that Hastelloy C-276 is susceptible to both pitting and crevice corrosion (Molecke et al., 1981).

Ti-Grade 12 Overpack. Recently, the Ti-Code12 (98.9%-Ti 0.8%-Ni 0.3%-Mo) alloy has received great attention with regard to waste containment in salt repositories. Ti-alloys would provide a virtually noncorrosive canister with a relatively thin, 0.001-m to 0.006-m wall thickness (Nowak, 1980). The superior high-temperature corrosion-resistance of Ti-Code12 to uniform and localized attack in saline solutions is due to the formation of an adherent, compact, and protective oxide film (Molecke et al., 1981, p. 17). No significant problems in the behavior of Grade-12 Ti-alloy have been measured in the laboratory, which would preclude its usage for nuclear waste containers in salt, or elsewhere (Molecke and Van Den Avyle, 1988, p. 20). Other corrosion tests involving Ti-alloys in salt that were subjected to high temperature (25 to 250 °C) and varied brine solution compositions (bedded type, dome type, and sea water) also have shown that both TiCode-12 and 99.8%-Ti 0.2%Pd were free of any pitting and crevice corrosion (Molecke et al., 1981, p. 16). More recently, a 5-year WIPP brine (at 90° C) immersion test on several overpack materials revealed that Ti-alloys exhibited very little corrosion (Sorensen and Molecke, 1993, p. 1). Other research suggests that titanium alloys are the most durable and cost-effective materials for use in a high-level nuclear waste repository in salt (Molecke and Van Den Avyle, 1988, p. 1). Because of titanium's extremely high resistivity to corrosion, it is generally agreed that titanium-alloys are the best suited overpack material of a waste parcel in a salt repository. Perhaps the only significant drawback to selecting Ti-alloy as an overpack reinforcement is that few data exist regarding Ti-alloy behavior when subjected to long-term tests, thus limiting our ability to predict its response in salt (Nowak, 1980).

Overpack Material for Salt and Granite Repositories. The waste overpack design chosen for the salt repository consists of a thin-walled layer of Ti-Code12 (98.9%-Ti 0.8%-Mo 0.3%-Ni). The purpose of the Ti-Code12 overpack is to provide enhanced corrosion resistance and durability, providing barrier protection for a long period of time. The overpack is not intended to provide structural support to the waste-form canister. Due to fabrication tolerances (out of roundness) of waste canister and waste canister deformations (localized bulging, nonlinearity over the length of the canister) caused by nonuniform heating during waste filling, a gap of about 12 to 25 mm may be necessary between the canister and overpack (Molecke et al., 1981, p. 31). This gap can then be filled with a molten metal, such as lead, that provides additional corrosion resistance and adds to structural strength.

The overpack material selected for the granite repository was Inconel (a nickel-based material).

D.1.4 Canister Internal Backfill

Internal canister backfill materials can be used to provide additional structural strength to waste canisters and fill any void space. A backfill material between the waste canister and the overpack provides transfer of stresses and distributes the load from the overpack to the waste canister (which is designed to withstand large stress fields associated with underground repositories, i.e., lithostatic pressure). Void space is generally undesirable because the heat of the

1 waste form might cause space filled by air or other gases to create an internal pressure on the waste canister that
2 would act against the external lithostatic loads already imposed on the canister/overpack. Other repository designs
3 offer different ideas on the function and role of canister internal backfill materials. The Swedish project proposes to
4 fill its copper waste canisters with spent fuel assemblies and use either lead or argon gas as an internal backfill (SKB
5 1992, p. 49-50). The YMP approach places SNF "hardware" (i.e., fuel rod spacers, pins, etc.) in the center section of
6 the canister. The YMP reference HLW canister design utilizes special fuel dividers to allow spent fuel "hardware" to
7 be separated from the rods inside the canister. The Canadians propose to use a granular particulate material (such as
8 fine-grain high-silica glass beads) as filler between a bundle of steel-encapsulated spent fuel rods and an ASTM-B265
9 Grade-2 titanium container (Teper, 1991, p. 961). The Canadian approach uses the titanium canister for corrosion
10 protection and external loads are transferred to the internal steel basket structure and surrounding (internal to the can-
11 ister) granular materials.
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15 For the proposed generic repositories, the waste parcel will include an internal backfill of a lead-based alloy.
16 This internal backfill allows transfer of pressure to the solid, self-supporting waste form container made of steel.*
17 Thus the waste parcel will act as a solid body in the repository. It is envisioned that a liquid lead alloy poured
18 between the overpack and waste canister would be in contact with the stainless steel and titanium for only a short
19 time before solidification occurs, thus minimizing corrosion effects associated of a liquid metal (Molecke et al.,
20 1981, p. 31). Goodwin and Westerman (1992, p.1681) describe an internal filler material between waste canister and
21 overpack as a "redundant barrier." These authors further state that the combination of a waste form canister, redun-
22 dant barrier, and overpack greatly increases the statistical confidence in the integrity of the waste parcel.
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63 * For the salt repository, the lithostatic pressure is the result of salt creep. In the saturated granite repository, pressures are due to both hydrostatic
64 pressures and confining stresses that are the result of swelling of the bentonite clay backfill-buffer.
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D.2 Waste Parcel Types

A total of 26 types of waste parcels were designed for this performance assessment based on information about the waste form and waste treatment options (Appendix A and Chapter 4). In general, the differences among the types of waste parcels were the form of the waste they enclosed, the size of the canister and/or overpack, and whether they were designed for the salt repository, granite repository, or both. Illustrations of each waste parcel are provided in Chapter 4.

Parameters and values for the layers of each waste parcel are provided on the following pages[†]. An explanation of the parameter names follows:

General

WDG	=	waste disposal group
Count	=	number of this type of waste parcel in repository
Nlayers	=	number of layers in this type of waste parcel
Nmatrix	=	number of matrices in this type of waste parcel

Layer

AreaL1	=	area of layer
CompL1	=	composition of layer
Note: 1 = titanium		
2 = lead		
3 = Inconel		
4 = stainless steel 304L		
5 = aluminum		
6 = uranium aluminite (UA13)		
7 = zircaloy		
8 = stainless steel HIP (Hot Isostatic Press)		
9 = graphite		
10 = metal carbide (MC ₂)		
DiamL1	=	diameter of layer (note: input for Layer 1 only)
LengthL1	=	length of layer (note: input for Layer 1 only)
MassL1	=	mass of layer

Note: L1 = Layer 1, L2 = Layer 2, etc.

Matrix

AreaM1	=	area of matrix
CompM1	=	composition of matrix
Note: 1 = glass		
2 = graphite		
3 = uranium dioxide (UO ₂)		
4 = calcine		
5 = glass-ceramic		
6 = uranium aluminite (UA13)		
7 = metal carbide (MC ₂)		

[†] Although the use of more than three significant digits in the parameter values may indicate a certainty beyond what is reasonable, we have maintained the data as it is stored in the database so that data output in subsequent calculations can be traced back to its original form.

1 CovLayM1 = The number of the metallic layer that covers the matrix, e.g., if CovLayM1 = 3.000, then
2 layer 3 covers matrix 1 (and therefore, layer 3 must be breached before matrix 1 is avail-
3 able).
4 InFrAl11 = fraction of the inventory that is contained in the total number of waste parcels of this type;
5 the numbers in the parameter name correspond to the inventory identification number
6 (InvID) (see below). For example, InFrAl11 is the fraction of the inventory identified as
7 parameter InvId11.
8 InvId11 = identification of the waste form in the matrix; the first number refers to the matrix number;
9 the second signifies the inventory identified. The inventories include the following:
10 1 = graphite
11 2 = highly enriched uranium
12 3 = low enriched uranium
13 4 = calcine
14 5 = high-level waste from Savannah River
15 6 = high-level waste from Hanford
16 7 = high-level waste from West Valley
17 8 = depleted uranium
18 For example, if InvId11 equals 3.0000, then the inventory 11 is for matrix 1 and is low
19 enriched uranium.
20 MassM1 = mass of matrix

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26 Note: M1 = Matrix 1, M2 = Matrix 2
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D.2.1 Waste Parcel 1

Waste Parcel 1 is in Waste Disposal Group 1. This waste parcel contains whole block graphite spent nuclear fuel from Fort St. Vrain and Peach Bottom and is intended for disposal in a salt repository.

WP01 - Block graphite in salt, 5FSV + 3.62PB

WDG	=	1
Count	=	0442
Nlayers	=	04
Nmatrix	=	02

Layer 1 - Outer Canister

AreaL1	=	8.4600 m ²
CompL1	=	4.0000 = sst 304
DiamL1	=	.6096 m
LengthL1	=	4.1148 m
MassL1	=	694.0000 kg

Layer 2 - Fuel Canisters -aluminum

AreaL2	=	1.5100 m ²
CompL2	=	5.0000 = Al
MassL2	=	5.5000 kg

Layer 3 - Graphite

AreaL3	=	18.7500 m ²
CompL3	=	9.0000 = Graphite
MassL3	=	569.0000 kg

Layer 4 - Interior of particles

CompL4	=	10.0000 = MC2
MassL4	=	82.6400 kg

Matrix 1 - graphite

AreaM1	=	18.7500 m ²
CompM1	=	2.0000 = Graphite
CovLayM1	=	1.0000
InFrAl11	=	0.00
InFrC_11	=	.9300
InvId11	=	1.0000 = Graphite
MassM1	=	569.0000 kg

Matrix 2 - MC

CompM2	=	7.0000 = MC2
CovLayM2	=	3.0000
InFrAl21	=	1.0000
InFrC_21	=	.0700
InvId21	=	1.0000 = Graphite
MassM2	=	69.2000 kg

D.2.2 Waste Parcel 2

Waste Parcel 2 is in Waste Disposal Group 1. This waste parcel contains whole or cut block graphite spent nuclear fuel from Fort St. Vrain and Peach Bottom and is intended for disposal in a granite repository.

WP02 - Single block FSV in granite

WDG	=	1
Count	=	2368
Nlayers	=	03
Nmatrix	=	02

Layer 1 - Outer canister

AreaL1	=	1.7500 m ²
CompL1	=	4.0000 = sst 304
DiamL1	=	.4576 m
LengthL1	=	.9860 m
MassL1	=	170.0000 kg

Layer 2 - Graphite

AreaL2	=	5.2000 m ²
CompL2	=	9.0000 = Graphite
MassL2	=	90.0000 kg

Layer 3 - Interior of particles

CompL3	=	10.0000 = MC2
MassL3	=	15.0800 kg

Matrix 1 - graphite

AreaM1	=	5.2000 m ²
CompM1	=	2.0000 = Graphite
CovLayM1	=	1.0000
InFrAl11	=	0.0
InFrC_11	=	.9300
InvId11	=	1.0000 = Graphite
MassM1	=	90.0000 kg

Matrix 2 - MC

CompM2	=	7.0000 = MC2
CovLayM2	=	2.0000
InFrAl21	=	1.0000
InFrC_21	=	.0700
InvId21	=	1.0000 = Graphite
MassM2	=	12.4000 kg

D.2.3 Waste Parcel 3

Waste Parcel 3 is in Waste Disposal Group 1. This waste parcel contains fuel assemblies from Peach Bottom and is intended for disposal in a granite repository.

WP03 - Dual elements PB in granite

WDG	=	1
Count	=	0800
Nlayers	=	04
Nmatrix	=	02

Layer 1 - Outer canister

AreaL1	=	3.5600 m ²
CompL1	=	4.0000 = sst 304
DiamL1	=	.2731 m
LengthL1	=	4.0090 m
MassL1	=	279.0000 kg

Layer 2 - Fuel canisters - aluminum

AreaL2	=	3.6200 m ²
CompL2	=	5.0000 = Al
MassL2	=	11.0000 kg

Layer 3 - Graphite

AreaL3	=	2.0400 m ²
CompL3	=	9.0000 = Graphite
MassL3	=	66.0000 kg

Layer 4 - Interior of particles

CompL4	=	10.0000 = MC2
MassL4	=	4.0000 kg

Matrix 1 - graphite

AreaM1	=	2.0400 m ²
CompM1	=	2.0000 = Graphite
CovLayM1	=	2.0000
InFrAl11	=	0.0
InFrC_11	=	.9300
InvId11	=	1.0000 = Graphite
MassM1	=	66.0000 kg

Matrix 2 - MC

CompM2	=	7.0000 = MC2
CovLayM2	=	3.0000
InFrAl21	=	1.0000
InFrC_21	=	.0700
InvId21	=	1.0000 = Graphite
MassM2	=	4.0000 kg

D.2.4 Waste Parcel 4

Waste Parcel 4 is in Waste Disposal Group 1. This waste parcel contains a single fuel assembly in lead from special highly enriched uranium spent nuclear fuel and is intended for disposal in both salt and granite repositories.

WP04 - Spent fuel single highly enriched uranium element in lead

WDG	=	1
Count	=	5064
Nlayers	=	04
Nmatrix	=	01

Layer 1 - Outer canister

AreaL1	=	.8900 m ²
CompL1	=	4.0000 = sst 304
DiamL1	=	.1683 m
LengthL1	=	1.6000 m
MassL1	=	63.0000 kg

Layer 2 - Canister fill

AreaL2	=	.8500 m ²
CompL2	=	2.0000 = Pb
MassL2	=	163.8000 kg

Layer 3 - Clad

AreaL3	=	3.8320 m ²
CompL3	=	5.0000 = Al
MassL3	=	1.2016 kg

Layer 4 - UA14 highly enriched U fuel

AreaL4	=	3.7200 m ²
CompL4	=	6.0000 = UA13
MassL4	=	1.0100 kg

Matrix 1 - UA14

AreaM1	=	6.1700 m ²
CompM1	=	6.0000 = UA13
CovLayM1	=	3.0000
InFrAl11	=	1.0000
InvId11	=	2.0000 = Heu
MassM1	=	1.0100 kg

D.2.5 Waste Parcel 5

Waste Parcel 5 is in Waste Disposal Group 1. This waste parcel contains a single fuel assembly in lead from special low enriched uranium spent nuclear fuel and is intended for disposal in both salt and granite repositories.

WP05 - Spent fuel single low enriched uranium element in lead

WDG	=	1
Count	=	0888
Nlayers	=	04
Nmatrix	=	01

Layer 1 - Outer canister

AreaL1	=	4.5600 m ²
CompL1	=	4.0000 = sst 304
DiamL1	=	.3239 m
LengthL1	=	4.3430 m
MassL1	=	366.0000 kg

Layer 2 - Canister fill

AreaL2	=	4.4400 m ²
CompL2	=	2.0000 = Pb
MassL2	=	3454.0000 kg

Layer 3 - Clad

AreaL3	=	31.8250 m ²
CompL3	=	7.0000 = Zircaloy
MassL3	=	72.5000 kg

Matrix 1 - UO₂

AreaM1	=	90.6000 m ²
CompM1	=	3.0000 = UO ₂
CovLayM1	=	3.0000
InFrAl11	=	1.0000
InvId11	=	3.0000 = Leu
MassM1	=	508.2000 kg

D.2.6 Waste Parcel 6

Waste Parcel 6 is in Waste Disposal Group 1. This waste parcel contains calcine high-level waste as a loose powder and is intended for disposal in both salt and granite repositories.

WP06 - Loose calcine

WDG	=	1
Count	=	6945
Nlayers	=	01
Nmatrix	=	01

Layer 1 - Outer canister

AreaL1	=	6.3200 m ²
CompL1	=	4.0000 = sst 304
DiamL1	=	.6096 m
LengthL1	=	2.9970 m
MassL1	=	500.0000 kg

Matrix 1 - loose

CompM1	=	4.0000 = Loose
CovLayM1	=	1.0000
InFrAl11	=	1.0000
InvId11	=	4.0000 = Calcine
MassM1	=	857.0000 kg

D.2.7 Waste Parcel 7

Waste Parcel 7 is in all disposal groups. This waste parcel contains high-level waste in borosilicate glass from the Defense Waste Processing Project at Savannah River, Georgia, and is intended for disposal in both salt and granite repositories.

WP07 - Savannah River HLW

WDG	=	All
Count	=	5283
Nlayers	=	01
Nmatrix	=	01

Layer 1 - Outer canister

AreaL1	=	6.3200 m ²
CompL1	=	4.0000 = sst 304
DiamL1	=	.6096 m
LengthL1	=	2.9970 m
MassL1	=	500.0000 kg

Matrix 1 - glass

AreaM1	=	5.6400 m ²
CompM1	=	1.0000 = Glass
CovLayM1	=	1.0000
InFrAl11	=	1.0000
Invld11	=	5.0000 = SRS
MassM1	=	1900.0000 kg

1 D.2.8 Waste Parcel 8

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4 Waste Parcel 8 is in all disposal groups. This waste parcel contains high-level waste in borosilicate glass from
5 Hanford Waste Vittrification Project at Hanford, Washington, and is intended for disposal in both salt and granite
6 repositories.
7

8 WP08 - Hanford HLW

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10 WDG = All
11 Count = 1960
12 Nlayers = 01
13 Nmatrix = 01
14

15 Layer 1 - Outer canister

16
17 AreaL1 = 6.3200 m²
18 CompL1 = 4.0000 = sst 304
19 DiamL1 = .6096 m
20 LengthL1 = 2.9970 m
21 MassL1 = 500.0000 kg
22
23

24 Matrix 1 - glass

25
26 AreaM1 = 5.6400 m²
27 CompM1 = 1.0000 = Glass
28 CovLayM1 = 1.0000
29 InFrAl11 = 1.0000
30 InvId11 = 6.0000 = Hanford
31 MassM1 = 1900.0000 kg
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D.2.9 Waste Parcel 9

Waste Parcel 9 is in all disposal groups. This waste parcel contains high-level waste in borosilicate glass from West Valley Development Project, at West Valley, New York, and is intended for disposal in both salt and granite repositories.

WP09 - West Valley Development Project

WDG	=	All
Count	=	0276
Nlayers	=	01
Nmatrix	=	01

Layer 1 - Outer canister

AreaL1	=	6.3200 m ²
CompL1	=	4.0000 = sst 304
DiamL1	=	.6096 m
LengthL1	=	2.9970 m
MassL1	=	500.0000 kg

Matrix 1 - glass

AreaM1	=	5.6400 m ²
CompM1	=	1.0000 = Glass
CovLayM1	=	1.0000
InFrAl11	=	1.0000
InvId11	=	7.0000 = WVDP
MassM1	=	1900.0000 kg

D.2.10 Waste Parcel 10

Waste Parcel 10 is in Waste Disposal Group 2. This waste parcel contains graphite spent fuel compacts (fuel rods) in clean glass from Fort St. Vrain and Peach Bottom reactors and is intended for disposal in both salt and granite repositories.

WP10 - Graphite compacts in capsule in glass

WDG	=	2
Count	=	1801
Nlayers	=	04
Nmatrix	=	02

Layer 1 - Outer Canister

AreaL1	=	2.6200 m ²
CompL1	=	4.0000 = sst 304
DiamL1	=	.2032 m
LengthL1	=	4.0090 m
MassL1	=	147.0000 kg

Layer 2 - Capsule

AreaL2	=	1.8900 m ²
CompL2	=	4.0000 = sst 304
MassL2	=	116.7900 kg

Layer 3 - Graphite

AreaL3	=	8.5500 m ²
CompL3	=	9.0000 = Graphite
MassL3	=	3.7500 kg

Layer 4 - Interior of particles

CompL4	=	10.0000 = MC2
MassL4	=	15.7700 kg

Matrix 1 - graphite

AreaM1	=	8.5500 m ²
CompM1	=	2.0000 = Graphite
CovLayM1	=	2.0000
InFrAl11	=	0.0
InFrC_11	=	.2000
InvId11	=	1.0000 = Graphite
MassM1	=	24.0000 kg

Matrix 2 - MC

CompM2	=	7.0000 = MC2
CovLayM2	=	3.0000
InFrAl21	=	1.0000
InFrC_21	=	.0700
InvId21	=	1.0000 = Graphite
MassM2	=	15.7700 kg

D.2.11 Waste Parcel 11

Waste Parcel 11 is in Waste Disposal Group 2. This waste parcel contains high-level waste in glass ceramic within 18 Hot Isostatic Press (HIP) packages and is intended for disposal in both salt and granite repositories.

WP11 - Calcine in glass ceramic: 18 HIPs

WDG	=	2
Count	=	0990
Nlayers	=	02
Nmatrix	=	01

Layer 1 - Outer Canister

AreaL1	=	6.3200 m ²
CompL1	=	4.0000 = sst 304
DiamL1	=	.6096 m
LengthL1	=	2.9970 m
MassL1	=	500.0000 kg

Layer 2 - 18 Steel HIP packages

AreaL2	=	15.3000 m ²
CompL2	=	4.0000 = sst 304
MassL2	=	111.8400 kg

Matrix 1 - glass ceramic

AreaM1	=	15.3000 m ²
CompM1	=	5.0000 = GCeramic
CovLayM1	=	2.0000
InFrAl11	=	.2050
InvId11	=	4.0000 = Calcine
MassM1	=	2065.0000 kg

D.2.12 Waste Parcel 12

Waste Parcel 12 is in Waste Disposal Group 2. This waste parcel contains a single fuel assembly from special highly enriched uranium spent fuel in 10 stainless steel Hot Isostatic Press (HIP) packages and is intended for disposal in both salt and granite repositories.

WP12 - Highly enriched SF in 10 glass ceramic HIPs

WDG	=	2
Count	=	5064
Nlayers	=	06
Nmatrix	=	02

Layer 1 - Outer Canister

AreaL1	=	3.9900 m ²
CompL1	=	4.0000 = sst 304
DiamL1	=	.6096 m
LengthL1	=	1.7780 m
MassL1	=	328.0000 kg

Layer 2 - 10 Steel HIPs

AreaL2	=	8.5000 m ²
CompL2	=	4.0000 = sst 304
MassL2	=	62.1400 kg

Layer 3 - Canister

AreaL3	=	1.0400 m ²
CompL3	=	8.0000 = HIP SSt
MassL3	=	79.5000 kg

Layer 4 - Lead Fill

AreaL4	=	.7830 m ²
CompL4	=	2.0000 = Pb
MassL4	=	263.0000 kg

Layer 5 - Al Clad

AreaL5	=	3.8200 m ²
CompL5	=	5.0000 = Al
MassL5	=	1.2010 kg

Layer 6 - UAl Fuel

AreaL6	=	3.7200 m ²
CompL6	=	6.0000 = UAl3
MassL6	=	1.0100 kg

Appendix D: Waste Parcel Design and Contents

1 Matrix 1 - glass ceramic
2 AreaM1 = 9.3500 m²
3 CompM1 = 5.0000 = GCeramic
4 CovLayM1 = 2.0000
5 InFrAl11 = .5811
6 InvId11 = 4.0000 = Calcine
7 MassM1 = 962.0000 kg
8
9

10
11 Matrix 2 - UA14 fuel
12 AreaM2 = 6.1700 m²
13 CompM2 = 6.0000 = UA13
14 CovLayM2 = 5.0000
15 InFrAl21 = 1.0000
16 InvId21 = 2.0000 = HEU
17 MassM2 = 1.0100 kg
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D.2.13 Waste Parcel 13

Waste Parcel 13 is in Waste Disposal Group 2. This waste parcel contains a single fuel assembly from special low enriched uranium spent fuel in 28 stainless steel Hot Isostatic Press (HIP) packages and is intended for disposal in both salt and granite repositories.

WP13 - Low enriched SF in 28 glass ceramic HIP units

WDG	=	2
Count	=	0888
Nlayers	=	05
Nmatrix	=	02

Layer 1 - Outer Canister

AreaL1	=	9.2400 m ²
CompL1	=	4.0000 = sst 304
DiamL1	=	.6096 m
LengthL1	=	4.5210 m
MassL1	=	715.0000 kg

Layer 2 - 28 Steel HIPs

AreaL2	=	22.4800 m ²
CompL2	=	4.0000 = sst 304
MassL2	=	164.3300 kg

Layer 3 - Canister

AreaL3	=	4.8600 m ²
CompL3	=	8.0000 = HIP SSt
MassL3	=	371.3400 kg

Layer 4 - Lead fill

AreaL4	=	4.4400 m ²
CompL4	=	2.0000 = Pb
MassL4	=	3454.0000 kg

Layer 5 - Zircaloy clad

AreaL5	=	31.8300 m ²
CompL5	=	7.0000 = Zircaloy
MassL5	=	72.5000 kg

Matrix 1 - glass ceramic

AreaM1	=	27.3400 m ²
CompM1	=	5.0000 = GCeramic
CovLayM1	=	2.0000
InFrAl11	=	.2143
InvId11	=	4.0000 = Calcine
MassM1	=	2023.0000 kg

Appendix D: Waste Parcel Design and Contents

Matrix 2 - UO₂ fuel

AreaM2	=	90.6000 m ²
CompM2	=	3.0000 = UO ₂
CovLayM2	=	5.0000
InFrAl21	=	1.0000
InvId21	=	3.0000 = LEU
MassM2	=	588.2000 kg

D.2.14 Waste Parcel 14

Waste Parcel 14 is in Waste Disposal Group 3. This waste parcel contains compacts of pulverized graphite from Fort St. Vrain and Peach Bottom reactors that has been burned off and emplaced in glass and is intended for disposal in a salt repository.

WP14 - Graphite particles in salt only

WDG	=	3
Count	=	0152
Nlayers	=	03
Nmatrix	=	02

Layer 1 - Outer Canister

AreaL1	=	9.2400 m ²
CompL1	=	4.0000 = sst 304
DiamL1	=	.6096 m
LengthL1	=	4.5200 m
MassL1	=	715.0000 kg

Layer 2 - Capsule

AreaL2	=	3.6700 m ²
CompL2	=	4.0000 = sst 304
MassL2	=	186.2600 kg

Layer 3 - Interior of particles

CompL3	=	10.0000 = MC2
MassL3	=	204.3000 kg

Matrix 1 - glass

AreaM1	=	1.8500 m ²
CompM1	=	1.0000 = Glass
CovLayM1	=	1.0000
InFrAl11	=	.0120
InvId11	=	4.0000 = Calcine
MassM1	=	2782.0000 kg

Matrix 2 - Graphite particles

CompM2	=	4.0000 = Loose
CovLayM2	=	2.0000
InFrAl21	=	1.0000
InFrC_21	=	.0700
InvId21	=	1.0000 = Graphite
MassM2	=	204.3000 kg

D.2.15 Waste Parcel 15

Waste Parcel 15 is in Waste Disposal Group 3. This waste parcel contains compacts of pulverized graphite from Fort St. Vrain and Peach Bottom reactors that has been burned off and emplaced in glass and is intended for disposal in a granite repository.

WP15 - Graphite particles in capsules in glass: granite only

WDG	=	3
Count	=	1801
Nlayers	=	03
Nmatrix	=	02

Layer 1 - Outer Canister

AreaL1	=	9.2400 m ²
CompL1	=	4.0000 = sst 304
DiamL1	=	.6096 m
LengthL1	=	4.5200 m
MassL1	=	715.0000 kg

Layer 2 - Capsule

AreaL2	=	.8000 m ²
CompL2	=	4.0000 = sst 304
MassL2	=	18.7100 kg

Layer 3 - Interior of particles

CompL3	=	10.0000 = MC2
MassL3	=	17.2400 kg

Matrix 1 - glass

AreaM1	=	.4200 m ²
CompM1	=	1.0000 = Glass
CovLayM1	=	1.0000
InFrAl11	=	.1840
InvId11	=	4.0000 = Calcine
MassM1	=	3383.0000 kg

Matrix 2 - Graphite particles

CompM2	=	4.0000 = Loose
CovLayM2	=	2.0000
InFrAl21	=	1.0000
InFrC_21	=	.0700
InvId21	=	1.0000 = Graphite
MassM2	=	17.2400 kg

D.2.16 Waste Parcel 16

Waste Parcel 16 is in Waste Disposal Group 3. This waste parcel contains 16 special highly enriched uranium spent fuel assemblies in glass and is intended for disposal in a salt repository.

WP16 - Highly enriched SF in glass - salt only

WDG	=	3
Count	=	0633
Nlayers	=	05
Nmatrix	=	02

Layer 1 - Outer Canister

AreaL1	=	8.2700 m ²
CompL1	=	4.0000 = sst 304
DiamL1	=	.6096 m
LengthL1	=	4.0120 m
MassL1	=	643.0000 kg

Layer 2 - Fuel canister - stainless

AreaL2	=	3.9900 m ²
CompL2	=	4.0000 = sst 304
MassL2	=	258.7000 kg

Layer 3 - Lead fill

AreaL3	=	3.5020 m ²
CompL3	=	2.0000 = Pb
MassL3	=	3307.0000 kg

Layer 4 - UAl Clad

AreaL4	=	61.3000 m ²
CompL4	=	5.0000 = Al
MassL4	=	19.2000 kg

Layer 5 - UAl Fuel

AreaL5	=	59.5000 m ²
CompL5	=	6.0000 = UAl3
MassL5	=	16.1600 kg

Matrix 1 - glass

AreaM1	=	272.0000 m ²
CompM1	=	1.0000 = Glass
CovLayM1	=	1.0000
InFrAl11	=	.0350
InvId11	=	4.0000 = Calcine
MassM1	=	2236.0000 kg

Appendix D: Waste Parcel Design and Contents

1	Matrix 2 - UA14 fuel		
2	AreaM2	=	6.1700 m ²
3	CompM2	=	6.0000 = UA13
4	CovLayM2	=	4.0000
5	InFrAl21	=	1.0000
6	InvId21	=	2.0000 = HEU
7	MassM2	=	1.0100 kg
8			
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1 D.2.17 Waste Parcel 17

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4 Waste Parcel 17 is in Waste Disposal Group 3. This waste parcel contains three special low enriched uranium
5 spent fuel assemblies in glass and is intended for disposal in a salt repository.
6

7 WP17 - Low enriched SF in glass - salt only

8 WDG = 3
9 Count = 0296
10 Nlayers = 04
11 Nmatrix = 02
12
13

14 Layer 1 - Outer Canister

15 AreaL1 = 9.7300 m²
16 CompL1 = 4.0000 = sst 304
17 DiamL1 = .6096 m
18 LengthL1 = 4.7740 m
19 MassL1 = 750.0000 kg
20
21
22

23 Layer 2 - Fuel cannisters

24 AreaL2 = 13.8100 m²
25 CompL2 = 4.0000 = sst 304
26 MassL2 = 917.5000 kg
27
28
29

30 Layer 3 - Lead Fill

31 AreaL3 = 12.6800 m²
32 CompL3 = 2.0000 = Pb
33 MassL3 = 1677.0000 kg
34
35
36

37 Layer 4 - Zr Clad

38 AreaL4 = 95.4570 m²
39 CompL4 = 7.0000 = Zircaloy
40 MassL4 = 217.5000 kg
41
42
43

44 Matrix 1 - glass

45 AreaM1 = 377.0000 m²
46 CompM1 = 1.0000 = Glass
47 CovLayM1 = 1.0000
48 InFrAl11 = .0130
49 InvId11 = 4.0000 = Calcine
50 MassM1 = 1482.0000 kg
51
52
53

54 Matrix 2 - UO2 fuel

55 AreaM2 = 272.0000 m²
56 CompM2 = 3.0000 = UO2
57 CovLayM2 = 4.0000
58 InFrAl21 = 1.0000
59 InvId21 = 3.0000 = Leu
60 MassM2 = 1524.6000 kg
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63
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D.2.18 Waste Parcel 18

Waste Parcel 18 is in Waste Disposal Group 3. This waste parcel contains high-level waste in Idaho Chemical Processing Plant glass and is intended for disposal in a salt repository.

WP18 - Left over ICPP glass - salt only

WDG	=	3
Count	=	5238
Nlayers	=	01
Nmatrix	=	01

Layer 1 - Outer Canister

AreaL1	=	6.3200 m ²
CompL1	=	4.0000 = sst 304
DiamL1	=	.6096 m
LengthL1	=	2.9970 m
MassL1	=	500.0000 kg

Matrix 1 - glass

AreaM1	=	5.6370 m ²
CompM1	=	1.0000 = Glass
CovLayM1	=	1.0000
InFrA111	=	.9400
InvId11	=	4.0000 = Calcine
MassM1	=	2274.0000 kg

D.2.19 Waste Parcel 19

Waste Parcel 19 is in Waste Disposal Group 3. This waste parcel contains a single special highly enriched uranium spent fuel assembly in glass ceramic and is intended for disposal in a granite repository.

WP19 - Highly enriched SF in glass - granite only

WDG	=	3
Count	=	5064
Nlayers	=	05
Nmatrix	=	02

Layer 1 - Outer Canister

AreaL1	=	3.9900 m ²
CompL1	=	4.0000 = sst 304
DiamL1	=	.6096 m
LengthL1	=	1.7780 m
MassL1	=	328.0000 kg

Layer 2 - Fuel Canister

AreaL2	=	.8500 m ²
CompL2	=	4.0000 = sst 304
MassL2	=	163.8000 kg

Layer 3 - Lead Fill

AreaL3	=	.7830 m ²
CompL3	=	2.0000 = Pb
MassL3	=	263.0000 kg

Layer 4 - Al Clad

AreaL4	=	3.8320 m ²
CompL4	=	5.0000 = Al
MassL4	=	1.2016 kg

Layer 5 - UAl Fuel

AreaL5	=	3.7200 m ²
CompL5	=	6.0000 = UAl3
MassL5	=	1.0100 kg

Matrix 1 - glass

AreaM1	=	272.0000 m ²
CompM1	=	1.0000 = Glass
CovLayM1	=	1.0000
InFrAl11	=	.1930
InvId11	=	4.0000 = Calcine
MassM1	=	1261.0000 kg

Appendix D: Waste Parcel Design and Contents

Matrix 2 - UA14 fuel

AreaM2	=	6.1700 m ²
CompM2	=	6.0000 = UA13
CovLayM2	=	4.0000
InFrAl21	=	1.0000
InvId21	=	2.0000 = HEU
MassM2	=	1.0100 kg

D.2.20 Waste Parcel 20

Waste Parcel 20 is in Waste Disposal Group 3. This waste parcel contains special low enriched uranium spent fuel assemblies in glass and is intended for disposal in a granite repository.

WP20 - Low enriched SF in glass - granite only

WDG	=	3
Count	=	0888
Nlayers	=	04
Nmatrix	=	02

Layer 1 - Outer Canister

AreaL1	=	9.2400 m ²
CompL1	=	4.0000 = sst 304
DiamL1	=	.6096 m
LengthL1	=	4.5210 m
MassL1	=	715.0000 kg

Layer 2 - Fuel Canister

AreaL2	=	4.5600 m ²
CompL2	=	4.0000 = sst 304
MassL2	=	366.0000 kg

Layer 3 - Lead Fill

AreaL3	=	4.4400 m ²
CompL3	=	2.0000 = Pb
MassL3	=	34548.0000 kg

Layer 4 - Zr Clad

AreaL4	=	31.8750 m ²
CompL4	=	7.0000 = Zircaloy
MassL4	=	72.5000 kg

Matrix 1 - glass

AreaM1	=	509.0000 m ²
CompM1	=	1.0000 = Glass
CovLayM1	=	1.0000
InFrAl11	=	.0670
InvId11	=	4.0000 = Calcine
MassM1	=	2518.0000 kg

Matrix 2 - UO2 fuel

AreaM2	=	90.6000 m ²
CompM2	=	3.0000 = UO2
CovLayM2	=	4.0000
InFrAl21	=	1.0000
InvId21	=	3.0000 = Leu
MassM2	=	508.2000 kg

D.2.21 Waste Parcel 21

Waste Parcel 21 is in Waste Disposal Group 3. This waste parcel contains high-level waste in Idaho Chemical Processing Plant glass and is intended for disposal in a granite repository.

WP21 - Left over ICPP glass - granite only

WDG	=	3
Count	=	2136
Nlayers	=	01
Nmatrix	=	01

Layer 1 - Outer Canister

AreaL1	=	6.3200 m ²
CompL1	=	4.0000 = sst 304
DiamL1	=	.6096 m
LengthL1	=	2.9970 m
MassL1	=	500.0000 kg

Matrix 1 - glass

AreaM1	=	5.6370 m ²
CompM1	=	1.0000 = Glass
CovLayM1	=	1.0000
InFrAl11	=	.5560
InvId11	=	4.0000 = Calcine
MassM1	=	1891.0000 kg

D.2.22 Waste Parcel 22

Waste Parcel 22 is in Waste Disposal Group 4. This waste parcel contains reprocessed graphite spent fuel from Fort St. Vrain and Peach Bottom reactors and calcine in glass ceramic within 18 Hot Isostatic Press packages and is intended for disposal in both salt and granite repositories.

WP22 - Reprocessed graphite and calcine in glass ceramic

WDG	=	4
Count	=	0038
Nlayers	=	02
Nmatrix	=	01

Layer 1 - Outer Canister

AreaL1	=	6.3200 m ²
CompL1	=	4.0000 = sst 304
DiamL1	=	.6096 m
LengthL1	=	2.9970 m
MassL1	=	500.0000 kg

Layer 2 - Steel HIP package

AreaL2	=	13.6000 m ²
CompL2	=	8.0000 = HIP SSt
MassL2	=	99.8000 kg

Matrix 1 - glass ceramic

AreaM1	=	13.6000 m ²
CompM1	=	5.0000 = GCeramic
CovLayM1	=	2.0000
InFrAl11	=	1.0000
InFrAl12	=	.0075
InFrPu11	=	.0050
InFrU_11	=	.0050
InFrC_11	=	0
InvId11	=	1.0000 = Graphite
InvId12	=	4.0000 = Calcine
MassM1	=	1902.1000 kg

D.2.23 Waste Parcel 23

Waste Parcel 23 is in Waste Disposal Group 4. This waste parcel contains reprocessed special spent fuel, low and highly enriched uranium, and calcine in glass ceramic within 18 Hot Isostatic Press packages and is intended for disposal in both salt and granite repositories.

WP23 - Reprocessed SF (HEU & LEU) and calcine in glass ceramic

WDG	=	4
Count	=	5061
Nlayers	=	02
Nmatrix	=	01

Layer 1 - Outer Canister

AreaL1	=	6.3200 m ²
CompL1	=	4.0000 = sst 304
DiamL1	=	.6096 m
LengthL1	=	2.9970 m
MassL1	=	500.0000 kg

Layer 2 - Steel HIP package

AreaL2	=	13.6000 m ²
CompL2	=	8.0000 = HIP SSt
MassL2	=	99.8000 kg

Matrix 1 - glass ceramic

AreaM1	=	13.6000 m ²
CompM1	=	5.0000 = GCeramic
CovLayM1	=	2.0000
InFrAl11	=	.9925
InFrAl12	=	1.0000
InFrAl13	=	1.0000
InFrC_12	=	0.0
InFrC_13	=	0.0
InFrPu12	=	.0050
InFrPu13	=	.0050
InFrU_12	=	.0050
InFrU_13	=	.0050
InvId11	=	4.0000 = Calcine
InvId12	=	2.0000 = Heu
InvId13	=	3.0000 = Leu
MassM1	=	1902.0000 kg

D.2.24 Waste Parcel 24

Waste Parcel 24 is in Waste Disposal Group 5. This waste parcel contains graphite spent fuel from Fort St. Vrain and Peach Bottom reactors processed and mixed with depleted uranium in glass and is intended for disposal in both salt and granite repositories.

WP24 - Graphite processed and mixed with DU in glass

WDG	=	5
Count	=	0202
Nlayers	=	01
Nmatrix	=	01

Layer 1 - Outer Canister

AreaL1	=	6.3200 m ²
CompL1	=	4.0000 = sst 304
DiamL1	=	.6096 m
LengthL1	=	2.9970 m
MassL1	=	500.0000 kg

Matrix 1 - glass

AreaM1	=	5.6370 m ²
CompM1	=	1.0000 = Glass
CovLayM1	=	1.0000
InFrAl11	=	1.0000
InFrAl12	=	1.7300
InFrC	=	0
InvId11	=	1.0000 = Graphite
InvId12	=	8.0000 = DU
MassM1	=	2383.3000 kg

D.2.25 Waste Parcel 25

Waste Parcel 25 is in Waste Disposal Group 5. This waste parcel contains special spent fuel, low and highly enriched uranium, processed and mixed with depleted uranium in glass and is intended for disposal in both salt and granite repositories.

WP25 - Spent fuel reprocessed and mixed with DU in glass

WDG	=	5
Count	=	1408
Nlayers	=	01
Nmatrix	=	01

Layer 1 - Outer Canister

AreaL1	=	6.3200 m ²
CompL1	=	4.0000 = sst 304
DiamL1	=	.6096 m
LengthL1	=	2.9970 m
MassL1	=	500.0000 kg

Matrix 1 - glass

AreaM1	=	5.6400 m ²
CompM1	=	1.0000 = Glass
CovLayM1	=	1.0000
InFrAl11	=	1.0000
InFrAl12	=	1.0000
InFrAl13	=	10.2800
InvId11	=	2.0000 = Heu
InvId12	=	3.0000 = Leu
InvId13	=	8.0000 = DU
MassM1	=	2383.3000 kg

D.2.26 Waste Parcel 26

Waste Parcel 26 is in Waste Disposal Group 5. This waste parcel contains calcine high-level waste in glass and is intended for disposal in both salt and granite repositories.

WP26 - INEL calcine in glass

WDG	=	5
Count	=	7301
Nlayers	=	01
Nmatrix	=	01

Layer 1 - outer canister

AreaL1	=	6.3200 m ²
CompL1	=	4.0000 = sst 304
DiamL1	=	.6096 m
LengthL1	=	2.9970 m
MassL1	=	500.0000 kg

Matrix 1 - glass

AreaM1	=	5.6370 m ²
CompM1	=	1.0000 = Glass
CovLayM1	=	1.0000
InFrAl11	=	1.0000
InvId11	=	4.0000 = Calcine
MassM1	=	2274.0000 kg

D.3 Waste Parcel Contents

Intrusions at six time intervals were used to discretize the 10,000-yr regulatory period for calculating release through drill cuttings. Tables D-1 through D-26 list the contents, in curies, of each waste parcel at the time of emplacement, and after 125 yr, 175 yr, 350 yr, 1000 yr, 3000 yr, and 7250 yr.[‡]

Figures D-2a through D-2h are plots of the normalized activity decay histories of all the radionuclides per waste type, based on 40 CFR 191 criterion.

[‡] Although the use of more than three significant digits in the parameter values may indicate a certainty beyond what is reasonable, we have maintained the data as it is stored in the database so that data output in subsequent calculations can be traced back to its original form.

Table D-1. Radioisotope Release History (Ci) for WP01 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	2.9280E-01	2.8840E-01	2.8666E-01	2.8065E-01	2.5942E-01	2.0364E-01	1.2174E-01
³⁶ Cl	3.2870E-03	3.2861E-03	3.2857E-03	3.2843E-03	3.2794E-03	3.2644E-03	3.2325E-03
⁵⁹ Ni	2.2940E-02	2.2913E-02	2.2903E-02	2.2866E-02	2.2729E-02	2.2312E-02	2.1452E-02
⁶³ Ni	4.9550E-01	2.0080E-01	1.3991E-01	3.9504E-02	3.6037E-04	1.9062E-10	0.0000E+00
⁷⁹ Se	3.5630E-02	3.5583E-02	3.5564E-02	3.5497E-02	3.5252E-02	3.4508E-02	3.2978E-02
⁸⁷ Rb	2.1220E-06	2.1220E-06	2.1220E-06	2.1220E-06	2.1220E-06	2.1220E-06	2.1220E-06
⁹⁰ Sr	2.3190E+03	1.1832E+02	3.5987E+01	5.5846E-01	1.0648E-07	0.0000E+00	0.0000E+00
⁹³ Zr	1.4110E+00	1.4109E+00	1.4109E+00	1.4108E+00	1.4104E+00	1.4091E+00	1.4064E+00
⁹⁹ Tc	8.9210E-01	8.9174E-01	8.9159E-01	8.9108E-01	8.8920E-01	8.8343E-01	8.7128E-01
¹⁰⁷ Pd	1.0470E-03	1.0470E-03	1.0470E-03	1.0470E-03	1.0469E-03	1.0467E-03	1.0462E-03
¹¹³ Cd	4.8420E-15	4.8420E-15	4.8420E-15	4.8420E-15	4.8420E-15	4.8420E-15	4.8420E-15
^{113m} Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	1.5530E-02	1.5517E-02	1.5511E-02	1.5492E-02	1.5423E-02	1.5210E-02	1.4768E-02
¹²⁹ I	2.6700E-03	2.6700E-03	2.6700E-03	2.6700E-03	2.6699E-03	2.6696E-03	2.6691E-03
¹³⁵ Cs	1.6970E-02	1.6969E-02	1.6969E-02	1.6968E-02	1.6965E-02	1.6955E-02	1.6933E-02
¹³⁷ Cs	2.5410E+03	1.4148E+02	4.4563E+01	7.8154E-01	2.3473E-07	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	2.3850E-06	2.3850E-06	2.3850E-06	2.3850E-06	2.3850E-06	2.3850E-06	2.3850E-06
¹⁴⁷ Pm	0.0000E+00	7.6846E-03	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	7.6830E-03	7.6830E-03	7.6830E-03	7.6830E-03	7.6830E-03	7.6830E-03	7.6830E-03
¹⁵¹ Sm	6.8510E+01	2.6142E+01	1.7782E+01	4.6153E+00	3.0794E-02	6.2214E-09	0.0000E+00
¹⁵² Eu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁶ Ra	0.0000E+00	1.9941E-05	3.8725E-05	1.5133E-04	1.1268E-03	7.8293E-03	2.9250E-02
²²⁸ Ra	0.0000E+00	6.1628E-03	6.1628E-03	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	0.0000E+00	6.1319E-02	8.5635E-02	1.6980E-01	4.6995E-01	1.2807E+00	2.5458E+00
²³⁰ Th	0.0000E+00	7.6223E-04	1.0712E-03	2.1594E-03	6.2033E-03	1.8458E-02	4.3581E-02
²³² Th	6.1650E-03	6.1650E-03	6.1650E-03	6.1650E-03	6.1650E-03	6.1650E-03	6.1650E-03
²³¹ Pa	0.0000E+00	8.2038E-06	1.1479E-05	2.2916E-05	6.5027E-05	1.9103E-04	4.4171E-04
²³² U	2.6000E+00	7.8001E-01	4.8189E-01	8.9314E-02	1.7060E-04	7.3444E-13	0.0000E+00
²³³ U	5.2260E+00	5.2232E+00	5.2220E+00	5.2180E+00	5.2033E+00	5.1581E+00	5.0635E+00
²³⁴ U	6.6700E-01	6.8562E-01	6.8917E-01	6.9448E-01	6.9508E-01	6.9116E-01	6.8287E-01
²³⁵ U	3.1060E-03	3.1060E-03	3.1060E-03	3.1061E-03	3.1062E-03	3.1067E-03	3.1076E-03
²³⁶ U	2.5720E-02	2.5722E-02	2.5722E-02	2.5725E-02	2.5732E-02	2.5753E-02	2.5785E-02
²³⁸ U	7.1560E-05	7.1560E-05	7.1560E-05	7.1560E-05	7.1560E-05	7.1560E-05	7.1560E-05
²³⁷ Np	1.8520E-02	1.8675E-02	1.8729E-02	1.8886E-02	1.9199E-02	1.9352E-02	1.9333E-02
²³⁸ Pu	8.3760E+01	3.1204E+01	2.1022E+01	5.2761E+00	3.1083E-02	5.7086E-09	0.0000E+00
²³⁹ Pu	2.5950E-01	2.5869E-01	2.5836E-01	2.5722E-01	2.5302E-01	2.4038E-01	2.1503E-01
²⁴⁰ Pu	4.6700E-01	4.6256E-01	4.6012E-01	4.5167E-01	4.2158E-01	3.4103E-01	2.1731E-01
²⁴¹ Pu	0.0000E+00	2.8965E-04	2.8911E-04	2.8508E-04	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	0.0000E+00	1.9321E-08	2.6949E-08	5.3207E-08	1.4497E-07	3.7754E-07	6.9107E-07
²⁴⁴ Pu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴¹ Am	4.2470E+00	3.4756E+00	3.2078E+00	2.4230E+00	8.5456E-01	3.4811E-02	2.0867E-04
^{242m} Am	5.3440E-04	3.0208E-04	2.4045E-04	1.0819E-04	5.5705E-06	6.0526E-10	2.2832E-18
²⁴³ Am	3.2850E-02	3.2466E-02	3.2313E-02	3.1785E-02	2.9899E-02	2.4769E-02	1.6603E-02
²⁴³ Cm	1.8730E-03	8.9410E-05	2.6481E-05	3.7438E-07	5.0505E-14	0.0000E+00	0.0000E+00
²⁴⁴ Cm	6.2710E-01	5.2427E-03	7.7348E-04	9.5402E-07	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	2.9280E-04	2.8983E-04	2.8865E-04	2.8456E-04	2.6986E-04	2.2923E-04	1.6206E-04
²⁴⁶ Cm	8.4710E-05	8.3172E-05	8.2564E-05	8.0473E-05	7.3158E-05	5.4566E-05	2.9263E-05
²⁴⁷ Cm	2.5110E-10	2.5110E-10	2.5110E-10	2.5110E-10	2.5109E-10	2.5107E-10	2.5102E-10
TOTAL	5.0296E+03	3.3110E+02	1.3269E+02	2.3351E+01	1.0716E+01	1.0452E+01	1.1369E+01

Table D-2. Radioisotope Release History (Ci) for WP02 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	5.4650E-02	5.3829E-02	5.3504E-02	5.2383E-02	4.8419E-02	3.8008E-02	2.2722E-02
³⁶ Cl	5.4000E-04	5.3984E-04	5.3978E-04	5.3956E-04	5.3876E-04	5.3628E-04	5.3105E-04
⁵⁹ Ni	3.7680E-03	3.7636E-03	3.7619E-03	3.7558E-03	3.7333E-03	3.6649E-03	3.5236E-03
⁶³ Ni	8.1390E-02	3.2983E-02	2.2981E-02	6.4889E-03	5.9194E-05	3.1311E-11	0.0000E+00
⁷⁹ Se	5.8530E-03	5.8452E-03	5.8421E-03	5.8312E-03	5.7909E-03	5.6686E-03	5.4173E-03
⁸⁷ Rb	3.4860E-07	3.4860E-07	3.4860E-07	3.4860E-07	3.4860E-07	3.4860E-07	3.4860E-07
⁹⁰ Sr	3.8090E+02	1.9434E+01	5.9110E+00	9.1729E-02	1.7490E-08	0.0000E+00	0.0000E+00
⁹³ Zr	2.3180E-01	2.3179E-01	2.3178E-01	2.3176E-01	2.3169E-01	2.3148E-01	2.3104E-01
⁹⁹ Tc	1.4650E-01	1.4644E-01	1.4642E-01	1.4633E-01	1.4602E-01	1.4508E-01	1.4308E-01
¹⁰⁷ Pd	1.7200E-04	1.7200E-04	1.7200E-04	1.7199E-04	1.7198E-04	1.7194E-04	1.7187E-04
¹¹³ Cd	7.9530E-16	7.9530E-16	7.9530E-16	7.9530E-16	7.9530E-16	7.9530E-16	7.9530E-16
^{113m} Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	2.5510E-03	2.5488E-03	2.5479E-03	2.5448E-03	2.5334E-03	2.4985E-03	2.4259E-03
¹²⁹ I	4.3850E-04	4.3850E-04	4.3850E-04	4.3849E-04	4.3848E-04	4.3844E-04	4.3836E-04
¹³⁵ Cs	2.7880E-03	2.7879E-03	2.7879E-03	2.7877E-03	2.7872E-03	2.7855E-03	2.7819E-03
¹³⁷ Cs	4.1730E+02	2.3235E+01	7.3185E+00	1.2835E-01	3.8549E-08	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	3.9170E-07	3.9170E-07	3.9170E-07	3.9170E-07	3.9170E-07	3.9170E-07	3.9170E-07
¹⁴⁷ Pm	0.0000E+00	1.2623E-03	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	1.2620E-03	1.2620E-03	1.2620E-03	1.2620E-03	1.2620E-03	1.2620E-03	1.2620E-03
¹⁵¹ Sm	1.1250E+01	4.2928E+00	2.9200E+00	7.5788E-01	5.0567E-03	1.0216E-09	0.0000E+00
¹⁵² Eu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁶ Ra	0.0000E+00	3.2767E-06	6.3633E-06	2.4866E-05	1.8516E-04	1.2865E-03	4.8063E-03
²²⁸ Ra	0.0000E+00	1.0126E-03	1.0126E-03	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	0.0000E+00	1.0072E-02	1.4066E-02	2.7891E-02	7.7192E-02	2.1035E-01	4.1816E-01
²³⁰ Th	0.0000E+00	1.2525E-04	1.7602E-04	3.5483E-04	1.0193E-03	3.0330E-03	7.1610E-03
²³² Th	1.0130E-03	1.0130E-03	1.0130E-03	1.0130E-03	1.0130E-03	1.0130E-03	1.0130E-03
²³¹ Pa	0.0000E+00	1.3476E-06	1.8856E-06	3.7643E-06	1.0682E-05	3.1379E-05	7.2557E-05
²³² U	4.2700E-01	1.2810E-01	7.9141E-02	1.4668E-02	2.8017E-05	1.2062E-13	0.0000E+00
²³³ U	8.5840E-01	8.5793E-01	8.5775E-01	8.5709E-01	8.5467E-01	8.4725E-01	8.3171E-01
²³⁴ U	1.0960E-01	1.1266E-01	1.1324E-01	1.1411E-01	1.1421E-01	1.1357E-01	1.1221E-01
²³⁵ U	5.1020E-04	5.1021E-04	5.1021E-04	5.1021E-04	5.1024E-04	5.1032E-04	5.1047E-04
²³⁶ U	4.2250E-03	4.2253E-03	4.2254E-03	4.2257E-03	4.2270E-03	4.2305E-03	4.2356E-03
²³⁸ U	1.1750E-05	1.1750E-05	1.1750E-05	1.1750E-05	1.1750E-05	1.1750E-05	1.1750E-05
²³⁷ Np	3.0420E-03	3.0675E-03	3.0763E-03	3.1022E-03	3.1536E-03	3.1787E-03	3.1755E-03
²³⁸ Pu	1.3760E+01	5.1261E+00	3.4535E+00	8.6674E-01	5.1062E-03	9.3778E-10	0.0000E+00
²³⁹ Pu	4.2620E-02	4.2486E-02	4.2433E-02	4.2246E-02	4.1556E-02	3.9481E-02	3.5316E-02
²⁴⁰ Pu	7.6700E-02	7.5970E-02	7.5570E-02	7.4182E-02	6.9241E-02	5.6010E-02	3.5691E-02
²⁴¹ Pu	0.0000E+00	4.7572E-05	4.7485E-05	4.6822E-05	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	0.0000E+00	3.1726E-09	4.4252E-09	8.7369E-09	2.3805E-08	6.1994E-08	1.1348E-07
²⁴⁴ Pu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴¹ Am	6.9750E-01	5.7081E-01	5.2683E-01	3.9793E-01	1.4035E-01	5.7172E-03	3.4272E-05
^{242m} Am	8.7780E-05	4.9619E-05	3.9496E-05	1.7771E-05	9.1500E-07	9.9419E-11	3.7504E-19
²⁴³ Am	5.3960E-03	5.3329E-03	5.3079E-03	5.2211E-03	4.9113E-03	4.0686E-03	2.7272E-03
²⁴³ Cm	3.0760E-04	1.4684E-05	4.3489E-06	6.1484E-08	8.2943E-15	0.0000E+00	0.0000E+00
²⁴⁴ Cm	1.0300E-01	8.6111E-04	1.2704E-04	1.5670E-07	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	4.8090E-05	4.7602E-05	4.7408E-05	4.6736E-05	4.4322E-05	3.7649E-05	2.6617E-05
²⁴⁶ Cm	1.3910E-05	1.3657E-05	1.3558E-05	1.3214E-05	1.2013E-05	8.9602E-06	4.8053E-06
²⁴⁷ Cm	4.1250E-11	4.1250E-11	4.1250E-11	4.1249E-11	4.1248E-11	4.1245E-11	4.1237E-11
TOTAL	8.2607E+02	5.4386E+01	2.1800E+01	3.8417E+00	1.7660E+00	1.7214E+00	1.8703E+00

Table D-3. Radioisotope Release History (Ci) for WP03 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	1.6170E-01	1.5927E-01	1.5831E-01	1.5499E-01	1.4326E-01	1.1246E-01	6.7231E-02
³⁶ Cl	2.1790E-04	2.1784E-04	2.1781E-04	2.1772E-04	2.1740E-04	2.1640E-04	2.1429E-04
⁵⁹ Ni	1.5210E-03	1.5192E-03	1.5185E-03	1.5161E-03	1.5070E-03	1.4794E-03	1.4224E-03
⁶³ Ni	3.2850E-02	1.3312E-02	9.2754E-03	2.6190E-03	2.3892E-05	1.2638E-11	0.0000E+00
⁷⁹ Se	2.3620E-03	2.3589E-03	2.3576E-03	2.3532E-03	2.3369E-03	2.2876E-03	2.1862E-03
⁸⁷ Rb	1.4070E-07	1.4070E-07	1.4070E-07	1.4070E-07	1.4070E-07	1.4070E-07	1.4070E-07
⁹⁰ Sr	1.5380E+02	7.8470E+00	2.3867E+00	3.7038E-02	7.0622E-09	0.0000E+00	0.0000E+00
⁹³ Zr	9.3570E-02	9.3565E-02	9.3563E-02	9.3555E-02	9.3528E-02	9.3443E-02	9.3263E-02
⁹⁹ Tc	5.9140E-02	5.9116E-02	5.9106E-02	5.9073E-02	5.8948E-02	5.8565E-02	5.7760E-02
¹⁰⁷ Pd	6.9440E-05	6.9439E-05	6.9439E-05	6.9437E-05	6.9433E-05	6.9418E-05	6.9386E-05
¹¹³ Cd	3.2100E-16	3.2100E-16	3.2100E-16	3.2100E-16	3.2100E-16	3.2100E-16	3.2100E-16
^{113m} Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	1.0300E-03	1.0291E-03	1.0288E-03	1.0275E-03	1.0229E-03	1.0088E-03	9.7949E-04
¹²⁹ I	1.7700E-04	1.7700E-04	1.7700E-04	1.7700E-04	1.7699E-04	1.7698E-04	1.7694E-04
¹³⁵ Cs	1.1250E-03	1.1250E-03	1.1249E-03	1.1249E-03	1.1247E-03	1.1240E-03	1.1225E-03
¹³⁷ Cs	1.6840E+02	9.3764E+00	2.9534E+00	5.1795E-02	1.5556E-08	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	1.5810E-07	1.5810E-07	1.5810E-07	1.5810E-07	1.5810E-07	1.5810E-07	1.5810E-07
¹⁴⁷ Pm	0.0000E+00	5.0951E-04	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	5.0940E-04	5.0940E-04	5.0940E-04	5.0940E-04	5.0940E-04	5.0940E-04	5.0940E-04
¹⁵¹ Sm	4.5420E+00	1.7331E+00	1.1789E+00	3.0598E-01	2.0415E-03	4.1246E-10	0.0000E+00
¹⁵² Eu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁶ Ra	0.0000E+00	1.3221E-06	2.5674E-06	1.0033E-05	7.4706E-05	5.1906E-04	1.9392E-03
²²⁸ Ra	0.0000E+00	4.0856E-04	4.0856E-04	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	0.0000E+00	4.0656E-03	5.6779E-03	1.1258E-02	3.1159E-02	8.4911E-02	1.6879E-01
²³⁰ Th	0.0000E+00	5.0533E-05	7.1020E-05	1.4316E-04	4.1126E-04	1.2237E-03	2.8893E-03
²³² Th	4.0870E-04	4.0870E-04	4.0870E-04	4.0870E-04	4.0870E-04	4.0870E-04	4.0870E-04
²³¹ Pa	0.0000E+00	5.4384E-07	7.6098E-07	1.5191E-06	4.3107E-06	1.2663E-05	2.9282E-05
²³² U	1.7230E-01	5.1690E-02	3.1934E-02	5.9188E-03	1.1305E-05	4.8671E-14	0.0000E+00
²³³ U	3.4650E-01	3.4631E-01	3.4624E-01	3.4597E-01	3.4499E-01	3.4200E-01	3.3573E-01
²³⁴ U	4.4220E-02	4.5454E-02	4.5690E-02	4.6042E-02	4.6081E-02	4.5822E-02	4.5272E-02
²³⁵ U	2.0590E-04	2.0590E-04	2.0590E-04	2.0591E-04	2.0592E-04	2.0595E-04	2.0601E-04
²³⁶ U	1.7050E-03	1.7051E-03	1.7052E-03	1.7053E-03	1.7058E-03	1.7072E-03	1.7093E-03
²³⁸ U	4.7440E-06	4.7440E-06	4.7440E-06	4.7440E-06	4.7440E-06	4.7440E-06	4.7440E-06
²³⁷ Np	1.2280E-03	1.2383E-03	1.2418E-03	1.2523E-03	1.2730E-03	1.2832E-03	1.2819E-03
²³⁸ Pu	5.5530E+00	2.0687E+00	1.3937E+00	3.4978E-01	2.0607E-03	3.7846E-10	0.0000E+00
²³⁹ Pu	1.7200E-02	1.7146E-02	1.7125E-02	1.7049E-02	1.6771E-02	1.5933E-02	1.4253E-02
²⁴⁰ Pu	3.0960E-02	3.0665E-02	3.0504E-02	2.9943E-02	2.7949E-02	2.2608E-02	1.4407E-02
²⁴¹ Pu	0.0000E+00	1.9201E-05	1.9166E-05	1.8898E-05	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	0.0000E+00	1.2809E-09	1.7866E-09	3.5274E-09	9.6110E-09	2.5029E-08	4.5816E-08
²⁴⁴ Pu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴¹ Am	2.8150E-01	2.3037E-01	2.1262E-01	1.6060E-01	5.6642E-02	2.3074E-03	1.3833E-05
^{242m} Am	3.5430E-05	2.0027E-05	1.5941E-05	7.1727E-06	3.6931E-07	4.0128E-11	1.5138E-19
²⁴³ Am	2.1780E-03	2.1525E-03	2.1424E-03	2.1074E-03	1.9824E-03	1.6422E-03	1.1008E-03
²⁴³ Cm	1.2420E-04	5.9289E-06	1.7559E-06	2.4826E-08	3.3490E-15	0.0000E+00	0.0000E+00
²⁴⁴ Cm	4.1580E-02	3.4762E-04	5.1285E-05	6.3256E-08	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	1.9410E-05	1.9213E-05	1.9135E-05	1.8864E-05	1.7889E-05	1.5196E-05	1.0743E-05
²⁴⁶ Cm	5.6160E-06	5.5140E-06	5.4737E-06	5.3351E-06	4.8502E-06	3.6176E-06	1.9401E-06
²⁴⁷ Cm	1.6650E-11	1.6650E-11	1.6650E-11	1.6650E-11	1.6649E-11	1.6648E-11	1.6645E-11
TOTAL	3.3359E+02	2.2090E+01	8.9360E+00	1.6845E+00	8.3653E-01	7.9195E-01	8.1298E-01

Table D-4. Radioisotope Release History (Ci) for WP04 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	1.4080E-06	1.3869E-06	1.3785E-06	1.3496E-06	1.2475E-06	9.7924E-07	5.8541E-07
³⁶ Cl	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁵⁹ Ni	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁶³ Ni	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁷⁹ Se	5.0750E-03	5.0682E-03	5.0655E-03	5.0561E-03	5.0211E-03	4.9151E-03	4.6972E-03
⁸⁷ Rb	3.4160E-07	3.4160E-07	3.4160E-07	3.4160E-07	3.4160E-07	3.4160E-07	3.4160E-07
⁹⁰ Sr	5.7660E+02	2.9419E+01	8.9479E+00	1.3886E-01	2.6476E-08	0.0000E+00	0.0000E+00
⁹³ Zr	2.6070E-02	2.6069E-02	2.6068E-02	2.6066E-02	2.6058E-02	2.6035E-02	2.5984E-02
⁹⁹ Tc	1.7100E-01	1.7093E-01	1.7090E-01	1.7081E-01	1.7044E-01	1.6934E-01	1.6701E-01
¹⁰⁷ Pd	1.9710E-04	1.9710E-04	1.9710E-04	1.9709E-04	1.9708E-04	1.9704E-04	1.9695E-04
¹¹³ Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
^{113m} Cd	3.9300E-02	6.6957E-05	5.2283E-06	6.9555E-10	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	4.5420E-03	4.5381E-03	4.5365E-03	4.5310E-03	4.5106E-03	4.4485E-03	4.3193E-03
¹²⁹ I	2.8040E-04	2.8040E-04	2.8040E-04	2.8040E-04	2.8039E-04	2.8036E-04	2.8031E-04
¹³⁵ Cs	1.3250E-03	1.3250E-03	1.3249E-03	1.3249E-03	1.3246E-03	1.3238E-03	1.3221E-03
¹³⁷ Cs	6.1220E+02	3.4087E+01	1.0737E+01	1.8830E-01	5.6553E-08	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	0.0000E+00	9.0853E-20	9.0853E-20	9.0853E-20	9.0853E-20	9.0853E-20	9.0853E-20
¹⁴⁷ Pm	7.1090E+00	9.5990E-08	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	9.5970E-08	9.6146E-08	9.6146E-08	9.6146E-08	9.6146E-08	9.6146E-08	9.6146E-08
¹⁵¹ Sm	3.5940E+00	1.3714E+00	9.3283E-01	2.4212E-01	1.6154E-03	3.2637E-10	0.0000E+00
¹⁵² Eu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	2.1920E-10	6.2423E-09	9.0336E-09	1.8395E-08	4.7455E-08	0.0000E+00	0.0000E+00
²²⁶ Ra	8.3930E-10	2.9888E-08	5.3235E-08	1.8581E-07	1.2830E-06	8.6745E-06	3.2164E-05
²²⁸ Ra	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	1.2860E-09	8.8205E-09	1.1808E-08	2.2150E-08	5.9030E-08	1.5864E-07	3.1408E-07
²³⁰ Th	1.3760E-07	1.8153E-06	2.7940E-06	6.7797E-06	2.3017E-05	7.2653E-05	1.7646E-04
²³² Th	0.0000E+00	3.5709E-11	5.0101E-11	1.0021E-10	2.8570E-10	8.5753E-10	2.0729E-09
²³¹ Pa	9.7950E-07	4.6959E-06	6.1798E-06	1.1361E-05	3.0437E-05	8.7515E-05	2.0108E-04
²³² U	2.4090E-05	7.2271E-06	4.4649E-06	8.2753E-07	1.5806E-09	6.8049E-18	0.0000E+00
²³³ U	6.4180E-07	2.5427E-06	3.3020E-06	5.9585E-06	1.5806E-05	4.5919E-05	1.0897E-04
²³⁴ U	7.3060E-04	2.0413E-03	2.2951E-03	2.6867E-03	2.8124E-03	2.7972E-03	2.7630E-03
²³⁵ U	1.4080E-03	1.4080E-03	1.4080E-03	1.4081E-03	1.4082E-03	1.4085E-03	1.4092E-03
²³⁶ U	5.7860E-03	5.7864E-03	5.7865E-03	5.7870E-03	5.7888E-03	5.7935E-03	5.8006E-03
²³⁸ U	2.5870E-05	2.5870E-05	2.5870E-05	2.5870E-05	2.5870E-05	2.5870E-05	2.5870E-05
²³⁷ Np	3.4760E-03	3.5197E-03	3.5355E-03	3.5820E-03	3.6749E-03	3.7209E-03	3.7178E-03
²³⁸ Pu	5.8250E+00	2.1701E+00	1.4620E+00	3.6696E-01	2.1649E-03	7.6514E-10	0.0000E+00
²³⁹ Pu	1.8500E-01	1.8434E-01	1.8407E-01	1.8315E-01	1.7977E-01	1.6973E-01	1.5022E-01
²⁴⁰ Pu	1.0550E-01	1.0413E-01	1.0358E-01	1.0168E-01	9.4908E-02	7.6773E-02	4.8922E-02
²⁴¹ Pu	8.8860E+00	2.1652E-02	1.9520E-03	1.7660E-06	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	1.5840E-04	1.5836E-04	1.5835E-04	1.5830E-04	1.5811E-04	1.5753E-04	1.5630E-04
²⁴⁴ Pu	2.8830E-11	2.8830E-11	2.8830E-11	2.8830E-11	2.8830E-11	2.8829E-11	2.8828E-11
²⁴¹ Am	9.4390E-01	1.0223E+00	9.4414E-01	7.1316E-01	2.4942E-01	1.0094E-02	1.1870E-05
^{242m} Am	1.7440E-04	9.8582E-05	7.8470E-05	3.5307E-05	1.8179E-06	1.9753E-10	7.4513E-19
²⁴³ Am	6.5360E-04	6.4596E-04	6.4292E-04	6.3242E-04	5.9489E-04	4.9281E-04	3.3034E-04
²⁴³ Cm	5.4500E-05	2.6016E-06	7.7052E-07	1.0894E-08	1.4696E-15	0.0000E+00	0.0000E+00
²⁴⁴ Cm	8.6890E-03	7.2643E-05	1.0717E-05	1.3219E-08	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	1.3740E-06	1.3601E-06	1.3545E-06	1.3353E-06	1.2663E-06	1.0757E-06	7.6050E-07
²⁴⁶ Cm	9.4590E-08	9.2872E-08	9.2194E-08	8.9859E-08	8.1691E-08	6.0930E-08	3.2677E-08
²⁴⁷ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
TOTAL	1.2157E+03	5.8602E+01	2.3535E+01	2.1568E+00	7.5025E-01	4.7775E-01	4.1769E-01

Table D-5. Radioisotope Release History (Ci) for WP05 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	3.4680E-01	3.4159E-01	3.3953E-01	3.3241E-01	3.0726E-01	2.4119E-01	1.4419E-01
³⁶ Cl	2.5680E-03	2.5673E-03	2.5670E-03	2.5659E-03	2.5621E-03	2.5503E-03	2.5255E-03
⁵⁹ Ni	1.1710E+00	1.1696E+00	1.1691E+00	1.1672E+00	1.1602E+00	1.1390E+00	1.0951E+00
⁶³ Ni	1.1940E+02	4.8386E+01	3.3713E+01	9.5193E+00	8.6839E-02	4.5934E-08	0.0000E+00
⁷⁹ Se	9.2120E-02	9.1997E-02	9.1948E-02	9.1777E-02	9.1142E-02	8.9218E-02	8.5262E-02
⁸⁷ Rb	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁹⁰ Sr	8.0630E+03	4.1138E+02	1.2512E+02	1.9417E+00	3.7024E-07	0.0000E+00	0.0000E+00
⁹³ Zr	4.0650E-01	4.0648E-01	4.0647E-01	4.0644E-01	4.0632E-01	4.0595E-01	4.0517E-01
⁹⁹ Tc	2.9390E+00	2.9378E+00	2.9373E+00	2.9357E+00	2.9294E+00	2.9104E+00	2.8704E+00
¹⁰⁷ Pd	2.5230E-02	2.5230E-02	2.5230E-02	2.5229E-02	2.5227E-02	2.5222E-02	2.5210E-02
¹¹³ Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
^{113m} Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	1.7450E-01	1.7435E-01	1.7429E-01	1.7408E-01	1.7329E-01	1.7091E-01	1.6594E-01
¹²⁹ I	7.1170E-03	7.1170E-03	7.1169E-03	7.1169E-03	7.1167E-03	7.1161E-03	7.1147E-03
¹³⁵ Cs	7.7700E-02	7.7697E-02	7.7696E-02	7.7692E-02	7.7677E-02	7.7630E-02	7.7530E-02
¹³⁷ Cs	1.1710E+04	6.5201E+02	2.0537E+02	3.6017E+00	1.0817E-06	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	0.0000E+00	2.6195E-22	2.6195E-22	2.6195E-22	2.6195E-22	2.6195E-22	2.6195E-22
¹⁴⁷ Pm	1.1240E+01	5.1032E-14	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	0.0000E+00	2.7830E-10	2.7830E-10	2.7830E-10	2.7830E-10	2.7830E-10	2.7830E-10
¹⁵¹ Sm	6.4530E+01	2.4624E+01	1.6749E+01	4.3472E+00	2.9005E-02	5.8600E-09	0.0000E+00
¹⁵² Eu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	1.4080E-07	3.6429E-06	5.2485E-06	1.0631E-05	2.7340E-05	0.0000E+00	0.0000E+00
²²⁶ Ra	5.3720E-07	1.3467E-05	2.3338E-05	7.8263E-05	5.2457E-04	3.5066E-03	1.2961E-02
²²⁸ Ra	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	3.3560E-08	1.8153E-07	2.4021E-07	4.4332E-07	1.1676E-06	3.1240E-06	6.1767E-06
²³⁰ Th	7.9170E-05	4.7243E-04	6.5222E-04	1.3221E-03	3.9088E-03	1.1777E-02	2.8047E-02
²³² Th	0.0000E+00	3.5654E-10	5.0023E-10	1.0005E-09	2.8527E-09	8.5628E-09	2.0700E-08
²³¹ Pa	5.2360E-06	1.5455E-05	1.9535E-05	3.3780E-05	8.6233E-05	2.4317E-04	5.5542E-04
²³² U	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²³³ U	1.2610E-05	5.3499E-05	6.9832E-05	1.2698E-04	3.3881E-04	9.8655E-04	2.3429E-03
²³⁴ U	2.9390E-01	3.9010E-01	4.0871E-01	4.3735E-01	4.4615E-01	4.4368E-01	4.3831E-01
²³⁵ U	3.8740E-03	3.8827E-03	3.8861E-03	3.8982E-03	3.9424E-03	4.0735E-03	4.3282E-03
²³⁶ U	5.7770E-02	5.7775E-02	5.7777E-02	5.7784E-02	5.7810E-02	5.7878E-02	5.7980E-02
²³⁸ U	7.1510E-02	7.1510E-02	7.1510E-02	7.1510E-02	7.1510E-02	7.1510E-02	7.1510E-02
²³⁷ Np	7.4770E-02	1.0826E-01	1.2042E-01	1.5606E-01	2.2758E-01	2.6435E-01	2.6554E-01
²³⁸ Pu	4.2790E+02	1.5941E+02	1.0739E+02	2.6953E+01	1.5873E-01	2.1840E-08	0.0000E+00
²³⁹ Pu	7.0500E+01	7.0260E+01	7.0165E+01	6.9831E+01	6.8602E+01	6.4939E+01	5.7730E+01
²⁴⁰ Pu	1.1820E+00	1.4627E+00	1.4571E+00	1.4307E+00	1.3346E+00	1.0796E+00	6.8797E-01
²⁴¹ Pu	6.7000E+03	1.6325E+01	1.4708E+00	3.2288E-04	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	3.9640E-01	3.9631E-01	3.9627E-01	3.9614E-01	3.9567E-01	3.9422E-01	3.9114E-01
²⁴⁴ Pu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴¹ Am	7.2410E+02	7.8095E+02	7.2124E+02	5.4479E+02	1.9056E+02	7.7112E+00	8.4567E-03
^{242m} Am	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴³ Am	3.8290E+00	3.7842E+00	3.7664E+00	3.7049E+00	3.4851E+00	2.8871E+00	1.9352E+00
²⁴³ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁴ Cm	1.0900E+02	9.1127E-01	1.3444E-01	1.6582E-04	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁶ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁷ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
TOTAL	2.8011E+04	2.1758E+03	1.2929E+03	6.7247E+02	2.7064E+02	8.2939E+01	6.6513E+01

Table D-6. Radioisotope Release History (Ci) for WP06 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
³⁶ Cl	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁵⁹ Ni	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁶³ Ni	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁷⁹ Se	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁸⁷ Rb	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁹⁰ Sr	6.7820E+02	3.4602E+01	1.0525E+01	1.6332E-01	3.1142E-08	0.0000E+00	0.0000E+00
⁹³ Zr	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁹⁹ Tc	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁰⁷ Pd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹¹³ Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
^{113m} Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁹ I	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹³⁵ Cs	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹³⁷ Cs	7.9050E+02	4.4015E+01	1.3864E+01	2.4314E-01	7.3024E-08	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	0.0000E+00	1.1325E-37	1.1325E-37	1.1325E-37	1.1325E-37	1.1325E-37	1.1325E-37
¹⁴⁷ Pm	1.0550E-08	4.7900E-23	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	0.0000E+00	2.6122E-19	2.6122E-19	2.6122E-19	2.6122E-19	2.6122E-19	2.6122E-19
¹⁵¹ Sm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁵² Eu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	0.0000E+00	3.9430E-07	5.9679E-07	1.2782E-06	3.3940E-06	0.0000E+00	0.0000E+00
²²⁶ Ra	0.0000E+00	8.7680E-07	1.4104E-06	4.0714E-06	2.3386E-05	1.4540E-04	5.2567E-04
²²⁸ Ra	0.0000E+00	2.9849E-11	2.9849E-11	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	0.0000E+00	5.6262E-06	7.8572E-06	1.5580E-05	4.3119E-05	1.1750E-04	2.3358E-04
²³⁰ Th	9.9570E-06	2.4086E-05	3.0054E-05	5.1504E-05	1.3236E-04	3.7776E-04	8.8246E-04
²³² Th	2.9860E-11	3.0866E-11	3.1257E-11	3.2664E-11	3.7970E-11	5.5058E-11	9.4171E-11
²³¹ Pa	0.0000E+00	1.9054E-07	2.6662E-07	5.3225E-07	1.5103E-06	4.4368E-06	1.0259E-05
²³² U	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²³³ U	4.7950E-04	5.3365E-04	5.5530E-04	6.3105E-04	9.1182E-04	1.7704E-03	3.5683E-03
²³⁴ U	1.1780E-02	1.3147E-02	1.3410E-02	1.3814E-02	1.3926E-02	1.3848E-02	1.3681E-02
²³⁵ U	7.2140E-05	7.2203E-05	7.2228E-05	7.2316E-05	7.2639E-05	7.3595E-05	7.5451E-05
²³⁶ U	1.6130E-04	1.6210E-04	1.6242E-04	1.6353E-04	1.6744E-04	1.7794E-04	1.9405E-04
²³⁸ U	4.0460E-06	4.0460E-06	4.0460E-06	4.0460E-06	4.0460E-06	4.0460E-06	4.0461E-06
²³⁷ Np	9.9570E-02	9.9573E-02	9.9574E-02	9.9576E-02	9.9570E-02	9.9513E-02	9.9377E-02
²³⁸ Pu	6.0910E+00	2.2691E+00	1.5287E+00	3.8366E-01	2.2594E-03	3.1088E-10	0.0000E+00
²³⁹ Pu	5.1400E-01	5.1217E-01	5.1143E-01	5.0888E-01	4.9951E-01	4.7173E-01	4.1765E-01
²⁴⁰ Pu	2.1890E-01	2.1602E-01	2.1488E-01	2.1093E-01	1.9688E-01	1.5926E-01	1.0148E-01
²⁴¹ Pu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	1.0730E-04	1.0728E-04	1.0727E-04	1.0723E-04	1.0710E-04	1.0671E-04	1.0588E-04
²⁴⁴ Pu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴¹ Am	1.9910E-01	1.6294E-01	1.5038E-01	1.1358E-01	4.0052E-02	1.6207E-03	1.7774E-06
^{242m} Am	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴³ Am	3.8930E-03	3.8475E-03	3.8294E-03	3.7668E-03	3.5433E-03	2.9353E-03	1.9676E-03
²⁴³ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁴ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁶ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁷ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
TOTAL	1.4758E+03	8.1895E+01	2.6911E+01	1.7417E+00	8.5721E-01	7.5168E-01	6.3977E-01

Table D-7. Radioisotope Release History (Ci) for WP07 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
³⁶ Cl	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁵⁹ Ni	2.3960E-02	2.3932E-02	2.3921E-02	2.3883E-02	2.3739E-02	2.3304E-02	2.2406E-02
⁶³ Ni	2.2710E+00	9.2030E-01	6.4123E-01	1.8106E-01	1.6517E-03	8.7367E-10	0.0000E+00
⁷⁹ Se	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁸⁷ Rb	8.7170E-07	8.7170E-07	8.7170E-07	8.7170E-07	8.7170E-07	8.7170E-07	8.7170E-07
⁹⁰ Sr	1.9250E+04	9.8215E+02	2.9873E+02	4.6358E+00	8.8392E-07	0.0000E+00	0.0000E+00
⁹³ Zr	1.1170E+00	1.1169E+00	1.1169E+00	1.1168E+00	1.1165E+00	1.1155E+00	1.1133E+00
⁹⁹ Tc	3.0780E+00	3.0767E+00	3.0762E+00	3.0745E+00	3.0680E+00	3.0481E+00	3.0062E+00
¹⁰⁷ Pd	1.4730E-02	1.4730E-02	1.4730E-02	1.4729E-02	1.4728E-02	1.4725E-02	1.4719E-02
¹¹³ Cd	5.0090E-14	5.0090E-14	5.0090E-14	5.0090E-14	5.0090E-14	5.0090E-14	5.0090E-14
^{113m} Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	4.4120E-01	4.4082E-01	4.4066E-01	4.4013E-01	4.3815E-01	4.3212E-01	4.1956E-01
¹²⁹ I	2.7860E-20	2.7860E-20	2.7860E-20	2.7860E-20	2.7859E-20	2.7856E-20	2.7851E-20
¹³⁵ Cs	9.9410E-02	9.9406E-02	9.9405E-02	9.9399E-02	9.9380E-02	9.9320E-02	9.9193E-02
¹³⁷ Cs	1.8350E+04	1.0217E+03	3.2182E+02	5.6439E+00	1.6951E-06	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	4.8630E-09	4.8630E-09	4.8630E-09	4.8630E-09	4.8630E-09	4.8630E-09	4.8630E-09
¹⁴⁷ Pm	1.2860E+00	2.5975E-06	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	2.5970E-06	2.5970E-06	2.5970E-06	2.5970E-06	2.5970E-06	2.5970E-06	2.5970E-06
¹⁵¹ Sm	1.8590E+02	7.0936E+01	4.8251E+01	1.2524E+01	8.3559E-02	1.6882E-08	0.0000E+00
¹⁵² Eu	5.3080E-01	7.9498E-04	5.8956E-05	6.5483E-09	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	0.0000E+00	1.4026E-06	2.1230E-06	4.5469E-06	1.2073E-05	0.0000E+00	0.0000E+00
²²⁶ Ra	0.0000E+00	6.9487E-06	1.2454E-05	4.3545E-05	2.9960E-04	2.0208E-03	7.4873E-03
²²⁸ Ra	0.0000E+00	2.1302E-12	2.1302E-12	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	5.5610E-09	4.1095E-08	5.5186E-08	1.0396E-07	2.7789E-07	7.4768E-07	1.4808E-06
²³⁰ Th	3.5420E-05	3.8898E-04	5.8874E-04	1.3941E-03	4.6566E-03	1.4624E-02	3.5448E-02
²³² Th	2.1310E-12	9.5719E-12	1.1976E-11	2.2246E-11	6.4206E-11	2.2360E-10	6.7681E-10
²³¹ Pa	1.2390E-07	5.4037E-07	7.0665E-07	1.2872E-06	3.4250E-06	9.8214E-06	2.2547E-05
²³² U	1.2870E-02	3.8612E-03	2.3855E-03	4.4213E-04	8.4449E-07	3.6357E-15	0.0000E+00
²³³ U	3.0340E-06	8.1600E-06	1.0209E-05	1.7379E-05	4.3958E-05	1.2523E-04	2.9542E-04
²³⁴ U	1.7000E-01	4.1864E-01	4.6676E-01	5.4103E-01	5.6482E-01	5.6178E-01	5.5491E-01
²³⁵ U	1.5780E-04	1.5938E-04	1.6002E-04	1.6222E-04	1.7031E-04	1.9428E-04	2.4085E-04
²³⁶ U	1.1370E-03	1.1698E-03	1.1828E-03	1.2279E-03	1.3882E-03	1.8172E-03	2.4759E-03
²³⁸ U	1.0500E-02	1.0500E-02	1.0500E-02	1.0500E-02	1.0500E-02	1.0500E-02	1.0500E-02
²³⁷ Np	9.3830E-03	1.1688E-02	1.2511E-02	1.4921E-02	1.9749E-02	2.2241E-02	2.2316E-02
²³⁸ Pu	1.1050E+03	4.1165E+02	2.7733E+02	6.9605E+01	4.1018E-01	8.9152E-08	0.0000E+00
²³⁹ Pu	1.2890E+01	1.2844E+01	1.2825E+01	1.2761E+01	1.2524E+01	1.1823E+01	1.0461E+01
²⁴⁰ Pu	8.8700E+00	8.8235E+00	8.7774E+00	8.6161E+00	8.0421E+00	6.5054E+00	4.1455E+00
²⁴¹ Pu	2.7790E+02	6.7712E-01	6.1012E-02	1.9909E-05	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	1.2240E-02	1.2237E-02	1.2236E-02	1.2232E-02	1.2217E-02	1.2173E-02	1.2078E-02
²⁴⁴ Pu	2.1470E-19	9.3436E-19	1.2221E-18	2.2294E-18	5.9667E-18	1.7436E-17	4.1651E-17
²⁴¹ Am	5.5010E+01	5.2831E+01	4.8780E+01	3.6845E+01	1.2928E+01	5.2316E-01	5.7764E-04
^{242m} Am	1.2210E-02	6.9019E-03	5.4938E-03	2.4719E-03	1.2727E-04	1.3829E-08	5.2168E-17
²⁴³ Am	5.7660E-03	5.6986E-03	5.6718E-03	5.5791E-03	5.2481E-03	4.3476E-03	2.9142E-03
²⁴³ Cm	2.2470E-03	1.0726E-04	3.1768E-05	4.4914E-07	6.0589E-14	0.0000E+00	0.0000E+00
²⁴⁴ Cm	2.5860E+01	2.1620E-01	3.1896E-02	3.9341E-05	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	6.6930E-06	6.6251E-06	6.5981E-06	6.5046E-06	6.1686E-06	5.2399E-06	3.7045E-06
²⁴⁶ Cm	5.3130E-07	5.2165E-07	5.1784E-07	5.0473E-07	4.5885E-07	3.4224E-07	1.8354E-07
²⁴⁷ Cm	6.6020E-13	6.6020E-13	6.6019E-13	6.6019E-13	6.6017E-13	6.6011E-13	6.5999E-13
TOTAL	3.9281E+04	2.5680E+03	1.0225E+03	1.5617E+02	3.9370E+01	2.4215E+01	1.9932E+01

Table D-8. Radioisotope Release History (Ci) for WP08 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
³⁶ Cl	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁵⁹ Ni	1.3600E-01	1.3584E-01	1.3578E-01	1.3556E-01	1.3475E-01	1.3228E-01	1.2718E-01
⁶³ Ni	1.1990E+01	4.8588E+00	3.3855E+00	9.5591E-01	8.7203E-03	4.6126E-09	0.0000E+00
⁷⁹ Se	3.8980E-03	3.8928E-03	3.8907E-03	3.8835E-03	3.8566E-03	3.7752E-03	3.6078E-03
⁸⁷ Rb	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁹⁰ Sr	1.7220E+04	8.7858E+02	2.6723E+02	4.1469E+00	7.9071E-07	0.0000E+00	0.0000E+00
⁹³ Zr	1.2900E+00	1.2899E+00	1.2899E+00	1.2898E+00	1.2894E+00	1.2882E+00	1.2858E+00
⁹⁹ Tc	9.3520E+00	9.3482E+00	9.3467E+00	9.3413E+00	9.3216E+00	9.2611E+00	9.1338E+00
¹⁰⁷ Pd	4.0700E-02	4.0699E-02	4.0699E-02	4.0698E-02	4.0696E-02	4.0687E-02	4.0669E-02
¹¹³ Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
^{113m} Cd	2.1840E+00	3.7210E-03	2.9055E-04	3.8653E-08	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	4.5990E-01	4.5950E-01	4.5934E-01	4.5879E-01	4.5672E-01	4.5043E-01	4.3735E-01
¹²⁹ I	1.6300E-05	1.6300E-05	1.6300E-05	1.6300E-05	1.6299E-05	1.6298E-05	1.6295E-05
¹³⁵ Cs	2.5100E-01	2.5099E-01	2.5099E-01	2.5097E-01	2.5092E-01	2.5077E-01	2.5045E-01
¹³⁷ Cs	2.1570E+04	1.2010E+03	3.7829E+02	6.6343E+00	1.9926E-06	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	1.1050E-11	1.1050E-11	1.1050E-11	1.1050E-11	1.1050E-11	1.1050E-11	1.1050E-11
¹⁴⁷ Pm	2.1110E+00	9.8341E-07	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	9.8320E-07	9.8325E-07	9.8321E-07	9.8325E-07	9.8325E-07	9.8325E-07	9.8325E-07
¹⁵¹ Sm	6.2760E+02	2.3948E+02	1.6289E+02	4.2280E+01	2.8209E-01	5.6993E-08	0.0000E+00
¹⁵² Eu	3.9440E-01	5.9070E-04	4.3806E-05	4.8656E-09	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	0.0000E+00	6.4469E-08	9.7577E-08	2.0899E-07	5.5493E-07	0.0000E+00	0.0000E+00
²²⁶ Ra	0.0000E+00	2.3197E-07	4.0269E-07	1.3381E-06	8.8309E-06	5.8564E-05	2.1593E-04
²²⁸ Ra	0.0000E+00	8.9768E-13	8.9768E-13	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	0.0000E+00	3.8040E-07	5.3124E-07	1.0534E-06	2.9154E-06	7.9446E-06	1.5793E-05
²³⁰ Th	1.6280E-06	7.2084E-06	9.4687E-06	1.7424E-05	4.6979E-05	1.3654E-04	3.2013E-04
²³² Th	8.9800E-13	3.8599E-12	5.0162E-12	9.1616E-12	2.4756E-11	7.4720E-11	1.8812E-10
²³¹ Pa	1.5520E-07	6.7512E-07	8.8271E-07	1.6075E-06	4.2763E-06	1.2262E-05	2.8149E-05
²³² U	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²³³ U	3.2420E-05	1.4487E-04	1.8983E-04	3.4712E-04	9.3017E-04	2.7131E-03	6.4465E-03
²³⁴ U	4.8900E-03	5.0175E-03	5.0418E-03	5.0780E-03	5.0815E-03	5.0528E-03	4.9923E-03
²³⁵ U	1.9700E-04	1.9717E-04	1.9724E-04	1.9748E-04	1.9837E-04	2.0099E-04	2.0608E-04
²³⁶ U	4.7560E-04	4.7770E-04	4.7854E-04	4.8143E-04	4.9171E-04	5.1924E-04	5.6148E-04
²³⁸ U	3.7200E-03	3.7200E-03	3.7200E-03	3.7200E-03	3.7200E-03	3.7200E-03	3.7200E-03
²³⁷ Np	2.0580E-01	2.2577E-01	2.3270E-01	2.5302E-01	2.9359E-01	3.1462E-01	3.1508E-01
²³⁸ Pu	5.7400E-01	2.1383E-01	1.4406E-01	3.6155E-02	2.1292E-04	2.9296E-11	0.0000E+00
²³⁹ Pu	1.4090E+00	1.4042E+00	1.4023E+00	1.3955E+00	1.3708E+00	1.2972E+00	1.1525E+00
²⁴⁰ Pu	5.6580E-01	5.6652E-01	5.6358E-01	5.5323E-01	5.1636E-01	4.1770E-01	2.6617E-01
²⁴¹ Pu	4.2940E+00	1.0462E-02	9.4264E-04	2.0693E-07	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	1.3100E-04	1.3097E-04	1.3096E-04	1.3092E-04	1.3076E-04	1.3028E-04	1.2926E-04
²⁴⁴ Pu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴¹ Am	5.4440E+02	4.4564E+02	4.1130E+02	3.1065E+02	1.0954E+02	4.4328E+00	4.8613E-03
^{242m} Am	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴³ Am	6.7350E-02	6.6562E-02	6.6250E-02	6.5167E-02	6.1300E-02	5.0782E-02	3.4040E-02
²⁴³ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁴ Cm	3.0040E+00	2.5114E-02	3.7052E-03	4.5700E-06	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁶ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁷ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
TOTAL	4.0000E+04	2.7836E+03	1.2370E+03	3.7851E+02	1.2358E+02	1.7953E+01	1.3068E+01

Table D-9. Radioisotope Release History (Ci) for WP09 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
³⁶ Cl	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁵⁹ Ni	4.1450E-01	4.1402E-01	4.1383E-01	4.1316E-01	4.1068E-01	4.0316E-01	3.8762E-01
⁶³ Ni	2.2990E+01	9.3165E+00	6.4914E+00	1.8329E+00	1.6720E-02	8.8444E-09	0.0000E+00
⁷⁹ Se	1.3740E-02	1.3722E-02	1.3714E-02	1.3689E-02	1.3594E-02	1.3307E-02	1.2717E-02
⁸⁷ Rb	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁹⁰ Sr	1.0800E+04	5.5103E+02	1.6760E+02	2.6009E+00	4.9591E-07	0.0000E+00	0.0000E+00
⁹³ Zr	1.0660E+00	1.0659E+00	1.0659E+00	1.0658E+00	1.0655E+00	1.0646E+00	1.0625E+00
⁹⁹ Tc	4.2640E-01	4.2623E-01	4.2616E-01	4.2591E-01	4.2501E-01	4.2225E-01	4.1645E-01
¹⁰⁷ Pd	4.3150E-02	4.3149E-02	4.3149E-02	4.3148E-02	4.3145E-02	4.3136E-02	4.3117E-02
¹¹³ Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
^{113m} Cd	1.2430E+00	2.1178E-03	1.6536E-04	2.1999E-08	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	4.0760E-01	4.0725E-01	4.0711E-01	4.0661E-01	4.0478E-01	3.9921E-01	3.8761E-01
¹²⁹ I	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹³⁵ Cs	6.3190E-01	6.3188E-01	6.3187E-01	6.3183E-01	6.3171E-01	6.3133E-01	6.3052E-01
¹³⁷ Cs	1.1920E+04	6.6370E+02	2.0905E+02	3.6663E+00	1.1011E-06	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	9.4600E-19	9.4600E-19	9.4600E-19	9.4600E-19	9.4600E-19	9.4600E-19	9.4600E-19
¹⁴⁷ Pm	1.8280E-02	8.5158E-09	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	8.5140E-09	8.5145E-09	8.5145E-09	8.5145E-09	8.5145E-09	8.5145E-09	8.5145E-09
¹⁵¹ Sm	2.4750E+02	9.4442E+01	6.4239E+01	1.6673E+01	1.1125E-01	2.2476E-08	0.0000E+00
¹⁵² Eu	2.0510E-01	3.0718E-04	2.2781E-05	2.5302E-09	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	1.5380E-06	1.3117E-05	1.7979E-05	3.4231E-05	8.4658E-05	0.0000E+00	0.0000E+00
²²⁶ Ra	3.7640E-06	1.6849E-05	2.2207E-05	4.1507E-05	1.1960E-04	4.0141E-04	1.0714E-03
²²⁸ Ra	6.4240E-03	6.4257E-03	6.4257E-03	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	9.4350E-04	1.3476E-03	1.5078E-03	2.0624E-03	4.0400E-03	9.3807E-03	1.7711E-02
²³⁰ Th	2.4110E-04	2.6625E-04	2.7760E-04	3.1963E-04	4.8130E-04	9.7290E-04	1.9885E-03
²³² Th	6.4280E-03	6.4280E-03	6.4280E-03	6.4280E-03	6.4280E-03	6.4280E-03	6.4280E-03
²³¹ Pa	5.9460E-02	5.9304E-02	5.9242E-02	5.9024E-02	5.8223E-02	5.5825E-02	5.1056E-02
²³² U	1.9030E-02	5.7090E-03	3.5271E-03	6.5371E-04	1.2486E-06	5.3756E-15	0.0000E+00
²³³ U	3.5380E-02	3.5412E-02	3.5425E-02	3.5470E-02	3.5636E-02	3.6143E-02	3.7206E-02
²³⁴ U	1.9410E-02	2.4913E-02	2.5977E-02	2.7614E-02	2.8113E-02	2.7957E-02	2.7619E-02
²³⁵ U	3.5290E-04	3.5368E-04	3.5399E-04	3.5508E-04	3.5908E-04	3.7091E-04	3.9389E-04
²³⁶ U	1.1010E-03	1.1184E-03	1.1252E-03	1.1491E-03	1.2338E-03	1.4605E-03	1.8087E-03
²³⁸ U	3.1290E-03	3.1290E-03	3.1290E-03	3.1290E-03	3.1290E-03	3.1290E-03	3.1290E-03
²³⁷ Np	9.3990E-02	1.0158E-01	1.0422E-01	1.1195E-01	1.2740E-01	1.3539E-01	1.3555E-01
²³⁸ Pu	2.4480E+01	9.5686E+00	6.6074E+00	1.8669E+00	3.2503E-02	2.6401E-06	0.0000E+00
²³⁹ Pu	6.3620E+00	6.3442E+00	6.3370E+00	6.3118E+00	6.2182E+00	5.9328E+00	5.3455E+00
²⁴⁰ Pu	4.7070E+00	4.6646E+00	4.6400E+00	4.5548E+00	4.2514E+00	3.4390E+00	2.1915E+00
²⁴¹ Pu	5.2570E+01	1.3149E-01	1.4934E-02	3.3489E-03	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	6.3480E-03	6.3466E-03	6.3461E-03	6.3442E-03	6.3370E-03	6.3148E-03	6.2670E-03
²⁴⁴ Pu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴¹ Am	2.0550E+02	1.6965E+02	1.5658E+02	1.1827E+02	4.1694E+01	1.6899E+00	3.8546E-03
^{242m} Am	9.8370E-01	5.5605E-01	4.4261E-01	1.9915E-01	1.0254E-02	1.1141E-06	4.2029E-15
²⁴³ Am	1.3500E+00	1.3342E+00	1.3279E+00	1.3063E+00	1.2287E+00	1.0179E+00	6.8231E-01
²⁴³ Cm	2.1210E-01	1.0125E-02	2.9987E-03	4.2395E-05	5.7192E-12	0.0000E+00	0.0000E+00
²⁴⁴ Cm	7.1850E+00	6.0069E-02	8.8621E-03	1.0931E-05	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	3.4370E-03	3.4021E-03	3.3883E-03	3.3402E-03	3.1677E-03	2.6908E-03	1.9023E-03
²⁴⁶ Cm	3.8950E-04	3.8243E-04	3.7963E-04	3.7002E-04	3.3639E-04	2.5090E-04	1.3455E-04
²⁴⁷ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
TOTAL	2.3299E+04	1.5141E+03	6.2660E+02	1.6054E+02	5.6832E+01	1.5343E+01	1.1456E+01

Table D-10. Radioisotope Release History (Ci) for WP10 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	1.9400E-02	1.9109E-02	1.8993E-02	1.8595E-02	1.7188E-02	1.3492E-02	8.0661E-03
³⁶ Cl	8.0680E-04	8.0657E-04	8.0647E-04	8.0615E-04	8.0494E-04	8.0124E-04	7.9343E-04
⁵⁹ Ni	5.6300E-03	5.6235E-03	5.6209E-03	5.6118E-03	5.5782E-03	5.4759E-03	5.2649E-03
⁶³ Ni	1.2160E-01	4.9277E-02	3.4335E-02	9.6946E-03	8.8439E-05	4.6780E-11	0.0000E+00
⁷⁹ Se	8.7450E-03	8.7333E-03	8.7287E-03	8.7124E-03	8.6522E-03	8.4695E-03	8.0940E-03
⁸⁷ Rb	5.2080E-07	5.2080E-07	5.2080E-07	5.2080E-07	5.2080E-07	5.2080E-07	5.2080E-07
⁹⁰ Sr	5.6910E+02	2.9036E+01	8.8315E+00	1.3705E-01	2.6132E-08	0.0000E+00	0.0000E+00
⁹³ Zr	3.4640E-01	3.4638E-01	3.4637E-01	3.4635E-01	3.4624E-01	3.4593E-01	3.4526E-01
⁹⁹ Tc	2.1890E-01	2.1881E-01	2.1878E-01	2.1865E-01	2.1819E-01	2.1677E-01	2.1379E-01
¹⁰⁷ Pd	2.5700E-04	2.5700E-04	2.5700E-04	2.5699E-04	2.5697E-04	2.5692E-04	2.5680E-04
¹¹³ Cd	1.1880E-15	1.1880E-15	1.1880E-15	1.1880E-15	1.1880E-15	1.1880E-15	1.1880E-15
^{113m} Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	3.8120E-03	3.8087E-03	3.8074E-03	3.8028E-03	3.7857E-03	3.7335E-03	3.6251E-03
¹²⁹ I	6.5520E-04	6.5520E-04	6.5519E-04	6.5519E-04	6.5517E-04	6.5511E-04	6.5499E-04
¹³⁵ Cs	4.1650E-03	4.1648E-03	4.1648E-03	4.1646E-03	4.1637E-03	4.1612E-03	4.1559E-03
¹³⁷ Cs	6.2350E+02	3.4716E+01	1.0935E+01	1.9177E-01	5.7597E-08	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	5.8520E-07	5.8520E-07	5.8520E-07	5.8520E-07	5.8520E-07	5.8520E-07	5.8520E-07
¹⁴⁷ Pm	0.0000E+00	1.8864E-03	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	1.8860E-03	1.8860E-03	1.8860E-03	1.8860E-03	1.8860E-03	1.8860E-03	1.8860E-03
¹⁵¹ Sm	1.6810E+01	6.4144E+00	4.3631E+00	1.1324E+00	7.5558E-03	1.5265E-09	0.0000E+00
¹⁵² Eu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁶ Ra	0.0000E+00	4.8942E-06	9.5043E-06	3.7140E-05	2.7656E-04	1.9215E-03	7.1787E-03
²²⁸ Ra	0.0000E+00	1.5125E-03	1.5125E-03	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	0.0000E+00	1.5054E-02	2.1024E-02	4.1687E-02	1.1537E-01	3.1440E-01	6.2499E-01
²³⁰ Th	0.0000E+00	1.8707E-04	2.6291E-04	5.2999E-04	1.5225E-03	4.5302E-03	1.0696E-02
²³² Th	1.5130E-03	1.5130E-03	1.5130E-03	1.5130E-03	1.5130E-03	1.5130E-03	1.5130E-03
²³¹ Pa	0.0000E+00	2.0137E-06	2.8177E-06	5.6250E-06	1.5962E-05	4.6889E-05	1.0842E-04
²³² U	6.3800E-01	1.9140E-01	1.1825E-01	2.1916E-02	4.1862E-05	1.8022E-13	0.0000E+00
²³³ U	1.2830E+00	1.2823E+00	1.2820E+00	1.2810E+00	1.2774E+00	1.2663E+00	1.2431E+00
²³⁴ U	1.6370E-01	1.6827E-01	1.6914E-01	1.7044E-01	1.7059E-01	1.6963E-01	1.6760E-01
²³⁵ U	7.6240E-04	7.6241E-04	7.6241E-04	7.6242E-04	7.6246E-04	7.6258E-04	7.6280E-04
²³⁶ U	6.3130E-03	6.3134E-03	6.3136E-03	6.3141E-03	6.3160E-03	6.3212E-03	6.3289E-03
²³⁸ U	1.7560E-05	1.7560E-05	1.7560E-05	1.7560E-05	1.7560E-05	1.7560E-05	1.7560E-05
²³⁷ Np	4.5460E-03	4.5840E-03	4.5972E-03	4.6359E-03	4.7126E-03	4.7502E-03	4.7454E-03
²³⁸ Pu	2.0560E+01	7.6593E+00	5.1601E+00	1.2951E+00	7.6297E-03	1.4011E-09	0.0000E+00
²³⁹ Pu	6.3690E-02	6.3490E-02	6.3410E-02	6.3131E-02	6.2101E-02	5.8998E-02	5.2776E-02
²⁴⁰ Pu	1.1460E-01	1.1351E-01	1.1291E-01	1.1084E-01	1.0345E-01	8.3686E-02	5.3328E-02
²⁴¹ Pu	0.0000E+00	7.1076E-05	7.0946E-05	6.9955E-05	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	0.0000E+00	4.7418E-09	6.6140E-09	1.3058E-08	3.5579E-08	9.2657E-08	1.6961E-07
²⁴⁴ Pu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴¹ Am	1.0420E+00	8.5274E-01	7.8704E-01	5.9447E-01	2.0967E-01	8.5409E-03	5.1204E-05
^{242m} Am	1.3110E-04	7.4106E-05	5.8987E-05	2.6541E-05	1.3666E-06	1.4848E-10	5.6013E-19
²⁴³ Am	8.0620E-03	7.9677E-03	7.9303E-03	7.8007E-03	7.3378E-03	6.0788E-03	4.0747E-03
²⁴³ Cm	4.5960E-04	2.1940E-05	6.4978E-06	9.1867E-08	1.2393E-14	0.0000E+00	0.0000E+00
²⁴⁴ Cm	1.5390E-01	1.2866E-03	1.8982E-04	2.3413E-07	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	7.1850E-05	7.1121E-05	7.0831E-05	6.9827E-05	6.6221E-05	5.6251E-05	3.9768E-05
²⁴⁶ Cm	2.7790E-05	2.0412E-05	2.0263E-05	1.9750E-05	1.7955E-05	1.3392E-05	7.1820E-06
²⁴⁷ Cm	6.1630E-11	6.1630E-11	6.1630E-11	6.1629E-11	6.1627E-11	6.1622E-11	6.1610E-11
TOTAL	1.2342E+03	8.1198E+01	3.2511E+01	5.6809E+00	2.5839E+00	2.5292E+00	2.7692E+00

Table D-11. Radioisotope Release History (Ci) for WP11 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
³⁶ Cl	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁵⁹ Ni	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁶³ Ni	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁷⁹ Se	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁸⁷ Rb	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁹⁰ Sr	9.7530E+02	4.9761E+01	1.5135E+01	2.3487E-01	4.4784E-08	0.0000E+00	0.0000E+00
⁹³ Zr	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁹⁹ Tc	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁰⁷ Pd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹¹³ Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
^{113m} Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁹ I	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹³⁵ Cs	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹³⁷ Cs	1.1370E+03	6.3308E+01	1.9940E+01	3.4971E-01	1.0503E-07	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	0.0000E+00	1.6285E-37	1.6285E-37	1.6285E-37	1.6285E-37	1.6285E-37	1.6285E-37
¹⁴⁷ Pm	1.5180E-08	6.8921E-23	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	0.0000E+00	3.7586E-19	3.7586E-19	3.7586E-19	3.7586E-19	3.7586E-19	3.7586E-19
¹⁵¹ Sm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁵² Eu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	0.0000E+00	5.6707E-07	8.5829E-07	1.8383E-06	4.8812E-06	0.0000E+00	0.0000E+00
²²⁶ Ra	0.0000E+00	1.2609E-06	2.0283E-06	5.8550E-06	3.3630E-05	2.0909E-04	7.5593E-04
²²⁸ Ra	0.0000E+00	4.2935E-11	4.2935E-11	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	0.0000E+00	8.0902E-06	1.1298E-05	2.2403E-05	6.2004E-05	1.6896E-04	3.3588E-04
²³⁰ Th	1.4320E-05	3.4637E-05	4.3220E-05	7.4066E-05	1.9034E-04	5.4323E-04	1.2690E-03
²³² Th	4.2950E-11	4.4396E-11	4.4958E-11	4.6981E-11	5.4609E-11	7.9178E-11	1.3541E-10
²³¹ Pa	0.0000E+00	2.7390E-07	3.8326E-07	7.6510E-07	2.1711E-06	6.3778E-06	1.4747E-05
²³² U	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²³³ U	6.8950E-04	7.6738E-04	7.9852E-04	9.0745E-04	1.3113E-03	2.5461E-03	5.1317E-03
²³⁴ U	1.6940E-02	1.8905E-02	1.9285E-02	1.9866E-02	2.0026E-02	1.9913E-02	1.9674E-02
²³⁵ U	1.0370E-04	1.0379E-04	1.0383E-04	1.0395E-04	1.0442E-04	1.0579E-04	1.0846E-04
²³⁶ U	2.3190E-04	2.3306E-04	2.3351E-04	2.3510E-04	2.4073E-04	2.5582E-04	2.7898E-04
²³⁸ U	5.8190E-06	5.8190E-06	5.8190E-06	5.8190E-06	5.8190E-06	5.8191E-06	5.8192E-06
²³⁷ Np	1.4320E-01	1.4320E-01	1.4321E-01	1.4321E-01	1.4320E-01	1.4312E-01	1.4292E-01
²³⁸ Pu	8.7590E+00	3.2630E+00	2.1983E+00	5.5171E-01	3.2490E-03	4.4705E-10	0.0000E+00
²³⁹ Pu	7.3920E-01	7.3656E-01	7.3551E-01	7.3184E-01	7.1836E-01	6.7841E-01	6.0064E-01
²⁴⁰ Pu	3.1470E-01	3.1056E-01	3.0891E-01	3.0324E-01	2.8304E-01	2.2896E-01	1.4590E-01
²⁴¹ Pu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	1.5430E-04	1.5426E-04	1.5425E-04	1.5420E-04	1.5402E-04	1.5345E-04	1.5225E-04
²⁴⁴ Pu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴¹ Am	2.8640E-01	2.3438E-01	2.1632E-01	1.6339E-01	5.7613E-02	2.3314E-03	2.5568E-06
^{242m} Am	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴³ Am	5.5990E-03	5.5335E-03	5.5075E-03	5.4176E-03	5.0961E-03	4.2216E-03	2.8298E-03
²⁴³ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁴ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁶ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁷ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
TOTAL	2.1226E+03	1.1778E+02	3.8704E+01	2.5048E+00	1.2327E+00	1.0809E+00	9.2002E-01

Table D-12. Radioisotope Release History (Ci) for WP12 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	1.4080E-06	1.3869E-06	1.3785E-06	1.3496E-06	1.2475E-06	9.7924E-07	5.8541E-07
³⁶ Cl	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁵⁹ Ni	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁶³ Ni	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁷⁹ Se	5.0750E-03	5.0682E-03	5.0655E-03	5.0561E-03	5.0211E-03	4.9151E-03	4.6972E-03
⁸⁷ Rb	3.4160E-07	3.4160E-07	3.4160E-07	3.4160E-07	3.4160E-07	3.4160E-07	3.4160E-07
⁹⁰ Sr	1.1170E+03	5.6990E+01	1.7334E+01	2.6900E-01	5.1290E-08	0.0000E+00	0.0000E+00
⁹³ Zr	2.6070E-02	2.6069E-02	2.6068E-02	2.6066E-02	2.6058E-02	2.6035E-02	2.5984E-02
⁹⁹ Tc	1.7100E-01	1.7093E-01	1.7090E-01	1.7081E-01	1.7044E-01	1.6934E-01	1.6701E-01
¹⁰⁷ Pd	1.9710E-04	1.9710E-04	1.9710E-04	1.9709E-04	1.9708E-04	1.9704E-04	1.9695E-04
¹¹³ Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
^{113m} Cd	3.9300E-02	6.6957E-05	5.2283E-06	6.9555E-10	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	4.5420E-03	4.5381E-03	4.5365E-03	4.5310E-03	4.5106E-03	4.4485E-03	4.3193E-03
¹²⁹ I	2.8040E-04	2.8040E-04	2.8040E-04	2.8040E-04	2.8039E-04	2.8036E-04	2.8031E-04
¹³⁵ Cs	1.3250E-03	1.3250E-03	1.3249E-03	1.3249E-03	1.3246E-03	1.3238E-03	1.3221E-03
¹³⁷ Cs	1.2420E+03	6.9154E+01	2.1782E+01	3.8200E-01	1.1473E-07	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	0.0000E+00	9.0853E-20	9.0853E-20	9.0853E-20	9.0853E-20	9.0853E-20	9.0853E-20
¹⁴⁷ Pm	7.1090E+00	9.5990E-08	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	9.5970E-08	9.6146E-08	9.6146E-08	9.6146E-08	9.6146E-08	9.6146E-08	9.6146E-08
¹⁵¹ Sm	3.5940E+00	1.3714E+00	9.3283E-01	2.4212E-01	1.6154E-03	3.2637E-10	0.0000E+00
¹⁵² Eu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	2.1920E-10	3.2049E-07	4.8466E-07	1.0371E-06	2.7524E-06	0.0000E+00	0.0000E+00
²²⁶ Ra	8.3930E-10	7.2870E-07	1.1773E-06	3.4308E-06	1.9922E-05	1.2457E-04	4.5116E-04
²²⁸ Ra	0.0000E+00	2.3792E-11	2.3792E-11	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	1.2860E-09	4.4929E-06	6.2740E-06	1.2439E-05	3.4425E-05	9.3808E-05	1.8648E-04
²³⁰ Th	8.0730E-06	2.1012E-05	2.6748E-05	4.7831E-05	1.2852E-04	3.7375E-04	8.7984E-04
²³² Th	2.3800E-11	6.0308E-11	7.5009E-11	1.2623E-10	3.1594E-10	9.0133E-10	2.1477E-09
²³¹ Pa	9.7950E-07	4.8465E-06	6.3904E-06	1.1781E-05	3.1630E-05	9.1020E-05	2.0918E-04
²³² U	2.4090E-05	7.2271E-06	4.4649E-06	8.2753E-07	1.5806E-09	6.8049E-18	0.0000E+00
²³³ U	3.8280E-04	4.2786E-04	4.4587E-04	5.0889E-04	7.4251E-04	1.4569E-03	2.9528E-03
²³⁴ U	1.0120E-02	1.2520E-02	1.2984E-02	1.3698E-02	1.3912E-02	1.3835E-02	1.3668E-02
²³⁵ U	1.4650E-03	1.4651E-03	1.4651E-03	1.4652E-03	1.4656E-03	1.4667E-03	1.4688E-03
²³⁶ U	5.9140E-03	5.9150E-03	5.9154E-03	5.9168E-03	5.9217E-03	5.9348E-03	5.9547E-03
²³⁸ U	2.9090E-05	2.9090E-05	2.9090E-05	2.9090E-05	2.9090E-05	2.9090E-05	2.9090E-05
²³⁷ Np	8.2830E-02	8.2876E-02	8.2893E-02	8.2941E-02	8.3029E-02	8.3030E-02	8.2918E-02
²³⁸ Pu	1.0680E+01	3.9787E+00	2.6805E+00	6.7277E-01	3.9658E-03	1.0129E-09	0.0000E+00
²³⁹ Pu	5.9470E-01	5.9258E-01	5.9173E-01	5.8877E-01	5.7792E-01	5.4574E-01	4.8313E-01
²⁴⁰ Pu	2.7990E-01	2.7624E-01	2.7478E-01	2.6973E-01	2.5176E-01	2.0366E-01	1.2978E-01
²⁴¹ Pu	8.8860E+00	2.1652E-02	1.9520E-03	1.7660E-06	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	2.4390E-04	2.4384E-04	2.4382E-04	2.4374E-04	2.4345E-04	2.4256E-04	2.4067E-04
²⁴⁴ Pu	2.8830E-11	2.8830E-11	2.8830E-11	2.8830E-11	2.8830E-11	2.8829E-11	2.8828E-11
²⁴¹ Am	1.1030E+00	1.1525E+00	1.0643E+00	8.0392E-01	2.8143E-01	1.1389E-02	1.3291E-05
^{242m} Am	1.7440E-04	9.8582E-05	7.8470E-05	3.5307E-05	1.8179E-06	1.9753E-10	7.4513E-19
²⁴³ Am	3.7570E-03	3.7131E-03	3.6956E-03	3.6353E-03	3.4195E-03	2.8328E-03	1.8988E-03
²⁴³ Cm	5.4500E-05	2.6016E-06	7.7052E-07	1.0894E-08	1.4696E-15	0.0000E+00	0.0000E+00
²⁴⁴ Cm	8.6890E-03	7.2643E-05	.0717E-05	1.3219E-08	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	1.3740E-06	1.3601E-06	1.3545E-06	1.3353E-06	1.2663E-06	1.0757E-06	7.6050E-07
²⁴⁶ Cm	9.4590E-08	9.2872E-08	9.2194E-08	8.9859E-08	8.1691E-08	6.0930E-08	3.2677E-08
²⁴⁷ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
TOTAL	2.3916E+03	1.3385E+02	4.4978E+01	3.5451E+00	1.4335E+00	1.0768E+00	9.2759E-01

Table D-13. Radioisotope Release History (Ci) for WP13 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	3.4680E-01	3.4159E-01	3.3953E-01	3.3241E-01	3.0726E-01	2.4119E-01	1.4419E-01
³⁶ Cl	2.5680E-03	2.5673E-03	2.5670E-03	2.5659E-03	2.5621E-03	2.5503E-03	2.5255E-03
⁵⁹ Ni	1.1710E+00	1.1696E+00	1.1691E+00	1.1672E+00	1.1602E+00	1.1390E+00	1.0951E+00
⁶³ Ni	1.1940E+02	4.8386E+01	3.3713E+01	9.5193E+00	8.6839E-02	4.5934E-08	0.0000E+00
⁷⁹ Se	9.2120E-02	9.1997E-02	9.1948E-02	9.1777E-02	9.1142E-02	8.9218E-02	8.5262E-02
⁸⁷ Rb	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁹⁰ Sr	9.2000E+03	4.6939E+02	1.4277E+02	2.2156E+00	4.2245E-07	0.0000E+00	0.0000E+00
⁹³ Zr	4.0650E-01	4.0648E-01	4.0647E-01	4.0644E-01	4.0632E-01	4.0595E-01	4.0517E-01
⁹⁹ Tc	2.9390E+00	2.9378E+00	2.9373E+00	2.9357E+00	2.9294E+00	2.9104E+00	2.8704E+00
¹⁰⁷ Pd	2.5230E-02	2.5230E-02	2.5230E-02	2.5229E-02	2.5227E-02	2.5222E-02	2.5210E-02
¹¹³ Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
^{113m} Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	1.7450E-01	1.7435E-01	1.7429E-01	1.7408E-01	1.7329E-01	1.7091E-01	1.6594E-01
¹²⁹ I	7.1170E-03	7.1170E-03	7.1169E-03	7.1169E-03	7.1167E-03	7.1161E-03	7.1147E-03
¹³⁵ Cs	7.7700E-02	7.7697E-02	7.7696E-02	7.7692E-02	7.7677E-02	7.7630E-02	7.7530E-02
¹³⁷ Cs	1.3040E+04	7.2606E+02	2.2869E+02	4.0107E+00	1.2046E-06	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	0.0000E+00	2.6195E-22	2.6195E-22	2.6195E-22	2.6195E-22	2.6195E-22	2.6195E-22
¹⁴⁷ Pm	1.1240E+01	5.1032E-14	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	0.0000E+00	2.7830E-10	2.7830E-10	2.7830E-10	2.7830E-10	2.7830E-10	2.7830E-10
¹⁵¹ Sm	6.4530E+01	2.4624E+01	1.6749E+01	4.3472E+00	2.9005E-02	5.8600E-09	0.0000E+00
¹⁵² Eu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	1.4080E-07	4.3035E-06	6.2482E-06	1.2773E-05	3.3025E-05	0.0000E+00	0.0000E+00
²²⁶ Ra	5.3720E-07	1.4938E-05	2.5704E-05	8.5098E-05	5.6386E-04	3.7510E-03	1.3844E-02
²²⁸ Ra	0.0000E+00	5.0032E-11	5.0032E-11	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	3.3560E-08	9.6104E-06	1.3408E-05	2.6553E-05	7.3431E-05	2.0005E-04	3.9763E-04
²³⁰ Th	9.5850E-05	5.1285E-04	7.0267E-04	1.4086E-03	4.1311E-03	1.2412E-02	2.9530E-02
²³² Th	5.0050E-11	4.0828E-10	5.5262E-10	1.0553E-09	2.9163E-09	8.6550E-09	2.0858E-08
²³¹ Pa	5.2360E-06	1.5774E-05	1.9982E-05	3.4673E-05	8.8766E-05	2.5062E-04	5.7263E-04
²³² U	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²³³ U	8.1620E-04	9.4787E-04	1.0005E-03	1.1846E-03	1.8672E-03	3.9543E-03	8.3249E-03
²³⁴ U	3.1370E-01	4.1219E-01	4.3124E-01	4.6056E-01	4.6954E-01	4.6694E-01	4.6130E-01
²³⁵ U	3.9950E-03	4.0038E-03	4.0073E-03	4.0195E-03	4.0643E-03	4.1970E-03	4.4548E-03
²³⁶ U	5.8040E-02	5.8046E-02	5.8049E-02	5.8058E-02	5.8090E-02	5.8176E-02	5.8305E-02
²³⁸ U	7.1520E-02	7.1520E-02	7.1520E-02	7.1520E-02	7.1520E-02	7.1520E-02	7.1520E-02
²³⁷ Np	2.4170E-01	2.7519E-01	2.8735E-01	3.2300E-01	3.9451E-01	4.3118E-01	4.3214E-01
²³⁸ Pu	4.3810E+02	1.6321E+02	1.0995E+02	2.7595E+01	1.6251E-01	2.2360E-08	0.0000E+00
²³⁹ Pu	7.1360E+01	7.1117E+01	7.1020E+01	7.0682E+01	6.9438E+01	6.5729E+01	5.8429E+01
²⁴⁰ Pu	1.5490E+00	1.8249E+00	1.8174E+00	1.7843E+00	1.6647E+00	1.3466E+00	8.5812E-01
²⁴¹ Pu	6.7000E+03	1.6325E+01	1.4708E+00	3.2288E-04	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	3.9660E-01	3.9651E-01	3.9647E-01	3.9634E-01	3.9587E-01	3.9442E-01	3.9134E-01
²⁴⁴ Pu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴¹ Am	7.2440E+02	7.8119E+02	7.2147E+02	5.4497E+02	1.9062E+02	7.7137E+00	8.4594E-03
^{242m} Am	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴³ Am	3.8350E+00	3.7901E+00	3.7724E+00	3.7107E+00	3.4905E+00	2.8916E+00	1.9383E+00
²⁴³ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁴ Cm	1.0900E+02	9.1127E-01	1.3444E-01	1.6582E-04	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁶ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁷ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
TOTAL	3.0490E+04	2.3133E+03	1.3380E+03	6.7537E+02	2.7207E+02	8.4197E+01	6.7584E+01

Table D-14. Radioisotope Release History (Ci) for WP14 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	5.9590E-02	5.8695E-02	5.8341E-02	5.7118E-02	5.2796E-02	4.1444E-02	2.4776E-02
³⁶ Cl	9.5590E-03	9.5562E-03	9.5551E-03	9.5513E-03	9.5370E-03	9.4931E-03	9.4006E-03
⁵⁹ Ni	6.6710E-02	6.6633E-02	6.6602E-02	6.6494E-02	6.6096E-02	6.4884E-02	6.2383E-02
⁶³ Ni	1.4410E+00	5.8395E-01	4.0688E-01	1.1488E-01	1.0480E-03	5.5436E-10	0.0000E+00
⁷⁹ Se	1.0360E-01	1.0346E-01	1.0341E-01	1.0321E-01	1.0250E-01	1.0034E-01	9.5888E-02
⁸⁷ Rb	6.1710E-06	6.1710E-06	6.1710E-06	6.1710E-06	6.1710E-06	6.1710E-06	6.1710E-06
⁹⁰ Sr	7.1150E+03	3.6301E+02	1.1041E+02	1.7134E+00	3.2671E-07	0.0000E+00	0.0000E+00
⁹³ Zr	4.1040E+00	4.1038E+00	4.1037E+00	4.1033E+00	4.1021E+00	4.0984E+00	4.0905E+00
⁹⁹ Tc	2.5940E+00	2.5929E+00	2.5925E+00	2.5910E+00	2.5856E+00	2.5688E+00	2.5335E+00
¹⁰⁷ Pd	3.0450E-03	3.0450E-03	3.0449E-03	3.0449E-03	3.0447E-03	3.0440E-03	3.0426E-03
¹¹³ Cd	1.4080E-14	1.4080E-14	1.4080E-14	1.4080E-14	1.4080E-14	1.4080E-14	1.4080E-14
^{113m} Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	4.5160E-02	4.5121E-02	4.5105E-02	4.5051E-02	4.4848E-02	4.4230E-02	4.2945E-02
¹²⁹ I	7.7630E-03	7.7630E-03	7.7629E-03	7.7629E-03	7.7627E-03	7.7620E-03	7.7605E-03
¹³⁵ Cs	4.9360E-02	4.9358E-02	4.9357E-02	4.9355E-02	4.9345E-02	4.9315E-02	4.9252E-02
¹³⁷ Cs	7.8220E+03	4.3552E+02	1.3718E+02	2.4058E+00	7.2257E-07	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	6.9340E-06	6.9340E-06	6.9340E-06	6.9340E-06	6.9340E-06	6.9340E-06	6.9340E-06
¹⁴⁷ Pm	5.7870E-09	2.2345E-02	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	2.2340E-02	2.2340E-02	2.2340E-02	2.2340E-02	2.2340E-02	2.2340E-02	2.2340E-02
¹⁵¹ Sm	1.9920E+02	7.6011E+01	5.1703E+01	1.3420E+01	8.9537E-02	1.8089E-08	0.0000E+00
¹⁵² Eu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	0.0000E+00	2.1618E-07	3.2719E-07	7.0078E-07	1.8608E-06	0.0000E+00	0.0000E+00
²²⁶ Ra	0.0000E+00	5.8468E-05	1.1338E-04	4.4227E-04	3.2895E-03	2.2846E-02	8.5342E-02
²²⁸ Ra	0.0000E+00	1.7924E-02	1.7924E-02	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	0.0000E+00	1.7835E-01	2.4907E-01	4.9387E-01	1.3669E+00	3.7248E+00	7.4045E+00
²³⁰ Th	5.4590E-06	2.2297E-03	3.1315E-03	6.3076E-03	1.8111E-02	5.3882E-02	1.2721E-01
²³² Th	1.7930E-02	1.7930E-02	1.7930E-02	1.7930E-02	1.7930E-02	1.7930E-02	1.7930E-02
²³¹ Pa	0.0000E+00	2.3962E-05	3.3529E-05	6.6933E-05	1.8993E-04	5.5795E-04	1.2902E-03
²³² U	7.5590E+00	2.2677E+00	1.4010E+00	2.5966E-01	4.9597E-04	2.1353E-12	0.0000E+00
²³³ U	1.5200E+01	1.5192E+01	1.5188E+01	1.5177E+01	1.5134E+01	1.5003E+01	1.4729E+01
²³⁴ U	1.9460E+00	2.0009E+00	2.0114E+00	2.0270E+00	2.0288E+00	2.0174E+00	1.9932E+00
²³⁵ U	9.0720E-03	9.0721E-03	9.0722E-03	9.0724E-03	9.0730E-03	9.0749E-03	9.0786E-03
²³⁶ U	7.4890E-02	7.4895E-02	7.4897E-02	7.4904E-02	7.4929E-02	7.4996E-02	7.5096E-02
²³⁸ U	2.1030E-04	2.1030E-04	2.1030E-04	2.1030E-04	2.1030E-04	2.1030E-04	2.1030E-04
²³⁷ Np	1.0850E-01	1.0895E-01	1.0911E-01	1.0957E-01	1.1048E-01	1.1089E-01	1.1076E-01
²³⁸ Pu	2.4690E+02	9.1979E+01	6.1966E+01	1.5552E+01	9.1622E-02	1.6770E-08	0.0000E+00
²³⁹ Pu	1.0360E+00	1.0326E+00	1.0313E+00	1.0266E+00	1.0092E+00	9.5727E-01	8.5394E-01
²⁴⁰ Pu	1.4780E+00	1.4635E+00	1.4558E+00	1.4290E+00	1.3339E+00	1.0790E+00	6.8756E-01
²⁴¹ Pu	0.0000E+00	8.4213E-04	8.4058E-04	8.2885E-04	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	5.8820E-05	5.8863E-05	5.8879E-05	5.8937E-05	5.9133E-05	5.9594E-05	6.0049E-05
²⁴⁴ Pu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴¹ Am	1.2460E+01	1.0197E+01	9.4113E+00	7.1085E+00	2.5071E+00	1.0212E-01	6.0769E-04
^{242m} Am	1.5540E-03	8.7842E-04	6.9921E-04	3.1460E-04	1.6199E-05	1.7601E-09	6.6395E-18
²⁴³ Am	9.7660E-02	9.6518E-02	9.6065E-02	9.4495E-02	8.8887E-02	7.3636E-02	4.9359E-02
²⁴³ Cm	5.4450E-03	2.5992E-04	7.6982E-05	1.0884E-06	1.4682E-13	0.0000E+00	0.0000E+00
²⁴⁴ Cm	1.8240E+00	1.5249E-02	2.2498E-03	2.7749E-06	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	8.5130E-04	8.4266E-04	8.3923E-04	8.2733E-04	7.8460E-04	6.6647E-04	4.7119E-04
²⁴⁶ Cm	2.4630E-04	2.4183E-04	2.4006E-04	2.3398E-04	2.1271E-04	1.5865E-04	8.5085E-05
²⁴⁷ Cm	7.3030E-10	7.3030E-10	7.3029E-10	7.3029E-10	7.3027E-10	7.3020E-10	7.3006E-10
TOTAL	1.5433E+04	1.0069E+03	3.9981E+02	6.8101E+01	3.0933E+01	3.0259E+01	3.3087E+01

Table D-15. Radioisotope Release History (Ci) for WP15 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	5.0290E-03	4.9535E-03	4.9236E-03	4.8204E-03	4.4556E-03	3.4976E-03	2.0909E-03
³⁶ Cl	8.0680E-04	8.0657E-04	8.0647E-04	8.0615E-04	8.0494E-04	8.0124E-04	7.9343E-04
⁵⁹ Ni	5.6300E-03	5.6235E-03	5.6209E-03	5.6118E-03	5.5782E-03	5.4759E-03	5.2649E-03
⁶³ Ni	1.2160E-01	4.9277E-02	3.4335E-02	9.6946E-03	8.8439E-05	4.6780E-11	0.0000E+00
⁷⁹ Se	8.7450E-03	8.7333E-03	8.7287E-03	8.7124E-03	8.6522E-03	8.4695E-03	8.0940E-03
⁸⁷ Rb	5.2080E-07	5.2080E-07	5.2080E-07	5.2080E-07	5.2080E-07	5.2080E-07	5.2080E-07
⁹⁰ Sr	1.0500E+03	5.3572E+01	1.6294E+01	2.5286E-01	4.8214E-08	0.0000E+00	0.0000E+00
⁹³ Zr	3.4640E-01	3.4638E-01	3.4637E-01	3.4635E-01	3.4624E-01	3.4593E-01	3.4526E-01
⁹⁹ Tc	2.1890E-01	2.1881E-01	2.1878E-01	2.1865E-01	2.1819E-01	2.1677E-01	2.1379E-01
¹⁰⁷ Pd	2.5700E-04	2.5700E-04	2.5700E-04	2.5699E-04	2.5697E-04	2.5692E-04	2.5680E-04
¹¹³ Cd	1.1880E-15	1.1880E-15	1.1880E-15	1.1880E-15	1.1880E-15	1.1880E-15	1.1880E-15
^{113m} Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	3.8120E-03	3.8087E-03	3.8074E-03	3.8028E-03	3.7857E-03	3.7335E-03	3.6251E-03
¹²⁹ I	6.5520E-04	6.5520E-04	6.5519E-04	6.5519E-04	6.5517E-04	6.5511E-04	6.5499E-04
¹³⁵ Cs	4.1650E-03	4.1648E-03	4.1648E-03	4.1646E-03	4.1637E-03	4.1612E-03	4.1559E-03
¹³⁷ Cs	1.1840E+03	6.5924E+01	2.0765E+01	3.6416E-01	1.0937E-07	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	5.8520E-07	5.8520E-07	5.8520E-07	5.8520E-07	5.8520E-07	5.8520E-07	5.8520E-07
¹⁴⁷ Pm	7.4890E-09	1.8864E-03	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	1.8860E-03	1.8860E-03	1.8860E-03	1.8860E-03	1.8860E-03	1.8860E-03	1.8860E-03
¹⁵¹ Sm	1.6810E+01	6.4144E+00	4.3631E+00	1.1324E+00	7.5558E-03	1.5265E-09	0.0000E+00
¹⁵² Eu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	0.0000E+00	2.7977E-07	4.2345E-07	9.0695E-07	2.4082E-06	0.0000E+00	0.0000E+00
²²⁶ Ra	0.0000E+00	5.5146E-06	1.0502E-05	4.0015E-05	2.9305E-04	2.0240E-03	7.5491E-03
²²⁸ Ra	0.0000E+00	1.5125E-03	1.5125E-03	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	0.0000E+00	1.5054E-02	2.1024E-02	4.1687E-02	1.1537E-01	3.1440E-01	6.2499E-01
²³⁰ Th	7.0650E-06	2.0410E-04	2.8415E-04	5.6635E-04	1.6158E-03	4.7967E-03	1.1318E-02
²³² Th	1.5130E-03	1.5130E-03	1.5130E-03	1.5130E-03	1.5130E-03	1.5130E-03	1.5130E-03
²³¹ Pa	0.0000E+00	2.1487E-06	3.0066E-06	6.0020E-06	1.7031E-05	5.0032E-05	1.1569E-04
²³² U	6.3800E-01	1.9140E-01	1.1825E-01	2.1916E-02	4.1862E-05	1.8022E-13	0.0000E+00
²³³ U	1.2830E+00	1.2823E+00	1.2821E+00	1.2812E+00	1.2777E+00	1.2673E+00	1.2453E+00
²³⁴ U	1.7200E-01	1.7754E-01	1.7860E-01	1.8019E-01	1.8041E-01	1.7940E-01	1.7725E-01
²³⁵ U	8.1350E-04	8.1355E-04	8.1357E-04	8.1365E-04	8.1392E-04	8.1471E-04	8.1625E-04
²³⁶ U	6.4280E-03	6.4290E-03	6.4294E-03	6.4307E-03	6.4354E-03	6.4480E-03	6.4671E-03
²³⁸ U	2.0430E-05	2.0430E-05	2.0430E-05	2.0430E-05	2.0430E-05	2.0430E-05	2.0430E-05
²³⁷ Np	7.5190E-02	7.5230E-02	7.5244E-02	7.5284E-02	7.5357E-02	7.5354E-02	7.5252E-02
²³⁸ Pu	2.4880E+01	9.2687E+00	6.2443E+00	1.5672E+00	9.2321E-03	1.6215E-09	0.0000E+00
²³⁹ Pu	4.2840E-01	4.2690E-01	4.2630E-01	4.2421E-01	4.1653E-01	3.9372E-01	3.4912E-01
²⁴⁰ Pu	2.6990E-01	2.6676E-01	2.6536E-01	2.6048E-01	2.4313E-01	1.9667E-01	1.2533E-01
²⁴¹ Pu	0.0000E+00	7.1076E-05	7.0946E-05	6.9955E-05	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	7.6110E-05	7.6097E-05	7.6092E-05	7.6074E-05	7.6006E-05	7.5783E-05	7.5270E-05
²⁴⁴ Pu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴¹ Am	1.1830E+00	9.6813E-01	8.9354E-01	6.7491E-01	2.3803E-01	9.6887E-03	5.2462E-05
^{242m} Am	1.3110E-04	7.4106E-05	5.8987E-05	2.6541E-05	1.3666E-06	1.4848E-10	5.6013E-19
²⁴³ Am	1.0820E-02	1.0693E-02	1.0643E-02	1.0469E-02	9.8481E-03	8.1583E-03	5.4686E-03
²⁴³ Cm	4.5960E-04	2.1940E-05	6.4978E-06	9.1867E-08	1.2393E-14	0.0000E+00	0.0000E+00
²⁴⁴ Cm	1.5390E-01	1.2866E-03	1.8982E-04	2.3413E-07	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	7.1850E-05	7.1121E-05	7.0831E-05	6.9827E-05	6.6221E-05	5.6251E-05	3.9768E-05
²⁴⁶ Cm	2.0790E-05	2.0412E-05	2.0263E-05	1.9750E-05	1.7955E-05	1.3392E-05	7.1820E-06
²⁴⁷ Cm	6.1630E-11	6.1630E-11	6.1630E-11	6.1629E-11	6.1627E-11	6.1622E-11	6.1610E-11
TOTAL	2.2806E+03	1.3925E+02	5.1579E+01	6.9020E+00	3.1789E+00	3.0521E+00	3.2166E+00

Table D-16. Radioisotope Release History (Ci) for WP16 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	1.1260E-05	1.1091E-05	1.1024E-05	1.0793E-05	9.9762E-06	7.8311E-06	4.6816E-06
³⁶ Cl	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁵⁹ Ni	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁶³ Ni	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁷⁹ Se	4.0600E-02	4.0546E-02	4.0524E-02	4.0449E-02	4.0169E-02	3.9321E-02	3.7578E-02
⁸⁷ Rb	2.7330E-06	2.7330E-06	2.7330E-06	2.7330E-06	2.7330E-06	2.7330E-06	2.7330E-06
⁹⁰ Sr	4.8730E+03	2.4862E+02	7.5621E+01	1.1735E+00	2.2376E-07	0.0000E+00	0.0000E+00
⁹³ Zr	2.0850E-01	2.0849E-01	2.0848E-01	2.0847E-01	2.0841E-01	2.0822E-01	2.0782E-01
⁹⁹ Tc	1.3680E+00	1.3674E+00	1.3672E+00	1.3664E+00	1.3636E+00	1.3547E+00	1.3361E+00
¹⁰⁷ Pd	1.5770E-03	1.5770E-03	1.5770E-03	1.5769E-03	1.5768E-03	1.5765E-03	1.5758E-03
¹¹³ Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
^{113m} Cd	3.1440E-01	5.3566E-01	4.1826E-05	5.5644E-09	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	3.6330E-02	3.6299E-02	3.6286E-02	3.6242E-02	3.6079E-02	3.5582E-02	3.4548E-02
¹²⁹ I	2.2430E-03	2.2430E-03	2.2430E-03	2.2430E-03	2.2429E-03	2.2427E-03	2.2423E-03
¹³⁵ Cs	1.0600E-02	1.0600E-02	1.0599E-02	1.0599E-02	1.0597E-02	1.0590E-02	1.0577E-02
¹³⁷ Cs	5.2010E+03	2.8959E+02	9.1214E+01	1.5997E+00	4.8045E-07	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	0.0000E+00	7.2675E-19	7.2675E-19	7.2675E-19	7.2675E-19	7.2675E-19	7.2675E-19
¹⁴⁷ Pm	5.6870E+01	7.6796E-07	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	7.6780E-01	7.6921E-07	7.6921E-07	7.6921E-07	7.6921E-07	7.6921E-07	7.6921E-07
¹⁵¹ Sm	2.8750E+01	1.0971E+01	7.4621E+00	1.9368E+00	1.2923E-02	2.6108E-09	0.0000E+00
¹⁵² Eu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	1.7540E-09	2.0138E-07	3.0148E-07	6.3808E-07	1.6832E-06	0.0000E+00	0.0000E+00
²²⁶ Ra	6.7140E-09	5.7588E-07	9.6762E-07	3.0503E-06	1.9247E-05	1.2525E-04	4.5925E-04
²²⁸ Ra	0.0000E+00	1.1466E-11	1.1466E-11	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	1.0280E-08	2.2315E-06	3.1123E-06	6.1610E-06	1.7033E-05	4.6400E-05	9.2226E-05
²³⁰ Th	4.9250E-06	2.3774E-05	3.3896E-05	7.4022E-05	2.3498E-04	7.2634E-04	1.7507E-03
²³² Th	1.1470E-11	2.9753E-10	4.1282E-10	8.1421E-10	2.3002E-09	6.8814E-09	1.6619E-08
²³¹ Pa	7.8360E-06	3.7636E-05	4.9534E-05	9.1077E-05	2.4404E-04	7.0172E-04	1.6123E-03
²³² U	1.9270E-04	5.7810E-05	3.5715E-05	6.6196E-06	1.2644E-08	5.4434E-17	0.0000E+00
²³³ U	1.8930E-04	2.2530E-04	2.3969E-04	2.9002E-04	4.7661E-04	1.0472E-03	2.2420E-03
²³⁴ U	1.0370E-02	2.1381E-02	2.3512E-02	2.6800E-02	2.7849E-02	2.7698E-02	2.7360E-02
²³⁵ U	1.1290E-02	1.1290E-02	1.1290E-02	1.1291E-02	1.1292E-02	1.1295E-02	1.1301E-02
²³⁶ U	4.6350E-02	4.6353E-02	4.6355E-02	4.6359E-02	4.6375E-02	4.6416E-02	4.6479E-02
²³⁸ U	2.0850E-04	2.0850E-04	2.0850E-04	2.0850E-04	2.0850E-04	2.0850E-04	2.0850E-04
²³⁷ Np	6.6040E-02	6.6391E-02	6.6518E-02	6.6890E-02	6.7631E-02	6.7977E-02	6.7900E-02
²³⁸ Pu	4.8940E+01	1.8232E+01	1.2283E+01	3.0831E+00	1.8187E-02	6.2400E-09	0.0000E+00
²³⁹ Pu	1.6780E+00	1.6720E+00	1.6696E+00	1.6612E+00	1.6305E+00	1.5396E+00	1.3627E+00
²⁴⁰ Pu	9.2760E-01	9.1558E-01	9.1074E-01	8.9399E-01	8.3446E-01	6.7501E-01	4.3014E-01
²⁴¹ Pu	7.1090E+01	1.7322E-01	1.5617E-02	1.4136E-05	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	1.3080E-03	1.3077E-03	1.3076E-03	1.3072E-03	1.3056E-03	1.3008E-03	1.2907E-03
²⁴⁴ Pu	2.3060E-10	2.3060E-10	2.3060E-10	2.3060E-10	2.3060E-10	2.3059E-10	2.3059E-10
²⁴¹ Am	7.6280E+00	8.2412E+00	7.6111E+00	5.7491E+00	2.0108E+00	8.1381E-02	9.5654E-05
^{242m} Am	1.3950E-03	7.8855E-04	6.2767E-04	2.8241E-04	1.4541E-05	1.5800E-09	5.9602E-18
²⁴³ Am	6.7240E-03	6.6454E-03	6.6142E-03	6.5061E-03	6.1200E-03	5.0699E-03	3.3984E-03
²⁴³ Cm	4.3600E-04	2.0813E-05	6.1642E-06	8.7149E-08	1.1757E-14	0.0000E+00	0.0000E+00
²⁴⁴ Cm	6.9510E-02	5.8112E-04	8.5735E-05	1.0575E-07	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	1.1000E-05	1.0888E-05	1.0844E-05	1.0690E-05	1.0138E-05	8.6118E-06	6.0884E-06
²⁴⁶ Cm	7.5670E-07	7.4296E-07	7.3753E-07	7.1885E-07	6.5351E-07	4.8743E-07	2.6141E-07
²⁴⁷ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
TOTAL	1.0292E+04	5.8024E+02	1.9861E+02	1.7924E+01	6.3314E+00	4.1108E+00	3.5874E+00

Table D-17. Radioisotope Release History (Ci) for WP17 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	1.0410E+00	1.0254E+00	1.0192E+00	9.9782E-01	9.2232E-01	7.2400E-01	4.3282E-01
³⁶ Cl	7.7030E-03	7.7008E-03	7.6999E-03	7.6968E-03	7.6853E-03	7.6499E-03	7.5754E-03
⁵⁹ Ni	3.5140E+00	3.5099E+00	3.5083E+00	3.5026E+00	3.4816E+00	3.4178E+00	3.2861E+00
⁶³ Ni	3.5810E+02	1.4512E+02	1.0111E+02	2.8550E+01	2.6044E-01	1.3776E-07	0.0000E+00
⁷⁹ Se	2.7640E-01	2.7603E-01	2.7588E-01	2.7537E-01	2.7347E-01	2.6769E-01	2.5582E-01
⁸⁷ Rb	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁹⁰ Sr	2.4400E+04	1.2449E+03	3.7865E+02	5.8760E+00	1.1204E-06	0.0000E+00	0.0000E+00
⁹³ Zr	1.2200E+00	1.2199E+00	1.2199E+00	1.2198E+00	1.2194E+00	1.2183E+00	1.2160E+00
⁹⁹ Tc	8.8180E+00	8.8144E+00	8.8130E+00	8.8080E+00	8.7893E+00	8.7323E+00	8.6123E+00
¹⁰⁷ Pd	7.5680E-02	7.5679E-02	7.5679E-02	7.5677E-02	7.5672E-02	7.5656E-02	7.5621E-02
¹¹³ Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
^{113m} Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	5.2360E-01	5.2315E-01	5.2296E-01	5.2233E-01	5.1998E-01	5.1282E-01	4.9792E-01
¹²⁹ I	2.1350E-02	2.1350E-02	2.1350E-02	2.1350E-02	2.1349E-02	2.1347E-02	2.1343E-02
¹³⁵ Cs	2.3310E-01	2.3309E-01	2.3309E-01	2.3308E-01	2.3303E-01	2.3289E-01	2.3259E-01
¹³⁷ Cs	3.5380E+04	1.9699E+03	6.2048E+02	1.0882E+01	3.2683E-06	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	0.0000E+00	7.8573E-22	7.8573E-22	7.8573E-22	7.8573E-22	7.8573E-22	7.8573E-22
¹⁴⁷ Pm	3.3720E+01	1.5310E-13	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	0.0000E+00	8.3491E-10	8.3491E-10	8.3491E-10	8.3491E-10	8.3491E-10	8.3491E-10
¹⁵¹ Sm	1.9360E+02	7.3874E+01	5.0249E+01	1.3042E+01	8.7020E-02	1.7581E-08	0.0000E+00
¹⁵² Eu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	4.2230E-07	1.1047E-05	1.5924E-05	3.2277E-05	8.3038E-05	0.0000E+00	0.0000E+00
²²⁶ Ra	1.6110E-06	4.0665E-05	7.0440E-05	2.3602E-04	1.5808E-03	1.0564E-02	3.9042E-02
²²⁸ Ra	0.0000E+00	9.1058E-12	9.1058E-12	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	1.0070E-07	2.2609E-06	3.1175E-06	6.0825E-06	1.6656E-05	4.5216E-05	8.9783E-05
²³⁰ Th	2.4050E-04	1.4247E-03	1.9659E-03	3.9824E-03	1.1768E-02	3.5451E-02	8.4421E-02
²³² Th	9.1090E-12	1.0793E-09	1.5106E-09	3.0122E-09	8.5717E-09	2.5711E-08	6.2145E-08
²³¹ Pa	1.5710E-05	4.6414E-05	5.8673E-05	1.0148E-04	2.5908E-04	7.3063E-04	1.6688E-03
²³² U	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²³³ U	1.8410E-04	3.2329E-04	3.7890E-04	5.7345E-04	1.2946E-03	3.4999E-03	8.1178E-03
²³⁴ U	8.8530E-01	1.1744E+00	1.2303E+00	1.3164E+00	1.3429E+00	1.3354E+00	1.3193E+00
²³⁵ U	1.1640E-02	1.1666E-02	1.1676E-02	1.1713E-02	1.1845E-02	1.2239E-02	1.3003E-02
²³⁶ U	1.7340E-01	1.7342E-01	1.7342E-01	1.7344E-01	1.7352E-01	1.7373E-01	1.7404E-01
²³⁸ U	2.1450E-01	2.1450E-01	2.1450E-01	2.1450E-01	2.1450E-01	2.1450E-01	2.1450E-01
²³⁷ Np	2.5470E-01	3.5515E-01	3.9162E-01	4.9855E-01	7.1308E-01	8.2336E-01	8.2689E-01
²³⁸ Pu	1.2860E+03	4.7908E+02	3.2275E+02	8.1003E+01	4.7703E-01	6.5637E-08	0.0000E+00
²³⁹ Pu	2.1160E+02	2.1088E+02	2.1059E+02	2.0959E+02	2.0590E+02	1.9491E+02	1.7327E+02
²⁴⁰ Pu	3.6140E+00	4.4552E+00	4.4381E+00	4.3576E+00	4.0651E+00	3.2883E+00	2.0954E+00
²⁴¹ Pu	2.0100E+04	4.8974E+01	4.4124E+00	9.6863E-04	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	1.1890E+00	1.1887E+00	1.1886E+00	1.1882E+00	1.1868E+00	1.1825E+00	1.1732E+00
²⁴⁴ Pu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴¹ Am	2.1720E+03	2.3426E+03	2.1635E+03	1.6342E+03	5.7161E+02	2.3131E+01	2.5368E-02
^{242m} Am	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴³ Am	1.1490E+01	1.1356E+01	1.1302E+01	1.1118E+01	1.0458E+01	8.6635E+00	5.8072E+00
²⁴³ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁴ Cm	3.2700E+02	2.7338E+00	4.0333E-01	4.9747E-04	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁶ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁷ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
TOTAL	8.4496E+04	6.5527E+03	3.8868E+03	2.0177E+03	8.1207E+02	2.4900E+02	1.9969E+02

Table D-18. Radioisotope Release History (Ci) for WP18 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
³⁶ Cl	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁵⁹ Ni	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁶³ Ni	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁷⁹ Se	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁸⁷ Rb	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁹⁰ Sr	8.4520E+02	4.3123E+01	1.3116E+01	2.0354E-01	3.8810E-08	0.0000E+00	0.0000E+00
⁹³ Zr	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁹⁹ Tc	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁰⁷ Pd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹¹³ Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
^{113m} Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁹ I	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹³⁵ Cs	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹³⁷ Cs	9.8520E+02	5.4855E+01	1.7278E+01	3.0302E-01	9.1009E-08	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	0.0000E+00	1.4115E-37	1.4115E-37	1.4115E-37	1.4115E-37	1.4115E-37	1.4115E-37
¹⁴⁷ Pm	1.3150E-08	5.9704E-23	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	0.0000E+00	3.2559E-19	3.2559E-19	3.2559E-19	3.2559E-19	3.2559E-19	3.2559E-19
¹⁵¹ Sm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁵² Eu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	0.0000E+00	4.9144E-07	7.4381E-07	1.5931E-06	4.2302E-06	0.0000E+00	0.0000E+00
²²⁶ Ra	0.0000E+00	1.0927E-06	1.7577E-06	5.0739E-06	2.9143E-05	1.8120E-04	6.5508E-04
²²⁸ Ra	0.0000E+00	3.7207E-11	3.7207E-11	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	0.0000E+00	7.0119E-06	9.7925E-06	1.9417E-05	5.3739E-05	1.4644E-04	2.9111E-04
²³⁰ Th	1.2410E-05	3.0017E-05	3.7454E-05	6.4186E-05	1.6495E-04	4.7076E-04	1.0997E-03
²³² Th	3.7220E-11	3.8473E-11	3.8960E-11	4.0714E-11	4.7326E-11	6.8621E-11	1.1736E-10
²³¹ Pa	0.0000E+00	2.3748E-07	3.3229E-07	6.6336E-07	1.8824E-06	5.5297E-06	1.2786E-05
²³² U	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²³³ U	5.9760E-04	6.6509E-04	6.9208E-04	7.8648E-04	1.1364E-03	2.2065E-03	4.4473E-03
²³⁴ U	1.4680E-02	1.6383E-02	1.6712E-02	1.7215E-02	1.7354E-02	1.7257E-02	1.7049E-02
²³⁵ U	8.9910E-05	8.9989E-05	9.0020E-05	9.0130E-05	9.0532E-05	9.1723E-05	9.4038E-05
²³⁶ U	2.0100E-04	2.0200E-04	2.0240E-04	2.0377E-04	2.0866E-04	2.2174E-04	2.4181E-04
²³⁸ U	5.0430E-06	5.0430E-06	5.0430E-06	5.0430E-06	5.0430E-06	5.0431E-06	5.0431E-06
²³⁷ Np	1.2410E-01	1.2410E-01	1.2411E-01	1.2411E-01	1.2410E-01	1.2403E-01	1.2386E-01
²³⁸ Pu	7.5910E+00	2.8279E+00	1.9051E+00	4.7814E-01	2.8158E-03	3.8744E-10	0.0000E+00
²³⁹ Pu	6.4070E-01	6.3841E-01	6.3750E-01	6.3432E-01	6.2264E-01	5.8801E-01	5.2061E-01
²⁴⁰ Pu	2.7280E-01	2.6921E-01	2.6778E-01	2.6286E-01	2.4536E-01	1.9847E-01	1.2647E-01
²⁴¹ Pu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	1.3370E-04	1.3367E-04	1.3366E-04	1.3361E-04	1.3345E-04	1.3296E-04	1.3193E-04
²⁴⁴ Pu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴¹ Am	2.4820E-01	2.0312E-01	1.8747E-01	1.4159E-01	4.9929E-02	2.0204E-03	2.2158E-06
^{242m} Am	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴³ Am	4.8530E-03	4.7962E-03	4.7737E-03	4.6957E-03	4.4171E-03	3.6592E-03	2.4528E-03
²⁴³ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁴ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁶ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁷ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
TOTAL	1.8393E+03	1.0206E+02	3.3539E+01	2.1708E+00	1.0684E+00	9.3691E-01	7.9742E-01

Table D-19. Radioisotope Release History (Ci) for WP19 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	1.4080E-06	1.3869E-06	1.3785E-06	1.3496E-06	1.2475E-06	9.7924E-07	5.8541E-07
³⁶ Cl	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁵⁹ Ni	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁶³ Ni	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁷⁹ Se	5.0750E-03	5.0682E-03	5.0655E-03	5.0561E-03	5.0211E-03	4.9151E-03	4.6972E-03
⁸⁷ Rb	3.4160E-07	3.4160E-07	3.4160E-07	3.4160E-07	3.4160E-07	3.4160E-07	3.4160E-07
⁹⁰ Sr	7.5610E+02	3.8577E+01	1.1733E+01	1.8208E-01	3.4719E-08	0.0000E+00	0.0000E+00
⁹³ Zr	2.6070E-02	2.6069E-02	2.6068E-02	2.6066E-02	2.6053E-02	2.6035E-02	2.5984E-02
⁹⁹ Tc	1.7100E-01	1.7093E-01	1.7090E-01	1.7081E-01	1.7044E-01	1.6934E-01	1.6701E-01
¹⁰⁷ Pd	1.9710E-04	1.9710E-04	1.9710E-04	1.9709E-04	1.9708E-04	1.9704E-04	1.9695E-04
¹¹³ Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
^{113m} Cd	3.9300E-02	6.6957E-05	5.2283E-06	6.9555E-10	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	4.5420E-03	4.5381E-03	4.5365E-03	4.5310E-03	4.5106E-03	4.4485E-03	4.3193E-03
¹²⁹ I	2.8040E-04	2.8040E-04	2.8040E-04	2.8040E-04	2.8039E-04	2.8036E-04	2.8031E-04
¹³⁵ Cs	1.3250E-03	1.3250E-03	1.3249E-03	1.3249E-03	1.3246E-03	1.3238E-03	1.3221E-03
¹³⁷ Cs	8.2140E+02	4.5735E+01	1.4405E+01	2.5264E-01	7.5878E-08	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	0.0000E+00	9.0853E-20	9.0853E-20	9.0853E-20	9.0853E-20	9.0853E-20	9.0853E-20
¹⁴⁷ Pm	7.1090E+00	9.5990E-08	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	9.5970E-08	9.6146E-08	9.6146E-08	9.6146E-08	9.6146E-08	9.6146E-08	9.6146E-08
¹⁵¹ Sm	3.5940E+00	1.3714E+00	9.3283E-01	2.4212E-01	1.6154E-03	3.2637E-10	0.0000E+00
¹⁵² Eu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	2.1920E-10	1.1060E-07	1.6699E-07	3.5671E-07	9.4577E-07	0.0000E+00	0.0000E+00
²²⁶ Ra	8.3930E-10	2.6194E-07	4.2651E-07	1.2633E-06	7.4718E-06	4.7153E-05	1.7128E-04
²²⁸ Ra	0.0000E+00	7.9012E-12	7.9012E-12	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	1.2860E-09	1.4985E-06	2.0922E-06	4.1472E-06	1.1476E-05	3.1270E-05	6.2160E-05
²³⁰ Th	2.7730E-06	8.1898E-06	1.0748E-05	2.0411E-05	5.8049E-05	1.7263E-04	4.1002E-04
²³² Th	7.9040E-12	4.3881E-11	5.8377E-11	1.0886E-10	2.9577E-10	8.7215E-10	2.0979E-09
²³¹ Pa	9.7950E-07	4.7461E-06	6.2500E-06	1.1501E-05	3.0835E-05	8.8683E-05	2.0378E-04
²³² U	2.4090E-05	7.2271E-06	4.4649E-06	8.2753E-07	1.5806E-09	6.8049E-18	0.0000E+00
²³³ U	1.2760E-04	1.4383E-04	1.5032E-04	1.7303E-04	2.5719E-04	5.1455E-04	1.0535E-03
²³⁴ U	3.8480E-03	5.5207E-03	5.8442E-03	6.3428E-03	6.4980E-03	6.4622E-03	6.3839E-03
²³⁵ U	1.4270E-03	1.4270E-03	1.4271E-03	1.4271E-03	1.4273E-03	1.4279E-03	1.4291E-03
²³⁶ U	5.8290E-03	5.8296E-03	5.8298E-03	5.8306E-03	5.8334E-03	5.8409E-03	5.8522E-03
²³⁸ U	2.6940E-05	2.6940E-05	2.6940E-05	2.6940E-05	2.6940E-05	2.6940E-05	2.6940E-05
²³⁷ Np	2.9830E-02	2.9875E-02	2.9891E-02	2.9938E-02	3.0029E-02	3.0060E-02	3.0021E-02
²³⁸ Pu	7.4380E+00	2.7710E+00	1.8668E+00	4.6856E-01	2.7632E-03	8.4746E-10	0.0000E+00
²³⁹ Pu	3.2110E-01	3.1995E-01	3.1949E-01	3.1790E-01	3.1203E-01	2.7464E-01	2.6081E-01
²⁴⁰ Pu	1.6340E-01	1.6127E-01	1.6042E-01	1.5747E-01	1.4698E-01	1.1890E-01	7.5765E-02
²⁴¹ Pu	8.8860E+00	2.1652E-02	1.9520E-03	1.7660E-06	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	1.8680E-04	1.8676E-04	1.8674E-04	1.8668E-04	1.8646E-04	1.8577E-04	1.8432E-04
²⁴⁴ Pu	2.8830E-11	2.8830E-11	2.8830E-11	2.8830E-11	2.8830E-11	2.8829E-11	2.8828E-11
²⁴¹ Am	9.9660E-01	1.0654E+00	9.8394E-01	7.4322E-01	2.6002E-01	1.0523E-02	1.2341E-05
^{242m} Am	1.7440E-04	9.8582E-05	7.8470E-05	3.5307E-05	1.8179E-06	1.9753E-10	7.4513E-19
²⁴³ Am	1.6840E-03	1.6643E-03	1.6565E-03	1.6294E-03	1.5327E-03	1.2697E-03	8.5112E-04
²⁴³ Cm	5.4500E-05	2.6016E-06	7.7052E-07	1.0894E-08	1.4696E-15	0.0000E+00	0.0000E+00
²⁴⁴ Cm	8.6890E-03	7.2643E-05	1.0717E-05	1.3219E-08	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	1.3740E-06	1.3601E-06	1.3545E-06	1.3353E-06	1.2663E-06	1.0757E-06	7.6050E-07
²⁴⁶ Cm	9.4590E-08	9.2872E-08	9.2194E-08	8.9859E-08	8.1691E-08	6.0930E-08	3.2677E-08
²⁴⁷ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
TOTAL	1.6063E+03	9.0276E+01	3.0658E+01	2.6179E+00	9.7716E-01	6.7673E-01	5.8705E-01

Table D-20. Radioisotope Release History (Ci) for WP20 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	3.4680E-01	3.4159E-01	3.3953E-01	3.3241E-01	3.0726E-01	2.4119E-01	1.4419E-01
³⁶ Cl	2.5680E-03	2.5673E-03	2.5670E-03	2.5659E-03	2.5621E-03	2.5503E-03	2.5255E-03
⁵⁹ Ni	1.1710E+00	1.1696E+00	1.1691E+00	1.1672E+00	1.1602E+00	1.1390E+00	1.0951E+00
⁶³ Ni	1.1940E+02	4.8386E+01	3.3713E+01	9.5193E+00	8.6839E-02	4.5934E-08	0.0000E+00
⁷⁹ Se	9.2120E-02	9.1997E-02	9.1948E-02	9.1777E-02	9.1142E-02	8.9218E-02	8.5262E-02
⁸⁷ Rb	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁹⁰ Sr	8.4180E+03	4.2949E+02	1.3063E+02	2.0272E+00	3.8654E-07	0.0000E+00	0.0000E+00
⁹³ Zr	4.0650E-01	4.0648E-01	4.0647E-01	4.0644E-01	4.0632E-01	4.0595E-01	4.0517E-01
⁹⁹ Tc	2.9390E+00	2.9378E+00	2.9373E+00	2.9357E+00	2.9294E+00	2.9104E+00	2.8704E+00
¹⁰⁷ Pd	2.5230E-02	2.5230E-02	2.5230E-02	2.5229E-02	2.5227E-02	2.5222E-02	2.5210E-02
¹¹³ Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
^{113m} Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	1.7450E-01	1.7435E-01	1.7429E-01	1.7408E-01	1.7329E-01	1.7091E-01	1.6594E-01
¹²⁹ I	7.1170E-03	7.1170E-03	7.1169E-03	7.1169E-03	7.1167E-03	7.1161E-03	7.1147E-03
¹³⁵ Cs	7.7700E-02	7.7697E-02	7.7696E-02	7.7692E-02	7.7677E-02	7.7630E-02	7.7530E-02
¹³⁷ Cs	1.2130E+04	6.7539E+02	2.1273E+02	3.7308E+00	1.1205E-06	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	0.0000E+00	2.6195E-22	2.6195E-22	2.6195E-22	2.6195E-22	2.6195E-22	2.6195E-22
¹⁴⁷ Pm	1.1240E+01	5.1032E-14	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	0.0000E+00	2.7830E-10	2.7830E-10	2.7830E-10	2.7830E-10	2.7830E-10	2.7830E-10
¹⁵¹ Sm	6.4530E+01	2.4624E+01	1.6749E+01	4.3472E+00	2.9005E-02	5.8600E-09	0.0000E+00
¹⁵² Eu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	1.4080E-07	3.8492E-06	5.5607E-06	1.1300E-05	2.9116E-05	0.0000E+00	0.0000E+00
²²⁶ Ra	5.3720E-07	1.3927E-05	2.4078E-05	8.0401E-05	5.3687E-04	3.5831E-03	1.3237E-02
²²⁸ Ra	0.0000E+00	1.5644E-11	1.5644E-11	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	3.3560E-08	3.1300E-06	4.3579E-06	8.6081E-06	2.3765E-05	6.4703E-05	1.2859E-04
²³⁰ Th	8.4380E-05	4.8507E-04	6.6801E-04	1.3492E-03	3.9784E-03	1.1976E-02	2.8512E-02
²³² Th	1.5650E-11	3.7269E-10	5.1657E-10	1.0176E-09	2.8724E-09	8.5910E-09	2.0748E-08
²³¹ Pa	5.2360E-06	1.5555E-05	1.9675E-05	3.4060E-05	8.7028E-05	2.4551E-04	5.6083E-04
²³² U	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²³³ U	2.6390E-04	3.3314E-04	3.6081E-04	4.5761E-04	8.1644E-04	1.9137E-03	4.2113E-03
²³⁴ U	3.0010E-01	3.9702E-01	4.1576E-01	4.4462E-01	4.5347E-01	4.5097E-01	4.4551E-01
²³⁵ U	3.9120E-03	3.9207E-03	3.9242E-03	3.9363E-03	3.9807E-03	4.1123E-03	4.3679E-03
²³⁶ U	5.7850E-02	5.7855E-02	5.7858E-02	5.7865E-02	5.7893E-02	5.7967E-02	5.8077E-02
²³⁸ U	7.1510E-02	7.1510E-02	7.1510E-02	7.1510E-02	7.1510E-02	7.1510E-02	7.1510E-02
²³⁷ Np	1.2690E-01	1.6039E-01	1.7255E-01	2.0820E-01	2.7971E-01	3.1645E-01	3.1757E-01
²³⁸ Pu	4.3110E+02	1.6060E+02	1.0820E+02	2.7154E+01	1.5991E-01	2.2003E-08	0.0000E+00
²³⁹ Pu	7.0760E+01	7.0519E+01	7.0423E+01	7.0088E+01	6.8855E+01	6.5178E+01	5.7941E+01
²⁴⁰ Pu	1.2970E+00	1.5762E+00	1.5700E+00	1.5415E+00	1.4381E+00	1.1633E+00	7.4128E-01
²⁴¹ Pu	6.7000E+03	1.6325E+01	1.4708E+00	3.2288E-04	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	3.9650E-01	3.9641E-01	3.9637E-01	3.9624E-01	3.9577E-01	3.9432E-01	3.9124E-01
²⁴⁴ Pu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴¹ Am	7.2420E+02	7.8103E+02	7.2132E+02	5.4485E+02	1.9058E+02	7.7120E+00	8.4576E-03
^{242m} Am	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴³ Am	3.8310E+00	3.7862E+00	3.7684E+00	3.7069E+00	3.4869E+00	2.8886E+00	1.9362E+00
²⁴³ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁴ Cm	1.0900E+02	9.1127E-01	1.3444E-01	1.6582E-04	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁶ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁷ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
TOTAL	2.8790E+04	2.2190E+03	1.3071E+03	6.7337E+02	2.7108E+02	8.3324E+01	6.6841E+01

Table D-21. Radioisotope Release History (Ci) for WP21 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
³⁶ Cl	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁵⁹ Ni	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁶³ Ni	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁷⁹ Se	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁸⁷ Rb	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁹⁰ Sr	1.2260E+03	6.2552E+01	1.9026E+01	2.9525E-01	5.6296E-08	0.0000E+00	0.0000E+00
⁹³ Zr	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁹⁹ Tc	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁰⁷ Pd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹¹³ Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
^{113m} Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁹ I	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹³⁵ Cs	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹³⁷ Cs	1.4290E+03	7.9566E+01	2.5061E+01	4.3952E-01	1.3201E-07	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	0.0000E+00	2.0470E-37	2.0470E-37	2.0470E-37	2.0470E-37	2.0470E-37	2.0470E-37
¹⁴⁷ Pm	1.9080E-08	8.6628E-23	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	0.0000E+00	4.7242E-19	4.7242E-19	4.7242E-19	4.7242E-19	4.7242E-19	4.7242E-19
¹⁵¹ Sm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁵² Eu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	0.0000E+00	7.1280E-07	1.0789E-06	2.3107E-06	6.1356E-06	0.0000E+00	0.0000E+00
²²⁶ Ra	0.0000E+00	1.5849E-06	2.5494E-06	7.3588E-06	4.2266E-05	2.6279E-04	9.5005E-04
²²⁸ Ra	0.0000E+00	5.3971E-11	5.3971E-11	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	0.0000E+00	1.0171E-05	1.4204E-05	2.8164E-05	7.7947E-05	2.1241E-04	4.2225E-04
²³⁰ Th	1.8000E-05	4.3535E-05	5.4321E-05	9.3090E-05	2.3923E-04	6.8274E-04	1.5949E-03
²³² Th	5.3990E-11	5.5808E-11	5.6514E-11	5.9057E-11	6.8646E-11	9.9529E-11	1.7022E-10
²³¹ Pa	0.0000E+00	3.4442E-07	4.8194E-07	9.6209E-07	2.7301E-06	8.0199E-06	1.8545E-05
²³² U	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²³³ U	8.6680E-04	9.6469E-04	1.0038E-03	1.1408E-03	1.6483E-03	3.2005E-03	6.4506E-03
²³⁴ U	2.1290E-02	2.3760E-02	2.4237E-02	2.4967E-02	2.5168E-02	2.5028E-02	2.4727E-02
²³⁵ U	1.3040E-04	1.3051E-04	1.3056E-04	1.3072E-04	1.3130E-04	1.3303E-04	1.3639E-04
²³⁶ U	2.9150E-04	2.9295E-04	2.9353E-04	2.9552E-04	3.0261E-04	3.2158E-04	3.5070E-04
²³⁸ U	7.3140E-06	7.3140E-06	7.3140E-06	7.3140E-06	7.3140E-06	7.3141E-06	7.3142E-06
²³⁷ Np	1.8000E-01	1.8001E-01	1.8001E-01	1.8001E-01	1.8000E-01	1.7990E-01	1.7965E-01
²³⁸ Pu	1.1010E+01	4.1016E+00	2.7632E+00	6.9349E-01	4.0840E-03	5.6194E-10	0.0000E+00
²³⁹ Pu	9.2930E-01	9.2599E-01	9.2466E-01	9.2005E-01	9.0311E-01	8.5288E-01	7.5511E-01
²⁴⁰ Pu	3.9570E-01	3.9049E-01	3.8843E-01	3.8128E-01	3.5589E-01	2.8789E-01	1.8345E-01
²⁴¹ Pu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	1.9390E-04	1.9386E-04	1.9384E-04	1.9378E-04	1.9354E-04	1.9283E-04	1.9133E-04
²⁴⁴ Pu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴¹ Am	3.6000E-01	2.9461E-01	2.7191E-01	2.0537E-01	7.2419E-02	2.9305E-03	3.2139E-06
^{242m} Am	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴³ Am	7.0390E-03	6.9567E-03	6.9240E-03	6.8109E-03	6.4067E-03	5.3074E-03	3.5576E-03
²⁴³ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁴ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁶ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁷ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
TOTAL	2.6679E+03	1.4804E+02	4.8648E+01	3.1487E+00	1.5497E+00	1.3589E+00	1.1566E+00

Table D-22. Radioisotope Release History (Ci) for WP22 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
³⁶ Cl	3.8240E-02	3.8229E-02	3.8225E-02	3.8209E-02	3.8152E-02	3.7977E-02	3.7606E-02
⁵⁹ Ni	2.6680E-01	2.6649E-01	2.6637E-01	2.6594E-01	2.6434E-01	2.5950E-01	2.4950E-01
⁶³ Ni	5.7630E+00	2.3354E+00	1.6272E+00	4.5946E-01	4.1914E-03	2.2171E-09	0.0000E+00
⁷⁹ Se	4.1450E-01	4.1395E-01	4.1373E-01	4.1295E-01	4.1010E-01	4.0144E-01	3.8364E-01
⁸⁷ Rb	2.4680E-05	2.4680E-05	2.4680E-05	2.4680E-05	2.4680E-05	2.4680E-05	2.4680E-05
⁹⁰ Sr	2.7900E+04	1.4235E+03	4.3296E+02	6.7189E+00	1.2811E-06	0.0000E+00	0.0000E+00
⁹³ Zr	1.6420E+01	1.6419E+01	1.6419E+01	1.6417E+01	1.6413E+01	1.6398E+01	1.6366E+01
⁹⁹ Tc	1.0380E+01	1.0376E+01	1.0374E+01	1.0368E+01	1.0346E+01	1.0279E+01	1.0138E+01
¹⁰⁷ Pd	1.2180E-02	1.2180E-02	1.2180E-02	1.2180E-02	1.2179E-02	1.2176E-02	1.2171E-02
¹¹³ Cd	5.6320E-14	5.6320E-14	5.6320E-14	5.6320E-14	5.6320E-14	5.6320E-14	5.6320E-14
^{113m} Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	1.8070E-01	1.8054E-01	1.8048E-01	1.8026E-01	1.7945E-01	1.7698E-01	1.7184E-01
¹²⁹ I	3.1050E-02	3.1050E-02	3.1050E-02	3.1050E-02	3.1049E-02	3.1046E-02	3.1040E-02
¹³⁵ Cs	1.9740E-01	1.9739E-01	1.9739E-01	1.9738E-01	1.9734E-01	1.9722E-01	1.9697E-01
¹³⁷ Cs	3.0640E+04	1.7060E+03	5.3736E+02	9.4240E+00	2.8304E-06	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	2.7740E-05	2.7740E-05	2.7740E-05	2.7740E-05	2.7740E-05	2.7740E-05	2.7740E-05
¹⁴⁷ Pm	1.4470E-08	8.9389E-02	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	8.9370E-02	8.9370E-02	8.9370E-02	8.9370E-02	8.9370E-02	8.9370E-02	8.9370E-02
¹⁵¹ Sm	7.9680E+02	3.0405E+02	2.0681E+02	5.3678E+01	3.5815E-01	7.2358E-08	0.0000E+00
¹⁵² Eu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	0.0000E+00	5.4054E-07	8.1814E-07	1.7523E-06	4.6528E-06	0.0000E+00	0.0000E+00
²²⁶ Ra	0.0000E+00	2.3614E-06	4.1851E-06	1.4380E-05	9.7576E-05	6.5454E-04	2.4213E-03
²²⁸ Ra	0.0000E+00	7.1685E-02	7.1685E-02	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	0.0000E+00	3.5740E-03	4.9913E-03	9.8969E-03	2.7391E-02	7.4643E-02	1.4838E-01
²³⁰ Th	1.3650E-05	7.7336E-05	1.0349E-04	5.4213E-04	1.5911E-03	3.7436E-03	7.1710E-02
²³² Th	7.1710E-02	7.1710E-02	7.1710E-02	7.1710E-02	7.1710E-02	7.1710E-02	7.1710E-02
²³¹ Pa	0.0000E+00	7.3823E-07	1.0330E-06	5.8516E-06	1.7190E-05	3.9748E-05	0.0000E+00
²³² U	1.5120E-01	4.5360E-02	2.8024E-02	9.9208E-06	4.2711E-14	0.0000E+00	0.0000E+00
²³³ U	3.0460E-01	3.0463E-01	3.0464E-01	3.0467E-01	3.0481E-01	3.0521E-01	3.0606E-01
²³⁴ U	5.4930E-02	5.7887E-02	5.8456E-02	5.9319E-02	5.9506E-02	5.9171E-02	5.8461E-02
²³⁵ U	2.7950E-04	2.7959E-04	2.7962E-04	2.7975E-04	2.8020E-04	2.8154E-04	2.8414E-04
²³⁶ U	1.7170E-03	1.7183E-03	1.7188E-03	1.7205E-03	1.7267E-03	1.7433E-03	1.7687E-03
²³⁸ U	9.7080E-06	9.7080E-06	9.7080E-06	9.7080E-06	9.7080E-06	9.7081E-06	9.7082E-06
²³⁷ Np	3.5190E-01	3.5371E-01	3.5434E-01	3.5617E-01	3.5980E-01	3.6150E-01	3.6109E-01
²³⁸ Pu	1.3220E+01	4.9300E+00	3.3223E+00	8.3513E-01	5.0541E-03	1.7350E-08	0.0000E+00
²³⁹ Pu	7.1970E-01	7.1852E-01	7.1804E-01	7.1633E-01	7.0979E-01	6.8803E-01	6.3639E-01
²⁴⁰ Pu	3.2720E-01	3.4272E-01	3.4105E-01	3.3481E-01	3.1246E-01	2.5275E-01	1.6106E-01
²⁴¹ Pu	0.0000E+00	3.3683E-03	3.3621E-03	3.3152E-03	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	1.4700E-04	1.4719E-04	1.4727E-04	1.4752E-04	1.4842E-04	1.5058E-04	1.5309E-04
²⁴⁴ Pu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴¹ Am	4.9670E+01	4.0648E+01	3.7517E+01	2.8337E+01	9.9944E+00	4.0711E-01	2.4291E-03
^{242m} Am	6.2160E-03	3.5137E-03	2.7968E-03	1.2584E-03	6.4794E-05	7.0402E-09	2.6558E-17
²⁴³ Am	3.8710E-01	3.8287E-01	3.8107E-01	3.7485E-01	3.5260E-01	2.9210E-01	1.9580E-01
²⁴³ Cm	2.1780E-02	1.0397E-03	3.0793E-04	4.3535E-06	5.8729E-13	0.0000E+00	0.0000E+00
²⁴⁴ Cm	7.2950E+00	6.0988E-02	8.9978E-03	1.1098E-05	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	3.4050E-03	3.3705E-03	3.3567E-03	3.3091E-03	3.1382E-03	2.6657E-03	1.8846E-03
²⁴⁶ Cm	9.8530E-04	9.6741E-04	9.6034E-04	9.3612E-04	8.5094E-04	6.3468E-04	3.4038E-04
²⁴⁷ Cm	2.9210E-09	2.9210E-09	2.9210E-09	2.9210E-09	2.9209E-09	2.9206E-09	2.9201E-09
TOTAL	5.9443E+04	3.5120E+03	1.2500E+03	1.2971E+02	4.0548E+01	3.0403E+01	2.9628E+01

Table D-23. Radioisotope Release History (Ci) for WP23 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
³⁶ Cl	4.5050E-04	4.5037E-04	4.5032E-04	4.5014E-04	4.4946E-04	4.4740E-04	4.4304E-04
⁵⁹ Ni	2.0550E-01	2.0526E-01	2.0517E-01	2.0484E-01	2.0361E-01	1.9988E-01	1.9217E-01
⁶³ Ni	2.0940E+01	8.4858E+00	5.9126E+00	1.6695E+00	1.5230E-02	8.0558E-09	0.0000E+00
⁷⁹ Se	2.1240E-02	2.1212E-02	2.1200E-02	2.1161E-02	2.1015E-02	2.0571E-02	1.9659E-02
⁸⁷ Rb	3.4180E-07	3.4180E-07	3.4180E-07	3.4180E-07	3.4180E-07	3.4180E-07	3.4180E-07
⁹⁰ Sr	2.9150E+03	1.4873E+02	4.5236E+01	7.0199E-01	1.3385E-07	0.0000E+00	0.0000E+00
⁹³ Zr	9.7410E-02	9.7404E-02	9.7402E-02	9.7395E-02	9.7366E-02	9.7278E-02	9.7090E-02
⁹⁹ Tc	6.8680E-01	6.8652E-01	6.8641E-01	6.8602E-01	6.8457E-01	6.8012E-01	6.7077E-01
¹⁰⁷ Pd	4.6230E-03	4.6229E-03	4.6229E-03	4.6228E-03	4.6225E-03	4.6215E-03	4.6194E-03
¹¹³ Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
^{113m} Cd	3.9320E-02	6.8991E-05	5.2309E-06	6.9590E-10	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	3.5170E-02	3.5140E-02	3.5127E-02	3.5085E-02	3.4927E-02	3.4446E-02	3.3445E-02
¹²⁹ I	1.5290E-03	1.5290E-03	1.5290E-03	1.5290E-03	1.5289E-03	1.5288E-03	1.5285E-03
¹³⁵ Cs	1.4960E-02	1.4959E-02	1.4959E-02	1.4958E-02	1.4955E-02	1.4946E-02	1.4927E-02
¹³⁷ Cs	3.7440E+03	2.0846E+02	6.5661E+01	1.1515E+00	3.4586E-07	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	0.0000E+00	9.0927E-20	9.0927E-20	9.0927E-20	9.0927E-20	9.0927E-20	9.0927E-20
¹⁴⁷ Pm	9.0850E+00	9.6050E-08	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	9.6030E-08	9.6255E-08	9.6255E-08	9.6255E-08	9.6255E-08	9.6255E-08	9.6255E-08
¹⁵¹ Sm	1.4920E+01	5.6932E+00	3.8725E+00	1.0051E+00	6.7063E-03	1.3549E-09	0.0000E+00
¹⁵² Eu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	2.4920E-08	1.1825E-06	1.7427E-06	3.6246E-06	9.4670E-06	0.0000E+00	0.0000E+00
²²⁶ Ra	9.5090E-08	2.0311E-06	3.0475E-06	7.6556E-06	3.7254E-05	2.1110E-04	7.3998E-04
²²⁸ Ra	0.0000E+00	4.0656E-11	4.0656E-11	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	7.1740E-09	7.6703E-06	1.0709E-05	2.1227E-05	5.8738E-05	1.6005E-04	3.1816E-04
²³⁰ Th	2.7590E-05	4.7164E-05	5.5447E-05	8.5238E-05	1.9760E-04	5.3863E-04	1.2401E-03
²³² Th	4.0670E-11	4.2531E-11	4.3262E-11	4.5868E-11	5.5646E-11	8.6789E-11	1.5678E-10
²³¹ Pa	1.8990E-06	2.1813E-06	2.2939E-06	2.6872E-06	4.1352E-06	8.4677E-06	1.7088E-05
²³² U	1.2050E-07	3.6150E-08	2.2334E-08	4.1394E-09	7.9065E-12	3.4039E-20	0.0000E+00
²³³ U	6.5310E-04	7.3593E-04	7.6904E-04	8.8488E-04	1.3143E-03	2.6274E-03	5.3770E-03
²³⁴ U	1.6300E-02	1.8252E-02	1.8629E-02	1.9207E-02	1.9366E-02	1.9258E-02	1.9026E-02
²³⁵ U	1.0870E-04	1.0879E-04	1.0883E-04	1.0896E-04	1.0944E-04	1.1086E-04	1.1361E-04
²³⁶ U	2.9930E-04	3.0056E-04	3.0107E-04	3.0284E-04	3.0920E-04	3.2610E-04	3.5205E-04
²³⁸ U	6.8370E-05	6.8370E-05	6.8370E-05	6.8370E-05	6.8370E-05	6.8370E-05	6.8370E-05
²³⁷ Np	1.5220E-01	1.5691E-01	1.5854E-01	1.6333E-01	1.7288E-01	1.7778E-01	1.7775E-01
²³⁸ Pu	8.7000E+00	3.2413E+00	2.1837E+00	5.4808E-01	3.2315E-03	9.1217E-10	0.0000E+00
²³⁹ Pu	7.6290E-01	7.6258E-01	7.6245E-01	7.6194E-01	7.5961E-01	7.4854E-01	7.1064E-01
²⁴⁰ Pu	2.9960E-01	3.4768E-01	3.4621E-01	3.3991E-01	3.1714E-01	2.5654E-01	1.6348E-01
²⁴¹ Pu	5.9230E+00	1.4433E-02	1.3016E-03	1.6242E-06	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	4.9460E-04	4.9449E-04	4.9444E-04	4.9428E-04	4.9369E-04	4.9188E-04	4.8804E-04
²⁴⁴ Pu	1.4420E-13	1.4420E-13	1.4420E-13	1.4420E-13	1.4420E-13	1.4420E-13	1.4419E-13
²⁴¹ Am	1.2830E+02	1.0516E+02	9.7059E+01	7.3309E+01	2.5849E+01	1.0460E+00	1.1479E-03
^{242m} Am	1.7450E-04	9.8639E-05	7.8515E-05	3.5327E-05	1.8189E-06	1.9764E-10	7.4556E-19
²⁴³ Am	6.7780E-01	6.6987E-01	6.6673E-01	6.5584E-01	6.1692E-01	5.1106E-01	3.4257E-01
²⁴³ Cm	5.4530E-05	2.6031E-06	7.7095E-07	1.0900E-08	1.4704E-15	0.0000E+00	0.0000E+00
²⁴⁴ Cm	1.9140E+01	1.6002E-01	2.3608E-02	2.9118E-05	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	1.3750E-06	1.3610E-06	1.3555E-06	1.3363E-06	1.2673E-06	1.0765E-06	7.6105E-07
²⁴⁶ Cm	9.4650E-08	9.2931E-08	9.2253E-08	8.9916E-08	8.1743E-08	6.0969E-08	3.2697E-08
²⁴⁷ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
TOTAL	6.8690E+03	4.8297E+02	2.2297E+02	8.1394E+01	2.8826E+01	3.8176E+00	2.4580E+00

Table D-24. Radioisotope Release History (Ci) for WP24 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
³⁶ Cl	7.1930E-03	7.1909E-03	7.1901E-03	7.1872E-03	7.1764E-03	7.1434E-03	7.0738E-03
⁵⁹ Ni	5.0200E-02	5.0142E-02	5.0119E-02	5.0038E-02	4.9738E-02	4.8826E-02	4.6944E-02
⁶³ Ni	1.0840E+00	4.3928E-01	3.0608E-01	8.6423E-02	7.8839E-04	4.1702E-10	0.0000E+00
⁷⁹ Se	7.7970E-02	7.7866E-02	7.7825E-02	7.7679E-02	7.7142E-02	7.5514E-02	7.2166E-02
⁸⁷ Rb	4.6440E-06	4.6440E-06	4.6440E-06	4.6440E-06	4.6440E-06	4.6440E-06	4.6440E-06
⁹⁰ Sr	5.0740E+03	2.5888E+02	7.8740E+01	1.2219E+00	2.3299E-07	0.0000E+00	0.0000E+00
⁹³ Zr	3.0880E+00	3.0878E+00	3.0878E+00	3.0875E+00	3.0866E+00	3.0838E+00	3.0779E+00
⁹⁹ Tc	1.9520E+00	1.9512E+00	1.9509E+00	1.9498E+00	1.9457E+00	1.9330E+00	1.9065E+00
¹⁰⁷ Pd	2.2920E-03	2.2920E-03	2.2920E-03	2.2919E-03	2.2918E-03	2.2913E-03	2.2902E-03
¹¹³ Cd	1.0590E-14	1.0590E-14	1.0590E-14	1.0590E-14	1.0590E-14	1.0590E-14	1.0590E-14
^{113m} Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	3.3990E-02	3.3961E-02	3.3949E-02	3.3908E-02	3.3755E-02	3.3290E-02	3.2323E-02
¹²⁹ I	5.8420E-03	5.8420E-03	5.8420E-03	5.8419E-03	5.8417E-03	5.8412E-03	5.8401E-03
¹³⁵ Cs	3.7140E-02	3.7139E-02	3.7138E-02	3.7136E-02	3.7129E-02	3.7106E-02	3.7059E-02
¹³⁷ Cs	5.5590E+03	3.0952E+02	9.7492E+01	1.7098E+00	5.1352E-07	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	5.2180E-06	5.2180E-06	5.2180E-06	5.2180E-06	5.2180E-06	5.2180E-06	5.2180E-06
¹⁴⁷ Pm	0.0000E+00	1.6814E-02	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	1.6810E-02	1.6810E-02	1.6810E-02	1.6810E-02	1.6810E-02	1.6810E-02	1.6810E-02
¹⁵¹ Sm	1.4990E+02	5.7199E+01	3.8907E+01	1.0098E+01	6.7377E-02	1.3612E-08	0.0000E+00
¹⁵² Eu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁶ Ra	0.0000E+00	4.6819E-05	9.0920E-05	3.5529E-04	2.6456E-03	1.8382E-02	6.8673E-02
²²⁸ Ra	0.0000E+00	1.3485E-02	1.3485E-02	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	0.0000E+00	1.3423E-01	1.8746E-01	3.7170E-01	1.0287E+00	2.8034E+00	5.5728E+00
²³⁰ Th	0.0000E+00	1.7876E-03	2.5116E-03	5.0600E-03	1.4527E-02	4.3215E-02	1.0202E-01
²³² Th	1.3490E-02	1.3490E-02	1.3490E-02	1.3490E-02	1.3490E-02	1.3490E-02	1.3490E-02
²³¹ Pa	0.0000E+00	3.5604E-05	4.9820E-05	9.9455E-05	2.8222E-04	8.2905E-04	1.9170E-03
²³² U	5.6880E+00	1.7064E+00	1.0542E+00	1.9539E-01	3.7321E-04	1.6067E-12	0.0000E+00
²³³ U	1.1440E+01	1.1434E+01	1.1431E+01	1.1423E+01	1.1390E+01	1.1291E+01	1.1084E+01
²³⁴ U	1.5660E+00	1.6067E+00	1.6145E+00	1.6260E+00	1.6271E+00	1.6180E+00	1.5986E+00
²³⁵ U	1.3480E-02	1.3480E-02	1.3480E-02	1.3480E-02	1.3481E-02	1.3482E-02	1.3484E-02
²³⁶ U	5.6290E-02	5.6294E-02	5.6295E-02	5.6300E-02	5.6317E-02	5.6363E-02	5.6432E-02
²³⁸ U	2.8710E-01	2.8710E-01	2.8710E-01	2.8710E-01	2.8710E-01	2.8710E-01	2.8710E-01
²³⁷ Np	4.0530E-02	4.0869E-02	4.0987E-02	4.1332E-02	4.2016E-02	4.2351E-02	4.2308E-02
²³⁸ Pu	1.8330E+02	6.8286E+01	4.6004E+01	1.1546E+01	6.8021E-02	1.2491E-08	0.0000E+00
²³⁹ Pu	5.6780E-01	5.6602E-01	5.6531E-01	5.6282E-01	5.5363E-01	5.2597E-01	4.7050E-01
²⁴⁰ Pu	1.0220E+00	1.0123E+00	1.0069E+00	9.8844E-01	9.2260E-01	7.4631E-01	4.7557E-01
²⁴¹ Pu	0.0000E+00	6.3370E-04	6.3254E-04	6.2371E-04	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	0.0000E+00	4.2263E-08	5.8950E-08	1.1639E-07	3.1712E-07	8.2585E-07	1.5117E-06
²⁴⁴ Pu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴¹ Am	9.2920E+00	7.6043E+00	7.0184E+00	5.3012E+00	1.8697E+00	7.6163E-02	4.5654E-04
^{242m} Am	1.1690E-03	6.6080E-04	5.2598E-04	2.3666E-04	1.2185E-05	1.3240E-09	4.9946E-18
²⁴³ Am	7.1880E-02	7.1039E-02	7.0706E-02	6.9551E-02	6.5423E-02	5.4198E-02	3.6329E-02
²⁴³ Cm	4.0980E-03	1.9562E-04	5.7938E-05	8.1912E-07	1.1050E-13	0.0000E+00	0.0000E+00
²⁴⁴ Cm	1.3720E+00	1.1470E-02	1.6922E-03	2.0872E-06	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	6.4060E-04	6.3410E-04	6.3152E-04	6.2257E-04	5.9041E-04	5.0152E-04	3.5457E-04
²⁴⁶ Cm	1.8530E-04	1.8194E-04	1.8061E-04	1.7603E-04	1.6003E-04	1.1936E-04	6.4013E-05
²⁴⁷ Cm	5.4950E-10	5.4950E-10	5.4950E-10	5.4949E-10	5.4948E-10	5.4943E-10	5.4932E-10
TOTAL	1.1004E+04	7.2419E+02	2.9010E+02	5.0887E+01	2.3287E+01	2.2835E+01	2.5029E+01

Table D-25. Radioisotope Release History (Ci) for WP25 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
³⁶ Cl	1.6190E-03	1.6185E-03	1.6183E-03	1.6177E-03	1.6153E-03	1.6078E-03	1.5922E-03
⁵⁹ Ni	7.3860E-01	7.3775E-01	7.3741E-01	7.3621E-01	7.3180E-01	7.1839E-01	6.9070E-01
⁶³ Ni	7.5280E+01	3.0507E+01	2.1256E+01	6.0018E+00	5.4751E-02	2.8961E-08	0.0000E+00
⁷⁹ Se	7.6350E-02	7.6248E-02	7.6208E-02	7.6065E-02	7.5540E-02	7.3945E-02	7.0666E-02
⁸⁷ Rb	1.2290E-06	1.2290E-06	1.2290E-06	1.2290E-06	1.2290E-06	1.2290E-06	1.2290E-06
⁹⁰ Sr	7.1590E+03	3.6526E+02	1.1110E+02	1.7240E+00	3.2873E-07	0.0000E+00	0.0000E+00
⁹³ Zr	3.5010E-01	3.5008E-01	3.5007E-01	3.5004E-01	3.4994E-01	3.4962E-01	3.4895E-01
⁹⁹ Tc	2.4690E+00	2.4680E+00	2.4676E+00	2.4662E+00	2.4610E+00	2.4450E+00	2.4114E+00
¹⁰⁷ Pd	1.6620E-02	1.6620E-02	1.6620E-02	1.6619E-02	1.6618E-02	1.6615E-02	1.6607E-02
¹¹³ Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
^{113m} Cd	1.4130E-01	2.4074E-01	1.8798E-05	2.5008E-09	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	1.2640E-01	1.2629E-01	1.2625E-01	1.2609E-01	1.2553E-01	1.2380E-01	1.2020E-01
¹²⁹ I	5.4970E-03	5.4970E-03	5.4970E-03	5.4969E-03	5.4968E-03	5.4963E-03	5.4952E-03
¹³⁵ Cs	5.3770E-02	5.3768E-02	5.3767E-02	5.3764E-02	5.3754E-02	5.3721E-02	5.3653E-02
¹³⁷ Cs	9.5880E+03	5.3385E+02	1.6815E+02	2.9490E+00	8.8570E-07	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	0.0000E+00	3.2687E-19	3.2687E-19	3.2687E-19	3.2687E-19	3.2687E-19	3.2687E-19
¹⁴⁷ Pm	3.2660E+01	3.4527E-07	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	3.4520E-07	3.4601E-07	3.4601E-07	3.4601E-07	3.4601E-07	3.4601E-07	3.4601E-07
¹⁵¹ Sm	5.3620E+01	2.0460E+01	1.3917E+01	3.6122E+00	2.4101E-02	4.8693E-09	0.0000E+00
¹⁵² Eu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	8.9570E-08	2.3197E-06	3.3422E-06	6.7704E-06	1.7411E-05	0.0000E+00	0.0000E+00
²²⁶ Ra	3.4180E-07	1.1331E-05	2.0211E-05	7.0744E-05	4.8972E-04	3.3146E-03	1.2294E-02
²²⁸ Ra	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	2.5790E-08	1.4620E-07	1.9395E-07	3.5922E-07	9.4860E-07	2.5405E-06	5.0246E-06
²³⁰ Th	5.0420E-05	4.0714E-04	5.6507E-04	1.1453E-03	3.3650E-03	1.0111E-02	2.4032E-02
²³² Th	0.0000E+00	3.5327E-10	4.9564E-10	9.9134E-10	2.8265E-09	8.4839E-09	2.0509E-08
²³¹ Pa	6.8250E-06	4.1672E-05	5.5586E-05	1.0416E-04	2.8304E-04	8.1824E-04	1.8831E-03
²³² U	8.6650E-05	2.5995E-05	1.6060E-05	2.9766E-06	5.6854E-09	2.4477E-17	0.0000E+00
²³³ U	1.0260E-05	4.2885E-05	5.5918E-05	1.0151E-04	2.7054E-04	7.8737E-04	1.8696E-03
²³⁴ U	2.7930E-01	3.4465E-01	3.5728E-01	3.7671E-01	3.8254E-01	3.8042E-01	3.7582E-01
²³⁵ U	1.3200E-02	1.3206E-02	1.3208E-02	1.3215E-02	1.3244E-02	1.3328E-02	1.3491E-02
²³⁶ U	5.7240E-02	5.7244E-02	5.7246E-02	5.7253E-02	5.7275E-02	5.7335E-02	5.7425E-02
²³⁸ U	2.8980E-01	2.8980E-01	2.8980E-01	2.8980E-01	2.8980E-01	2.8980E-01	2.8980E-01
²³⁷ Np	5.9660E-02	8.0939E-02	8.8665E-02	1.1132E-01	1.5676E-01	1.8012E-01	1.8086E-01
²³⁸ Pu	2.9080E+02	1.0833E+02	7.2984E+01	1.8317E+01	1.0788E-01	1.6525E-08	0.0000E+00
²³⁹ Pu	4.5130E+01	4.4976E+01	4.4915E+01	4.4701E+01	4.3914E+01	4.1568E+01	3.6951E+01
²⁴⁰ Pu	1.1250E+00	1.2971E+00	1.2916E+00	1.2681E+00	1.1832E+00	9.5710E-01	6.0989E-01
²⁴¹ Pu	4.2580E+03	1.0375E+01	9.3474E-01	2.1001E-04	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	2.5060E-01	2.5054E-01	2.5052E-01	2.5044E-01	2.5014E-01	2.4922E-01	2.4728E-01
²⁴⁴ Pu	1.0370E-10	1.0370E-10	1.0370E-10	1.0370E-10	1.0370E-10	1.0370E-10	1.0369E-10
²⁴¹ Am	4.6010E+02	4.9624E+02	4.5830E+02	3.4618E+02	1.2109E+02	4.9000E+00	5.3766E-03
^{242m} Am	6.2710E-04	3.5448E-04	2.8216E-04	1.2695E-04	6.5368E-06	7.1025E-10	2.6793E-18
²⁴³ Am	2.4170E+00	2.3887E+00	2.3775E+00	2.3387E+00	2.1999E+00	1.8224E+00	1.2216E+00
²⁴³ Cm	1.9600E-04	9.3563E-06	2.7711E-06	3.9177E-08	5.2851E-15	0.0000E+00	0.0000E+00
²⁴⁴ Cm	6.8780E+01	5.7502E-01	8.4834E-02	1.0464E-04	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	4.9430E-06	4.8928E-06	4.8729E-06	4.8038E-06	4.5557E-06	3.9698E-06	2.7359E-06
²⁴⁶ Cm	3.4020E-07	3.3402E-07	3.3158E-07	3.2318E-07	2.9381E-07	2.1914E-07	1.1752E-07
²⁴⁷ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
TOTAL	2.2040E+04	1.6191E+03	9.0021E+02	4.3203E+02	1.7355E+02	5.4221E+01	4.3712E+01

Table D-26. Radioisotope Release History (Ci) for WP26 (CUTTINGS)

Radioisotope	Initial Inventory	Intrusion Time					
		125 yr	175 yr	350 yr	1000 yr	3000 yr	7250 yr
¹⁴ C	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
³⁶ Cl	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁵⁹ Ni	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁶³ Ni	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁷⁹ Se	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁸⁷ Rb	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁹⁰ Sr	6.4510E+02	3.2914E+01	1.0011E+01	1.5535E-01	2.9622E-08	0.0000E+00	0.0000E+00
⁹³ Zr	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
⁹⁹ Tc	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁰⁷ Pd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹¹³ Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
^{113m} Cd	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁶ Sn	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹²⁹ I	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹³⁵ Cs	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹³⁷ Cs	7.5200E+02	4.1871E+01	1.3188E+01	2.3129E-01	6.9467E-08	0.0000E+00	0.0000E+00
¹⁴⁴ Nd	0.0000E+00	1.0773E-37	1.0773E-37	1.0773E-37	1.0773E-37	1.0773E-37	1.0773E-37
¹⁴⁷ Pm	1.0040E-08	4.5584E-23	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁴⁷ Sm	0.0000E+00	2.4859E-19	2.4859E-19	2.4859E-19	2.4859E-19	2.4859E-19	2.4859E-19
¹⁵¹ Sm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
¹⁵² Eu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²¹⁰ Pb	0.0000E+00	3.7505E-07	5.6766E-07	1.2158E-06	3.2283E-06	0.0000E+00	0.0000E+00
²²⁶ Ra	0.0000E+00	8.3385E-07	1.3413E-06	3.8715E-06	2.2236E-05	1.3824E-04	4.9979E-04
²²⁸ Ra	0.0000E+00	2.8400E-11	2.8400E-11	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²²⁹ Th	0.0000E+00	5.3516E-06	7.4738E-06	1.4819E-05	4.1015E-05	1.1177E-04	2.2218E-04
²³⁰ Th	9.4710E-06	2.2904E-05	2.8579E-05	4.8975E-05	1.2586E-04	3.5919E-04	8.3908E-04
²³² Th	2.8410E-11	2.9367E-11	2.9738E-11	3.1077E-11	3.6122E-11	5.2374E-11	8.9572E-11
²³¹ Pa	0.0000E+00	1.8124E-07	2.5361E-07	5.0628E-07	1.4366E-06	4.2203E-06	9.7586E-06
²³² U	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²³³ U	4.5610E-04	5.0761E-04	5.2820E-04	6.0025E-04	8.6732E-04	1.6840E-03	3.3941E-03
²³⁴ U	1.1200E-02	1.2500E-02	1.2751E-02	1.3135E-02	1.3241E-02	1.3167E-02	1.3009E-02
²³⁵ U	6.8620E-05	6.8680E-05	6.8704E-05	6.8788E-05	6.9095E-05	7.0004E-05	7.1770E-05
²³⁶ U	1.5340E-04	1.5416E-04	1.5447E-04	1.5552E-04	1.5924E-04	1.6923E-04	1.8455E-04
²³⁸ U	3.8490E-06	3.8490E-06	3.8490E-06	3.8490E-06	3.8490E-06	3.8490E-06	3.8491E-06
²³⁷ Np	9.4710E-02	9.4713E-02	9.4714E-02	9.4716E-02	9.4710E-02	9.4656E-02	9.4526E-02
²³⁸ Pu	5.7940E+00	2.1584E+00	1.4541E+00	3.6495E-01	2.1492E-01	2.9572E-01	0.0000E+00
²³⁹ Pu	4.8900E-01	4.8726E-01	4.8656E-01	4.8413E-01	4.7522E-01	4.4879E-01	3.9734E-01
²⁴⁰ Pu	2.0820E-01	2.0546E-01	2.0437E-01	2.0062E-01	1.8725E-01	1.5147E-01	9.6524E-02
²⁴¹ Pu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴² Pu	1.0200E-04	1.0198E-04	1.0197E-04	1.0193E-04	1.0181E-04	1.0144E-04	1.0065E-04
²⁴⁴ Pu	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴¹ Am	1.8940E-01	1.5500E-01	1.4305E-01	1.0805E-01	3.8100E-02	1.5418E-03	1.6908E-06
^{242m} Am	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴³ Am	3.7040E-03	3.6607E-03	3.6435E-03	3.5840E-03	3.3713E-03	2.7928E-03	1.8721E-03
²⁴³ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁴ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁵ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁶ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
²⁴⁷ Cm	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
TOTAL	1.4039E+03	7.7902E+01	2.5599E+01	1.6568E+00	8.1544E-01	7.1506E-01	6.0860E-01

TRI-6342-4201-0

TRI-6342-4202-0

(a) Graphite spent fuel

Figure D-2. Normalized activity decay history of individual radionuclides, based on 40 CFR 191 criterion, per waste type. (a) Graphite spent fuel, (b) highly enriched uranium special spent fuel, (c) low enriched uranium special spent fuel, (d) calcine high level-waste, (e) high-level waste from Savannah River, (f) high-level waste from Hanford, (g) high-level waste from West Valley, and (h) special spent fuel with depleted uranium.

TRI-6342-4203-0

TRI-6342-4204-0

(b) Highly enriched uranium special spent fuel

Figure D-2. Normalized activity decay history of individual radionuclides, based on 40 CFR 191 criterion, per waste type. (a) Graphite spent fuel, (b) highly enriched uranium special spent fuel, (c) low enriched uranium special spent fuel, (d) calcine high level-waste, (e) high-level waste from Savannah River, (f) high-level waste from Hanford, (g) high-level waste from West Valley, and (h) special spent fuel with depleted uranium (continued).

TRI-6342-4205-0

TRI-6342-4206-0

(c) Low enriched uranium special spent fuel

Figure D-2. Normalized activity decay history of individual radionuclides, based on 40 CFR 191 criterion, per waste type. (a) Graphite spent fuel, (b) highly enriched uranium special spent fuel, (c) low enriched uranium special spent fuel, (d) calcine high level-waste, (e) high-level waste from Savannah River, (f) high-level waste from Hanford, (g) high-level waste from West Valley, and (h) special spent fuel with depleted uranium (continued).

TRI-6342-4207-0

TRI-6342-4208-0

(d) Calcine High-level waste

Figure D-2. Normalized activity decay history of individual radionuclides, based on 40 CFR 191 criterion, per waste type. (a) Graphite spent fuel, (b) highly enriched uranium special spent fuel, (c) low enriched uranium special spent fuel, (d) calcine high level-waste, (e) high-level waste from Savannah River, (f) high-level waste from Hanford, (g) high-level waste from West Valley, and (h) special spent fuel with depleted uranium (continued).

TRI-6342-4209-0

TRI-6342-4210-0

(e) High-level waste from Savannah River

Figure D-2. Normalized activity decay history of individual radionuclides, based on 40 CFR 191 criterion, per waste type. (a) Graphite spent fuel, (b) highly enriched uranium special spent fuel, (c) low enriched uranium special spent fuel, (d) calcine high level-waste, (e) high-level waste from Savannah River, (f) high-level waste from Hanford, (g) high-level waste from West Valley, and (h) special spent fuel with depleted uranium (continued).

TRI-6342-4211-0

TRI-6342-4212-0

(f) High-level waste from Hanford

Figure D-2. Normalized activity decay history of individual radionuclides, based on 40 CFR 191 criterion, per waste type. (a) Graphite spent fuel, (b) highly enriched uranium special spent fuel, (c) low enriched uranium special spent fuel, (d) calcine high level-waste, (e) high-level waste from Savannah River, (f) high-level waste from Hanford, (g) high-level waste from West Valley, and (h) special spent fuel with depleted uranium (continued).

TRI-6342-4213-0

TRI-6342-4214-0

(g) High-level waste from West Valley

Figure D-2. Normalized activity decay history of individual radionuclides, based on 40 CFR 191 criterion, per waste type. (a) Graphite spent fuel, (b) highly enriched uranium special spent fuel, (c) low enriched uranium special spent fuel, (d) calcine high level-waste, (e) high-level waste from Savannah River, (f) high-level waste from Hanford, (g) high-level waste from West Valley, and (h) special spent fuel with depleted uranium (continued).

TRI-6342-4215-0

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(h) Special spent fuel with depleted uranium

Figure D-2. Normalized activity decay history of individual radionuclides, based on 40 CFR 191 criterion, per waste type. (a) Graphite spent fuel, (b) highly enriched uranium special spent fuel, (c) low enriched uranium special spent fuel, (d) calcine high level-waste, (e) high-level waste from Savannah River, (f) high-level waste from Hanford, (g) high-level waste from West Valley, and (h) special spent fuel with depleted uranium (continued).

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Appendix E: Demonstration Calculation for Dose to Humans

Appendix E: Demonstration Calculation for Dose to Humans

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Appendix E: Demonstration Calculation for Dose to Humans

R. D. McCurley

This study was done in response to a request by the Technical Peer and Program Review Panels for dose calculations of contaminants released from hypothetical salt disposal systems storing spent nuclear fuel and high-level radioactive waste currently stored at the Idaho National Engineering Laboratory (INEL), because of interest by the general scientific community, and also in anticipation of evolving changes in regulatory requirements for repositories. The scope of this calculation was limited by the time period allocated for its completion and also because of its purpose; that is, to *demonstrate* the capability for dose assessment for future consideration.

For this study, a borehole intrusion into a salt repository, coupled with transport to a remote ranch that accesses groundwater from an aquifer, was considered (see Figure 7-5 and Figure 3-1, in Volume 1 of this report). The transport modeling of groundwater to the remote site used dual porosity with no retardation (see Section 12.2). This part of the analysis (biosphere modeling) is schematically indicated in Figure 1-1 of Volume 1 of this report.

The primary computer code used for this purpose is GENII-S, developed originally at Pacific Northwest Laboratories (PNL) and modified more recently for specific performance assessment needs at Sandia National Laboratories (Napier et al., 1988; Leigh et al., 1993). This program was developed to estimate "potential radiation doses to humans from radionuclides in the environment" (Leigh et al., 1993). The most recent modifications included the addition of the SUNS software (Sensitivity and Uncertainty Analysis Shell). GENII-S is *not* part of CAMCON (Rechard, Ed., 1992; Rechard et al., 1993); it is supported only on personal computers. It has its own sampling output display and input preprocessing capability. SUNS offers several useful features including a fully interactive menu-driven environment, a flexible data editor, Monte Carlo and Latin hypercube sampling, the ability to perform both repeated trials and parametric studies in a single run, and statistical and graphical output.

GENII-S uses radioactive concentration source terms, exposure pathway specification (i.e., transport mechanism, inhalation, external exposure, ingestion), and biological transfer models to determine the potential harm or damage to humans due to radiological effects. GENII-S incorporates radioactive decay using Bateman's equations for chains. GENII-S can calculate surface water, air, and biotic transport, but not groundwater transport. However, it does use concentrations in groundwater for drinking water by animal stock or humans and for irrigation.

E.1 Quality Assurance

The GENII package of codes was developed under a stringent quality assurance (QA) plan based on the American National Standards Institute (ANSI) standard NQA-1 (ASME, 1986) as implemented in the PNL Quality Assurance Manual PNL-MA-70 (a). At steps of the code, development has been documented and tested. Extensive hand calculations have been performed and are available for review. An external peer review of the internal dosimetry portions of the code was held in Richland, Washington, the week of September 14, 1987. Copies of the reports issued by these reviewers are available. Recommendations of these review committees were then incorporated into the final product (PNL, 1991, excerpted from Vol. 1 of GENII-The Hartford Environmental Radiation Dosimetry Software System, Napier et al., 1988).

E.2 Calculation Description

The exposure pathway considered here consists of (1) groundwater being pumped into a stockpond and made available for drinking water for cattle, (2) cattle drinking the contaminated water, and (3) cattle used for meat, which is consumed by humans who live on a ranch near the waste repository. For the purposes of this calculation, it is assumed that this scenario occurs about 10,000 yr after an intrusion borehole event. The calculations that produce the concentration in the drinking water are not done by GENII but are described below.

A succinct description of the exposure pathway considered here is as follows:

Groundwater → animal stock → man

Other pathway groups that could be accommodated might include:

1. Resuspension of contaminated dust producing an air plume with deposition on a remote farm where a human family resides.
2. Direct external exposure to contaminated cuttings collected by a human driller for examination.

The term pathway group refers to a collection of exposure pathways. The collection may consist of only one pathway (e.g., meat ingestion by a human), or many pathways (e.g., a hypothetical scenario where ingestion, inhalation, and external exposure might all be considered). All pathway groups presume an inadvertent borehole intrusion into the repository at some time after loss of institutional control.

To supply the GENII code with appropriate input and to use its Latin hypercube sampling (LHS) capabilities, a software tool named PREGENII was developed to compute concentrations of radionuclides (in appropriate units) for critical pathways, waste types, and scenarios. (See Figure E-1 for a schematic representation of this process.) In addition, this software tool can supply discrete, uniform, normal, or lognormal fits to a sample of concentrations. In the case of normal or lognormal fits, the number of standard deviations (from the mean) from which significant data can be included is a specifiable item. (This capability gives the analyst more flexibility in using judgment regarding the nature or integrity of the data.)

An examination of contour plots of plume concentrations for 20 distinct parameterizations (or vectors) using the groundwater transport results at the 2.5-km boundary, "downstream" from a hypothetical borehole intrusion of the waste disposal site, suggested locations that would be representative of maximal concentrations (in space). Three locations spaced equidistant within this bounded region on the 2.5-km boundary, were selected as representative of maximal concentrations for the 20 vectors. These three locations corresponded to nodes (of the local domain mesh used for groundwater transport, see Section 12.3) where a hypothetical stockwell might be located.

Using PREGENII, calculations were performed to determine concentrations (in pico curies/liter) in the stockpond using the following procedures. Once a site (i.e., one of the three locations mentioned above) for the stockpond was selected, it was necessary to determine how much water needed to be pumped from the aquifer to satisfy the drinking requirements of the cattle; the requirements are based on the amount supplied by a typical stockpond in that area. In general, a cow in a semi-arid environment drinks about 10 m^3 of water per year. Assuming that a stockwell supplies about 25 cattle and that the evaporation rate is 25 m^3 per year means that about 275 m^3 of water need to be pumped per year. Typically, pumping occurs about 7 months each year or about 0.6 years. Thus during the actual time of pumping, the rate is about $4.67 \times 10^{-7} \text{ m}^3/\text{sec}$ (personal communication, Tom Blaine, September 1991). In addition, the water consumed by the cattle should not have a brine concentration greater than 7 parts per 1000 or a specific gravity greater than 1.007. Many of the vectors showed concentrations greater than this so that another source of water was hypothetically supplied to dilute the brine concentrations adequately (i.e., so that the brine concentration was the maximum allowed value).

The local grid on which transport calculations were done has been described in previous parts of this report (Section 12.3). An assumption made was that the influence of the stockwell would extend to a locus defined by a cell of this mesh centered at that node (in the aquifer) where the stockwell intrusion was located. Because the elements are squares 125 m per side, the radius of influence is 62.5 m. Thus, this assumption appears to be conservative because the effects on the movement of brine from the presence of the pumping well are very small at a 50-m radius. To sustain the flow at the well, assuming a steady-state flow and an aquifer thickness of about 1 m, a flow rate of about $4 \times 10^{-9} \text{ m}^3/\text{s}$ is required, which is less than the typical flow rates found in the aquifer at the locations chosen. Thus the concentrations and necessary flow rates at the chosen nodal locations are reasonable for determining the amount of water available for pumping and the extent of radiological contamination in the water.

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Figure E-1. Schematic representation of modeling units and flow of information in dose calculations.

The concentration of radionuclides ingested by humans depends on (1) the concentration of radionuclides in the stockpond, (2) the drinking rate from the pond of the stock cattle, (3) transfer factors that determine how much of the radioactive contaminants actually pass into the meat of the animal ingested by man, and, of course, (4) how much meat is actually ingested by the humans residing at the ranch site. The extent of the radiological damage to various organs of the human body also depends, calculationally, on transfer factors that determine how much of the ingested contamination is transferred into various organs and on factors called "internal dose factors." Internal dose factors relate radiation types (γ , β , α) and energies, e.g., from nuclear disintegration to biological effects, measured in units of rem or sieverts (Sv).

The transfer factors needed by GENII-S come from ICRP-30 and ICRP-48 (updated actinide data). The internal dose factors are calculated by GENII using the ICRP lung model, and these agree closely with those published in U.S. DOE, 1988.

The effective dose equivalent (EDE) for the whole body, is therefore:

$$EDE = \sum_j w_j H_j, \quad (E.1)$$

where j is an organ index, w_j is a weighing factor for a particular organ, and H_j is a "dose equivalent" for that same organ. H_j is calculated for the period of integration shown in Figure E-2, which illustrates conceptually a 50-yr committed dose. H_j also depends on the radiation types and energies from radioactive disintegrations in the body organs. A detailed description of this can be found in Volume 1 of GENII-1 (Leigh et al., 1993) and the Hanford Environmental Radiation Dosimetry Software System (Napier et al., 1988).

For the calculations, the various human organs are weighted as listed below:

gonads	0.25
breast:	0.15
red bone marrow	0.12

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Figure E-2. Situation showing committed dose (area under curve) from 1-yr internal intake dose.

1	lung:	0.12
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3	thyroid:	0.03
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5	bone:	0.03
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7	remainder:	0.30
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10 To obtain the committed dose to the whole body, the committed effective dose equivalent (CEDE) is summed over all
11 nuclides. Thus,

$$12 \quad CEDE = \sum_k \sum_j w_j H_j^k, \quad (E.2)$$

13
14 where H_j^k depends on organ j and nuclide k . Here, the annual EDE (the abscissa in the complementary cumulative
15 distribution function [CCDF]) is the average annual CEDE.
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19 E.3 Results

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21 The radionuclides that were transported (using STAFF2D) through the Snake aquifer to the point of extraction at
22 the stockpond are; ^{241}Am , ^{14}C , ^{237}Np , ^{239}Pu , ^{240}Pu , ^{226}Ra , ^{229}Th , ^{230}Th , ^{233}U , ^{234}U , and ^{238}U . Except for ^{14}C ,
23 these actinides comprised two decay chains. In addition, for dose calculations, ^{210}Pb , which is a descendant of ^{226}Ra ,
24 makes a contribution and is included by assuming that it is in secular equilibrium with ^{226}Ra , which is long-lived in
25 comparison. (The half lives for ^{226}Ra and ^{210}Pb are 1.0×10^3 and 22.3 yr, respectively.)
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30 The predicted mean annual, whole-body effective dose equivalent (EDE) (averaged over a 50 yr period) from a
31 1-yr internal ingestion of meat from stock that had drunk contaminated water from a well 2.4-km from a human intru-
32 sion into the INEL salt disposal system is about 4×10^{-9} rem (Table E-1). The plume concentrations for the INEL
33 disposal system at a maximum 5 km boundary are extremely small, so small that most values of the 20 simulations
34 are lost in numerical noise. The human intrusion is assumed to occur through exploratory drilling 1000 yr after
35 emplacement of the INEL spent nuclear fuel and high-level radioactive waste into the repository. The exposure to the
36 stock is assumed to occur about 8500 yr later or 9500 yr after emplacement (the peak plume concentration at a 2.4 km
37 boundary within the assumed 10,000 yr regulatory period). Included in Table E-1 are two committed doses (EDE) of
38 the organs (kidneys and testes) significantly contributing to the predicted whole body EDE.
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44 **Table E-1. Comparison between mean annual 50-yr committed effective dose equivalents (EDE) from INEL,**
45 **WIPP, and PSAG salt disposal systems predicted by GENII-S**

Salt Disposal System	Mean annual 50-yr committed effective dose equivalents			
	Whole Body (rem)	Kidneys (rem)	Testes (rem)	Liver (rem)
INEL (2.4 km boundary)	4.49×10^{-7}	1.69×10^{-11}	9.62×10^{-9}	6.78×10^{-8}
WIPP (5-km boundary)	7.84×10^{-7}	4.95×10^{-7}	6.98×10^{-8}	5.84×10^{-7}
PSAG (5-km boundary)	5.84×10^{-6}	Not Available	6.98×10^{-8}	5.84×10^{-7}

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51 In Table E-1, the EDE from the hypothetical INEL salt disposal system are compared to the dose calculations
52 performed in 1992 for the Waste Isolation Pilot Plant (WIPP), which will potentially dispose of transuranic waste
53 from several U.S. Department of Energy (DOE) defense facilities, and calculations for a hypothetical salt disposal
54 system conducted in 1992 by the Probabilistic System Assessment Group (PSAG) of the Nuclear Energy Agency
55 (NEA). Both the WIPP and PSAG results are representative of the largest doses that occur at a 5-km boundary from
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the center of the waste disposal region. In both cases, the whole body EDE is larger than from the INEL salt disposal system. Furthermore, the whole body EDE estimate from the INEL salt disposal system is about 4 orders of magnitude smaller than the other common sources of radiation doses to humans (Figure E-3).

E.4 Acknowledgments

I want to express my gratitude to Dr. Leo Rahal for his assistance in producing the results from GENII-S.

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Appendix F: Disposition of Events and Processes as a Result of Screening for a Disposal Facility in Bedded Salt and Saturated Granite

Appendix F: Disposition of Events and Processes

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Appendix F: Disposition of Events and Processes

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Appendix F: Disposition of Events and Processes as a Result of Screening for a Disposal Facility in Bedded Salt and Saturated Granite

R. V. Guzowski

F.1 Hypothetical Disposal Facility in Bedded Salt

The screening step of the scenario-development procedure for events and processes at a hypothetical disposal facility in bedded salt is implemented below. Figure F-1 provides the potentially disruptive scenarios for a bedded salt location.

F.1.1 Naturally Occurring Events and Processes

Celestial Bodies

Name: METEORITE IMPACT

Screening Criterion: Probability of occurrence

Disposition: Screened out from scenario development

Discussion: Several estimates have been made of the probability of an impact from a sufficiently large meteorite to significantly disrupt the performance of a deep nuclear-waste disposal facility. Depending on the assumptions of facility size, depth, and impact frequency, these estimates range from approximately $1 \times 10^{-13}/\text{yr}$ to $1 \times 10^{-12}/\text{yr}$ (Claiborne and Gera, 1974; Kärn-Bränsle-Säkerhet, 1978; Logan and Berbano, 1978; Hartmann, 1979; Cranwell et al., 1990). Each of these estimates is below the limit established in the Standard of 1 chance in 10,000 in 10,000 yr for those events and processes that must be included in performance assessments.

Surficial Events and Processes

Name: EROSION/SEDIMENTATION

Screening Criterion: Base-case conditions

Disposition: Part of base-case scenario

Discussion: Both erosion and sedimentation resulting from water and/or wind action will occur to some extent at any nuclear-waste disposal location during the 10,000 yr of regulatory concern. The possible consequences of erosion and sedimentation need to be considered in the base-case scenario. For the bedded-salt location in this exercise, neither erosion nor sedimentation will be of sufficient magnitude to have an impact on disposal-system performance.

Name: GLACIATION

Screening Criterion: Physical reasonableness

Disposition: Screened out from scenario development

Discussion: Based on the extent of past glaciations and projections of future climatic changes within the time frame of regulatory concern, the bedded-salt location will be selected to exclude the likelihood of both continen-

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Figure F-1. Potentially disruptive scenarios for a bedded-salt location.

tal and alpine glaciations affecting disposal-system performance. Changes in future climatic conditions that would result in glaciations within the next 10,000 yr that are far more extensive than past glaciations are not physically reasonable.

Name: PLUVIAL PERIODS [Climate Change]

Screening Criterion: Base-case condition

Disposition: Part of base-case scenario

Discussion: Pluvial periods refers to time intervals characterized by relatively high amounts of precipitation when compared to current conditions. Because climatic conditions are continuously changing when viewed on a sufficiently long time scale, any disposal location is certain to experience both wetter and drier climatic conditions during the 10,000 yr of regulatory concern. Whether the wetter conditions at the bedded-salt location will be "pluvial" or simply wetter is an academic exercise. For the purposes of performance assessments, climatic changes that include wetter conditions will occur, and the effects of these changes on the performance of the disposal system are part of the base-case conditions.

Name: SEA-LEVEL VARIATIONS

Screening Criterion: Physical reasonableness

Disposition: Screened out from scenario development

Discussion: Changes in sea level are the result of climatic changes adding to or reducing the amount of water in the oceans through increases or decreases in the size of continental glaciers, eustatic adjustments due to changes in response to the load upon the crust, and/or tectonic activity. The bedded-salt site has been selected to preclude the possibility of inundation resulting from melting of the world's continental glaciers. Eustatic and tectonic contributions to sea-level changes that could result in inundation of the disposal location are not physically reasonable within the time frame of regulatory concern.

Name: HURRICANES

Screening Criterion: Physical reasonableness

Disposition: Screened out from scenario development

Discussion: Hurricanes are storms that originate over ocean water in the tropics of the northern hemisphere (cyclones in the southern hemisphere). These storms are characterized by high wind speeds and heavy rainfall. Although these storms may migrate out of the tropics, they dissipate quickly over cold water or land.

Hurricanes can be eliminated from performance assessments, because the survival of storms in this category is not physically reasonable as far inland as any disposal facility in bedded salt. Intense storms are certain to occur at any disposal location, and for this reason, intense storms are part of normal climatic conditions.

Name: SEICHES

Screening Criterion: Physical reasonableness

Disposition: Screened out from scenario development

Discussion: A seiche is a "free or standing-wave oscillation of the surface of water in an enclosed or semi-enclosed basin...that is initiated chiefly by local changes in atmospheric pressure, aided by winds, tidal currents, and small earthquakes; and that continues, pendulum fashion, for a time after cessation of the originating force"

(Bates and Jackson, 1987, p. 600). Seiches range in height from several centimeters to a few meters. The location of any disposal facility in bedded salt is far enough from large bodies of water for the mechanism to generate seiches of sufficient size to reach an inland location to be physically unreasonable.

Name: TSUNAMIS

Screening Criterion: Physical reasonableness

Disposition: Screened out from scenario development

Discussion: A tsunami is a "gravitational sea wave produced by any large-scale, short-duration disturbance of the ocean floor, principally by a shallow submarine earthquake, but also by submarine earth movement, subsidence, or volcanic eruption" (Bates and Jackson, 1987, p. 704). The location of the disposal facility is far enough from the ocean that an occurrence of a geologic disturbance on a scale that would result in a tsunami large enough to inundate a major part of a continent to reach the disposal-facility location is not physically reasonable.

Name: REGIONAL SUBSIDENCE OR UPLIFT

Screening Criterion: Physical reasonableness

Disposition: Screened out from scenario development

Discussion: Bedded-salt deposits typically are located in regions where tectonic activity is extremely low to nonexistent. The apparent long-term tectonic stability of regions that contain bedded salt suggests that changes in the tectonic setting that would result in regional subsidence or uplift on a scale that would have a significant effect on disposal-system performance during the period of regulatory concern is not physically reasonable given the current understanding of tectonic processes and their origin.

Name: MASS WASTING [including Landslides]

Screening Criterion: Base-case condition

Disposition: Part of base-case scenario

Discussion: Mass wasting is the dislodgment and downslope movement of soil and rock under the direct application of gravitational body forces (Bates and Jackson, 1987). Because of the tectonic stability, the geomorphic features in areas of bedded-salt deposits typically have more gentle rather than precipitous relief, except possibly along streams. Large-scale mass movements, such as entire hill slopes, are not likely to occur. Mass wasting is likely to occur along perennial and ephemeral streams as banks are undercut by channel migration and collapse into the stream channel. This type of small-scale mass wasting is common and is included in the base-case conditions. Any damming of the streams by the collapse of banks will tend to be short lived, because the material will be eroded by the stream and incorporated into the stream's bedload. No perennial rivers or streams exist at the location of the hypothetical disposal facility.

Name: FLOODING

Screening Criterion: Physical reasonableness

Disposition: Screened out from scenario development

Discussion: Flooding refers to the overbank flow of a stream or river. No streams or rivers exist at the location of the hypothetical disposal facility or are likely to exist at this location under wetter climatic conditions. The existing river in the region is at a sufficiently lower elevation and at a distance far enough from the disposal-facility location that the conditions for flooding, even under wetter climatic conditions, to reach the facility location are not physically reasonable.

Subsurface Events and Processes

Name: DIAPIRISM [Halokinesis]

Screening Criterion: Physical reasonableness

Disposition: Screened out from scenario development

Discussion: Halokinesis refers to autonomous, isostatic salt movement (Trusheim, 1960). Because of the low density of salt, a stratigraphic sequence in which salt is overlain by more dense sediments is unstable. This instability can be corrected if the sequence is rearranged so that the less dense salt is higher in the sequence than the more dense sediments. The buoyancy of the salt can be accommodated by solid-state flow (salt creep) and the formation of salt anticlines and diapirs. Certain conditions are necessary for salt creep to form these structures. In German salt basins, a sedimentary cover overlying the salt unit of at least 1,000 m and a salt thickness of at least 300 m are necessary for salt-dome or diapir formation (Trusheim, 1960). Similar conditions have been noted for the Gulf of Mexico (Halbouty, 1979).

The presence of undeformed bedded salt at the location of the disposal facility indicates that the conditions necessary for halokinesis to occur have not been met since the deposition of the salt. Changes in the geologic setting that would initiate halokinesis within the time period of regulatory concern are not physically reasonable.

Name: SEISMIC ACTIVITY

Screening Criterion: Base-case condition

Disposition: Part of base-case scenario

Discussion: Sedimentary basins that contain undeformed bedded-salt deposits typically are subjected to low levels of seismic activity. Epicenters of earthquakes are more likely to occur along the edges of the basin or in adjacent tectonic provinces rather than within the central portions of the basin. The certainty that the basin will be subjected to seismic waves places this phenomenon in the base-case scenario, although low-level seismic activity will have no effect on the performance of the disposal system.

Name: VOLCANIC ACTIVITY

Screening Criterion: Physical reasonableness

Disposition: Screened out from scenario development

Discussion: Sedimentary basins in the U.S. that contain bedded-salt deposits typically are not tectonically active at the present time, do not contain anomalously high geothermal gradients (Muffler, 1979), and contain few if any units of volcanic origin. The overall stability of these basins suggests that geologic changes within the time frame of regulatory concern that would result in volcanic activity within the basins are not physically reasonable.

Name: MAGMATIC ACTIVITY

Screening Criterion: Physical reasonableness

Disposition: Screened out from scenario development

Discussion: Sedimentary basins in the U.S. that contain bedded-salt deposits do not have the anomalous heat flow and seismic activity characteristic of actively mobile magma bodies at depth. The changes in the geologic setting necessary for magmatic activity to become an active process and to affect disposal-system performance is not physically reasonable within the time frame of regulatory concern.

1 Name: FORMATION OF DISSOLUTION CAVITIES

2
3 *Screening Criterion:* Physical reasonableness

4
5
6 *Disposition:* Screened out from scenario development

7
8 *Discussion:* Through the site-selection procedure, the location of the disposal facility and the surrounding area
9 do not contain evidence (e.g., breccia chimneys) of deep salt dissolution. A change of the geologic setting to an
10 extent that deep dissolution of the salt deposits would be initiated within the time frame of regulatory concern is
11 not physically reasonable.
12

13
14 Through the site-selection procedure, the location of the disposal facility and the surrounding area do not contain
15 evidence (e.g., karst features) of shallow salt dissolution on a scale that could affect disposal-system perfor-
16 mance. A change of the geologic setting in which shallow dissolution of the salt deposits could be initiated or
17 increased to an extent that would affect performance within the time frame of regulatory concern is not physi-
18 cally reasonable.
19
20

21 Name: FORMATION OF INTERCONNECTED FRACTURE SYSTEMS

22
23 *Screening Criterion:* Physical reasonableness

24
25
26 *Disposition:* Screened out from scenario development

27
28 *Discussion:* Fracture systems do not spontaneously occur but are the product of the occurrence of some event or
29 process. Under the pressure, temperature, and stress environment of the candidate salt unit, salt creep is the
30 exclusive mechanism of deformation. Based on the tectonic stability of the basin and the absence of detectable
31 deep dissolution features, the initiation of one or more naturally occurring events or processes that would impose
32 a sufficiently high strain rate to cause fracturing of the salt instead of salt creep is not physically reasonable
33 within the time frame of regulatory concern.
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37 Name: FAULTING

38
39 *Screening Criterion:* Physical reasonableness

40
41
42 *Disposition:* Screened out from scenario development

43
44 *Discussion:* Faulting refers to either the creation of a new fault or the renewed movement of an existing fault. To
45 reduce the accumulation of tectonically induced stresses, movement generally occurs along existing faults with
46 the formation of a new fault that is not subsidiary to an existing fault being a rare event (Bonilla, 1979). The
47 location of faults within a depositional basin containing bedded salt usually can be detected during site character-
48 ization with either remote-sensing techniques, or the recognition of surface lineaments resulting from the dis-
49 placement of units or from subsidence associated with dissolution of salt along the fault. Because of the tectonic
50 stability of the region identified for the hypothetical disposal system and the absence of faults that penetrate the
51 salt formation, changes in the geologic setting that would result in the formation of faults within the time frame
52 of regulatory concern are not physically reasonable.
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F.1.2 Human-Induced Events and Processes

Inadvertent Intrusion

Name: EXPLOSIONS

Screening Criterion: Probability of occurrence

Disposition: Screened out from scenario development

Discussion: Explosions can be detonated either at the surface or in the subsurface and can be either chemical or nuclear. For scenario development, all explosions are eliminated from consideration because of low probability of occurrence. Specific probability values are not estimated in reaching this conclusion, but the combination of low probability requirements strongly suggests that explosions can be eliminated from consideration based on probability arguments.

The first question to address about surface explosions is whether such explosions will occur. Because surface explosions have no particular geographic requirements, the next question is why would the location of the disposal facility in bedded salt be selected over all other possible locations, with the added constraint that the location is selected without knowledge that the facility is present. The third question to address is the size of the explosion. Only an explosion of sufficient magnitude to either crater or deeply fracture the underlying geologic strata would likely significantly affect disposal-system performance. Chemical explosions seem to be unlikely because of the volumes of explosives required, and nuclear explosions are unlikely because of perceived health risks to the population. As explained in Guzowski and Helton (1991, Section 4.1.4), regulatory restrictions on the severity of types of intrusion would eliminate this mode of directly penetrating the disposal facility, so the disruptive effects would have to be on transportation of radionuclides rather than on their release from the facility. The combined probabilities of surface detonations occurring in the future, of the appropriate location being selected, and of the required magnitude of an explosion arguably results in a probability of occurrence below the cutoff provided in the Standard.

Underground explosions present similar questions to those for surface explosions. The first question is whether subsurface explosions of either chemical or nuclear origin, exclusive of those in mining, will occur in the future. Whereas bedded salt would seem to be a good candidate medium for underground explosions because of salt creep's ability to seal fractures, only one of the hundreds of nuclear tests conducted by the United States was in bedded salt (NVO, 1987). No records are readily available (if any exist) on underground chemical explosions. The question of location also arises. With vast areas of the contiguous United States underlain by bedded salt, why would a location near the bedded-salt disposal facility be selected? Location is an important factor in the probability argument because of the restrictions placed on the severity of intrusion type by the Standard. As with surface explosions, the subsurface explosion would have to affect the transport of radionuclides without directly intruding the disposal facility. The location question is further complicated by limits to the magnitude of an explosion. Bedded salt may have favorable physical properties for containing explosions, but the magnitude would have to be restricted so as not to rupture the salt unit and thereby negate the medium's favorable characteristics. The combined probabilities of whether subsurface explosions will occur in the future, of the appropriate location, and of the required magnitude of the explosion arguably result in a probability of occurrence below the cutoff provided in the Standard.

Name: DRILLING (For Resources)

Screening Criterion: None

Disposition: Retained for scenario development

Discussion: The section on *Institutional Controls* of Appendix B of the Standard states that "...the Agency [EPA] believes that passive institutional controls can never be assumed to eliminate the chance of inadvertent and

intermittent human intrusion into these disposal sites" (EPA, 1985, p. 38088). Within the section of Appendix B entitled *Consideration of Inadvertent Human Intrusion into Geologic Repositories*, the statement is made

The Agency [EPA] believes that the most productive consideration of inadvertent intrusion concerns those realistic possibilities that may be usefully mitigated by repository design, site selection, or use of passive controls (although passive controls should not be assumed to completely rule out the possibility of intrusion). Therefore, inadvertent and intermittent intrusion by exploratory drilling for resources (other than any provided by the disposal system itself) can be the most severe intrusion scenario assumed by the implementing agencies. (EPA, 1985, p. 38089).

These statements in the Standard are interpreted to mean that exploratory drilling for resources has a probability of occurrence greater than the cutoff stated in EPA's standard and must be included in the development of scenarios for performance assessments.

Name: MINING

Screening Criterion: None

Disposition: Retained for scenario development

Discussion: Bedded-salt deposits can contain other minerals in sufficient quantity to be economically minable. Mineral deposits in locally economic concentrations are assumed to be present in the region of the disposal facility. Mining can occur by either conventional underground methods or by injection and dissolution. Salt itself can be mined as a resource under the appropriate geologic and economic conditions, but the salt within and near the hypothetical disposal system can be discounted as a possible resource because of the extensive deposits of more readily accessible salt at other locations.

Appendix B of the Standard provides guidance on the limits to the severity of human intrusion at the disposal location. As discussed under the topic of Drilling (above), exploratory drilling for resources is the most severe type of human intrusion that must be included in performance assessments. Mining for resources would have a more disruptive effect on the disposal system than drilling. Whereas mining is retained for scenario development, the location of the mining must be beyond the boundaries of the disposal facility so as to have no direct effect on facility integrity.

Name: (Drilling of) INJECTION WELLS

Screening Criterion: None

Disposition: Retained for scenario development

Discussion: Injection of liquid waste, such as brine associated with hydrocarbon production, does occur in basins where bedded-salt deposits are located and may occur in the future. The severity of drilling injection wells on the disposal system is not necessarily greater than drilling exploratory boreholes for resources, so drilling injection wells cannot be eliminated from consideration on severity or consequence arguments. Probability estimates of drilling injection wells at the hypothetical repository are not available, so this screening criterion cannot be used to eliminate this event from consideration.

Although drilling injection wells is retained for scenario development at this time, this event may be combined with drilling for resources for the single event of drilling in order to reduce the number of scenarios that will be developed for analyses.

1 Name: (Drilling of) WITHDRAWAL WELLS

2
3 *Screening Criterion:* Physical reasonableness

4
5
6 *Disposition:* Screened out from scenario development

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9 *Discussion:* All ground water within a 5-km radius of the hypothetical disposal facility has a salinity that is too
10 high for consumption by humans or livestock or for use in irrigation. Because of the hydrologic properties of the
11 geologic units at the bedded-salt location, recharge of sufficient quantities to create sources of potable water
12 under projected wetter climatic conditions is not physically reasonable.

13 14 **Hydrologic Stresses**

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16 Name: IRRIGATION

17
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19 *Screening Criterion:* Physical reasonableness

20
21
22 *Disposition:* Screened out from scenario development

23
24 *Discussion:* At the location of the hypothetical disposal facility, water of sufficient quality and in sufficient quan-
25 tities is not available to support irrigation of crops. Under wetter climatic conditions, farming may be possible at
26 this location, although the climate change will not be of sufficient magnitude or duration to have a significant
27 effect on the groundwater flow system. Irrigation is eliminated from consideration, because changes in the cli-
28 matic and/or groundwater flow systems on a scale that would make irrigation feasible are not physically reason-
29 able.

30
31
32 Name: DAMMING OF STREAMS OR RIVERS

33
34
35 *Screening Criterion:* Physical reasonableness

36
37
38 *Disposition:* Screened out from scenario development

39
40 *Discussion:* No streams or rivers are located in the immediate vicinity of the hypothetical disposal-facility loca-
41 tion, and these features are not likely to form under wetter climatic conditions. The only river in the region is at
42 a distance from the facility and at an elevation sufficiently lower than the surface elevation at the facility to pre-
43 clude impounded water at any location along the length of the river from reaching the disposal-facility location
44 or to significantly affect regional groundwater flow patterns. Because of the geologic stability of the region,
45 changes in the river morphology that would allow damming of the river to affect the regional hydrology are not
46 physically reasonable within the time frame of regulatory concern.

47 48 49 50 **F.1.3 Waste- and Repository-Induced Events and Processes**

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52 Name: SUBSIDENCE AND CAVING

53
54
55 *Screening Criterion:* None

56
57
58 *Disposition:* Retained for scenario development

1 Name: SHAFT AND BOREHOLE SEAL DEGRADATION

2
3 *Screening Criterion:* Base-case condition

4
5
6 *Disposition:* Part of base-case scenario

7
8 *Discussion:* Alteration of the nonsalt backfill material in either the shafts or boreholes will occur during the
9 10,000 yr of regulatory concern. As a result, this process is part of the base-case scenario. A catastrophic failure
10 of seals in these locations could be included in scenario development, although no mechanism for this type of
11 failure has been recognized as physically reasonable under current or projected conditions.
12

13
14 Name: THERMALLY INDUCED STRESSES/FRACTURING IN HOST ROCK

15
16 *Screening Criterion:* Base-case condition/consequence

17
18
19 *Disposition:* Part of base-case condition/screened out from scenario development

20
21 *Discussion:* Spent fuel and high-level waste will generate heat and, as a result, thermally induced stresses will
22 contribute to the stress field within the host rock surrounding the hypothetical disposal facility. Because heat is
23 certain to be generated, thermally induced stresses in the host rock are part of the base-case scenario.
24

25
26 In the early time periods after closure of the facility when differential stresses between the host rock and the
27 excavation are large, thermal effects may cause some fracturing of the host rock. Because elevated temperatures
28 increase the strain rate of salt creep, the fractures will heal as a result of salt creep once the stresses in the host
29 rock and the excavation approach equilibrium. Thermally induced fractures are short-term features that will be
30 of no consequence to the performance of the disposal facility.
31

32
33 Name: EXCAVATION-INDUCED STRESS/FRACTURING IN HOST ROCK

34
35
36 *Screening Criterion:* Base-case condition

37
38
39 *Disposition:* Part of base-case scenario

40
41 *Discussion:* In deep excavations in bedded salt, the differential stress between the host rock and the excavation
42 results in deformation with a high strain rate that produces fractures rather than salt creep in the salt. A disturbed
43 zone, which has a higher porosity than the intact host rock because of the presence of fractures, typically forms
44 around an excavation. After the facility is closed, the increasing pressure exerted by the host rock on the backfill
45 in the rooms and drifts as the salt deforms into the excavations will reduce the porosity of the backfill to a point
46 where the backfill will begin to resist further compaction. As the differential stress between the host rock and the
47 backfill approaches equilibrium, the deformation mechanism of salt creep will replace fracturing, and the frac-
48 tures will heal unless some noncompressible fluid (noncompressible from a practical viewpoint) is present within
49 the fractures. The stress field at the depth of the spent-fuel disposal facility along with the elevated temperatures
50 produced by radionuclide decay assure that salt creep will rapidly close the excavations and compact the backfill.
51 No source of fluid, either liquid or gas, has been recognized as available to fill in the pore space within the dis-
52 turbed zone to an extent that would prevent salt creep from sealing the fractures and pores within the short time
53 period until stresses approximate pre-excavation conditions.
54
55

56
57 Name: NUCLEAR CRITICALITY

58
59
60 *Screening Criterion:* None

61
62
63 *Disposition:* Retained for scenario development

64
65
66

Discussion: In an analysis of criticality issues associated with the disposal of highly enriched nuclear fuel, Palmer (1992) determined that certain configurations of fuel elements under certain environmental conditions (primarily the availability of water to act as a moderator) could result in criticality of the intact stored fuel. The potential for critical conditions developing by the concentration of radionuclides along possible radionuclide transport pathways was not evaluated. Until the particular environmental conditions of the stored waste and the transported radionuclides are determined, the question of whether criticality is physically reasonable cannot be answered with any assurance. Criticality is retained for scenario development pending further analysis.

F.2 Hypothetical Disposal Facility in Saturated Granite

The screening step of the scenario-development procedure for events and processes at a hypothetical disposal facility in granite is implemented below. Figure F-2 shows the potentially disruptive scenarios for a granite location.

F.2.1 Naturally Occurring Events and Processes

Celestial Bodies

Name: METEORITE IMPACT

Screening Criterion: Probability of occurrence

Disposition: Screened out from scenario development

Discussion: Several estimates have been made of the probability of the impact of a sufficiently large meteorite to significantly disrupt the performance of a deep nuclear-waste disposal facility. Depending on the assumptions of facility size, depth, and impact frequency, these estimates range from approximately $1 \times 10^{-13}/\text{yr}$ to $1 \times 10^{-12}/\text{yr}$ (Claiborne and Gera, 1974; Kärn-Bränsle-Säkerhet, 1978; Logan and Berbano, 1978; Hartmann, 1979; Cranwell et al., 1990). Each of these estimates is below the limit established in the Standard of 1 chance in 10,000 in 10,000 yr for those events and processes that must be included in performance assessments.

Surficial Events and Processes

Name: EROSION/SEDIMENTATION

Screening Criterion: Base-case conditions

Disposition: Part of base-case scenario

Discussion: Both erosion and sedimentation resulting from either water and/or wind action will occur to some extent at any nuclear-waste disposal location during the 10,000 yr of regulatory concern. The possible consequences of erosion and sedimentation need to be considered in the base-case scenario. For the saturated granite location in this exercise, neither erosion nor sedimentation will be of sufficient magnitude to have an impact on disposal-system performance.

Name: GLACIATION

Screening Criterion: None

Disposition: Retained for scenario development

G Glaciation
I Irrigation
S Subsidence Over the Engineered Facility
M Mining Outside the Boundaries of the Disposal Facility
D Drilling of Boreholes

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Figure F-2. Potentially disruptive scenarios for a granite location.

Discussion: Based on the geographic regions that the U.S. Department of Energy (DOE) has investigated in the past for commercial high-level waste repositories (Battelle, 1985a,b,c), locations that have experienced continental glaciations in the past were not automatically excluded from consideration. The possibility of an area being covered by a future glaciation is unlikely to exclude a saturated granite site from future considerations for a defense high-level waste repository. Most locations within regions subjected to continental glaciations are not topographically suitable for alpine glaciers, or the effects of alpine glaciers would be insignificant when compared to the effects of continental glaciers.

The most recent period of continental glaciations in North America and Eurasia started approximately 1.6 million years ago, and the last glaciation in North America ended approximately 10,000 yr ago. This time period was characterized by relatively long periods during which continental glaciers existed, separated by relatively short periods of more moderate climates during which the continental glaciers were greatly reduced in extent or totally absent. The causes of continental glaciations and their cyclicity are not known, although some of the factors that can affect climate on a global scale are (1) variations in the Earth's rotation and orbital distance from the Sun, (2) volcanic activity, (3) variations in ocean temperatures, (4) solar disturbances, and (5) topographic relief and orientation of the continents (Lamb, 1972a,b). Because of the cyclicity of continental glaciations, the Earth's climatic regime for the past 10,000 yr may be interglacial rather than being at the end of the glacial cycle.

The possibility exists that a granite site will be selected in an area that has been covered by continental glaciers in the past. Such a site may be revisited by continental glaciers in the future. The time frame of reglaciation relative to the 10,000 yr of regulatory concern is not known. Because continental glaciation cannot be eliminated from consideration using any of the screening criteria, glaciation is retained for scenario development.

Name: PLUVIAL PERIODS [Climate Change]

Screening Criterion: Base-case condition

Disposition: Part of base-case scenario

Discussion: Pluvial periods refers to time intervals characterized by relatively high amounts of precipitation when compared to current conditions. Because climatic conditions are continuously changing when viewed on a sufficiently long time scale, any disposal location is certain to experience both wetter and drier climatic conditions during the 10,000 yr of regulatory concern. Whether the wetter conditions at the bedded-salt location will be "pluvial" or simply wetter is an academic exercise. For the purposes of performance assessments, climatic changes that include wetter conditions will occur, and the effects of these changes on the performance of the disposal system are part of the base-case conditions.

Name: SEA-LEVEL VARIATIONS

Screening Criterion: Physical reasonableness

Disposition: Screened out from scenario development

Discussion: Changes in sea level are the result of climatic changes adding to or reducing the amount of water in the oceans through increases or decreases in the size of continental glaciers, eustatic adjustments due to changes in response to the load upon the crust, and/or tectonic activity. The bedded-salt site has been selected to preclude the possibility of inundation resulting from melting of the world's continental glaciers. Eustatic and tectonic contributions to sea-level changes that could result in inundation of the disposal location are not physically reasonable within the time frame of regulatory concern.

Name: HURRICANES

Screening Criterion: Physical reasonableness

Disposition: Screened out from scenario development

Discussion: Hurricanes are storms that originate over ocean water in the tropics of the northern hemisphere (cyclones in the southern hemisphere). These storms are characterized by high wind speeds and heavy rainfall. Although these storms may migrate out of the tropics, they dissipate quickly over cold water or land.

Hurricanes can be eliminated from performance assessments, because the survival of storms in this category is not physically reasonable as far inland as any disposal facility in bedded salt. Intense storms are certain to occur at any disposal location, and for this reason, intense storms are part of normal climatic conditions.

Name: SEICHES

Screening Criterion: Physical reasonableness

Disposition: Screened out from scenario development

Discussion: A seiche is a "free or standing-wave oscillation of the surface of water in an enclosed or semi-enclosed basin...that is initiated chiefly by local changes in atmospheric pressure, aided by winds, tidal currents, and small earthquakes; and that continues, pendulum fashion, for a time after cessation of the originating force" (Bates and Jackson, 1987, p. 600). Seiches range in height from several centimeters to a few meters. The location of any disposal facility in bedded salt is far enough from large bodies of water for the mechanism to generate seiches of sufficient size to reach an inland location to be physically unreasonable.

Name: TSUNAMIS

Screening Criterion: Physical reasonableness

Disposition: Screened out from scenario development

Discussion: A tsunami is a "gravitational sea wave produced by any large-scale, short-duration disturbance of the ocean floor, principally by a shallow submarine earthquake, but also by submarine earth movement, subsidence, or volcanic eruption" (Bates and Jackson, 1987, p. 704). The location of the disposal facility is far enough from the ocean that an occurrence of a geologic disturbance on a scale that would result in a tsunami large enough to inundate a major part of a continent to reach the disposal-facility location is not physically reasonable.

Name: REGIONAL SUBSIDENCE OR UPLIFT

Screening Criterion: Base-case condition

Disposition: Part of base-case scenario

Discussion: Granite terrains in the southeastern, northeastern, and north-central U.S. have been undergoing and continue to undergo regional uplift (Battelle, 1985a,b,c). The specific mechanism producing this uplift can be either or both tectonic in origin or eustatic response to glacial unloading at the end of the last ice age. Rates of uplift can be determined by denudation rates of streams, sedimentation volumes in ocean basins, changes in shoreline elevations of lakes, and geodetic leveling. The rate of uplift can vary by one to two orders of magnitude for the same physiographic province depending on which techniques are used. For the three regions studied for preliminary repository siting, uplift rates ranged from 0.02-0.03 mm/yr for the crystalline Appalachians using stream denudation rates (Ahnert, 1970) to 4.0 ± 0.9 mm/yr for the crystalline terrains in North Carolina and

Georgia using correlations between uplift rate and topography (Citron and Brown, 1979). No consensus exists on the "correct" technique to use to determine uplift rates.

Name: MASS WASTING [including Landslides]

Screening Criterion: Base-case condition

Disposition: Part of base-case scenario

Discussion: Mass wasting is the dislodgment and downslope movement of soil and rock under the direct application of gravitational body forces (Bates and Jackson, 1987). Mass wasting is likely to occur along perennial and ephemeral streams as banks are undercut by channel migration and collapse into the stream channel. This type of small-scale mass wasting is common and is included in the base-case conditions. Any damming of the streams by the collapse will tend to be short lived, because the material will be eroded by the stream and incorporated into the stream's bedload. Site selection will preclude a disposal-facility location being sited where large-scale mass movements could threaten surface facilities during the operational phase of waste disposal.

Name: FLOODING

Screening Criterion: Physical reasonableness

Disposition: Screened out from scenario development

Discussion: Flooding refers to the overbank flow of a stream or river. No rivers exist at the location of the hypothetical disposal facility or are likely to exist at this location under wetter climatic conditions. The existing river in the region is at a sufficient distance from the disposal-facility location that the conditions for flooding, even under wetter climatic conditions, are not physically reasonable. Perennial streams are of a size that the flooding of these water channels would not be of sufficient duration to affect performance of the disposal system. Flooding of streams to an extent that would affect performance is not physically reasonable.

Subsurface Events and Processes

Name: DIAPIRISM

Screening Criterion: Physical reasonableness

Disposition: Screened out from scenario development

Discussion: Diapirism refers to "the process of piercing or rupturing of domed or uplifted rocks by mobile core material, by tectonic stresses as in anticlinal folds, by the effect of geostatic load in sedimentary strata as in salt domes and shale diapirs, or by igneous intrusion, forming diapiric structures such as plugs" (Bates and Jackson, 1987, p. 181). The fact that the granite has crystallized precludes diapirism from continuing to occur, and a change in the geologic conditions to remobilize the granite body within the time frame of regulatory concern is not physically reasonable.

Name: SEISMIC ACTIVITY

Screening Criterion: Base-case condition

Disposition: Part of base-case scenario

Discussion: Areas that have been investigated to identify candidate granite repository sites typically are subjected to low levels of seismic activity (Battelle, 1985a,b,c). This seismic activity is the result of either or both tectonic activity and eustatic response to deglaciation and is certain to occur throughout the time frame of

regulatory concern. The certainty that any site will be subjected to seismic waves places this phenomenon in the base-case scenario, although low-level seismic activity is unlikely to affect the performance of the disposal system.

Name: VOLCANIC ACTIVITY

Screening Criterion: Physical reasonableness

Disposition: Screened out from scenario development

Discussion: The areas that have been investigated for candidate sites for repositories in granite have not been subjected to volcanic activity for many millions of years. Changes in the geologic setting within the time frame of regulatory concern that would result in volcanic activity within the investigated areas are not physically reasonable.

Name: MAGMATIC ACTIVITY

Screening Criterion: Physical reasonableness

Disposition: Screened out from scenario development

Discussion: The granite that is being investigated for a repository is of magmatic origin, although the rocks crystallized many millions of years ago. The physiographic areas in the United States that contain these rock do not have the anomalous heat flow in conjunction with seismic activity characteristic of actively mobile magma bodies at depth. Changes in the geologic setting necessary for magmatic activity to become an active process and to affect disposal-system performance are not physically reasonable within the time frame of regulatory concern.

Name: FORMATION OF DISSOLUTION CAVITIES

Screening Criterion: Physical reasonableness

Disposition: Screened out from scenario development

Discussion: The mineral composition of the saturated granite considered for a repository is not subjected to detectable levels of dissolution by groundwater. As a result, the formation of dissolution cavities is not a physically reasonable process

Name: FORMATION OF INTERCONNECTED FRACTURE SYSTEMS

Screening Criterion: Physical reasonableness

Disposition: Screened out from scenario development

Discussion: Fracture systems do not spontaneously occur but are the product of the occurrence of some event or process. At shallow depths, saturated granite contains fractures with the fracture density and orientations being dependent on the particular geologic history of the rock body. At depth, most granite bodies are not fractured, because the in-situ stress field does not allow the fractures to open. Changes in the geologic setting that would allow new fracture sets to form or that would alter the stress field to an extent that fractures would form in currently unfractured rock through either uplift or erosion are not physically reasonable within the time frame of regulatory concern.

Name: FAULTING

Screening Criterion: Physical reasonableness

Disposition: Screened out from scenario development

Discussion: Faulting refers to either the creation of a new fault or the renewed movement of an existing fault. To reduce the accumulation of tectonically induced stresses, movement generally occurs along existing faults with the formation of a new fault that is not subsidiary to an existing fault being a rare event (Bonilla, 1979). The faults within a body of granite are old features, and changes in the geologic setting to an extent that new faults would form are not physically reasonable within the time frame of regulatory concern.

Site selection will eliminate from consideration those sites where active faults could affect the performance of a disposal facility. Changes in the geologic setting to an extent that would reactivate currently inactive faults are not physically reasonable within the time frame of regulatory concern.

F.2.2 Human-Induced Events and Processes

Inadvertent Intrusion

Name: EXPLOSIONS

Screening Criterion: Probability of occurrence

Disposition: Screened out from scenario development

Discussion: Explosions can be detonated either at the surface or in the subsurface and can be either chemical or nuclear. For scenario development, all explosions are eliminated from consideration because of low probability of occurrence. Specific probability values are not estimated in reaching this conclusion, but the combination of low probability requirements strongly suggests that explosions can be eliminated from consideration based on probability arguments.

The first question to address about surface explosions is whether such explosions will occur. Because surface explosions have no particular geographic requirements, the next question is why would the location of a disposal facility in granite be selected over all other possible locations, with the added constraint that the location is selected without knowledge that the facility is present. The third question to address is the size of the explosion. Only an explosion of sufficient magnitude to either crater or deeply fracture the underlying geologic strata would likely significantly affect disposal-system performance. Chemical explosions seem to be unlikely because of the volumes of explosives required, and nuclear explosions are unlikely because of perceived health risks to the population. As explained in Guzowski and Helton (1991, Section 4.1.4), regulatory restrictions on the severity of types of intrusion would eliminate this mode of directly penetrating the disposal facility, so the disruptive effects would have to be on transportation of radionuclides rather than on their release from the facility. The combined probabilities of surface detonations occurring in the future, the appropriate location being selected, and the required magnitude of an explosion arguably results in a probability of occurrence below the cutoff provided in the Standard.

Underground explosions present similar questions to those for surface explosions. The first question is whether subsurface explosions of either chemical or nuclear origin, exclusive of those in mining, will occur in the future. Although the specific medium in which each underground test conducted by the United States has been conducted was not reported by NVO (1987), few if any of the tests have been in granite. No records are readily available (if any exist) on underground chemical explosions at any granite location. With no documented history of the use of granite for underground testing of chemical or nuclear explosions, the probability of this medium being used to contain future tests is likely to be low. The question of location also arises. With relatively large areas of the contiguous United States underlain by granite, why would a location near the saturated granite

disposal facility be selected? Location is an important factor in the probability argument because of the restrictions placed on the severity of intrusion type by the Standard. As with surface explosions, the subsurface explosion would have to affect the transport of radionuclides without directly intruding the disposal facility. The location question is further complicated by limits to the magnitude of an explosion. Granite may have favorable physical properties for containing explosions, but the magnitude would have to be restricted so as not to rupture the rock and thereby negate the medium's favorable characteristics. The combined probabilities of whether subsurface explosions will occur in the future, the appropriate location, and the required magnitude of the explosion arguably result in a probability of occurrence below the cutoff provided in the Standard.

Name: DRILLING (For Resources)

Screening Criterion: None

Disposition: Retained for scenario development

Discussion: The section on *Institutional Controls* of Appendix B of the Standard states that "...the Agency [EPA] believes that passive institutional controls can never be assumed to eliminate the chance of inadvertent and intermittent human intrusion into these disposal sites" (EPA, 1985, p. 38088). Within the section of Appendix B entitled *Consideration of Inadvertent Human Intrusion into Geologic Repositories*, the statement is made

The Agency [EPA] believes that the most productive consideration of inadvertent intrusion concerns those realistic possibilities that may be usefully mitigated by repository design, site selection, or use of passive controls (although passive controls should not be assumed to completely rule out the possibility of intrusion). Therefore, inadvertent and intermittent intrusion by exploratory drilling for resources (other than any provided by the disposal system itself) can be the most severe intrusion scenario assumed by the implementing agencies. (EPA, 1985, p. 38089).

These statements in the Standard are interpreted to mean that exploratory drilling for resources has a probability of occurrence greater than the cutoff stated in EPA's standard and must be included in the development of scenarios for performance assessments.

Name: MINING

Screening Criterion: None

Disposition: Retained for scenario development

Discussion: Granite can contain certain metallic minerals in sufficient quantity to be economically minable. Mineral deposits in locally economic concentrations are assumed to be present in the region of the disposal facility. Some granitic rocks themselves can be mined as an industrial resource under the appropriate geologic and economic conditions, but the granitic rock within and near the hypothetical disposal system can be discounted as a possible resource because of the extensive occurrence of more readily accessible rock at other locations.

Appendix B of the Standard provides guidance on the limits to the severity of human intrusion at the disposal location. As discussed under the topic of *Drilling* (above), exploratory drilling for resources is the most severe type of human intrusion that must be included in performance assessments. Mining for resources would have a more disruptive effect on the disposal system than drilling. Whereas mining is retained for scenario development, the location of the mining must be beyond the boundaries of the disposal facility so as to have no direct effect on facility integrity.

1 Name: (Drilling of) INJECTION WELLS

3 *Screening Criterion:* Physical reasonableness

6 *Disposition:* Screened out from scenario development

8 *Discussion:* Injection of liquid waste requires the existence of a medium at depth with sufficient permeability and interconnected porosity to receive the injected liquid in a timely manner. For isolation purposes, the zone of injection generally is relatively deep. With the possible exceptions of fault zones and fractured zones near the surface, bodies of granite do not contain the permeability and porosity necessary to receive injected fluids. Fractured zones near the surface and fault zones are not suitable for injection because of the problem of restricting contaminant migration. Changes in the properties of any granitic rock that would make this medium suitable for injecting liquid waste are not physically reasonable.

18 Name: (Drilling of) WITHDRAWAL WELLS

20 *Screening Criterion:* None

23 *Disposition:* Retained for scenario development

25 *Discussion:* Consideration of this event is limited to water wells. Fault zones and fractured zones in granite can be sources of potable water. Intact rock is not a source of water. With the restrictions of depth and degree of fracturing, drilling of withdrawal wells is a realistic event for granite terrains. This event is retained for scenario development.

31 **Hydrologic Stresses**

34 Name: IRRIGATION

36 *Screening Criterion:* None

38 *Disposition:* Retained for scenario development

41 *Discussion:* Irrigation at and near the disposal facility is feasible using water from a nearby river. The addition of water at the surface can affect the water table by increasing recharge. Alteration of the groundwater flow patterns in the unconfined aquifer has the potential to affect radionuclide transport in the event that radionuclides escape from the disposal facility and migrate to the unconfined aquifer.

47 Name: DAMMING OF STREAMS OR RIVERS

49 *Screening Criterion:* Physical reasonableness

52 *Disposition:* Screened out from scenario development

54 *Discussion:* The topographic relief in the vicinity of rivers near the hypothetical disposal facility is not appropriate for damming the river to create reservoirs. Changes in the local topography to make damming of the river feasible are not physically reasonable within the time frame of regulatory concern.

F.2.3 Waste- and Repository-Induced Events and Processes

Name: SUBSIDENCE AND CAVING

Screening Criterion: None

Disposition: Retained for scenario development

Discussion: [To be provided]

Name: SHAFT AND BOREHOLE SEAL DEGRADATION

Screening Criterion: Base-case condition

Disposition: Part of base-case scenario

Discussion: Alteration of the backfill material in either the shafts or boreholes will occur during the 10,000 yr of regulatory concern. As a result, this process is part of the base-case scenario. A catastrophic failure of seals in these locations could be included in scenario development, although no mechanism for this type of failure has been recognized as physically reasonable under current or projected conditions.

Name: THERMALLY INDUCED STRESSES/FRACTURING IN HOST ROCK

Screening Criterion: Base-case condition

Disposition: Part of base-case condition

Discussion: Spent fuel and high-level waste will generate heat and, as a result, thermally induced stresses will contribute to the stress field within the host rock surrounding the hypothetical disposal facility. Because heat is certain to be generated, thermally induced stresses in the host rock are part of the base-case scenario. The extent of fractures generated by these stresses will depend on the post-excavation stress field, the thermal effects of the waste-generated heat, and the distribution of naturally occurring and excavation-induced fractures.

Name: EXCAVATION-INDUCED STRESS/FRACTURING IN HOST ROCK

Screening Criterion: Base-case condition

Disposition: Part of base-case scenario

Discussion: In deep excavations, the differential stress between the host rock and the excavation results in deformation with a high strain rate that produces fractures. A disturbed zone, which has a higher porosity than the intact host rock because of the presence of fractures, typically forms around an excavation.

Name: NUCLEAR CRITICALITY

Screening Criterion: None

Disposition: Retained for scenario development

Discussion: In an analysis of criticality issues associated with the disposal of highly enriched nuclear fuel, Palmer (1992) determined that certain configurations of fuel elements under certain environmental conditions (primarily the availability of water to act as a moderator) could result in criticality of the intact stored fuel. The potential for critical conditions developing by the concentration of radionuclides along possible radionuclide transport pathways was not evaluated. Until the particular environmental conditions of the stored waste and the

transported radionuclides are determined, the question of whether criticality is physically reasonable cannot be answered with any assurance. Criticality is retained for scenario development pending further analysis.

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Appendix G: CAMCON Modeling Setup and Software

Appendix G: CAMCON Modeling Setup and Software

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Appendix G: CAMCON Modeling Setup and Software¹

R. P. Rechard and D. K. Rudeen

This document discusses software used by the Compliance Assessment Methodology CONTroller (CAMCON) for the complex (Figure G-1) and simple (Figure G-2) performance assessment of spent nuclear fuel and high-level waste stored at Idaho National Engineering Laboratory in either a generic salt or granite repository. CAMCON includes four main software groups: Modeling Setup; Repository Consequence Modeling; Snake Member Consequence Modeling; and Probability Modeling, CCDF Construction, and Results Analyses. The brief explanations of the functions for codes previously used in WIPP PA are from Rechard (1992). Abstracts for the codes are in Supplement G1 to Appendix G. During the execution of this PA, a new computer system was installed. Therefore, much of the software exists and was used on two computer systems; a VAX 8810 running VMS and a DEC ALPHA 3000 running open VMS. The version designators for the two systems are VV and ZO, respectively.

G.1 Modeling Setup System

The main function of the Modeling Setup System is parameter selection. The processes for parameter selection are

- select computational model parameters (x) consistent with manner of use in models
- place in secondary database file with database manager (parameter values extracted with MATSET or PRELHS)
- select parameters (x) to vary (parameter uncertainty)
- sample parameters with Latin hypercube sampling (LHS).

The software used for the modeling setup are INGRES, PRELHS, and LHS.

Complex and Simple PA for Salt and Granite Site

1. INGRES®

Function: Generally, INGRES stores and manipulates data.

2. PRELHS (QA Version C-2.00VV)

Function: Generally, PRELHS translates from the secondary database to LHS. Specifically, PRELHS extracts parameter distribution data requested by the user from the secondary database file, PROPERTYSDB, and sets up the LHS input file.

3. LHS (QA Version C-2.31VV)

Function: Generally, the code samples using Latin hypercube or Monte Carlo sampling techniques. Specifically, LHS samples distributions of input parameters using either normal Monte Carlo sampling or efficient Latin hypercube sampling. LHS also permits correlations (restricted pairings) between parameters. Latin hypercube sampling reduces the number of sample vectors $[nK]$ required to about $4/3 \cdot n_\alpha$, where n_α is the number of varying parameters (Iman and Shortencarier, 1984).

¹ Key for Software

Regular type = Support Codes

Bold type = Main Modeling Codes

Complex Modeling Setup for INEL Waste Snake Member Consequence Modeling

Figure G-1. Software used by CAMCON for the Complex PA.

Simple Modeling Setup for INEL Waste Snake Member Consequence Modeling

Salt Repository Consequence Modeling

Figure G-2. Software used by CAMCON for the Simple PA.

G.2 Repository Consequence Modeling System

The Repository Consequence Modeling System consists of three main groups of software: drill cuttings, two-phase flow, and source concentration. The Repository Consequence module develops a source term for transport calculations by incorporating the complex processes in the waste container, disposal room, drifts, shafts, and backfill. Specifically, the Repository Consequence module calculates

- fluid flux through the disposal area
- contaminant concentrations in fluids (gas or liquid) flowing through the disposal area of the repository
- contaminant releases to the accessible environment from drilling into the repository.

G.2.1 Drill Cutting Software

The software used for drill cuttings in the Repository Consequence Modeling System are GENNET, MATSET, POSTLHS, and CUTTINGS.

Complex and Simple PA for Salt and Granite Site (Figures G-1 and G-2)

1. GENNET (QA Version C-2.02VV)

Function: Constructs simple one-, two-, and three-dimensional networks from a geometry specification input file. The output is stored in a computational database file, CAMDAT.

2. MATSET (QA Version C-8.05VV)

Function: MATSET sets material names to specified regions (e.g., defined by GENMESH or GENNET), sets material property values, and sets attribute values into the performance-assessment computational database, CAMDAT. Both property and attribute values are obtained from either the property secondary database (usually median values read from PROPERTY.SDB) or the user-supplied MATSET input text file. Later, the user may overwrite a subset of these parameter values by running the Monte Carlo module, LHS.

3. POSTLHS (QA Version C-4.03VV)

Function: In general, POSTLHS translates from LHS output to a CAMDAT file format. Specifically, POSTLHS replicates all data in a PRECAMDAT file nK times, where nK is the number of sample vectors generated by LHS, and inserts one distinct sample of the varied parameters from the output of LHS into each replication of CAMDAT.

4. CUTTINGS (QA Version C-3.00VV)

Function: CUTTINGS calculates the quantity of radioactive material (in curies) brought to the surface as cuttings by a drilling operation that penetrates the repository. The code determines the amount of cuttings removed from a borehole by drilling and ages the cuttings to the intrusion time.

G.2.2 Two-Phase Flow Software

The software used for the two-phase flow in the Repository Consequence Modeling System are GENMESH, MATSET, POSTLHS, ALGEBRA, BCSET, ICSET, PREBRAG_T, BRAGFLO_T, POSTBRAG_T, SALFLOW, and GRANFLOW. The two-phase flow code, BRAGFLO_T, is used primarily in the repository domain of the disposal system where corrosion of metals may produce gas within the disposal area and cause two-phase flow conditions.

Complex PA for Salt and Granite Site (Figure G-1)

1. GENMESH (QA Version A-6.03VV)

Function: In general, GENMESH generates a rectilinear finite element or finite difference mesh. Specifically, GENMESH constructs a right-handed, Cartesian product, rectangular one-, two-, or three-dimensional finite-difference grid from a user input file specifying the geometry. In addition to setting the node coordinates and mesh connectivity, GENMESH sets material regions, identifies (flags) nodes or elements for boundary conditions, and sets elevations of elements. All information is then stored as a computational database file in CAMDAT.

2. MATSET (QA Version C-8.05VV)

3. POSTLHS (QA Version C-4.03VV)

4. ALGEBRACDB (QA Version C-2.26ZO)

Function: The code generates additional data (or removes unnecessary data) in CAMDAT by algebraically manipulating data already stored. With ALGEBRA, an analyst can generate pertinent data external to a code by combining data already stored in CAMDAT rather than by modifying a code and thereby invoking a new quality assessment.

5. BCSET (QA Version C-2.03VV)

Function: BCSET sets nodal boundary flags, element boundary flags, and values at boundary flags. The flags are those defined either in the mesh generation module or by BCSET in a user input file.

6. ICSET (QA Version C-2.08VV)

Function: Generally, ICSET sets up initial conditions. Specifically, ICSET sets CAMDAT analysis array variables: history, global, nodal, and/or element variable values, at the first time step obtained from a user file. In addition, any nodal or element variables (existing or new) can be linearly interpolated by specifying interpolation tables in the ICSET input text file.

7. PREBRAG_T (QA Version C-1.04VV)

Function: PREBRAG_T translates the information in the CAMDAT file format into the BRAGFLO_T input file format.

8. BRAGFLO_T (QA Version X-1.00ZO)

Function: Generally, BRAGFLO_T models components of gas and liquid flow using an implicit mathematical formulation based on the pressure, temperature, and saturation state variables of each component and a finite-difference solution technique. Specifically, BRAGFLO_T is a mathematical model that describes the simultaneous flow of two phases through a porous heterogeneous reservoir. BRAGFLO_T's fully implicit finite difference formulation makes it well suited for simulating convergent flow. This version of BRAGFLO_T includes reaction (corrosion) source and sink terms to the gas and brine components. Reaction rate is calculated as a function of brine saturation. Reactions are completed when the reactants (e.g., iron) are consumed. The BRAGFLO_T formulation is designed to be robust and numerically stable over a wide range of conditions and input property values, making it well-suited to perform as a component of a probabilistic computation system. BRAGFLO_T differs from BRAGFLO, its parent code, primarily by the addition of the energy equation for temperature. However, omission of some features and addition of others means that, except for the solver, the codes share few common subroutines. This version contains the INEL spent nuclear fuel gas generation model.

9. POSTBRAG_T (QA Version C-2.00VV)

Function: POSTBRAG_T translates binary output from BRAGFLO_T into the CAMDAT file format.

Simple PA for Salt and Granite Site (Figure G-2)

1. GENMESH (QA Version A-6.03VV)
2. MATSET (QA Version C-8.05VV)
3. POSTLHS (QA Version C-4.03VV)
4. PREBRAG_T (QA Version C-1.04VV)

5a. SALFLOW (QA Version X-1.00ZO)

Function: SALFLOW calculates liquid flow rates through an arbitrary number of boreholes penetrating a repository in salt and through a confined conductive zone in direct communication with the repository. SALFLOW is internally coupled to the canister simulation model CONCSNF.

5b. GRFLOW (QA Version X-1.00ZO)

Function: GRFLOW calculates liquid and gas flow rates through an arbitrary number of connections to overlying and underlying liquid reservoirs. GRFLOW is internally coupled to the canister simulation model CONCSNF.

G.2.3 Source Concentration Software

The software used for the source concentration in the Repository Consequence Modeling System are MATSET, ALGEBRA, PRESTAFF, PRECONCSNF, CLAMVD, STAFF2D, CONCSNF, POSTSTAFF, ALGEBRA, GENMESH, POSTLHS, and CLAMVD.

Complex PA for Salt and Granite Site (Figure G-1)

1. MATSET (QA Version C-8.05VV)
2. ALGEBRACDB (QA Version C-2.26ZO)
3. PRESTAFF (QA Version C-4.01ZO)

Function: PRESTAFF creates an input file for STAFF2D by translating data from the CAMDAT database.

4. PRESOR (QA Version X-1.00ZO)

Function: Generates input data file for the INEL source term model in STAFF2D.

5. STAFF2D (QA Version C-4.01ZO)

Function: In general, STAFF2D models single or dual porosity fluid flow and radionuclide transport using a finite-element solution technique. Specifically, STAFF2D is a two-dimensional finite-element code designed to simulate confined and unconfined groundwater flow and single- or multiple-component solute transport in fractured or granular aquifers. Fractured porous media are represented using both discrete-fracture and dual-porosity approaches. This version contains the INEL source term model CONCSNF.

6. POSTSTAFF (QA Version C-3.00ZO)

Function: POSTSTAFF places total head and interstitial velocity components or solute concentrations (nodal variable) from STAFF2D output in CAMDAT.

7. ALGEBRACBD (QA Version C-2.26ZO)

Simple PA for Salt and Granite Site (Figure G-2)

1. GENNET (QA Version C-2.02VV)
2. MATSET (QA Version C-8.05VV)
3. POSTLHS (QA Version C-4.03VV)
4. CLAMVD (QA Version X-1.00ZO)

Function: CLAMVD is used to simulate transport within and from the INEL granite and salt repositories following interconnection of the repository and an external source. Radionuclide sources are provided by internal coupling to the CONCSNF model.

G.3 Snake Member Consequence Modeling System

The Snake Member Consequence Modeling System consists of three main groups of software: transmissivity fields, groundwater flow, and groundwater transport.

G.3.1 Transmissivity Fields Software for Salt Repository

The software used for the transmissivity fields in the Snake Member Consequence Modeling System are CON-SIM, AKRIP, PAREST, PILOTL, GRASP II, TUBA, SWIFT II, STLINE, ORDER.TT(DOL), SAP, UNSWIFT, RELATE, ALGEBRA, ORDER.SMPL, TRACKER, and TABLE.

Complex PA for Salt Site. This software was used for evaluating transmissivity fields at the WIPP in 1992. For the INEL SNF Project, the results were directly used without rerunning the simulations. The descriptions of the functions and versions are in LaVenue and RamaRao (1992).

Simple PA for Salt Site. Essentially, the results used for establishing flow fields in the Complex PA were used in the Simple PA except that intermediate results from which particle paths and travel times could be evaluated were used. The travel times were then stored as model parameters for later use by the flow codes of the Simple PA modeling system.

G.3.2 Groundwater Flow Software

The software used for the groundwater flow in the Snake Member Modeling System are GENMESH, MATSET, RELATE, PRESTAFF, STAFF2D, and POSTSTAFF. PRESTAFF, STAFF2D, and POSTSTAFF are repeated for each scenario *Sj* or conceptual model. TRACKER is used in an intermediate result check. The groundwater flow module establishes regional and local fluid flow conditions within and directly surrounding the disposal system. This software was used for evaluating flow fields at the WIPP in 1992. For the INEL SNF project, the results were used directly without rerunning the simulations.

G.3.3 Groundwater Transport Software

The software used for groundwater transport in the Snake Member Consequence Modeling System are RELATE, POSTLHS, ALGEBRA, PRESTAFF, STAFF2D, POSTSTAFF, GENMESH, MATSET, and CLAMVD. The groundwater transport module predicts radionuclide migration from the repository source to the disposal system boundary. The simulations were run using the flow fields from G.3.2 and the source concentrations from G.2.3.

Complex PA for Salt and Granite Site (Figure G-1)

1. RELATE (QA Version C-1.42VV)
2. POSTLHS (QA Version C-4.03VV)
3. ALGEBRACDB (QA Version C-2.26ZO)
4. PRESTAFF (QA Version C-4.01ZO)
5. STAFF2D (QA Version C-4.01ZO)
6. POSTSTAFF (QA Version C-3.00ZO)

Simple PA for Salt and Granite Site (Figure G-2)

1. GENMESH (QA Version A-6.03VV)
2. MATSET (QA Version C-8.05VV)
3. POSTLHS (QA Version C-4.03VV)
4. CLAMVD (QA Version X-1.00ZO)

G.4 Probability Modeling, CCDF Construction, and Results Analyses System

The Probability Modeling, CCDF Construction, and Results Analyses System consists of two main groups of software: (1) probability modeling and CCDF construction, and (2) sensitivity analyses. Using results from the groundwater contaminant transport module, the compliance module calculates various performance metrics depending upon the EPA environmental standard being considered:

- containment calculations
- groundwater concentration and dose calculations.

G.4.1 Probability Modeling and CCDF Construction Software

Probability modeling and CCDF construction software includes CCDFCALC, NUCPLOT, and CCDFPERM. CCDFPLOT is used in an intermediate result check on conditional CCDFs.

Complex and Simple PA for Salt and Granite Site (Figure G-1 and G-2)

1. CCDFCALC (QA Version C-4.12ZO)

Function: In general, CCDFCALC preprocesses radionuclide time histories for CCDFs. Specifically, for each vector of all scenarios, CCDFCALC accesses the appropriate CAMDAT database file, integrates to find cumulative releases for each radionuclide, and calculate the EPA sum. The results are written to a transfer file used by CCDFPLOT, NUCPLOT, or CCDFPERM.

2. NUCPLOT (QA Version C-1.12ZO)

Function: Interactively plots Tukey Box plots showing distribution of individual radionuclide release or normalized discharge.

3A. CCDFPERMP (QA Version X-3.20ZO)

Function: In general, CCDFPERMP calculates decomposed scenario probabilities. Specifically, CCDFPERMP provides a scenario probability. Because the calculation of all possible vectors for up to 15 hits on a repository is beyond the capabilities of the available computing resources, the vectors are built from discharges and cuttings from single and double intrusions by assuming the results are independent (salt repository version).

3B. CCDFPERMF (QA Version X-3.20ZO) (Granite repository version of CCDFPERMP.)**4. CCDFPLOT (QA Version C-4.11ZO)**

Function: Interactively plots the CCDF data read from the CCDFCALC transfer file.

G.4.2 Sensitivity Analysis Software

Sensitivity analysis software includes SUMMARIZE, LHS2STEP, PCCSRC, CCD2STEP, and STEPWISE.

Complex and Simple PA for Salt and Granite Site (Figures G-1 and G-2)

1. SUMMARIZE (QA Version C-1.00VV)

Function: SUMMARIZE reads specified variable values from multiple CAMDAT databases (one for each vector) or transfer files. The data is then written to a text transfer file suitable for input to an X-Y plot program.

2. GRAPH (QA Version X-1.01ZO)

Function: Interactive x-, y-graphing package.

3. LHS2STEP (QA Version C-1.01VV)

Function: LHS2STEP reads an LHS sampled output file containing the dependent variables and writes an output file for either STEPWISE or PCCSRC.

4. PCCSRC (QA Version C-2.19VV)

Function: PCCSRC evaluates variable importance by reporting the partial correlation coefficients (PCC) and standardized regression coefficients (SRC) on either the rank or raw data. Provided useful regression models can be developed, the absolute values of the standardized regression coefficients (or mathematically related partial correlation coefficients) can be used to rank variable importance.

5. CCD2STEP (QA Version C-1.04VV)

Function: CCD2STEP reads CCDFCALC files containing radionuclide release data for the dependent variables and writes an output file for either STEPWISE or PCCSRC.

6. STEPWISE (QA Version C-2.18VV)

Function: STEPWISE evaluates variable importance by developing regression models between the observed response and input variables using a forward, backward, or stepwise regression procedure on the raw or ranked data. Provided useful regression models can be developed, the absolute values of the standardized regression coefficients (or mathematically related partial correlation coefficients) can be used to rank variable importance.

G.5 Other Miscellaneous Software

G.5.1 Repository Design Software

1. SUMQ_3D (QA Version C-1.01VV)

Function: Calculates temperature change at a point of interest from a three-dimensional layout of linear heat sources (canisters).

2. COND_1D (QA Version X-1.00VV)

Function: Calculates the temperature distribution for 1D spherical or cylindrical conduction heat flow with a decaying central heat source. Permits spatially varying thermal properties and spatially varying heat generation.

G.5.2 Inventory Analysis Software

1. GNRDP (QA Version X-1.02ZO)

Function: GNRDP decays the DOE high-level waste (from Savannah River, Hanford, and West Valley) inventory (in curies) of each radioisotope to the year 2030. Then each SNF/HLW supplier of radioisotopes is decayed for 10,000 years. Each radioisotope history is then normalized by the EPA 40 CFR 191 limit and history plots are then generated using the HJILOT graphics library routines.

2. INVENTOR (QA Version X-1.02ZO)

Function: Compiles useful data regarding radioisotope inventory grouped by waste disposal groups (WDG) and by waste parcels (WP).

3. DECAY (QA Version X-1.04ZO)

Function: Decays radioisotopes according to a simplified Bateman's equation model.

G.5.3 Dose Calculation

1. PREGENII (QA Version C-1.00VV)

Function: PREGENII creates input for GENII code (SNL PC version) by translating data from CAMDAT and a user-supplied PREGENII text input file.

2. GENII-S (QA Version C-1.00VV)

Function: Calculates internal and external radiation doses to humans from routine and accidental releases of radionuclides to air, water, or residing in soil.

G.5.4 Miscellaneous Tools

1. GROPECDB (QA Version C-2.07ZO)

Function: GROPECDB allows the user to interactively examine a CAMDAT database.

2. ABSGEN (QA Version C-4.10VV)

Function: To extract QA-related information from the program help files. This information is used to generate abstracts that can be viewed on the screen or printed to a file.

3. BLOTADB (QA Version C-1.30ZO)

Function: The plotting support module, BLOTADB, plots all intermediate and final results from all main modules. BLOTADB directly reads CAMDAT and plots (1) the computational mesh with contoured analysis results, (2) grid distance versus any variable, and/or (3) any variable versus any other variable.

4. PLOTSDB (QA Version C-4.01VV)

Function: Displays selected data from the secondary (material property) database.

5. DVRANK (QA Version X-1.00ZO)

Function: DVRANK ranks the integrated release (calculated by CCDFCALC) of any subset of scenarios or design alternatives for all sample vectors.

6. BRTSUM (QA Version X-1.00ZO)

Function: BRTSUM is used to create a table of breach times for all canister layers and all sample vectors using the CONCSNF breach time transfer files created during execution of GRFLOW or SALFLOW.

G.6 References

Iman, R.L., and M.J. Shortencarier. 1984. *A FORTRAN 77 Program and User's Guide for the Generation of Latin Hypercube and Random Samples for Use with Computer Models*. NUREG/CR-3624, SAND83-2365. Albuquerque, NM: Sandia National Laboratories.

LaVenue, A.M., and B.S. RamaRao. 1992. *A Modeling Approach to Address Spatial Variability within the Culebra Dolomite Transmissivity Field*. SAND92-7306. Albuquerque, NM: Sandia National Laboratories.

Rechard, R.P., ed. 1992. *User's Reference Manual for CAMCON: Compliance Assessment Methodology Controller Version 3.0*. SAND90-1983. Albuquerque, NM: Sandia National Laboratories.

Appendix G: CAMCON Modeling Setup and Software

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Supplement G1 to Appendix G: Code Abstracts

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INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: ABSGEN	2a. Version ID: C-4.10VV	2b. Version Date: 11/04/93	3. Host Computer: VAX/VMS 8810
4. Software Title: ABSGEN extracts data from help files and creates abstracts.			
5. Software Description: To extract QA related information from the program help files. This information is used to generate abstracts which can be viewed on the screen or printed to a file.			
6. Limitations: In order to extract abstract related data from the help file, the program enforces a rigid structure and sequence on the program help file. It requires that all level numbers start in column 1, and all identifiers start in column 3. Further more it imposes a restriction on the maximum length of data in most fields. For more information on the structure of the help file, see the Programmer's Manual for CAMCON or the help file. Or, run the program "HLP2CHK" on your help file.			
7. Keyword/Phrases: UPDATE_HISTORY, QA_INFO, EXECUTION, Title, Purpose, Authors, Sponsor, Version, Date, QA_Level, Limitations, Origin, Verification, Classification			
8. Origin/Source: Sandia National Laboratories, Division 6342, Albuquerque, NM 87185-5800			
9. Authors: Dave W. Osborn, Tracy A. Wilson, Samuel M. Gershin			
10. Parent Program: None			
11. Computer Languages: C			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2
Date: 12/27/93		
Software Name: ABSGEN	Version ID: C-4.10VV	Version Date: 11/04/93
13. Classification: Sandia National Laboratories,tribution.		
14. Software Sponsor: Samuel M. Gershin	15. Software Consultant: Samuel M. Gershin	
16. Verification: <div style="display: flex; flex-wrap: wrap; margin-top: 10px;"> <div style="width: 50%;"> ☐ Hand Calculations </div> <div style="width: 50%;"> ☐ Parametric Sensitivity Analysis </div> <div style="width: 50%;"> ☒ In Use Test </div> <div style="width: 50%;"> ☐ Other: None </div> <div style="width: 50%;"> ☐ Comparison with Analytic Solutions </div> <div style="width: 50%;"> ☐ Comparison with Other Verified Programs </div> </div>		
17. Documentation: <div style="margin-top: 10px;"> User's Reference: Rechard, R.P., 1992. User's Reference Manual for CAMCON: Compliance Assessment Methodology Controller. Version 3.0. SAND90-1983. Albuquerque, NM: Sandia National Laboratories. </div> <div style="margin-top: 20px;"> Theoretical Reference: None </div> <div style="margin-top: 20px;"> Verification/Sample Problems Reference: None </div> <div style="margin-top: 20px;"> Other Reference: None </div>		
18. Primary Reference: Rechard, R.P., 1992. User's Reference Manual for CAMCON: Compliance Assessment Methodology Controller. Version 3.0. SAND90-1983. Albuquerque, NM: Sandia National Laboratories.		
19. Approved by: <div style="display: flex; justify-content: space-between; margin-top: 20px;"> <div style="width: 45%;"> PA QA Coordinator </div> <div style="width: 45%; text-align: right;"> <u>12-27-93</u> Date </div> </div>		

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: ALGEBRACDB	2a. Version ID: C-2.26ZO	2b. Version Date: 12/02/93	3. Host Computer: ALPHA/VMS
4. Software Title: ALGEBRACDB: A program module that algebraically manipulates data in CAMDAT			
5. Software Description: ALGEBRACDB generates additional data (or removes unnecessary data) in CAMDAT by manipulating data already stored. With ALGEBRACDB, an analyst can generate pertinent data external to a program module by combining data already stored in the CAMDAT rather than modifying the program module and, thereby, invoking a new quality assessment of the module.			
6. Limitations: 1000 equations, 1000 variables (input, output, temp)			
7. Keyword/Phrases: WIPP, Performance Assessment, Data Base Manipulation			
8. Origin/Source: Sandia National Laboratories, Applied Mechanics Division 1523, Albuquerque, NM 87185-5800			
9. Authors: Amy P. Gilkey			
10. Parent Program: ALGEBRA			
11. Computer Languages: FORTRAN 77			
12. Status: .			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2								
Date: 12/27/93										
Software Name: ALGEBRACDB	Version ID: C-2.26ZO	Version Date: 12/02/93								
13. Classification: Government owned, Limited distribution.										
14. Software Sponsor: Amy P. Gilkey	15. Software Consultant: Amy P. Gilkey									
16. Verification: <table style="width: 100%; margin-top: 10px;"> <tr> <td>☐ Hand Calculations</td> <td>☐ Parametric Sensitivity Analysis</td> </tr> <tr> <td>☐ In Use Test</td> <td>☐ Other: None</td> </tr> <tr> <td>☐ Comparison with Analytic Solutions</td> <td></td> </tr> <tr> <td>☒ Comparison with Other Verified Programs</td> <td></td> </tr> </table>			☐ Hand Calculations	☐ Parametric Sensitivity Analysis	☐ In Use Test	☐ Other: None	☐ Comparison with Analytic Solutions		☒ Comparison with Other Verified Programs	
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☐ In Use Test	☐ Other: None									
☐ Comparison with Analytic Solutions										
☒ Comparison with Other Verified Programs										
17. Documentation: <p style="margin-top: 20px;">User's Reference: Rechard, R.P., et al., 1991. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND90-1983, Version 3.0. Albuquerque, NM: Sandia National Laboratories.</p> <p style="margin-top: 20px;">Theoretical Reference: None</p> <p style="margin-top: 20px;">Verification/Sample Problems Reference: See code notebook (on file with SWCF).</p> <p style="margin-top: 20px;">Other Reference: Gilkey, A.P., 1988. ALGEBRA - A program that algebraically manipulates the output of a finite element analysis (EXODUS version). SAND88-1431. Albuquerque, NM: Sandia National Laboratories.</p>										
18. Primary Reference: Rechard, R.P., et al., 1991. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND90-1983, Version 3.0. Albuquerque, NM: Sandia National Laboratories.										
19. Approved by: <div style="display: flex; justify-content: space-between; align-items: flex-end; margin-top: 20px;"> <div style="text-align: center;"> _____ PA QA Coordinator </div> <div style="text-align: center;"> <u>12-27-93</u> _____ Date </div> </div>										

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: BCSET	2a. Version ID: C-2.03VV	2b. Version Date: 02/25/93	3. Host Computer: DEC-VMS Alpha
4. Software Title: BCSET: A simple translator which sets nodal and/or element boundary conditions to 1-,2-, or 3-D rectilinear finite difference grids/finite eleme			
5. Software Description: Sets nodal boundary flags, element boundary flags, and values to new or existing boundary flags. All new boundary flag geometry and values are obtained from a user input file. Values can also be set to previously defined boundary flags (i.e. from GENMESH or GENNET). In addition, any boundary flag (existing or new) can be linearly interpolated by specifying interpolation tables in the BCSET input text file.			
6. Limitations:			
7. Keyword/Phrases:			
8. Origin/Source:			
9. Authors: J. S. Rath,			
10. Parent Program:			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	<div style="border: 1px solid black; padding: 2px; display: inline-block;"></div> Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2								
Date: 12/27/93										
Software Name: BCSET	Version ID: C-2.03VV	Version Date: 02/25/93								
13. Classification: Government owned, Limited distribution.										
14. Software Sponsor: J. S. Rath	15. Software Consultant: J. S. Rath									
16. Verification: <table style="width: 100%; margin-top: 10px;"> <tr> <td>☒ Hand Calculations</td> <td>☐ Parametric Sensitivity Analysis</td> </tr> <tr> <td>☐ In Use Test</td> <td>☐ Other: None</td> </tr> <tr> <td colspan="2">☐ Comparison with Analytic Solutions</td> </tr> <tr> <td colspan="2">☒ Comparison with Other Verified Programs</td> </tr> </table>			☒ Hand Calculations	☐ Parametric Sensitivity Analysis	☐ In Use Test	☐ Other: None	☐ Comparison with Analytic Solutions		☒ Comparison with Other Verified Programs	
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17. Documentation: <p style="margin-top: 20px;">User's Reference: Rechard, R.P., ed. 1992. User's Reference Manual for CAMCON: Compliance Assessment Methodology Controller, Version 3.0. SAND90-1983. Albuquerque, NM: Sandia National Laboratories.</p> <p style="margin-top: 20px;">Theoretical Reference: None</p> <p style="margin-top: 20px;">Verification/Sample Problems Reference: Computer/Node: Sandia VAX/VMS 8810 / Jimmy Directory: CAMCON\$ROOT:[CAMCON.BCSET.TEST] Input Files: BCSET_TEST_1.INP, BCSET_TEST_2.INP, BCSET_TEST_3.INP Command Files: QA_BCSET.COM</p> <p style="margin-top: 20px;">Other Reference: None</p>										
18. Primary Reference: Rechard, R.P., ed. 1992. User's Reference Manual for CAMCON: Compliance Assessment Methodology Controller, Version 3.0. SAND90-1983. Albuquerque, NM: Sandia National Laboratories.										
19. Approved by: <div style="display: flex; justify-content: space-between; align-items: flex-end; margin-top: 20px;"> <div style="text-align: center;"> _____ PA QA Coordinator </div> <div style="text-align: center;"> <u>12-27-93</u> Date </div> </div>										

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: BLOTCDB	2a. Version ID: C-1.30ZO	2b. Version Date: 12/16/93	3. Host Computer: ALPHA/VMS
4. Software Title: BLOTCDB: A code to plot the mesh and results from finite element and finite difference analysis programs			
5. Software Description: The plotting support module, BLOTCDB, plots all intermediate and final results from all main modules. BLOTCDB directly reads CAMDAT and plots (1) the computational mesh with contoured analysis results, (2) grid distance versus any variable, and/or (3) any variable versus any other variable.			
6. Limitations: BLOTCDB can only plot to devices which have a Sandia VDI driver. BLOTCDB plots 2-node and 4-node 2D elements and 8-node brick 3D elements; other element types are not implemented.			
7. Keyword/Phrases: WIPP, Performance Assessment, Plot Program			
8. Origin/Source: Sandia National Laboratories, Applied Mechanics Division 1523, Albuquerque, NM 87185-5800			
9. Authors: Amy P. Gilkey			
10. Parent Program: BLOT			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2								
Date: 12/27/93										
Software Name: BLOTADB	Version ID: C-1.30ZO	Version Date: 12/16/93								
13. Classification: Government owned, Limited distribution.										
14. Software Sponsor: Amy P. Gilkey		15. Software Consultant: Amy P. Gilkey								
16. Verification: <table style="width: 100%; margin-top: 10px;"> <tr> <td>☐ Hand Calculations</td> <td>☐ Parametric Sensitivity Analysis</td> </tr> <tr> <td>☐ In Use Test</td> <td>☐ Other: None</td> </tr> <tr> <td>☐ Comparison with Analytic Solutions</td> <td></td> </tr> <tr> <td>☒ Comparison with Other Verified Programs</td> <td></td> </tr> </table>			☐ Hand Calculations	☐ Parametric Sensitivity Analysis	☐ In Use Test	☐ Other: None	☐ Comparison with Analytic Solutions		☒ Comparison with Other Verified Programs	
☐ Hand Calculations	☐ Parametric Sensitivity Analysis									
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☐ Comparison with Analytic Solutions										
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17. Documentation: <p style="margin-top: 20px;">User's Reference: Rechard, R.P., et al., 1991. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND90-1983, Version 3.0. Albuquerque, NM: Sandia National Laboratories.</p> <p style="margin-top: 20px;">Theoretical Reference: None</p> <p style="margin-top: 20px;">Verification/Sample Problems Reference: See code notebook (on file with SWCF).</p> <p style="margin-top: 20px;">Other Reference: Gilkey, A.P. and J.H. Glick, 1989. BLOT - A mesh and curve plot program for the output of a finite element analysis. SAND88-1432. Albuquerque, NM: Sandia National Laboratories.</p>										
18. Primary Reference: Rechard, R.P., et al., 1991. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND90-1983, Version 3.0. Albuquerque, NM: Sandia National Laboratories.										
19. Approved by: <div style="display: flex; justify-content: space-between; align-items: flex-end; margin-top: 20px;"> <div style="text-align: center;"> _____ PA QA Coordinator </div> <div style="text-align: center;"> <u>12-27-93</u> _____ Date </div> </div>										

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: BRAGFLO_T	2a. Version ID: X-1.00ZO	2b. Version Date: 08/01/93	3. Host Computer: ALPHA/VMS
4. Software Title: BRAGFLO_T: A 3-dimensional finite difference, 2-phase flow simulator with energy balance and thermal effects.			
5. Software Description: BRAGFLO_T simulates simultaneous 2-phase (brine and gas) flow in porous media with heat transport.			
6. Limitations: No water vaporization/condensation. No solution gas. No restart capability.			
7. Keyword/Phrases: INEL, Performance Assessment, BRAGFLO_T, ground water flow, multiphase fluid flow, gas generation, gas flow, CAMCON.			
8. Origin/Source: Sandia National Laboratories Performance Assessment Division 6342 Albuquerque, NM 87185			
9. Authors: Palmer Vaughn, Michael Lord Applied Physics, Inc. (API) API: 823-2334, GEOCENTERS: 766-9629			
10. Parent Program: BRAGFLO			
11. Computer Languages: FORTTRAN 77			
12. Status: X			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2
Date: 12/27/93		
Software Name: BRAGFLO_T	Version ID: X-1.00ZO	Version Date: 08/01/93
13. Classification: Government owned, Limited Distribution.		
14. Software Sponsor: Michael Lord		15. Software Consultant: P. Vaughn, M. Lord
16. Verification: <div style="display: flex; flex-wrap: wrap; padding: 5px;"> <div style="width: 50%;">☐ Hand Calculations</div> <div style="width: 50%;">☐ Parametric Sensitivity Analysis</div> <div style="width: 50%;">☐ In Use Test</div> <div style="width: 50%;">☐ Other: None</div> <div style="width: 50%;">☐ Comparison with Analytic Solutions</div> <div style="width: 50%;">☐ Comparison with Other Verified Programs</div> </div>		
17. Documentation: <p>User's Reference: Rechard, R.P., ed., 1992. "User's Reference Manual for CAMCON: Compliance Assessment Methodology Controller Version 3.0," SAND90-1983 (UC-721), Sandia National Laboratories, Albuquerque, NM.</p> <p>Theoretical Reference: Crookston, R.B., W.E. Culham, and W.H. Chen, 1979. "Numerical Simulation Model for Thermal Recovery Process", Soc. of Petroleum Engr. J., Vol. 19, no. 1, 37-58.</p> <p>Verification/Sample Problems Reference: Ross, B., J.W. Mercer, S.D. Thomas, and B.H. Lester, "Benchmark Problems for Repository Siting Models," NUREG/CR-3097, Geotrans, Inc., Reston, VA, Dec. 1982.</p> <p>Other Reference: None Primary Reference: Rechard, R.P., ed., 1992. "User's Reference Manual for CAMCON: Compliance Assessment Methodology Controller Version 3.0," SAND90-1983 (UC-721), Sandia National Laboratories, Albuquerque, NM.</p>		
18. Primary Reference: Rechard, R.P., ed., 1992. "User's Reference Manual for CAMCON: Compliance Assessment Methodology Controller Version 3.0," SAND90-1983 (UC-721), Sandia National Laboratories, Albuquerque, NM.		
19. Approved by: <div style="display: flex; justify-content: space-between; align-items: flex-end; margin-top: 20px;"> <div style="text-align: center;"> _____ PA QA Coordinator </div> <div style="text-align: center;"> <u>12-27-93</u> _____ Date </div> </div>		

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: BRTSUM	2a. Version ID: X-1.00ZO	2b. Version Date: 09/29/93	3. Host Computer: ALPHA/VMS
4. Software Title: BRTSUM: A code to produce a summary table of canister layer breach times			
5. Software Description: BRTSUM is used to create a table of breach times for all canister layers and all sample vectors using the CONCNF breach time transfer files created during execution of GRFLOW or SALFLOW.			
6. Limitations: Only breach times for the first location listed in the CONCNF breach time table are extracted.			
7. Keyword/Phrases: INEL, Performance Assessment, Utility			
8. Origin/Source: Sandia National Laboratories Performance Assessment Division 6342 Albuquerque, NM 87185 - 5800			
9. Authors: Walt Beyeler			
10. Parent Program: None			
11. Computer Languages: FORTRAN 77			
12. Status: X			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2								
Date: 12/27/93										
Software Name: BRTSUM	Version ID: X-1.00ZO	Version Date: 09/29/93								
13. Classification: Government Owned, Limited Distribution.										
14. Software Sponsor: Walt Beyeler	15. Software Consultant:									
16. Verification: <table style="width: 100%; margin-top: 10px;"> <tr> <td>☐ Hand Calculations</td> <td>☐ Parametric Sensitivity Analysis</td> </tr> <tr> <td>☐ In Use Test</td> <td>☐ Other: None</td> </tr> <tr> <td colspan="2">☐ Comparison with Analytic Solutions</td> </tr> <tr> <td colspan="2">☒ Comparison with Other Verified Programs</td> </tr> </table>			☐ Hand Calculations	☐ Parametric Sensitivity Analysis	☐ In Use Test	☐ Other: None	☐ Comparison with Analytic Solutions		☒ Comparison with Other Verified Programs	
☐ Hand Calculations	☐ Parametric Sensitivity Analysis									
☐ In Use Test	☐ Other: None									
☐ Comparison with Analytic Solutions										
☒ Comparison with Other Verified Programs										
17. Documentation: <div style="margin-top: 10px;">User's Reference: None</div> <div style="margin-top: 20px;">Theoretical Reference: None</div> <div style="margin-top: 20px;">Verification/Sample Problems Reference: None</div> <div style="margin-top: 20px;">Other Reference: None</div>										
18. Primary Reference: None										
19. Approved by: <div style="display: flex; justify-content: space-between; margin-top: 20px;"> <div style="width: 45%;"> _____ PA QA Coordinator </div> <div style="width: 45%; text-align: right;"> 6-27-93 _____ Date </div> </div>										

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: CCD2STEP	2a. Version ID: C-1.04VV	2b. Version Date: 12/01/92	3. Host Computer: DEC-VMS Alpha
4. Software Title: CCD2STEP: Transfers CCDFCALC releases to STEPWISE or PCCSRC input.			
5. Software Description: CCD2STEP reads CCDFCALC files containing the radionuclide release data for the dependent variables and writes an output file for either STEPWISE or PCCSRC.			
6. Limitations: CCD2STEP has the following input maximums: 40 radionuclides 100 scenarios			
7. Keyword/Phrases: WIPP, Performance Assessment, Plot Program			
8. Origin/Source: Sandia National Laboratories, Safety and Reliability Analysis Division 6415, Albuquerque, NM 87185-5800			
9. Authors: Amy P. Gilkey			
10. Parent Program: None			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2
Date: 12/27/93		
Software Name: CCD2STEP	Version ID: C-1.04VV	Version Date: 12/01/92
13. Classification: Government owned, Limited distribution.		
14. Software Sponsor: Amy P. Gilkey	15. Software Consultant:	
16. Verification: <div style="display: flex; flex-wrap: wrap; margin-top: 10px;"> <div style="width: 50%;">☐ Hand Calculations</div> <div style="width: 50%;">☐ Parametric Sensitivity Analysis</div> <div style="width: 50%;">☐ In Use Test</div> <div style="width: 50%;">☐ Other: None</div> <div style="width: 50%;">☐ Comparison with Analytic Solutions</div> <div style="width: 50%;">☒ Comparison with Other Verified Programs</div> </div>		
17. Documentation: <div style="margin-top: 10px;"> User's Reference: Rechard, R.P., et al., 1991. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND90-1983, Version 3.0. Albuquerque, NM: Sandia National Laboratories. </div> <div style="margin-top: 20px;"> Theoretical Reference: None </div> <div style="margin-top: 20px;"> Verification/Sample Problems Reference: None </div> <div style="margin-top: 20px;"> Other Reference: None </div>		
18. Primary Reference: Rechard, R.P., et al., 1991. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND90-1983, Version 3.0. Albuquerque, NM: Sandia National Laboratories.		
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INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: CCDFCALC	2a. Version ID: C-4.12ZO	2b. Version Date: 12/06/93	3. Host Computer: ALPHA/VMS
4. Software Title: CCDFCALC: A code to calculate a complementary cumulative distribution function from CAMDAT files to be plotted by CCDFPLOT			
5. Software Description: For each vector of all scenarios, CCDFCALC, a code in the compliance module, accesses the appropriate CAMDAT file, integrates to find cumulative releases for each radionuclide, calculates the EPA sum, and combines with other EPA sums to generate the CCDF. The CCDF is written to a transfer file used by CCDFPLOT.			
6. Limitations: CCDFCALC is limited to 10 scenarios, 100 vectors each.			
7. Keyword/Phrases: WIPP, Performance Assessment			
8. Origin/Source: Sandia National Laboratories, Waste Management Systems Division 6416, Albuquerque, NM 87185-5800			
9. Authors: Amy P. Gilkey, M.J. Shortencarier			
10. Parent Program: CCDF, CCDFPLT			
11. Computer Languages: FORTTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2
Date: 12/27/93		
Software Name: CCDFCALC	Version ID: C-4.12ZO	Version Date: 12/06/93
13. Classification: Government owned, Limited distribution.		
14. Software Sponsor: Amy P. Gilkey	15. Software Consultant: Amy P. Gilkey	
16. Verification: <div style="display: flex; flex-wrap: wrap; margin-top: 10px;"> <div style="width: 50%;"> ☐ Hand Calculations </div> <div style="width: 50%;"> ☐ Parametric Sensitivity Analysis </div> <div style="width: 50%;"> ☐ In Use Test </div> <div style="width: 50%;"> ☐ Other: None </div> <div style="width: 50%;"> ☐ Comparison with Analytic Solutions </div> <div style="width: 50%;"> ☒ Comparison with Other Verified Programs </div> </div>		
17. Documentation: <div style="margin-top: 10px;"> User's Reference: Rechard, R.P., et al., 1991. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND90-1983, Version 3.0. Albuquerque, NM: Sandia National Laboratories. </div> <div style="margin-top: 20px;"> Theoretical Reference: None </div> <div style="margin-top: 20px;"> Verification/Sample Problems Reference: See code notebook (on file with SWCF). </div> <div style="margin-top: 20px;"> Other Reference: None </div>		
18. Primary Reference: Rechard, R.P., et al., 1991. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND90-1983, Version 3.0. Albuquerque, NM: Sandia National Laboratories.		
19. Approved by: <div style="display: flex; justify-content: space-between; margin-top: 20px;"> <div style="width: 45%;"> PA QA Coordinator </div> <div style="width: 45%; text-align: right;"> <div style="margin-bottom: 5px;">12-27-93</div> Date </div> </div>		

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: CCDFPERMF	2a. Version ID: X-3.20ZO	2b. Version Date: 12/14/93	3. Host Computer: ALPHA/VMS
4. Software Title: CCDFPERMF: Build a CCDF for a FRACTURED medium.			
5. Software Description: Since the calculation of all possible vectors for up to 10 hits on a repository is beyond the capabilities of the available computing resources, the vectors are built from discharges and cuttings from single and double intrusions by assuming the results are independent.			
6. Limitations: This program assumes flow discharges are dominated by the fractures, i.e., flow is the same no matter what the number of drilling intrusions is, even no intrusions. The cuttings are proportional to the number of intrusions. At most 10 intrusions, 2 intrusion types, 60 nuclides, 100 vectors per intrusion time, 6 intrusion times, and 10 waste package types.			
7. Keyword/Phrases: complementary cumulative discharge function, Latin Hypercube sampling.			
8. Origin/Source: Sandia National Laboratories			
9. Authors: Harold J. Iuzzolino, GEO-CENTERS, INC.			
10. Parent Program: None.			
11. Computer Languages: FORTRAN 77			
12. Status: X			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	<div style="border: 1px solid black; padding: 2px; display: inline-block;"></div> Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2
Date: 12/27/93		
Software Name: CCDFPERMF	Version ID: X-3.20ZO	Version Date: 12/14/93
13. Classification: Government owned, Limited distribution.		
14. Software Sponsor: Harold J. Iuzzolino, GEO	15. Software Consultant:	
16. Verification: <div style="display: flex; flex-wrap: wrap; margin-top: 10px;"> <div style="width: 50%;">☐ Hand Calculations</div> <div style="width: 50%;">☐ Parametric Sensitivity Analysis</div> <div style="width: 50%;">☐ In Use Test</div> <div style="width: 50%;">☐ Other: None</div> <div style="width: 50%;">☐ Comparison with Analytic Solutions</div> <div style="width: 50%;">☒ Comparison with Other Verified Programs</div> </div>		
17. Documentation: <div style="margin-top: 20px;"> User's Reference: None. </div> <div style="margin-top: 40px;"> Theoretical Reference: Construction of Complementary Cumulative Distribution Functions for Comparison with the EPA Release Limits for Radioactive Waste Disposal, Jon C. Helton, Department of Mathematics, Arizona State University, and Harold J. Iuzzolino, GEO-CENTERS, INC., Albuquerque, NM, 1992 </div> <div style="margin-top: 20px;"> Verification/Sample Problems Reference: None </div> <div style="margin-top: 40px;"> Other Reference: Drilling Intrusion Probabilities for Use In Performance Assessment for Radioactive Waste Disposal, Jon C. Helton, Department of Mathematics, Arizona State University, 1992 </div>		
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Date: 12/27/93			
1. Software Name: CCDFPERMP	2a. Version ID: X-3.20ZO	2b. Version Date: 12/14/93	3. Host Computer: ALPHA/VMS
4. Software Title: CCDFPERMP: Build a CCDF for a PERMEABLE medium.			
5. Software Description: Since the calculation of all possible vectors for up to 10 hits on a repository is beyond the capabilities of the available computing resources, the vectors are built from discharges and cuttings from single and double intrusions by assuming the results are independent.			
6. Limitations: This program assumes that the medium is permeable enough that flow discharges and cuttings are each proportional to the number of drilling intrusions. At most 10 intrusions, 2 intrusion types, 60 nuclides, 100 vectors per intrusion time, 6 intrusion times, and 10 waste package types.			
7. Keyword/Phrases: complementary cumulative discharge function, Latin Hypercube sampling.			
8. Origin/Source: Sandia National Laboratories			
9. Authors: Harold J. Iuzzolino, GEO-CENTERS, INC.			
10. Parent Program: None.			
11. Computer Languages: FORTRAN 77			
12. Status: X			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	 Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2								
Date: 12/27/93										
Software Name: CCDFPERMP	Version ID: X-3.20ZO	Version Date: 12/14/93								
13. Classification: Government owned, Limited distribution.										
14. Software Sponsor: Harold J. Iuzzolino, GEO	15. Software Consultant:									
16. Verification: <table style="width: 100%; margin-top: 10px;"> <tr> <td>☐ Hand Calculations</td> <td>☐ Parametric Sensitivity Analysis</td> </tr> <tr> <td>☐ In Use Test</td> <td>☐ Other: None</td> </tr> <tr> <td>☐ Comparison with Analytic Solutions</td> <td></td> </tr> <tr> <td>☒ Comparison with Other Verified Programs</td> <td></td> </tr> </table>			☐ Hand Calculations	☐ Parametric Sensitivity Analysis	☐ In Use Test	☐ Other: None	☐ Comparison with Analytic Solutions		☒ Comparison with Other Verified Programs	
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17. Documentation: <p style="margin-top: 20px;">User's Reference: None.</p> <p style="margin-top: 40px;">Theoretical Reference: Construction of Complementary Cumulative Distribution Functions for Comparison with the EPA Release Limits for Radioactive Waste Disposal, Jon C. Helton, Department of Mathematics, Arizona State University, and Harold J. Iuzzolino, GEO-CENTERS, INC., Albuquerque, NM, 1992</p> <p style="margin-top: 20px;">Verification/Sample Problems Reference: None</p> <p style="margin-top: 40px;">Other Reference: Drilling Intrusion Probabilities for Use In Performance Assessment for Radioactive Waste Disposal, Jon C. Helton, Department of Mathematics, Arizona State University, 1992</p>										
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19. Approved by: <div style="display: flex; justify-content: space-between; align-items: flex-end; margin-top: 20px;"> <div style="text-align: center;"> _____ PA QA Coordinator </div> <div style="text-align: center;"> 12-27-93 _____ Date </div> </div>										

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Date: 12/27/93			
1. Software Name: CCDFPLOT	2a. Version ID: C-4.11ZO	2b. Version Date: 09/27/93	3. Host Computer: ALPHA/VMS
4. Software Title: CCDFPLOT: A code to plot a complementary cumulative distribution function calculated by CCDFCALC			
5. Software Description: CCDFPLOT interactively plots the CCDF function read from the CCDFCALC transfer file.			
6. Limitations: CCDFPLOT is limited to 10 scenarios, 100 vectors each.			
7. Keyword/Phrases: WIPP, Performance Assessment, Plot Program			
8. Origin/Source: Sandia National Laboratories, Waste Management Systems Division 6416, Albuquerque, NM 87185-5800			
9. Authors: Amy P. Gilkey			
10. Parent Program: CCDFPLT			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2								
Date: 12/27/93										
Software Name: CCDFPLOT	Version ID: C-4.11ZO	Version Date: 09/27/93								
13. Classification: Government owned, Limited distribution.										
14. Software Sponsor: Amy P. Gilkey		15. Software Consultant: Amy P. Gilkey								
16. Verification: <table style="width: 100%; margin-top: 10px;"> <tr> <td>☐ Hand Calculations</td> <td>☐ Parametric Sensitivity Analysis</td> </tr> <tr> <td>☐ In Use Test</td> <td>☐ Other: None</td> </tr> <tr> <td>☐ Comparison with Analytic Solutions</td> <td></td> </tr> <tr> <td>☒ Comparison with Other Verified Programs</td> <td></td> </tr> </table>			☐ Hand Calculations	☐ Parametric Sensitivity Analysis	☐ In Use Test	☐ Other: None	☐ Comparison with Analytic Solutions		☒ Comparison with Other Verified Programs	
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☒ Comparison with Other Verified Programs										
17. Documentation: <p style="margin-top: 20px;">User's Reference: Rechard, R.P., et al., 1991. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND90-1983, Version 3.0. Albuquerque, NM: Sandia National Laboratories.</p> <p style="margin-top: 40px;">Theoretical Reference: None</p> <p style="margin-top: 40px;">Verification/Sample Problems Reference: See code notebook (on file with SWCF).</p> <p style="margin-top: 40px;">Other Reference: None</p>										
18. Primary Reference: Rechard, R.P., et al., 1991. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND90-1983, Version 3.0. Albuquerque, NM: Sandia National Laboratories.										
19. Approved by: <div style="display: flex; justify-content: space-between; align-items: flex-end; margin-top: 10px;"> <div style="text-align: center;"> <hr style="width: 100%;"/> PA QA Coordinator </div> <div style="text-align: center;"> <hr style="width: 100%;"/> 12-27-93 Date </div> </div>										

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: CLAMVD	2a. Version ID: X-1.00ZO	2b. Version Date: 08/20/93	3. Host Computer: ALPHA/VMS
4. Software Title: CLAMVD: A code to simulate coupled lateral advection with multiple vertical diffusion in a granite or salt repository.			
5. Software Description: CLAMVD is used to simulate transport within and from the INEL granite and salt repositories following interconnection or the repository and an external fluid source. Radionuclide sources are provided by internal coupling to the CONCNF model.			
6. Limitations: Transport only occurs through the repository, WP hole backfill, and fluid pathways (fracture connections or boreholes). Transport through the WP hole backfill is due to molecular diffusion alone; transport through the drift is due to advection and dispersion. Linear equilibrium sorbtion on the repository backfill may be included.			
7. Keyword/Phrases: INEL, Performance Assessment, Transport Model			
8. Origin/Source: Sandia National Laboratories Performance Assessment Division 6342 Albuquerque, NM 87185 - 5800			
9. Authors: Walt Beyeler			
10. Parent Program: None			
11. Computer Languages: FORTRAN 77			
12. Status: X			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	<div style="border: 1px solid black; padding: 2px; display: inline-block;"> </div> Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2
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Date: 12/27/93

Software Name: CLAMVD	Version ID: X-1.00ZO	Version Date: 08/20/93
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13. Classification:
Government Owned, Limited Distribution.

14. Software Sponsor: Walt Beyeler	15. Software Consultant:
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16. Verification:

☐ Hand Calculations	☐ Parametric Sensitivity Analysis
☐ In Use Test	☐ Other: None
☐ Comparison with Analytic Solutions	
☒ Comparison with Other Verified Programs	

17. Documentation:

User's Reference: **None**

Theoretical Reference: **None**

Verification/Sample Problems Reference: **None**

Other Reference: **None**

18. Primary Reference:
None

19. Approved by:

 PA QA Coordinator

12-27-93

 Date

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: COND_1D	2a. Version ID: X-1.00VV	2b. Version Date: 12/03/93	3. Host Computer: VAX/VMS 8810
4. Software Title: COND_1D: Finite difference heat conduction solution with uneven node spacing.			
5. Software Description: Calculates the temperature distribution for 1D spherical or cylindrical conduction heat flow with a decaying central heat source. Permits spatially varying thermal properties and spatially varying heat generation.			
6. Limitations:			
7. Keyword/Phrases:			
8. Origin/Source:			
9. Authors: G. R. Durka, J. W. Berglund			
10. Parent Program:			
11. Computer Languages: FORTRAN 77			
12. Status: X			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	<div style="display: flex; align-items: center;"> <div> Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2 </div> </div>
Date: 12/27/93		
Software Name: COND_1D	Version ID: X-1.00VV	Version Date: 12/03/93
13. Classification: Government owned, Limited distribution.		
14. Software Sponsor: G. R. Durka	15. Software Consultant: J. W. Berglund	
16. Verification: <div style="display: flex; flex-wrap: wrap; margin-top: 10px;"> <div style="width: 50%;"> ☒ Hand Calculations </div> <div style="width: 50%;"> ☐ Parametric Sensitivity Analysis </div> <div style="width: 50%;"> ☐ In Use Test </div> <div style="width: 50%;"> ☐ Other: None </div> <div style="width: 50%;"> ☒ Comparison with Analytic Solutions </div> <div style="width: 50%;"> ☐ Comparison with Other Verified Programs </div> </div>		
17. Documentation: <div style="margin-top: 10px;"> User's Reference: None </div> <div style="margin-top: 20px;"> Theoretical Reference: None </div> <div style="margin-top: 20px;"> Verification/Sample Problems Reference: None </div> <div style="margin-top: 20px;"> Other Reference: None </div>		
18. Primary Reference: None		
19. Approved by: <div style="display: flex; justify-content: space-between; margin-top: 20px;"> <div style="width: 45%;"> </div> <div style="width: 45%; text-align: center;"> <u>12-27-93</u> Date </div> </div> <div style="display: flex; justify-content: space-between; margin-top: 5px;"> <div style="width: 45%;">PA QA Coordinator</div> <div style="width: 45%;"></div> </div>		

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: CUTTINGS	2a. Version ID: C-3.01ZO	2b. Version Date: 12/07/93	3. Host Computer: VAX/VMS 8810
4. Software Title: CUTTINGS: A borehole drilling program			
5. Software Description: Calculates the quantity of radioactive material (in Curies) brought to the surface as cuttings generated by a drilling operation that penetrates a rad-waste repository. The code determines the amount of cuttings removed by drilling and mud erosion, and ages the cuttings to the intrusion time.			
6. Limitations:			
7. Keyword/Phrases:			
8. Origin/Source:			
9. Authors: J. W. Berglund and J. S. Rath			
10. Parent Program:			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2
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Date: 12/27/93

Software Name: CUTTINGS	Version ID: C-3.0120	Version Date: 12/07/93
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13. Classification:
Government owned, Limited distribution.

14. Software Sponsor: J. W. Berglund	15. Software Consultant: J. W. Berglund
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16. Verification:

☐ Hand Calculations	☐ Parametric Sensitivity Analysis
☐ In Use Test	☐ Other: None
☐ Comparison with Analytic Solutions	
☒ Comparison with Other Verified Programs	

17. Documentation:

User's Reference: **Rechard, R. P. ed. 1994. "User's Manual for CAMCON: Compliance Assessment Methodology Controller, Version 4.0." SAND90-19XX. Albuquerque, NM: Sandia National Laboratories.**

Theoretical Reference: **None** Other Reference: **None**

Verification/Sample Problems Reference: **Computer: Sandia National Laboratories "ALPHA" Directory: NFS3:[JSRATH.CUTTINGS.TEST] Input Files: CUTTINGS_WP01.INP, CUTTINGS_WIPP.INP Command Files: CUTTINGS1.COM, CUTTINGS_WIPP.COM**

Other Reference: **None**

18. Primary Reference:
J. W. Berglund. "Mechanisms Governing the Direct Removal of Wastes from the WIPP Repository Caused by Exploratory Drilling." SAND92-7295. Albuquerque, NM: Sandia National Laboratories.

19. Approved by:

PA QA Coordinator

12-27-93
 Date

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: DECAY	2a. Version ID: X-1.04ZO	2b. Version Date: 12/06/93	3. Host Computer: ALPHA/VMS
4. Software Title: DECAY: A code which decays the inventory (in Curies) of each radioisotope from each D.O.E. HLW supplier to the year 2030.			
5. Software Description: Decays radioisotopes according to a simplified bateman's equation model.			
6. Limitations:			
7. Keyword/Phrases:			
8. Origin/Source:			
9. Authors: J. S. Rath			
10. Parent Program:			
11. Computer Languages: FORTRAN 77			
12. Status: X			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2
Date: 12/27/93		
Software Name: DECAY	Version ID: X-1.04ZO	Version Date: 12/06/93
13. Classification: Government owned, Limited distribution.		
14. Software Sponsor: J. S. Rath	15. Software Consultant: J. S. Rath	
16. Verification: <div style="display: flex; flex-wrap: wrap; margin-top: 10px;"> <div style="width: 50%;"> ☒ Hand Calculations </div> <div style="width: 50%;"> ☐ Parametric Sensitivity Analysis </div> <div style="width: 50%;"> ☐ In Use Test </div> <div style="width: 50%;"> ☐ Other: None </div> <div style="width: 50%;"> ☐ Comparison with Analytic Solutions </div> <div style="width: 50%;"> ☒ Comparison with Other Verified Programs </div> </div>		
17. Documentation: <div style="margin-top: 20px;"> User's Reference: None </div> <div style="margin-top: 40px;"> Theoretical Reference: None </div> <div style="margin-top: 40px;"> Verification/Sample Problems Reference: Computer/Node: Sandia "IKE" Directory: NFS3:[JSRATH.INEL.DECAY] Input Files: Command Files: </div> <div style="margin-top: 40px;"> Other Reference: None </div>		
18. Primary Reference: None		
19. Approved by: <div style="display: flex; justify-content: space-between; align-items: flex-end; margin-top: 20px;"> <div style="text-align: center;"> PA QA Coordinator </div> <div style="text-align: center;"> <u>12-27-93</u> Date </div> </div>		

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: DVRANK	2a. Version ID: X-1.00ZO	2b. Version Date: 09/29/93	3. Host Computer: ALPHA/VMS
4. Software Title: DVRANK: A code to calculate ranks of integrated release			
5. Software Description: DVRANK ranks the integrated release (calculated by CCDFCALC) of any subset of scenarios or design alternatives for all sample vectors.			
6. Limitations: All integrated release values must be in the same STEPWISE transfer file.			
7. Keyword/Phrases: INEL, Performance Assessment, Utility			
8. Origin/Source: Sandia National Laboratories Performance Assessment Division 6342 Albuquerque, NM 87185 - 5800			
9. Authors: Walt Beyeler			
10. Parent Program: None			
11. Computer Languages: FORTRAN 77			
12. Status: X			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2								
Date: 12/27/93										
Software Name: DVRANK	Version ID: X-1.00ZO	Version Date: 09/29/93								
13. Classification: Government Owned, Limited Distribution.										
14. Software Sponsor: Walt Beyeler	15. Software Consultant:									
16. Verification: <table style="width: 100%; margin-top: 10px;"> <tr> <td>☐ Hand Calculations</td> <td>☐ Parametric Sensitivity Analysis</td> </tr> <tr> <td>☐ In Use Test</td> <td>☐ Other: None</td> </tr> <tr> <td>☐ Comparison with Analytic Solutions</td> <td></td> </tr> <tr> <td>☒ Comparison with Other Verified Programs</td> <td></td> </tr> </table>			☐ Hand Calculations	☐ Parametric Sensitivity Analysis	☐ In Use Test	☐ Other: None	☐ Comparison with Analytic Solutions		☒ Comparison with Other Verified Programs	
☐ Hand Calculations	☐ Parametric Sensitivity Analysis									
☐ In Use Test	☐ Other: None									
☐ Comparison with Analytic Solutions										
☒ Comparison with Other Verified Programs										
17. Documentation: <div style="margin-top: 10px;"> User's Reference: None </div> <div style="margin-top: 20px;"> Theoretical Reference: None </div> <div style="margin-top: 20px;"> Verification/Sample Problems Reference: None </div> <div style="margin-top: 20px;"> Other Reference: None </div>										
18. Primary Reference: None										
19. Approved by: <div style="display: flex; justify-content: space-between; align-items: flex-end; margin-top: 20px;"> <div style="text-align: center;"> _____ PA QA Coordinator </div> <div style="text-align: center;"> 12-27-93 _____ Date </div> </div>										

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: GENII-S	2a. Version ID: C-1.00'VV	2b. Version Date: 07/26/93	3. Host Computer: VAX/VMS 8810
4. Software Title: GENII-S: A code for statistical and deterministic simulations of radiation doses to humans from radio-nuclides in the environment			
5. Software Description: Calculates internal and external radiation doses to Man from routine and accidental releases of radionuclides to air, water, or residing in soil			
6. Limitations:			
7. Keyword/Phrases: WIPP, Performance Assessment, GENII, Safety, Dose Calculations, Exposure Pathways			
8. Origin/Source: Pacific Northwest Laboratories, Richland, Wa Sandia National Laboratories, Performance Assessment Division 6342, Albuquerque NM, 87185-5800			
9. Authors: J. E. Campbell, R. A. Kennedy, C. D. Leigh, D. E. Longsine, B. A. Napier, R. A. Peloquin, D. L. Strenge, J. V. Ramsdell, B. M. Thompson			
10. Parent Program: GENII			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2
Date: 12/27/93		
Software Name: GENII-S	Version ID: C-1.00VV	Version Date: 07/26/93
13. Classification: Government owned, unlimited distribution		
14. Software Sponsor: Ron D. McCurley	15. Software Consultant:	
16. Verification: <div style="display: flex; flex-wrap: wrap; margin-top: 10px;"> <div style="width: 50%;">☐ Hand Calculations</div> <div style="width: 50%;">☐ Parametric Sensitivity Analysis</div> <div style="width: 50%;">☐ In Use Test</div> <div style="width: 50%;">☐ Other: None</div> <div style="width: 50%;">☐ Comparison with Analytic Solutions</div> <div style="width: 50%;">☒ Comparison with Other Verified Programs</div> </div>		
17. Documentation: <div style="margin-top: 10px;"> <p>User's Reference: User's Guide for GENII-S: A Code for Statistical and Deterministic Simulations of Radiation Doses to Humans from Radionuclides in the Environment SAND91-0561; C.D. Leigh, B.M. Thompson, J.E. Campbell, D.E. Longsine, R.A. Kennedy, B.A. Napier</p> <p>Theoretical Reference: GENII-The Hanford Environmental Radiation Dosimetry Software System, Volume 1 by B.A. Napier, R.A. Peloquin, D.L. Strenge, J.V. Ramsdell for Pacific Northwest Laboratory</p> <p>Verification/Sample Problems Reference: User's Guide for GENII-S: A Code for Statistical and Deterministic Simulations of Radiation Doses to Humans from Radionuclides in the Environment SAND91-0561; C.D. Leigh, B.M. Thompson, J.E. Campbell, D.E. Longsine, R.A. Kennedy, B.A. Napier for Sandia National Laboratory</p> <p>Other Reference: GENII-The Hanford Environmental Radiation Dosimetry Software System, Volumes 2,3 by B.A. Napier, R.A. Peloquin, D.L. Strenge, J.V. Ramsdell for Pacific Northwest Laboratory</p> </div>		
18. Primary Reference: User's Guide for GENII-S: A Code for Statistical and Deterministic Simulations of Radiation Doses to Humans from Radionuclides in the Environment SAND91-0561; C.D. Leigh, B.M. Thompson, J.E. Campbell, D.E. Longsine, R.A. Kennedy, B.A. Napier for Sandia National Laboratory		
19. Approved by: <div style="display: flex; justify-content: space-between; margin-top: 20px;"> <div style="width: 45%;"> PA QA Coordinator </div> <div style="width: 45%; text-align: center;"> <u>12-27-93</u> Date </div> </div>		

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: GENMESH	2a. Version ID: A-6.03VV	2b. Version Date: 02/25/92	3. Host Computer: VAX/VMS 8810
4. Software Title: GENMESH: Rectilinear one-, two-, and three-dimensional finite difference grid or finite element mesh generator program.			
5. Software Description: Constructs a right-handed, Cartesian product, rectangular one-, two- or three-dimensional finite-difference grid from a geometry specification user input file. In addition to establishing the mesh connectivity and node coordinates, the program sets: material regions, geometry flags (identifies) for node or element boundary conditions, and elc .ations of elements. All information is then stored in a WIPP PA computational data base file, CAMDAT.			
6. Limitations:			
7. Keyword/Phrases:			
8. Origin/Source:			
9. Authors: J. S. Rath			
10. Parent Program:			
11. Computer Languages: FORTRAN 77			
12. Status: X			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2
Date: 12/27/93		
Software Name: GENMESH	Version ID: A-6.03VV	Version Date: 02/25/92
13. Classification: Government owned, Limited distribution.		
14. Software Sponsor: J. S. Rath	15. Software Consultant: J. S. Rath	
16. Verification: <div style="display: flex; flex-wrap: wrap; margin-top: 10px;"> <div style="width: 50%;"> ☒ Hand Calculations </div> <div style="width: 50%;"> ☐ Parametric Sensitivity Analysis </div> <div style="width: 50%;"> ☐ In Use Test </div> <div style="width: 50%;"> ☐ Other: None </div> <div style="width: 50%;"> ☐ Comparison with Analytic Solutions </div> <div style="width: 50%;"> ☒ Comparison with Other Verified Programs </div> </div>		
17. Documentation: <div style="margin-top: 10px;"> <p>User's Reference: Rechard, R.P., ed. 1992. User's Reference Manual for CAMCON: Compliance Assessment Methodology Controller, Version 3.0. SAND90-1383. Albuquerque, NM: Sandia National Laboratories.</p> <p>Theoretical Reference: None</p> <p>Verification/Sample Problems Reference: Computer/Node: Sandia VAX/VMS 8810 / Jimmy Directory: CAMCON\$ROOT:[CAMCON.GENMESH.A_LEVEL.TEST] Input Files: GENMESH_TEST_1D.INP, GENMESH_TEST_2D.INP, GENMESH_TEST_3D.INP, GENMESH_TEST_4.INP, GENMESH_TEST_5.INP, GENMESH_TEST_6.INP Command Files: QA_GENMESH.COM</p> <p>Other Reference: None</p> </div>		
18. Primary Reference: Rechard, R.P., ed. 1992. User's Reference Manual for CAMCON: Compliance Assessment Methodology Controller, Version 3.0. SAND90-1983. Albuquerque, NM: Sandia National Laboratories.		
19. Approved by: <div style="display: flex; justify-content: space-between; align-items: flex-end; margin-top: 20px;"> <div style="text-align: center;"> _____ PA QA Coordinator </div> <div style="text-align: center;"> <u>12-27-93</u> _____ Date </div> </div>		

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: GENNET	2a. Version ID: C-2.02VV	2b. Version Date: 05/12/93	3. Host Computer: VAX/VMS 8810
4. Software Title: GENNET: Network grid generator program.			
5. Software Description: Constructs simple one-, two-, or three-dimensional grid networks using two-node elements from a geometry specification user input file. In addition to establishing the mesh connectivity and node coordinates, the program sets: material regions, geometry flags (identifies) for node or element boundary conditions, and cross-sectional areas of elements. All information is then stored in a WIPP PA computational data base file, CAMDAT.			
6. Limitations:			
7. Keyword/Phrases:			
8. Origin/Source:			
9. Authors: J. S. Rath			
10. Parent Program:			
11. Computer Languages: FORTRAN 77			
12. Status: C

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INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	<div style="display: flex; align-items: center;"> Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2 </div>
Date: 12/27/93		
Software Name: GENNET	Version ID: C-2.02VV	Version Date: 05/12/93
13. Classification: Government owned, Limited distribution.		
14. Software Sponsor: J. S. Rath	15. Software Consultant: J. S. Rath	
16. Verification: <div style="display: flex; flex-wrap: wrap; margin-top: 10px;"> <div style="width: 50%;"> ☒ Hand Calculations </div> <div style="width: 50%;"> ☐ Parametric Sensitivity Analysis </div> <div style="width: 50%;"> ☐ In Use Test </div> <div style="width: 50%;"> ☐ Other: None </div> <div style="width: 50%;"> ☐ Comparison with Analytic Solutions </div> <div style="width: 50%;"> ☒ Comparison with Other Verified Programs </div> </div>		
17. Documentation: <div style="margin-top: 10px;"> User's Reference: Rechard, R.P., ed. 1992. User's Reference Manual for CAMCON: Compliance Assessment Methodology Controller, Version 3.0. SAND90-1983. Albuquerque, NM: Sandia National Laboratories. </div> <div style="margin-top: 20px;"> Theoretical Reference: None </div> <div style="margin-top: 20px;"> Verification/Sample Problems Reference: Computer/Node: Sandia VAX/VMS 8810 / Jimmy Directory: CAMCON\$ROOT:[CAMCON.GENNET.TEST] Input Files: GENNET_TEST_1D.INP, GENNET_TEST_2D.INP, GENNET_TEST_3D.INP, GENNET_TEST_4.INP, GENNET_TEST_5.INP, GENNET_TEST_6.INP Command Files: QA_GENNET.COM </div> <div style="margin-top: 10px;"> Other Reference: None </div>		
18. Primary Reference: Rechard, R.P., ed. 1992. User's Reference Manual for CAMCON: Compliance Assessment Methodology Controller, Version 3.0. SAND90-1983. Albuquerque, NM: Sandia National Laboratories.		
19. Approved by: <div style="display: flex; justify-content: space-between; margin-top: 20px;"> <div style="width: 45%;"> PA QA Coordinator </div> <div style="width: 45%; text-align: right;"> <u>12-27-93</u> Date </div> </div>		

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: GNRDP	2a. Version ID: X-1.02ZO	2b. Version Date: 12/15/93	3. Host Computer: ALPHA/VMS
4. Software Title: GNRDP: Generates Normalized Radioactive Decay Plots			
5. Software Description: GNRDP decays the D.O.E. HLW (from Hanford, Savannah River, and West Valley) Inventory (in Curies) of each radioisotope to the year 2030. Then each SNF/HLW supplier of radioisotopes are decayed for 10,000 years. Each radioisotope history is then normalized by the EPA 40CFR191 limit and history plots are then generated using the HJILOT graphics library routines.			
6. Limitations:			
7. Keyword/Phrases:			
8. Origin/Source:			
9. Authors: J. S. Rath			
10. Parent Program:			
11. Computer Languages: FORTRAN 77			
12. Status: X			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2
Date: 12/27/93		
Software Name: GNRDP	Version ID: X-1.02ZO	Version Date: 12/15/93
13. Classification: Government owned, Limited distribution.		
14. Software Sponsor: J. S. Rath	15. Software Consultant: J. S. Rath	
16. Verification: <div style="display: flex; flex-wrap: wrap; padding: 5px;"> <div style="width: 50%;"> ☒ Hand Calculations </div> <div style="width: 50%;"> ☐ Parametric Sensitivity Analysis </div> <div style="width: 50%;"> ☐ In Use Test </div> <div style="width: 50%;"> ☐ Other: None </div> <div style="width: 50%;"> ☐ Comparison with Analytic Solutions </div> <div style="width: 50%;"> ☒ Comparison with Other Verified Programs </div> </div>		
17. Documentation: <div style="margin-top: 10px;"> User's Reference: None </div> <div style="margin-top: 20px;"> Theoretical Reference: None </div> <div style="margin-top: 20px;"> Verification/Sample Problems Reference: Computer/Node: Sandia National Laboratories "IKE" Directory: NFS3:[JSRATH.INEL.GNRDP] Input Files: Command Files: </div> <div style="margin-top: 20px;"> Other Reference: None </div>		
18. Primary Reference: None		
19. Approved by: <div style="display: flex; justify-content: space-between; align-items: flex-end; margin-top: 20px;"> <div style="text-align: center;"> _____ PA QA Coordinator </div> <div style="text-align: center;"> 12-27-93 _____ Date </div> </div>		

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: GRAPH	2a. Version ID: X-1.01ZO	2b. Version Date: 10/04/93	3. Host Computer: ALPHA/VMS
4. Software Title: GRAPH: A general curve plotting utility			
5. Software Description: Curve plotting			
6. Limitations: Reads data from columnar ASCII file			
7. Keyword/Phrases: Graph, plot, INEL, CAMCON, WIPP			
8. Origin/Source: Sandia National Laboratories Performance Assessment Division 6342 Albuquerque, New Mexico			
9. Authors: Colin Selek			
10. Parent Program: none			
11. Computer Languages: FORTRAN 77			
12. Status: X			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	<div style="display: flex; align-items: center;"> <div> Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2 </div> </div>
Date: 12/27/93		
Software Name: GRAPH	Version ID: X-1.01ZO	Version Date: 10/04/93
13. Classification: Government owned, Limited Distribution.		
14. Software Sponsor: D. K. Rudeen	15. Software Consultant:	
16. Verification: <div style="display: flex; flex-wrap: wrap; margin-top: 10px;"> <div style="width: 50%;"> ☒ Hand Calculations </div> <div style="width: 50%;"> ☐ Parametric Sensitivity Analysis </div> <div style="width: 50%;"> ☐ In Use Test </div> <div style="width: 50%;"> ☐ Other: None </div> <div style="width: 50%;"> ☐ Comparison with Analytic Solutions </div> <div style="width: 50%;"> ☒ Comparison with Other Verified Programs </div> </div>		
17. Documentation: <div style="margin-top: 10px;"> User's Reference: None </div> <div style="margin-top: 20px;"> Theoretical Reference: None </div> <div style="margin-top: 20px;"> Verification/Sample Problems Reference: None </div> <div style="margin-top: 20px;"> Other Reference: None </div>		
18. Primary Reference: None		
19. Approved by: <div style="display: flex; justify-content: space-between; margin-top: 20px;"> <div style="text-align: center;"> _____ PA QA Coordinator </div> <div style="text-align: center;"> <u>12-27-93</u> _____ Date </div> </div>		

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: GRFLOW	2a. Version ID: X-1.00ZO	2b. Version Date: 08/19/93	3. Host Computer: ALPHA/VMS
4. Software Title: GRFLOW: A code to estimate fluid flow rates in a granite repository			
5. Software Description: GRFLOW calculates liquid and gas flow rates through an arbitrary number of connections to overlying and underlying liquid reservoirs. GRFLOW is internally coupled to the canister simulation model CONCNF			
6. Limitations: Fluid flow is allowed only through the repository and through connections to overlying and underlying liquid reservoirs; gas and liquid pressures in the repository are assumed to be equal, and the two phases are assumed to be separated; fluid pressure is assumed to be uniform throughout the repository; the ideal gas law is assumed; flow through the repository/reservoir connections is assumed to obey Darcy's law.			
7. Keyword/Phrases: INEL, Performance Assessment, Flow Model			
8. Origin/Source: Sandia National Laboratories Performance Assessment Division 6342 Albuquerque, NM 87185 - 5800			
9. Authors: Walt Beyeler			
10. Parent Program: None			
11. Computer Languages: FORTRAN 77			
12. Status: X			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2
Date: 12/27/93		
Software Name: GRFLOW	Version ID: X-1.00ZO	Version Date: 08/19/93
13. Classification: Government Owned, Limited Distribution.		
14. Software Sponsor: Walt Beyeler	15. Software Consultant:	
16. Verification: <div style="display: flex; flex-wrap: wrap; margin-top: 10px;"> <div style="width: 50%;">☐ Hand Calculations</div> <div style="width: 50%;">☐ Parametric Sensitivity Analysis</div> <div style="width: 50%;">☐ In Use Test</div> <div style="width: 50%;">☐ Other: None</div> <div style="width: 50%;">☐ Comparison with Analytic Solutions</div> <div style="width: 50%;">☒ Comparison with Other Verified Programs</div> </div>		
17. Documentation: <div style="margin-top: 10px;"> User's Reference: None </div> <div style="margin-top: 20px;"> Theoretical Reference: None </div> <div style="margin-top: 20px;"> Verification/Sample Problems Reference: None </div> <div style="margin-top: 20px;"> Other Reference: None </div>		
18. Primary Reference: None		
19. Approved by: <div style="display: flex; justify-content: space-between; margin-top: 20px;"> <div style="width: 45%;"> PA QA Coordinator </div> <div style="width: 45%; text-align: right;"> <u>12-27-93</u> Date </div> </div>		

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: GROPECDB	2a. Version ID: C-2.07ZO	2b. Version Date: 11/28/93	3. Host Computer: ALPHA/VMS
4. Software Title: GROPECDB: A CAMDAT Database Examination Program			
5. Software Description: GROPECDB allows the user to interactively examine a CAMDAT database.			
6. Limitations: None			
7. Keyword/Phrases: WIPP, Performance Assessment, CAMDAT			
8. Origin/Source: Sandia National Laboratories, Applied Mechanics Division 1523, Albuquerque, NM 87185-5800			
9. Authors: Amy P. Gilkey			
10. Parent Program: GROPE			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2								
Date: 12/27/93										
Software Name: GROPECDB	Version ID: C-2.07ZO	Version Date: 11/28/93								
13. Classification: Government owned, Limited distribution.										
14. Software Sponsor: Amy P. Gilkey	15. Software Consultant: Amy P. Gilkey									
16. Verification: <table style="width: 100%; margin-top: 10px;"> <tr> <td>☐ Hand Calculations</td> <td>☐ Parametric Sensitivity Analysis</td> </tr> <tr> <td>☐ In Use Test</td> <td>☐ Other: None</td> </tr> <tr> <td>☐ Comparison with Analytic Solutions</td> <td></td> </tr> <tr> <td>☒ Comparison with Other Verified Programs</td> <td></td> </tr> </table>			☐ Hand Calculations	☐ Parametric Sensitivity Analysis	☐ In Use Test	☐ Other: None	☐ Comparison with Analytic Solutions		☒ Comparison with Other Verified Programs	
☐ Hand Calculations	☐ Parametric Sensitivity Analysis									
☐ In Use Test	☐ Other: None									
☐ Comparison with Analytic Solutions										
☒ Comparison with Other Verified Programs										
17. Documentation: <p style="margin-top: 10px;">User's Reference: Rechard, R.P., et al., 1991. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND90-1983, Version 3.0. Albuquerque, NM: Sandia National Laboratories.</p> <p style="margin-top: 20px;">Theoretical Reference: None</p> <p style="margin-top: 20px;">Verification/Sample Problems Reference: See code notebook (on file with SWCF).</p> <p style="margin-top: 20px;">Other Reference: Gilkey, A.P., 1988. GROPE - A GENESIS/EXODUS Database Examination Program. RS1523/88/02. Albuquerque, NM: Sandia National Laboratories.</p>										
18. Primary Reference: Rechard, R.P., et al., 1991. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND90-1983, Version 3.0. Albuquerque, NM: Sandia National Laboratories.										
19. Approved by: <div style="display: flex; justify-content: space-between; margin-top: 20px;"> <div style="width: 45%;"> PA QA Coordinator </div> <div style="width: 45%; text-align: right;"> <u>12-27-93</u> Date </div> </div>										

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: ICSET	2a. Version ID: C-2.08VV	2b. Version Date: 07/15/93	3. Host Computer: VAX/VMS 8810
4. Software Title: ICSET: A code which sets initial conditions in a WIPP Performance Assessment Computational Data Base, CAMDAT, file.			
5. Software Description: Sets CAMDAT analysis array variables: history, global, nodal, and element variable values, at the first time step (NSTEP=1). Both analysis array names and values are obtained from a user input file. In addition, any nodal or element variable (existing or new), can be linearly interpolated by specifying interpolation tables in the ICSET input text file.			
6. Limitations: ICSET is an initial conditions (I.C.) translator for use with: regular rectilinear 1-,2-, or 3-D finite difference/finite element grids/meshes; or 1-,2-, or 3-D NETWORK-type grids compatible with the WIPP CAMCON Computational Data base, CAMDAT. The program sets History, Global, Nodal, and Element analysis array variables at the first time step (hence, I.C.). These values can also be linearly interpolated by specifying interpolation tables in the input text file.			
7. Keyword/Phrases: Element,Global,History,Initial Condition,Finite Difference,Finite Element,Network,Nodal,Rectilinear, Performance Assessment,Waste Isolation Pil			
8. Origin/Source: Sandia National Laboratories, Performance Assessment Department 6342, Albuquerque, NM 87185-5800			
9. Authors: J. S. Rath			
10. Parent Program: None			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2
Date: 12/27/93		
Software Name: ICSET	Version ID: C-2.08VV	Version Date: 07/15/93
13. Classification: Government owned, Limited distribution.		
14. Software Sponsor: J. S. Rath	15. Software Consultant: J. S. Rath	
16. Verification: <div style="display: flex; flex-wrap: wrap; margin-top: 10px;"> <div style="width: 50%;"> ☒ Hand Calculations </div> <div style="width: 50%;"> ☐ Parametric Sensitivity Analysis </div> <div style="width: 50%;"> ☐ In Use Test </div> <div style="width: 50%;"> ☐ Other: None </div> <div style="width: 50%;"> ☐ Comparison with Analytic Solutions </div> <div style="width: 50%;"> ☒ Comparison with Other Verified Programs </div> </div>		
17. Documentation: <div style="margin-top: 20px;"> User's Reference: Rechard, R.P., ed. 1992. User's Reference Manual for CAMCON: Compliance Assessment Methodology Controller, Version 3.0. SAND90-1983. Albuquerque, NM: Sandia National Laboratories. </div> <div style="margin-top: 20px;"> Theoretical Reference: None </div> <div style="margin-top: 20px;"> Verification/Sample Problems Reference: Computer/Node: Sandia VAX/VMS 8810 / Jimmy Directory: CAMCON\$ROOT:[CAMCON.ICSET.TEST] Input Files: ICSET_TEST_1.INP, ICSET_TEST_2.INP, ICSET_TEST_3.INP Command Files: QA_ICSET.COM </div> <div style="margin-top: 20px;"> Other Reference: None </div>		
18. Primary Reference: Rechard, R.P., ed. 1992. User's Reference Manual for CAMCON: Compliance Assessment Methodology Controller, Version 3.0. SAND90-1983. Albuquerque, NM: Sandia National Laboratories.		
19. Approved by: <div style="display: flex; justify-content: space-between; margin-top: 20px;"> <div style="text-align: center;"> PA QA Coordinator </div> <div style="text-align: center;"> <u>12-27-93</u> Date </div> </div>		

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: INVENTOR	2a. Version ID: X-1.02ZO	2b. Version Date: 11/18/93	3. Host Computer: ALPHA/VMS
4. Software Title: INVENTOR: Program for computing the inventory (in Curies) of each radioisotope for each waste disposal group of the INEL/WINCO HLW/SNF Halite (			
5. Software Description: Compiles useful data regarding radioisotope inventory grouped by waste disposal groups (WDG) and by waste parcels (WP).			
6. Limitations:			
7. Keyword/Phrases:			
8. Origin/Source:			
9. Authors: J. S. Rath			
10. Parent Program:			
11. Computer Languages: FORTRAN 77			
12. Status: X			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	<div style="border: 1px solid black; padding: 2px; display: inline-block;"> </div> Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2								
Date: 12/27/93										
Software Name: INVENTOR	Version ID: X-1.02ZO	Version Date: 11/18/93								
13. Classification: Government owned, Limited distribution.										
14. Software Sponsor: J. S. Rath	15. Software Consultant: J. S. Rath									
16. Verification: <table style="width: 100%; margin-top: 10px;"> <tr> <td>☒ Hand Calculations</td> <td>☐ Parametric Sensitivity Analysis</td> </tr> <tr> <td>☐ In Use Test</td> <td>☐ Other: None</td> </tr> <tr> <td colspan="2">☐ Comparison with Analytic Solutions</td> </tr> <tr> <td colspan="2">☒ Comparison with Other Verified Programs</td> </tr> </table>			☒ Hand Calculations	☐ Parametric Sensitivity Analysis	☐ In Use Test	☐ Other: None	☐ Comparison with Analytic Solutions		☒ Comparison with Other Verified Programs	
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☐ In Use Test	☐ Other: None									
☐ Comparison with Analytic Solutions										
☒ Comparison with Other Verified Programs										
17. Documentation: <div style="margin-top: 10px;"> User's Reference: None </div> <div style="margin-top: 20px;"> Theoretical Reference: None </div> <div style="margin-top: 20px;"> Verification/Sample Problems Reference: Computer/Node: Sandia National Laboratories "IKE" Directory: NFS3:[JSRATH.INEL.INVENTOR] Input Files: Command Files: </div> <div style="margin-top: 20px;"> Other Reference: None </div>										
18. Primary Reference: None										
19. Approved by: <div style="display: flex; justify-content: space-between; align-items: flex-end; margin-top: 20px;"> <div style="text-align: center;"> _____ PA QA Coordinator </div> <div style="text-align: center;"> <u>12-27-93</u> _____ Date </div> </div>										

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: LHS	2a. Version ID: C-2.31VV	2b. Version Date: 08/09/93	3. Host Computer: VAX/VMS 8810
4. Software Title: LHS: Latin Hypercube Sampling, an efficient, constrained Monte Carlo sampling code			
5. Software Description: The purpose of the LHS program is to sample distributions of input parameters using either normal Monte Carlo sampling or efficient Latin Hypercube Sampling. LHS permits correlations (restricted pairings) between parameters. Latin Hypercube sampling reduces the minimum number of sample vectors [nv] required to about $4/3 * n_a$ where n_a is number of varying parameters (Iman and Shortencarier, 1984).			
6. Limitations: 100 variables, 10000 output vectors.			
7. Keyword/Phrases: Latin Hypercube Sampling			
8. Origin/Source: Sandia National Laboratories			
9. Authors: Original version: R. Iman, Davenport, and Zeigler, 1980. Current version: R. Iman and M. Shortencarier, 1983.			
10. Parent Program: None.			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2
Date: 12/27/93		
Software Name: LHS	Version ID: C-2.31VV	Version Date: 08/09/93
13. Classification: Government owned, Limited distribution.		
14. Software Sponsor: Lanny N. Smith	15. Software Consultant: Harold J. Iuzzolino	
16. Verification: <div style="display: flex; flex-wrap: wrap; margin-top: 10px;"> <div style="width: 50%;">☐ Hand Calculations</div> <div style="width: 50%;">☐ Parametric Sensitivity Analysis</div> <div style="width: 50%;">☐ In Use Test</div> <div style="width: 50%;">☐ Other: None</div> <div style="width: 50%;">☐ Comparison with Analytic Solutions</div> <div style="width: 50%;">☒ Comparison with Other Verified Programs</div> </div>		
17. Documentation: <div style="margin-top: 10px;"> <p>User's Reference: Ronald L. Iman and Michael J. Shortencarier, "A FORTRAN 77 Program and User's Guide for the Generation of Latin Hypercube and Random Samples for Use With Computer Models", NUREG/CR-3624, SAND83-2365, Sandia National Laboratories, Albuquerque, NM., March 1984,</p> <p>Theoretical Reference: None</p> <p>Verification/Sample Problems Reference: See Code Notebook (on file with SWCF)</p> <p>Other Reference: Rechard, R.P., et al., "Programmer's Reference Manual for CAMCON: Compliance Assessment Methodology Controller", SAND90-1984, Sandia National Laboratories, Albuquerque, NM.</p> </div>		
18. Primary Reference: Ronald L. Iman and Michael J. Shortencarier, "A FORTRAN 77 Program and User's Guide for the Generation of Latin Hypercube and Random Samples for Use With Computer Models", NUREG/CR-3624, SAND83-2365, Sandia National Laboratories, Albuquerque, NM., March 1984		
19. Approved by: <div style="display: flex; justify-content: space-between; margin-top: 20px;"> <div style="width: 45%;"> PA QA Coordinator </div> <div style="width: 45%; text-align: center;"> <u>12-27-93</u> Date </div> </div>		

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: LHS2STEP	2a. Version ID: C-1.01VV	2b. Version Date: 07/22/93	3. Host Computer: VAX/VMS 8810
4. Software Title: LHS2STEP: Transfers LHS data to STEPWISE or PCCSRC input			
5. Software Description: LHS2STEP reads LHS sampled output file containing the dependent variables and writes an output file for either STEPWISE or PCCSRC.			
6. Limitations: None			
7. Keyword/Phrases: WIPP, Performance Assessment			
8. Origin/Source: Sandia National Laboratories, Performance Assessment Division 6342, Albuquerque, NM 87185-5800			
9. Authors: R.D. McCurley, Amy P. Gilkey			
10. Parent Program: None			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2								
Date: 12/27/93										
Software Name: LHS2STEP	Version ID: C-1.01VV	Version Date: 07/22/93								
13. Classification: Government owned, Limited distribution.										
14. Software Sponsor: Amy P. Gilkey	15. Software Consultant: Amy P. Gilkey									
16. Verification: <table style="width: 100%; margin-top: 10px;"> <tr> <td>☐ Hand Calculations</td> <td>☐ Parametric Sensitivity Analysis</td> </tr> <tr> <td>☐ In Use Test</td> <td>☐ Other: None</td> </tr> <tr> <td>☐ Comparison with Analytic Solutions</td> <td></td> </tr> <tr> <td>☒ Comparison with Other Verified Programs</td> <td></td> </tr> </table>			☐ Hand Calculations	☐ Parametric Sensitivity Analysis	☐ In Use Test	☐ Other: None	☐ Comparison with Analytic Solutions		☒ Comparison with Other Verified Programs	
☐ Hand Calculations	☐ Parametric Sensitivity Analysis									
☐ In Use Test	☐ Other: None									
☐ Comparison with Analytic Solutions										
☒ Comparison with Other Verified Programs										
17. Documentation: <p style="margin-top: 20px;">User's Reference: Rechard, R.P., et al., 1991. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND90-1983, Version 3.0. Albuquerque, NM: Sandia National Laboratories.</p> <p style="margin-top: 40px;">Theoretical Reference: None</p> <p style="margin-top: 40px;">Verification/Sample Problems Reference: See code notebook (on file with SWCF).</p> <p style="margin-top: 40px;">Other Reference: None</p>										
18. Primary Reference: Rechard, R.P., et al., 1991. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND90-1983, Version 3.0. Albuquerque, NM: Sandia National Laboratories.										
19. Approved by: <div style="display: flex; justify-content: space-between; margin-top: 10px;"> <div style="width: 45%;"> PA QA Coordinator </div> <div style="width: 45%; text-align: right;"> <u>12-27-93</u> Date </div> </div>										

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: MATSET	2a. Version ID: C-8.05VV	2b. Version Date: 07/08/93	3. Host Computer: VAX/VMS 8810
4. Software Title: MATSET: A code to insert parameter values into the computational data base (CAMDAT).			
5. Software Description: MATSET sets material names to specified regions (e.g., defined by GENMESH or GENNET), sets material property values, and sets attribute values into the WIPP Performance Assessment Computational data base, CAMDAT. Both property and attribute values are obtained from either the property Secondary Data Base (usually median values read from PROPERTY.SDB), or the user supplied MATSET input text file.			
6. Limitations: MATSET sets material names, material properties, and attributes into the WIPP computational data base, CAMDAT. Both property and attribute data can be retrieved from a secondary data base, or a user supplied input text file.			
7. Keyword/Phrases: Attribute,,Material,Names, Performance Assessment, Property,Secondary,Values,Waste Isolation Pilot Plant			
8. Origin/Source: Sandia National Laboratories, Performance Assessment Department 6342, Albuquerque, NM 87185-5800			
9. Authors: J. S. Rath			
10. Parent Program: None			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2								
Date: 12/27/93										
Software Name: MATSET	Version ID: C-8.05VV	Version Date: 07/08/93								
13. Classification: Government owned, Limited distribution.										
14. Software Sponsor: J. S. Rath	15. Software Consultant: J. S. Rath									
16. Verification: <table style="width: 100%; margin-top: 10px;"> <tr> <td>☐ Hand Calculations</td> <td>☐ Parametric Sensitivity Analysis</td> </tr> <tr> <td>☐ In Use Test</td> <td>☐ Other: None</td> </tr> <tr> <td>☐ Comparison with Analytic Solutions</td> <td></td> </tr> <tr> <td>☒ Comparison with Other Verified Programs</td> <td></td> </tr> </table>			☐ Hand Calculations	☐ Parametric Sensitivity Analysis	☐ In Use Test	☐ Other: None	☐ Comparison with Analytic Solutions		☒ Comparison with Other Verified Programs	
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☒ Comparison with Other Verified Programs										
17. Documentation: <p style="margin-top: 20px;">User's Reference: Rechard, R.P., ed. 1992. User's Reference Manual for CAMCON: Compliance Assessment Methodology Controller, Version 3.0. SAND90-1983. Albuquerque, NM: Sandia National Laboratories.</p> <p style="margin-top: 20px;">Theoretical Reference: None</p> <p style="margin-top: 20px;">Verification/Sample Problems Reference: Computer/Node: Sandia VAX/VMS 8810 / Jimmy Directory: CAMCON\$ROOT:[CAMCON.MATSET.TEST] Input Files: MATSET_TEST_1.INP, MATSET_TEST_2.INP, MATSET_TEST_3.INP, MATSET_TEST_4.INP, MATSET_TEST_5.INP Command Files: QA_MATSET.COM</p> <p style="margin-top: 20px;">Other Reference: None</p>										
18. Primary Reference: Rechard, R.P., ed. 1992. User's Reference Manual for CAMCON: Compliance Assessment Methodology Controller, Version 3.0. SAND90-1983. Albuquerque, NM: Sandia National Laboratories.										
19. Approved by: <div style="display: flex; justify-content: space-between; align-items: flex-end; margin-top: 20px;"> <div style="width: 45%;"> PA QA Coordinator </div> <div style="width: 45%; text-align: center;"> <u>12-27-93</u> Date </div> </div>										

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Date: 12/27/93			
1. Software Name: NUCPLOT	2a. Version ID: C-1.12ZO	2b. Version Date: 10/28/93	3. Host Computer: ALPHA/VMS
4. Software Title: NUCPLOT: A code to plot "Tukey box plots" of nuclide release data as calculated by CCDFCALC			
5. Software Description: NUCPLOT interactively plots Tukey box plots of release data read from the CCDFCALC transfer file. It will plot either total release, cuttings only, ground water release only or percentages of total of the same.			
6. Limitations: NUCPLOT is limited to 20 scenarios, 100 vectors each.			
7. Keyword/Phrases: WIPP, Performance Assessment, Plot Program			
8. Origin/Source: Sandia National Laboratories, Waste Management Systems Division 6416, Albuquerque, NM 87185-5800			
9. Authors: Amy P. Gilkey			
10. Parent Program: None			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2								
Date: 12/27/93										
Software Name: NUCLOT	Version ID: C-1.12ZO	Version Date: 10/28/93								
13. Classification: Government owned, Limited distribution.										
14. Software Sponsor: Amy P. Gilkey		15. Software Consultant: Amy P. Gilkey								
16. Verification: <table style="width: 100%; margin-top: 10px;"> <tr> <td>☐ Hand Calculations</td> <td>☐ Parametric Sensitivity Analysis</td> </tr> <tr> <td>☐ In Use Test</td> <td>☐ Other: None</td> </tr> <tr> <td>☐ Comparison with Analytic Solutions</td> <td></td> </tr> <tr> <td>☒ Comparison with Other Verified Programs</td> <td></td> </tr> </table>			☐ Hand Calculations	☐ Parametric Sensitivity Analysis	☐ In Use Test	☐ Other: None	☐ Comparison with Analytic Solutions		☒ Comparison with Other Verified Programs	
☐ Hand Calculations	☐ Parametric Sensitivity Analysis									
☐ In Use Test	☐ Other: None									
☐ Comparison with Analytic Solutions										
☒ Comparison with Other Verified Programs										
17. Documentation: <p style="margin-top: 20px;">User's Reference: Rechard, R.P., et al., 1991. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND90-1983, Version 3.0. Albuquerque, NM: Sandia National Laboratories.</p> <p style="margin-top: 20px;">Theoretical Reference: None</p> <p style="margin-top: 20px;">Verification/Sample Problems Reference: See code notebook (on file with SWCF).</p> <p style="margin-top: 20px;">Other Reference: None</p>										
18. Primary Reference: Rechard, R.P., et al., 1991. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND90-1983, Version 3.0. Albuquerque, NM: Sandia National Laboratories.										
19. Approved by: <div style="display: flex; justify-content: space-between; align-items: flex-end; margin-top: 20px;"> <div style="text-align: center;"> _____ PA QA Coordinator </div> <div style="text-align: center;"> <u>12-27-93</u> _____ Date </div> </div>										

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: PCCSRC	2a. Version ID: C-2.19VV	2b. Version Date: 09/27/93	3. Host Computer: VAX/VMS 8810
4. Software Title: PCCSRC: A statistical module to calculate partial correlation and standardized regression coefficients			
5. Software Description: The statistical module, PCCSRC, evaluates variable importance by reporting the partial correlation coefficients (PCC) and standardized regression coefficients (SRC) on either the rank or raw data. Provided useful regression models can be developed, the absolute values of the standardized regression coefficients (or mathematically related partial correlation coefficients) can be used to rank variable importance.			
6. Limitations: None			
7. Keyword/Phrases: WIPP, Performance Assessment, Plot Program			
8. Origin/Source: Sandia National Laboratories, Safety and Reliability Analysis Division 6415, Albuquerque, NM 87185-5800			
9. Authors: R.L. Iman, M.J. Shortencarier, J.D. Johnson, Amy P. Gilkey			
10. Parent Program: None			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2								
Date: 12/27/93										
Software Name: PCCSRC	Version ID: C-2.19VV	Version Date: 09/27/93								
13. Classification: Government owned, Limited distribution.										
14. Software Sponsor: Amy P. Gilkey	15. Software Consultant: Amy P. Gilkey									
16. Verification: <table style="width: 100%; margin-top: 10px;"> <tr> <td>☐ Hand Calculations</td> <td>☐ Parametric Sensitivity Analysis</td> </tr> <tr> <td>☐ In Use Test</td> <td>☐ Other: None</td> </tr> <tr> <td>☐ Comparison with Analytic Solutions</td> <td></td> </tr> <tr> <td>☒ Comparison with Other Verified Programs</td> <td></td> </tr> </table>			☐ Hand Calculations	☐ Parametric Sensitivity Analysis	☐ In Use Test	☐ Other: None	☐ Comparison with Analytic Solutions		☒ Comparison with Other Verified Programs	
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☐ In Use Test	☐ Other: None									
☐ Comparison with Analytic Solutions										
☒ Comparison with Other Verified Programs										
17. Documentation: <p style="margin-top: 20px;">User's Reference: Rechard, R.P., et al., 1991. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND90-1983, Version 3.0. Albuquerque, NM: Sandia National Laboratories.</p> <p style="margin-top: 20px;">Theoretical Reference: None</p> <p style="margin-top: 20px;">Verification/Sample Problems Reference: See code notebook (on file with SWCF).</p> <p style="margin-top: 20px;">Other Reference: R.L. Iman, M.J. Shortencarier, J.D. Johnson, 1985. A FORTRAN 77 Program and User's Guide for the Calculation of Partial Correlation and Standardized Regression Coefficients. SAND85-0044, NUREG/CR-4122. Albuquerque, NM: Sandia National Laboratories.</p>										
18. Primary Reference: Rechard, R.P., et al., 1991. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND90-1983, Version 3.0. Albuquerque, NM: Sandia National Laboratories.										
19. Approved by: <div style="display: flex; justify-content: space-between; align-items: flex-end; margin-top: 20px;"> <div style="text-align: center;"> _____ PA QA Coordinator </div> <div style="text-align: center;"> 12-27-93 _____ Date </div> </div>										

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: PLOTSDB	2a. Version ID: C-4.01VV	2b. Version Date: 02/28/92	3. Host Computer: VAX/VMS 8810
4. Software Title: PLOTSDB: PLOT distributions from the Secondary DataBase.			
5. Software Description: Display selected data from the Secondary (material property) database.			
6. Limitations:			
7. Keyword/Phrases: distribution plot, probability density, cumulative probability			
8. Origin/Source: Program DISTRPLT was adapted to plot data on the CAMCON secondary database. The resulting program was named PLOTSDB.			
9. Authors: Harold J. Iuzzolino and Bill D. Harrison			
10. Parent Program: DISTRPLT, a distribution-plotting program written by			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2								
Date: 12/27/93										
Software Name: PLOTSDB	Version ID: C-4.01VV	Version Date: 02/28/92								
13. Classification: Government owned, Limited distribution.										
14. Software Sponsor: Harold J. Iuzzolino		15. Software Consultant:								
16. Verification: <table style="width: 100%; margin-top: 10px;"> <tr> <td>☐ Hand Calculations</td> <td>☐ Parametric Sensitivity Analysis</td> </tr> <tr> <td>☐ In Use Test</td> <td>☐ Other: None</td> </tr> <tr> <td>☐ Comparison with Analytic Solutions</td> <td></td> </tr> <tr> <td>☐ Comparison with Other Verified Programs</td> <td></td> </tr> </table>			☐ Hand Calculations	☐ Parametric Sensitivity Analysis	☐ In Use Test	☐ Other: None	☐ Comparison with Analytic Solutions		☐ Comparison with Other Verified Programs	
☐ Hand Calculations	☐ Parametric Sensitivity Analysis									
☐ In Use Test	☐ Other: None									
☐ Comparison with Analytic Solutions										
☐ Comparison with Other Verified Programs										
17. Documentation: <p style="margin-top: 10px;">User's Reference: Rechard, R.P., et al., "Programmer's Reference Manual for CAMCON: Compliance Assessment Methodology Controller", SAND90-1984, Sandia National Laboratories, Albuquerque, NM.</p> <p style="margin-top: 20px;">Theoretical Reference: None.</p> <p style="margin-top: 20px;">Verification/Sample Problems Reference: None.</p> <p style="margin-top: 20px;">Other Reference: None.</p>										
18. Primary Reference: Rechard, R.P., et al., "Programmer's Reference Manual for CAMCON: Compliance Assessment Methodology Controller", SAND90-1984, Sandia National Laboratories, Albuquerque, NM.										
19. Approved by: <div style="display: flex; justify-content: space-between; align-items: flex-end; margin-top: 20px;"> <div style="text-align: center;"> PA QA Coordinator </div> <div style="text-align: center;"> <u>12-27-93</u> Date </div> </div>										

INELIDAHO NATIONAL
ENGINEERING
LABORATORY**TITLE: SOFTWARE ABSTRACT
FOR INEL PERFORMANCE
ASSESSMENT SOFTWARE**

Effective Date: 3/22/93

Sandia National Laboratories
Procedure PAP02
Revision: 0
Form No. PA001
Page 1 of 2

Date: 12/27/93

1. Software Name:
POSTBRAG_T2a. Version ID:
C-2.00ZO2b. Version Date:
08/05/933. Host Computer:
ALPHA/VMS4. Software Title:
POSTBRAG_T: A post-processor that formats BRAGFLO_T output for CAMDAT.

5. Software Description:

POSTBRAG_T transfers BRAGFLO_T output data into a CAMDAT. The BRAGFLO_T output includes global mass balances (global variables) and any of 63 user-specified element variable distributions, including fluid pressures, phase saturations, Darcy velocities of each fluid phase in each (x,y,z) direction, interblock fluid flow rates of each fluid phase in each direction, reactant concentrations, physical properties (e.g., porosity and viscosity of each phase), and phase mass balances.

6. Limitations:

7. Keyword/Phrases:

8. Origin/Source:

9. Authors:

J. S. Rath & J. D. Schreiber

10. Parent Program:

11. Computer Languages:

FORTRAN 77

12. Status:

C

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2
Date: 12/27/93		
Software Name: POSTBRAG_T	Version ID: C-2.00ZO	Version Date: 08/05/93
13. Classification: Government owned, Limited distribution.		
14. Software Sponsor: M. E. Lord	15. Software Consultant:	
16. Verification: <div style="display: flex; flex-wrap: wrap; margin-top: 10px;"> <div style="width: 50%;">☐ Hand Calculations</div> <div style="width: 50%;">☐ Parametric Sensitivity Analysis</div> <div style="width: 50%;">☐ In Use Test</div> <div style="width: 50%;">☐ Other: None</div> <div style="width: 50%;">☐ Comparison with Analytic Solutions</div> <div style="width: 50%;">☐ Comparison with Other Verified Programs</div> </div>		
17. Documentation: <div style="margin-top: 10px;"> <p>User's Reference: Rechard, R.P., ed., 1992. "User's Reference Manual for CAMCON: Compliance Assessment Methodology Controller Version 3.0," SAND90-1983 (UC-721), Sandia National Laboratories, Albuquerque, NM.</p> <p>Theoretical Reference: None Verification/Sample Problems Reference: None</p> <p>Verification/Sample Problems Reference: None</p> <p>Other Reference: None Primary Reference: Rechard, R.P., ed., 1992. "User's Reference Manual for CAMCON: Compliance Assessment Methodology Controller Version 3.0," SAND90-1983 (UC-721), Sandia National Laboratories, Albuquerque, NM.</p> </div>		
18. Primary Reference: Rechard, R.P., ed., 1992. "User's Reference Manual for CAMCON: Compliance Assessment Methodology Controller Version 3.0," SAND90-1983 (UC-721), Sandia National Laboratories, Albuquerque, NM.		
19. Approved by: <div style="display: flex; justify-content: space-between; margin-top: 20px;"> <div style="width: 45%;"> PA QA Coordinator </div> <div style="width: 45%; text-align: right;"> <u>12-27-93</u> Date </div> </div>		

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: POSTLHS	2a. Version ID: C-4.03VV	2b. Version Date: 06/15/92	3. Host Computer: VAX/VMS 8810
4. Software Title: POSTLHS: A translator of output from LHS Code to CAMDAT			
5. Software Description: The POSTLHS translator replicates PRECAMDAT N_v times where N_v is number of sample vectors generated by LHS and inserts one distinct sample vector of the varied parameters from the output of LHS into each replication of CAMDAT. Identical parameters previously inserted into PRECAMDAT by MATSET are overwritten by POSTLHS.			
6. Limitations: 100 sampled variables, 1000 vectors.			
7. Keyword/Phrases: Latin Hypercube Sampling, Performance Assessment, WIPP			
8. Origin/Source: Sandia National Laboratories Division 6342 Albuquerque, NM 87185-5800			
9. Authors: J. S. Rath and R. D. McCurley			
10. Parent Program: None			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	<div style="display: flex; align-items: center;"> <div> Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2 </div> </div>								
Date: 12/27/93										
Software Name: POSTLHS	Version ID: C-4.03VV	Version Date: 06/15/92								
13. Classification: Government owned, Limited distribution.										
14. Software Sponsor: Lanny N. Smith	15. Software Consultant: Lanny N. Smith									
16. Verification: <table style="width: 100%; margin-top: 10px;"> <tr> <td>☐ Hand Calculations</td> <td>☐ Parametric Sensitivity Analysis</td> </tr> <tr> <td>☐ In Use Test</td> <td>☐ Other: None</td> </tr> <tr> <td>☐ Comparison with Analytic Solutions</td> <td></td> </tr> <tr> <td>☒ Comparison with Other Verified Programs</td> <td></td> </tr> </table>			☐ Hand Calculations	☐ Parametric Sensitivity Analysis	☐ In Use Test	☐ Other: None	☐ Comparison with Analytic Solutions		☒ Comparison with Other Verified Programs	
☐ Hand Calculations	☐ Parametric Sensitivity Analysis									
☐ In Use Test	☐ Other: None									
☐ Comparison with Analytic Solutions										
☒ Comparison with Other Verified Programs										
17. Documentation: <p style="margin-top: 10px;">User's Reference: Rechard, R.P., ed. 1992. User's Reference Manual for CAMCON: Compliance Assessment Methodology Controller. Version 3.0. SAND90-1983. Albuquerque, NM: Sandia National Laboratories</p> <p style="margin-top: 20px;">Theoretical Reference: None</p> <p style="margin-top: 20px;">Verification/Sample Problems Reference: See Code Notebook (on file with SWCF)</p> <p style="margin-top: 20px;">Other Reference: None</p>										
18. Primary Reference: Rechard, R.P., ed. 1992. User's Reference Manual for CAMCON: Compliance Assessment Methodology Controller. Version 3.0. SAND90-1983. Albuquerque, NM: Sandia National Laboratories										
19. Approved by: <div style="display: flex; justify-content: space-between; margin-top: 20px;"> <div style="width: 45%;"> PA QA Coordinator </div> <div style="width: 45%; text-align: right;"> <u>12-27-93</u> Date </div> </div>										

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: POSTSTAFF	2a. Version ID: C-3.00ZO	2b. Version Date: 09/20/93	3. Host Computer: ALPHA/VMS
4. Software Title: POSTSTAFF: A post-processor that formats STAFF2D output for CAMDAT			
5. Software Description: POSTSTAFF places total head and interstitial velocity components or solute concentrations (nodal variable) from STAFF2D output into CAMDAT. This version also outputs saturation, porosity, fluid density and INEL source term data.			
6. Limitations: Rectangular grid with a rectangular i,j index space parallel to the global x, y coordinate system. Assumes grid and element and connectivity data exactly match an input CAMDAT database. Currently verified for transport mode only.			
7. Keyword/Phrases: WIPP, Performance Assessment, Postprocessor, STAFF2D, Ground water flow, Radionuclide transport			
8. Origin/Source: New Mexico Engineering Research Institute, Applied Mechanics Division, University of New Mexico, (under contract to Sandia National Laboratory)			
9. Authors: David K. Rudeen			
10. Parent Program: STAFF2D			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2
Date: 12/27/93		
Software Name: POSTSTAFF	Version ID: C-3.00ZO	Version Date: 09/20/93
13. Classification: Govt owned, Limited Distribution.		
14. Software Sponsor: David K. Rudeen	15. Software Consultant:	
16. Verification: <div style="display: flex; flex-wrap: wrap; margin-top: 10px;"> <div style="width: 50%;"> ☐ Hand Calculations </div> <div style="width: 50%;"> ☐ Parametric Sensitivity Analysis </div> <div style="width: 50%;"> ☐ In Use Test </div> <div style="width: 50%;"> ☐ Other: None </div> <div style="width: 50%;"> ☐ Comparison with Analytic Solutions </div> <div style="width: 50%;"> ☒ Comparison with Other Verified Programs </div> </div>		
17. Documentation: <div style="margin-top: 10px;"> User's Reference: Rechard, R.P., et al, (in preparation) User's Manual for CAMCON: Compliance Assessment Methodology Controller, Sandia National Laboratories, Albuquerque, NM. </div> <div style="margin-top: 20px;"> Theoretical Reference: None </div> <div style="margin-top: 20px;"> Verification/Sample Problems Reference: None </div> <div style="margin-top: 20px;"> Other Reference: None </div>		
18. Primary Reference: Huyakorn, et al, 1984. STAFF2D A Two-Dimensional Finite Element Code For Simulating Fluid Flow and Transport of Radionuclides in Fractured Porous Media with Water Table Boundary Conditions, HydroGeoLogic, Inc., Herndon, VA.		
19. Approved by: <div style="display: flex; justify-content: space-between; margin-top: 20px;"> <div style="width: 45%;"> PA QA Coordinator </div> <div style="width: 45%; text-align: right;"> 12-27-93 Date </div> </div>		

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: PREBRAG_T	2a. Version ID: C-1.04VV	2b. Version Date: 08/25/93	3. Host Computer: VAX/VMS 8810
4. Software Title: PREBRAG_T: A pre-processor that formats CAMDAT data for BRAGFLO_T			
5. Software Description: PREBRAG_T creates an input file for the BRAGFLO_T code by translating data from CAMDAT			
6. Limitations: Rectangular grid with a regular i,j,k index space parallel to the global x,y,z coordinate system. Certain data is presumed to exist in the CAMDAT. PREBRAG_T does not give the user access to a complete set of BRAGFLO_T capabilities.			
7. Keyword/Phrases: BRAGFLO_T, Performance Assessment, Pre-Processor, WIPP			
8. Origin/Source: Sandia National Laboratories, Performance Assessment Department 6342, Albuquerque, NM 87185-5800			
9. Authors: J. S. Rath			
10. Parent Program: BRAGFLO_T			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2								
Date: 12/27/93										
Software Name: PREBRAG_T	Version ID: C-1.04VV	Version Date: 08/25/93								
13. Classification: Government owned, Limited distribution.										
14. Software Sponsor: M. E. Lord	15. Software Consultant: J. S. Rath									
16. Verification: <table style="width: 100%; margin-top: 10px;"> <tr> <td>☐ Hand Calculations</td> <td>☐ Parametric Sensitivity Analysis</td> </tr> <tr> <td>☐ In Use Test</td> <td>☐ Other: None</td> </tr> <tr> <td>☐ Comparison with Analytic Solutions</td> <td></td> </tr> <tr> <td>☒ Comparison with Other Verified Programs</td> <td></td> </tr> </table>			☐ Hand Calculations	☐ Parametric Sensitivity Analysis	☐ In Use Test	☐ Other: None	☐ Comparison with Analytic Solutions		☒ Comparison with Other Verified Programs	
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☐ Comparison with Analytic Solutions										
☒ Comparison with Other Verified Programs										
17. Documentation: <p style="margin-top: 10px;">User's Reference: Rechard, R.P., ed. 1992. User's Reference Manual for CAMCON: Compliance Assessment Methodology Controller, Version 3.0. SAND90-1983. Albuquerque, NM: Sandia National Laboratories.</p> <p style="margin-top: 20px;">Theoretical Reference: None</p> <p style="margin-top: 20px;">Verification/Sample Problems Reference: Computer/Node: Sandia VAX/VMS 8810 / Jimmy Directory: CAMCON\$ROOT:[CAMCON.PREBRAG_T.TEST] Input Files: PREBRAG_T_TEST_1.INP Command Files: QA_PREBRAG_T.COM</p> <p style="margin-top: 20px;">Other Reference: None</p>										
18. Primary Reference: Rechard, R.P., ed. 1992. User's Reference Manual for CAMCON: Compliance Assessment Methodology Controller, Version 3.0. SAND90-1983. Albuquerque, NM: Sandia National Laboratories.										
19. Approved by: <div style="display: flex; justify-content: space-between; align-items: flex-end; margin-top: 20px;"> <div style="text-align: center;"> PA QA Coordinator </div> <div style="text-align: center;"> <u>12-27-93</u> Date </div> </div>										

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: PREGENII	2a. Version ID: C-1.00VV	2b. Version Date: 05/14/92	3. Host Computer: VAX/VMS 8810
4. Software Title: PREGENII: A pre-processor for input to GENII			
5. Software Description: PREGENII creates input for GENII code(SNL PC version) by translating data from CAMDAT and a user supplied PREGENII text input file.			
6. Limitations: Only creates concentration data and X/Q for GENII Certain data is presumed to exist in a CAMDAT file. Does not give user access to a complete set of GENII capabilities nor does it create any other GENII input.			
7. Keyword/Phrases: WIPP, Performance Assessment, Preprocessor, GENII, Safety, Dose Calculations			
8. Origin/Source: Sandia National Laboratories, Performance Assessment Division 6342, Albuquerque NM, 87185-5800			
9. Authors: R. D. McCurley, D. K. Rudeen			
10. Parent Program: GENII			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2
Date: 12/27/93		
Software Name: PREGENII	Version ID: C-1.00VV	Version Date: 05/14/92
13. Classification: Government owned, Limited distribution.		
14. Software Sponsor: Ron D. McCurley	15. Software Consultant:	
16. Verification: <div style="display: flex; flex-wrap: wrap; margin-top: 10px;"> <div style="width: 50%;">☐ Hand Calculations</div> <div style="width: 50%;">☐ Parametric Sensitivity Analysis</div> <div style="width: 50%;">☐ In Use Test</div> <div style="width: 50%;">☐ Other: None</div> <div style="width: 50%;">☐ Comparison with Analytic Solutions</div> <div style="width: 50%;">☒ Comparison with Other Verified Programs</div> </div>		
17. Documentation: <div style="margin-top: 10px;"> User's Reference: Rechard, R.P., et al. 1989. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND88-1496. Albuquerque, NM: Sandia National Laboratories. </div> <div style="margin-top: 20px;"> Theoretical Reference: None </div> <div style="margin-top: 20px;"> Verification/Sample Problems Reference: None </div> <div style="margin-top: 20px;"> Other Reference: None </div>		
18. Primary Reference: Rechard, R.P., et al. 1989. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND88-1496. Albuquerque, NM: Sandia National Laboratories. Leigh, C., 1991. GENII		
19. Approved by: <div style="display: flex; justify-content: space-between; margin-top: 20px;"> <div style="width: 45%;"> PA QA Coordinator </div> <div style="width: 45%; text-align: right;"> <u>12-27-93</u> Date </div> </div>		

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2	
Date: 12/27/93			
1. Software Name: PRELHS	2a. Version ID: C-2.00VV	2b. Version Date: 06/30/93	3. Host Computer: VAX/VMS 8810
4. Software Title: PRELHS: A preprocessor for the LHS code.			
5. Software Description: The PRELHS translator extracts parameter distribution data requested by user from the secondary data base file, PROPERTY.SDB, and sets up the L_atin H_ypercube S_ampler (LHS) program input file.			
6. Limitations: 100 sampled variables.			
7. Keyword/Phrases: Latin Hypercube Sampling			
8. Origin/Source: Sandia National Laboratories			
9. Authors: R. D. McCurley (NMERI) and J. S. Rath (NMERI)			
10. Parent Program: None.			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2								
Date: 12/27/93										
Software Name: PRELHS	Version ID: C-2.00VV	Version Date: 06/30/93								
13. Classification: Government owned, Limited distribution.										
14. Software Sponsor: Lanny N. Smith	15. Software Consultant: Lanny N. Smith									
16. Verification: <table style="width: 100%; margin-top: 10px;"> <tr> <td>☐ Hand Calculations</td> <td>☐ Parametric Sensitivity Analysis</td> </tr> <tr> <td>☐ In Use Test</td> <td>☐ Other: None</td> </tr> <tr> <td>☐ Comparison with Analytic Solutions</td> <td></td> </tr> <tr> <td>☒ Comparison with Other Verified Programs</td> <td></td> </tr> </table>			☐ Hand Calculations	☐ Parametric Sensitivity Analysis	☐ In Use Test	☐ Other: None	☐ Comparison with Analytic Solutions		☒ Comparison with Other Verified Programs	
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☐ In Use Test	☐ Other: None									
☐ Comparison with Analytic Solutions										
☒ Comparison with Other Verified Programs										
17. Documentation: <p style="margin-top: 20px;">User's Reference: Rechard, R.P., et al., "Programmer's Reference Manual for CAMCON: Compliance Assessment Methodology Controller", SAND90-1984, Sandia National Laboratories, Albuquerque, NM.</p> <p style="margin-top: 20px;">Theoretical Reference: None</p> <p style="margin-top: 20px;">Verification/Sample Problems Reference: See Code Notebook (on file with SWCF)</p> <p style="margin-top: 20px;">Other Reference: None</p>										
18. Primary Reference: Rechard, R.P., et al., "Programmer's Reference Manual for CAMCON: Compliance Assessment Methodology Controller", SAND90-1984, Sandia National Laboratories, Albuquerque, NM.										
19. Approved by: <div style="display: flex; justify-content: space-between; align-items: flex-end; margin-top: 20px;"> <div style="width: 45%;"> PA QA Coordinator </div> <div style="width: 45%; text-align: right;"> <u>12-27-93</u> Date </div> </div>										

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: PRESOR	2a. Version ID: X-1.00ZO	2b. Version Date: 09/17/93	3. Host Computer: ALPHA/VMS
4. Software Title: PRESOR: A pre-processor for input data to the source term routines of STAFF2D			
5. Software Description: PRESOR creates an input file for STAFF2D from files created by PREBRAG, BRAGFLO_T, CPVEXT, and INGRESS.			
6. Limitations: All input files must be present.			
7. Keyword/Phrases: Performance assessment, Preprocessor, STAFF2D, WIPP			
8. Origin/Source: Sandia National Laboratories, Performance Assessment, Division 6342, Albuquerque, NM 87185-5800			
9. Authors: Christine Stockman			
10. Parent Program: None			
11. Computer Languages: FORTRAN 77			
12. Status: X			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2								
Date: 12/27/93										
Software Name: PRESOR	Version ID: X-1.00ZO	Version Date: 09/17/93								
13. Classification: Government Owned, Limited distribution.										
14. Software Sponsor: Christine Stockman		15. Software Consultant:								
16. Verification: <table style="width: 100%; margin-top: 10px;"> <tr> <td>☐ Hand Calculations</td> <td>☐ Parametric Sensitivity Analysis</td> </tr> <tr> <td>☐ In Use Test</td> <td>☐ Other: None</td> </tr> <tr> <td>☐ Comparison with Analytic Solutions</td> <td></td> </tr> <tr> <td>☒ Comparison with Other Verified Programs</td> <td></td> </tr> </table>			☐ Hand Calculations	☐ Parametric Sensitivity Analysis	☐ In Use Test	☐ Other: None	☐ Comparison with Analytic Solutions		☒ Comparison with Other Verified Programs	
☐ Hand Calculations	☐ Parametric Sensitivity Analysis									
☐ In Use Test	☐ Other: None									
☐ Comparison with Analytic Solutions										
☒ Comparison with Other Verified Programs										
17. Documentation: <div style="margin-top: 20px;">User's Reference: one</div> <div style="margin-top: 40px;">Theoretical Reference: None</div> <div style="margin-top: 40px;">Verification/Sample Problems Reference: None</div> <div style="margin-top: 40px;">Other Reference: None</div>										
18. Primary Reference: None										
19. Approved by: <div style="display: flex; justify-content: space-between; align-items: flex-end; margin-top: 20px;"> <div style="text-align: center;"> PA QA Coordinator </div> <div style="text-align: center;"> <u>12-27-93</u> Date </div> </div>										

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: PRESTAFF	2a. Version ID: C-4.01ZO	2b. Version Date: 10/19/93	3. Host Computer: ALPHA/VMS
4. Software Title: PRESTAFF: A pre-processor for input data to STAFF2D			
5. Software Description: PRESTAFF creates an input file for the STAFF2D computer program by translating data from CAMDAT. This version provides a user interface to the INEL source term model and BRAGFLO_T flow field including saturation.			
6. Limitations: For dual porosity simulations matrix material properties must exist at the nodes. In CAMDAT, these properties exist at the element centers. PRESTAFF uses bilinear interpolation to calculate the value of the property at the nodes using the four surrounding elements. This version requires a rectangular elements with regular i-, j- index node numbering.			
7. Keyword/Phrases: Performance assessment, Preprocessor, STAFF2D, WIPP			
8. Origin/Source: Sandia National Laboratories, Performance Assessment, Division 6342, Albuquerque, NM 87185-5800			
9. Authors: Rebecca L. Blaine, D. K. Rudeen			
10. Parent Program: None			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	<div style="display: flex; align-items: center;"> <div> Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2 </div> </div>
Date: 12/27/93		
Software Name: PRESTAFF	Version ID: C-4.01ZO	Version Date: 10/19/93
13. Classification: Government Owned, Limited distribution.		
14. Software Sponsor: David Rudeen	15. Software Consultant:	
16. Verification: <div style="display: flex; flex-wrap: wrap; margin-top: 10px;"> <div style="width: 50%;"> ☐ Hand Calculations </div> <div style="width: 50%;"> ☐ Parametric Sensitivity Analysis </div> <div style="width: 50%;"> ☐ In Use Test </div> <div style="width: 50%;"> ☐ Other: None </div> <div style="width: 50%;"> ☐ Comparison with Analytic Solutions </div> <div style="width: 50%;"> ☒ Comparison with Other Verified Programs </div> </div>		
17. Documentation: <div style="margin-top: 10px;"> User's Reference: Rechard, R.P. et al, 1989. User's Manual for CAMCON: Compliance Assessment Methodology Controller, SAND88-1496, Sandia National Laboratories, Albuquerque, NM. </div> <div style="margin-top: 20px;"> Theoretical Reference: None </div> <div style="margin-top: 20px;"> Verification/Sample Problems Reference: None </div> <div style="margin-top: 20px;"> Other Reference: None </div>		
18. Primary Reference: Rechard, R.P. et al, 1989. User's Manual for CAMCON: Compliance Assessment Methodology Controller, SAND88-1496, Sandia National Laboratories, Albuquerque, NM.		
19. Approved by: <div style="display: flex; justify-content: space-between; margin-top: 20px;"> <div style="width: 45%;"> _____ PA QA Coordinator </div> <div style="width: 45%; text-align: right;"> 12-27-93 _____ Date </div> </div>		

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: RELATE	2a. Version ID: C-1.42VV	2b. Version Date: 08/09/93	3. Host Computer: ALPHA/VMS
4. Software Title: RELATE: Geometry-dependent database transfer utility			
5. Software Description: RELATE is used to create new components of one database (OBJECT database) from existing components of a second (REFERENCE database). In general, object component values depend on the relative locations of the object and reference meshes.			
6. Limitations: Mesh elements must be quadrilateral or cube-like			
7. Keyword/Phrases: WIPP, Performance Assessment, Computer Code, Interpolation, Transfer			
8. Origin/Source: Sandia National Laboratories Division 6342 Albuquerque, NM 87185 - 5800			
9. Authors: Walt Beyeler			
10. Parent Program: None			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	<div style="display: flex; align-items: center;"> <div> Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2 </div> </div>
Date: 12/27/93		
Software Name: RELATE	Version ID: C-1.42VV	Version Date: 08/09/93
13. Classification: Government owned, Limited distribution.		
14. Software Sponsor: Walt Beyeler	15. Software Consultant:	
16. Verification: <div style="display: flex; flex-wrap: wrap; margin-top: 10px;"> <div style="width: 50%;"> ☐ Hand Calculations </div> <div style="width: 50%;"> ☐ Parametric Sensitivity Analysis </div> <div style="width: 50%;"> ☐ In Use Test </div> <div style="width: 50%;"> ☐ Other: None </div> <div style="width: 50%;"> ☐ Comparison with Analytic Solutions </div> <div style="width: 50%;"> ☒ Comparison with Other Verified Programs </div> </div>		
17. Documentation: <div style="margin-top: 10px;"> User's Reference: Rechard, R.P., et al., 1991. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND90-1983. Albuquerque, NM: Sandia National Laboratories. </div> <div style="margin-top: 20px;"> Theoretical Reference: None </div> <div style="margin-top: 20px;"> Verification/Sample Problems Reference: None </div> <div style="margin-top: 20px;"> Other Reference: None </div>		
18. Primary Reference: Rechard, R.P., et al., 1991. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND90-1983. Albuquerque, NM: Sandia National Laboratories.		
19. Approved by: <div style="display: flex; justify-content: space-between; align-items: flex-end; margin-top: 20px;"> <div style="text-align: center;"> _____ PA QA Coordinator </div> <div style="text-align: center;"> 12-27-93 _____ Date </div> </div>		

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: SALFLOW	2a. Version ID: X-1.00ZO	2b. Version Date: 09/28/93	3. Host Computer: ALPHA/VMS
4. Software Title: SALFLOW: A code to estimate fluid flow rates in a salt repository			
5. Software Description: SALFLOW calculates liquid flow rates through an arbitrary number of boreholes penetrating a repository in salt, and through a confined conductive zone in direct communication with the repository. SALFLOW is internally coupled to the canister simulation model CONCNF			
6. Limitations: Fluid flow is allowed only through the repository, boreholes, conductive layer, and sub-repository brine reservoir; flow in all components is described by Darcy's law; gas and liquid pressures in the repository are assumed to be equal, and the two phases are assumed to be separated; the ideal gas law is assumed.			
7. Keyword/Phrases: INEL, Performance Assessment, Flow Model			
8. Origin/Source: Sandia National Laboratories Performance Assessment Division 6342 Albuquerque, NM 87185 - 5800			
9. Authors: Walt Beyeler			
10. Parent Program: None			
11. Computer Languages: FORTRAN 77			
12. Status: X			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	<div style="display: flex; align-items: center;"> <div> Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2 </div> </div>
Date: 12/27/93		
Software Name: SALFLOW	Version ID: X-1.00ZO	Version Date: 09/28/93
13. Classification: Government Owned, Limited Distribution.		
14. Software Sponsor: Walt Beyeler	15. Software Consultant:	
16. Verification: <div style="display: flex; flex-wrap: wrap; margin-top: 10px;"> <div style="width: 50%;"> ☐ Hand Calculations </div> <div style="width: 50%;"> ☐ Parametric Sensitivity Analysis </div> <div style="width: 50%;"> ☐ In Use Test </div> <div style="width: 50%;"> ☐ Other: None </div> <div style="width: 50%;"> ☐ Comparison with Analytic Solutions </div> <div style="width: 50%;"> ☒ Comparison with Other Verified Programs </div> </div>		
17. Documentation: <div style="margin-top: 20px;"> User's Reference: None </div> <div style="margin-top: 40px;"> Theoretical Reference: None </div> <div style="margin-top: 40px;"> Verification/Sample Problems Reference: None </div> <div style="margin-top: 40px;"> Other Reference: None </div>		
18. Primary Reference: None		
19. Approved by: <div style="display: flex; justify-content: space-between; margin-top: 20px;"> <div style="text-align: center;"> _____ PA QA Coordinator </div> <div style="text-align: center;"> 12-27-93 _____ Date </div> </div>		

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: STAFF2D	2a. Version ID: C-4.01ZO	2b. Version Date: 10/19/93	3. Host Computer: ALPHA/VMS
4. Software Title: STAFF2D: Solute Transport And Fractured Flow in 2 Dimensions			
5. Software Description: STAFF2D is a two-dimensional finite element code designed to simulate confined and unconfined groundwater flow and single or multiple component solute transport in fractured or granular aquifers. Fractured porous media are represented using both discrete-fracture and dual porosity approaches. This version allows time dependent saturation, fluid density and porosity. It allows contains the INEL source term model.			
6. Limitations: The code handles both unfractured or fractured flow systems. The fractured flow systems is represented using either the dual porosity or the discrete-fracture approach or a combination of these two approaches. Only single-phase flow is modeled. The effects of concentration and temperature changes on fluid properties are not modeled. Combining user specified time dependent BC,s and INEL source term model not verified. INEL source term not implemented for dual porosity.			
7. Keyword/Phrases: Dual porosity, Flow, Fractures, Groundwater, Performance Assessment, Radionuclide transport, WIPP			
8. Origin/Source: Huyakorn, P.S., H.O. White, Jr., S. Panday HydroGeoLogic, Inc., 503 Carlise Drive, Suite 250 Herndon, VA 22070			
9. Authors: P.S. Huyakorn, H.O. White, Jr., S. Panday			
10. Parent Program: None			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2
Date: 12/27/93		
Software Name: STAFF2D	Version ID: C-4.01ZO	Version Date: 10/19/93
13. Classification: Limited distribution		
14. Software Sponsor: David K. Rudeen		15. Software Consultant:
16. Verification: <div style="display: flex; justify-content: space-between; margin-top: 10px;"> <div style="width: 45%;"> ☐ Hand Calculations ☒ In Use Test ☒ Comparison with Analytic Solutions ☐ Comparison with Other Verified Programs </div> <div style="width: 45%;"> ☐ Parametric Sensitivity Analysis ☐ Other: None </div> </div>		
17. Documentation: <div style="margin-top: 10px;"> User's Reference: P.S. Huyakorn, H.O. White, Jr., S. Panday, STAFF2D, HydroGeoLogic, Inc., 503 Carlisle Drive, Suite 250, Herndon, VA 22070, May 1989. </div> <div style="margin-top: 20px;"> Theoretical Reference: P.S. Huyakorn, H.O. White, Jr., S. Panday, STAFF2D, HydroGeoLogic, Inc., 503 Carlisle Drive, Suite 250, Herndon, VA 22070, May 1989. </div> <div style="margin-top: 20px;"> Verification/Sample Problems Reference: None </div> <div style="margin-top: 20px;"> Other Reference: None </div>		
18. Primary Reference: P.S. Huyakorn, H.O. White, Jr., S. Panday, STAFF2D, HydroGeoLogic, Inc., 503 Carlisle Drive, Suite 250, Herndon, VA 22070, May 1989.		
19. Approved by: <div style="display: flex; justify-content: space-between; margin-top: 20px;"> <div style="width: 45%;"> PA QA Coordinator </div> <div style="width: 45%; text-align: right;"> <u>12-27-93</u> Date </div> </div>		

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: STEPWISE	2a. Version ID: C-2.18VV	2b. Version Date: 09/27/93	3. Host Computer: VAX/VMS 8810
4. Software Title: STEPWISE: A statistical program module using stepwise regression with rank regression and predicted error sum of squares analysis techniques			
5. Software Description: The statistical module, STEPWISE, evaluates variable importance by developing regression models between the observed response and input variables using either a forward, backward, or stepwise regression procedure on the raw or ranked data. Provided useful regression models can be developed, the absolute values of the standardized regression coefficients (or mathematically related partial correlation coefficients) can be used to rank variable importance.			
6. Limitations: None			
7. Keyword/Phrases: WIPP, Performance Assessment, Statistical Program, Stepwise Regression, Regression Analysis			
8. Origin/Source: Sandia National Laboratories, Safety and Reliability Analysis Division 6415, Albuquerque, NM 87185-5800			
9. Authors: R.L. Iman, J.M. Davenport, E.L. Frost, M.J. Shortencarier, Amy P. Gilkey			
10. Parent Program: None			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2
Date: 12/27/93		
Software Name: STEPWISE	Version ID: C-2.18VV	Version Date: 09/27/93
13. Classification: Government owned, Limited distribution.		
14. Software Sponsor: Amy P. Gilkey	15. Software Consultant: Amy P. Gilkey	
16. Verification: <div style="display: flex; flex-wrap: wrap; margin-top: 10px;"> <div style="width: 50%;"> ☐ Hand Calculations </div> <div style="width: 50%;"> ☐ Parametric Sensitivity Analysis </div> <div style="width: 50%;"> ☐ In Use Test </div> <div style="width: 50%;"> ☐ Other: None </div> <div style="width: 50%;"> ☐ Comparison with Analytic Solutions </div> <div style="width: 50%;"> ☒ Comparison with Other Verified Programs </div> </div>		
17. Documentation: <div style="margin-top: 10px;"> User's Reference: Rechard, R.P., et al., 1991. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND90-1983, Version 3.0. Albuquerque, NM: Sandia National Laboratories. </div> <div style="margin-top: 20px;"> Theoretical Reference: None </div> <div style="margin-top: 20px;"> Verification/Sample Problems Reference: See code notebook (on file with SWCF). </div> <div style="margin-top: 20px;"> Other Reference: R.L. Iman, J.M. Davenport, E.L. Frost, M.J. Shortencarier, 1980. Stepwise regression with PRESS and rank regression (program user's guide). SAND79-1472. Albuquerque, NM: Sandia National Laboratories. </div>		
18. Primary Reference: Rechard, R.P., et al., 1991. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND90-1983, Version 3.0. Albuquerque, NM: Sandia National Laboratories.		
19. Approved by: <div style="display: flex; justify-content: space-between; margin-top: 20px;"> <div style="width: 45%;"> PA QA Coordinator </div> <div style="width: 45%; text-align: right;"> 12-27-93 Date </div> </div>		

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: SUMMARIZE	2a. Version ID: C-1.00ZO	2b. Version Date: 09/08/93	3. Host Computer: ALPHA/VMS
4. Software Title: SUMMARIZE: Summarize variable values from multiple files in tabular form			
5. Software Description: SUMMARIZE reads specified variable values from multiple CAMDAT data bases (one for each vector) or transfer files. The data may be read from a single time or multiple times. Some simple processing (such as interpolation or integration) of the data may be done. The data is then written to a text transfer file suitable for input to an X-Y plot program.			
6. Limitations: SUMMARIZE is limited to 1000 variables			
7. Keyword/Phrases: WIPP, Performance Assessment			
8. Origin/Source: Sandia National Laboratories, Waste Management Systems Division 6416, Albuquerque, NM 87185-5800			
9. Authors: Amy P. Gilkey			
10. Parent Program: None			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2								
Date: 12/27/93										
Software Name: SUMMARIZE	Version ID: C-1.00ZO	Version Date: 09/08/93								
13. Classification: Government owned, Limited distribution.										
14. Software Sponsor: Amy P. Gilkey	15. Software Consultant: Amy P. Gilkey									
16. Verification: <table style="width: 100%; margin-top: 10px;"> <tr> <td>☐ Hand Calculations</td> <td>☐ Parametric Sensitivity Analysis</td> </tr> <tr> <td>☐ In Use Test</td> <td>☐ Other: None</td> </tr> <tr> <td>☐ Comparison with Analytic Solutions</td> <td></td> </tr> <tr> <td>☒ Comparison with Other Verified Programs</td> <td></td> </tr> </table>			☐ Hand Calculations	☐ Parametric Sensitivity Analysis	☐ In Use Test	☐ Other: None	☐ Comparison with Analytic Solutions		☒ Comparison with Other Verified Programs	
☐ Hand Calculations	☐ Parametric Sensitivity Analysis									
☐ In Use Test	☐ Other: None									
☐ Comparison with Analytic Solutions										
☒ Comparison with Other Verified Programs										
17. Documentation: <p style="margin-top: 20px;">User's Reference: Rechard, R.P., et al., 1991. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND90-1983, Version 3.0. Albuquerque, NM: Sandia National Laboratories.</p> <p style="margin-top: 20px;">Theoretical Reference: None</p> <p style="margin-top: 20px;">Verification/Sample Problems Reference: None</p> <p style="margin-top: 20px;">Other Reference: None</p>										
18. Primary Reference: Rechard, R.P., et al., 1991. User's Manual for CAMCON: Compliance Assessment Methodology Controller. SAND90-1983, Version 3.0. Albuquerque, NM: Sandia National Laboratories.										
19. Approved by: <div style="display: flex; justify-content: space-between; align-items: flex-end; margin-top: 20px;"> <div style="text-align: center;"> <hr style="width: 200px; margin: 0 auto;"/> PA QA Coordinator </div> <div style="text-align: center;"> <hr style="width: 150px; margin: 0 auto;"/> 12-27-93 Date </div> </div>										

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: SUMQ_3D	2a. Version ID: C-1.01VV	2b. Version Date: 11/20/92	3. Host Computer: VAX/VMS 8810
4. Software Title: SUMQ_3D: Sum of q's for a three-dimensional grid.			
5. Software Description: Calculates temperature change at a point of interest from a three- dimensional layout of linear heat sources (canisters).			
6. Limitations:			
7. Keyword/Phrases:			
8. Origin/Source:			
9. Authors: G. R. Durka, J. W. Berglund			
10. Parent Program:			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2
Date: 12/27/93		
Software Name: SUMQ_3D	Version ID: C-1.01VV	Version Date: 11/20/92
13. Classification: Government owned, Limited distribution.		
14. Software Sponsor: G. R. Durka	15. Software Consultant: J. W. Berglund	
16. Verification: <div style="display: flex; flex-wrap: wrap; margin-top: 10px;"> <div style="width: 50%;"> ☒ Hand Calculations </div> <div style="width: 50%;"> ☐ Parametric Sensitivity Analysis </div> <div style="width: 50%;"> ☐ In Use Test </div> <div style="width: 50%;"> ☐ Other: None </div> <div style="width: 50%;"> ☒ Comparison with Analytic Solutions </div> <div style="width: 50%;"> ☐ Comparison with Other Verified Programs </div> </div>		
17. Documentation: <div style="margin-top: 10px;"> User's Reference: None </div> <div style="margin-top: 20px;"> Theoretical Reference: None </div> <div style="margin-top: 20px;"> Verification/Sample Problems Reference: None </div> <div style="margin-top: 20px;"> Other Reference: None </div>		
18. Primary Reference: None		
19. Approved by: <div style="display: flex; justify-content: space-between; margin-top: 20px;"> <div style="width: 45%;"> PA QA Coordinator </div> <div style="width: 45%; text-align: center;"> <u>12-27-93</u> Date </div> </div>		

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93		Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 1 of 2
Date: 12/27/93			
1. Software Name: TRACKER	2a. Version ID: C-4.01VV	2b. Version Date: 06/14/93	3. Host Computer: VAX/VMS 8810
4. Software Title: TRACKER: A program to estimate the pathway of a particle released in a fluid velocity field			
5. Software Description: Tracker estimates the migration path of a neutrally buoyant particle from a user-defined point in an input fluid velocity field. Besides the coordinates of the pathway, material properties and average velocities of connecting legs are output.			
6. Limitations: Rectangular grid with a 2D i,j index space parallel to the global x, y coordinate system. Interstitial velocities must be element variables in the CAMDAT Data base which are averaged(distant weighted) to the nodal coordinates.			
7. Keyword/Phrases: WIPP, Performance Assessment, neutrally buoyant particle, interstitial velocity, Runge-Kutta-Fehlberg			
8. Origin/Source: Sandia National Laboratories Performance Assessment Division 6342 Albuquerque, New Mexico			
9. Authors: P. J. Roache			
10. Parent Program: none			
11. Computer Languages: FORTRAN 77			
12. Status: C			

INEL IDAHO NATIONAL ENGINEERING LABORATORY	TITLE: SOFTWARE ABSTRACT FOR INEL PERFORMANCE ASSESSMENT SOFTWARE Effective Date: 3/22/93	<div style="display: flex; align-items: center;"> <div> Sandia National Laboratories Procedure PAP02 Revision: 0 Form No. PA001 Page 2 of 2 </div> </div>
Date: 12/27/93		
Software Name: TRACKER	Version ID: C-4.01VV	Version Date: 06/14/93
13. Classification: Government owned, Limited Distribution.		
14. Software Sponsor: D. K. Rudeen	15. Software Consultant:	
16. Verification: <div style="display: flex; flex-wrap: wrap; margin-top: 10px;"> <div style="width: 50%;"> ☒ Hand Calculations </div> <div style="width: 50%;"> ☐ Parametric Sensitivity Analysis </div> <div style="width: 50%;"> ☐ In Use Test </div> <div style="width: 50%;"> ☐ Other: None </div> <div style="width: 50%;"> ☐ Comparison with Analytic Solutions </div> <div style="width: 50%;"> ☒ Comparison with Other Verified Programs </div> </div>		
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Appendix H: Summary of Code Enhancements

Appendix H: Summary of Code Enhancements

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Appendix H: Summary of Code Enhancements

M. E. Lord and J. W. Berglund

This appendix summarizes the enhancements for specific consequence modeling codes in the Compliance Assessment Methodology Controller (CAMCON) system (Appendix G). The codes were enhanced specifically to assess the performance of various processing options of spent nuclear fuel and high-level waste at the Idaho National Engineering Laboratory (INEL). The enhancements described here include adding the thermal equation in the multiphase, single continuum code, BRAGFLO; developing simple heat-conduction codes for assisting in repository design, and enhancing two transport codes, STAFF2D and SUTRA, to make their capabilities more similar and thus assist in benchmarking the codes.

H.1 Addition of Thermal Equation to BRAGFLO

H.1.1 Purpose

BRAGFLO_T is the code that is used to determine the flow, saturation, and pressure fields throughout the generic repositories for spent nuclear fuel and high-level waste. BRAGFLO_T is an enhanced version of BRAGFLO, which has been used to evaluate the isothermal flow of brine and waste-generated gas throughout the Waste Isolation Pilot Plant (WIPP) repository in performance assessment calculations. Enhancements to BRAGFLO were necessary to better simulate non-isothermal flow and account for the unique characteristics associated with the spent nuclear fuel and high-level waste forms.

The enhancements to BRAGFLO fall into three major areas:

1. The addition of the conservation of energy equation and its coupling to the existing multi-phase mass conservation equations,
2. The addition and modification of fluid and formation physical properties to account for temperature dependency,
3. The addition of gas and heat-generation descriptions associated with specific waste forms (see Chapter 11).

Details of these areas of code development and modifications to BRAGFLO are described in Chapter 12. Section H.1.2 summarizes the preliminary verification exercise for BRAGFLO_T.

H.1.2 Verification of BRAGFLO_T Code Enhancements

Benchmarking of BRAGFLO was documented against other multiphase reservoir codes (WIPP PA Division, 1991). The problem presented here verifies the heat-transport properties of BRAGFLO_T. This proposed test problem considers the solution of the one-dimensional heat transport problem, described in Ross et al., 1982. A description of the test problem, together with comparisons of the analytic solution and several heat-driven flow codes, is found in Updegraff, 1989.

The reservoir is semi-infinite in one-dimension with cross-section area normal to flow 100 m^2 . At initial time, the reservoir is at temperature $170^\circ\text{C} = 443 \text{ K}$. At the left boundary ($x=0$), cold water at $160^\circ\text{C} = 433 \text{ K}$ is injected at a fixed rate of 10 kg/s . It is assumed that there is no heat loss to the overburden/underburden.

The convective-diffusive process for heat transport is governed by the equation

$$\rho_m c_m \frac{\partial T}{\partial t} + u_w \rho_w c_w \frac{\partial T}{\partial x} = K_m \frac{\partial^2 T}{\partial x^2} , \quad (\text{H-1})$$

where

- ρ_m = density of saturated porous medium (fluid + rock),
- c_m = specific heat of the saturated porous medium,
- K_m = thermal conductivity of the saturated porous medium,
- u_w = Darcy velocity of the water,
- ρ_w = density of the water,
- c_w = specific heat of the water.

The initial and boundary conditions are given by

$$T(x, 0) = 443 \text{ K}, \quad 0 \leq x,$$

$$T(0, t) = 433 \text{ K}, \quad 0 < t.$$

The solution of the differential equation, together with initial and boundary conditions, is given by Ross et al., 1982:

$$\frac{T(x, t) - T(x, 0)}{T(0, t) - T(x, 0)} = \frac{1}{2} \left[\operatorname{erfc} \left(\frac{x - vt}{\sqrt{4Dt}} \right) + \exp \left(\frac{vx}{D} \right) \operatorname{erfc} \left(\frac{x + vt}{\sqrt{4Dt}} \right) \right] \quad (\text{H-2})$$

where

- $v = u_w \rho_w c_w / (\rho_m c_m),$
- $D = K_m / (\rho_m c_m).$

The following values were specified for the saturated medium and the water:

- $\rho_m = 2500 \text{ kg/m}^3,$
- $c_m = 1000 \text{ J/kg/K},$
- $K_m = 20 \text{ W/m/K},$
- $\rho_w = 919 \text{ kg/m}^3,$
- $c_w = 4185 \text{ J/kg/K}.$

In order to specify the Darcy velocity of water, it is necessary to describe further the fluid flow field. The formation permeability and water viscosity are given as

$$k = 1.74 \times 10^{-11} \text{ m}^2$$

$$\mu_w = 1.6 \times 10^{-4} \text{ Pa}\cdot\text{s}.$$

Further, it is assumed that the pressure gradient across the reservoir is a constant

$$\frac{dP}{dx} = -1000 \text{ Pa/m}.$$

Then the Darcy velocity is

$$u_w = -\frac{k}{\mu_w} \frac{dP}{dx} = 1.088 \times 10^{-4} \text{ m/s}.$$

This data determines the analytical solution. The analytical solution evaluated at time 130,000 s is shown in Figure H-1.

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Figure H-1. Solution of one-dimensional heat transport problem at time = 130,000 seconds.

For comparison using BRAGFLO_T, the reservoir was realized as 50 m in length. The reservoir was discretized into 500 grid blocks of length 0.1 m. Two additional grid blocks were added at the left and right boundary in order to maintain steady-state flow and boundary temperature at $x = 0$. A well located in the first (leftmost) grid block injects water at a rate of 10 kg/s with temperature 160°C (433 K). The well at the last (rightmost) grid block produces water at a rate of 10 kg/s.

Initial pressure is assigned as linear variation along the reservoir and satisfying

$$\begin{aligned} P &= 5.05 \text{ MPa at } x = 0, \\ P &= 5.00 \text{ MPa at } x = 50, \end{aligned}$$

with resulting pressure gradient

$$\frac{dP}{dx} = -1000 \text{ Pa/m}.$$

This pressure gradient, together with the injection/production well rates, maintains a steady-state fluid flow.

For porosity of 0.20, the rock density and specific heat are assigned so that the values of density and specific heat of the saturated medium used in the analytic solution are honored. BRAGFLO_T was run with a constant time-step size of 100 s to the end of run time 130,000 s. The comparison of the numerical results to the analytical solution is shown in Figure H-1. Good agreement is obtained with some evidence of numerical dispersion, which results from the spacial and time discretization.

A verification of the treatment in BRAGFLO_T of the boundary heat loss to the surrounding rock was done in a stand-alone program. The following one-dimensional problem was used to test the above treatment. A finite-difference solution was obtained for the heat conduction in a one-dimensional rod with constant initial temperature and

with constant heat source in the first (leftmost) grid block. The resulting temperature solution was used to compute a heat flux between the 1st and 2nd grid block. The temperature history in the first grid block and the resulting cumulative heat flux at the 1st and 2nd grid block interface are shown in Figures H-2 and H-3, respectively, as the numerical solution. An energy balance was then obtained using only the first grid block (0-dimension) with the specified heat injection and heat loss at the right boundary computed from the analytical heat loss treatment. Figures H-2 and H-3 show this temperature and cumulative boundary heat loss history as the analytical solution. The comparison demonstrates that the analytical treatment gives good estimates for the boundary heat loss.

H.2 New Thermal-Conduction Codes for Repository Design

H.2.1 Purpose

The design of a repository for the disposal of spent fuel and high-level radioactive wastes must take into consideration a number of factors not the least of which is the production of heat by the decaying radionuclides. The heat generated by the waste within the disposal canisters must be able to be transported into the surrounding geology so that local temperatures do not exceed the temperature limits of the waste forms or of the engineered backfill and host rock that surrounds each canister. To determine the time-dependent temperature distributions near a canister and within a field of canisters, two thermal-conduction codes were developed.

H.2.2 Code Development and Modification

Two computer codes were developed to study heat conduction in a waste repository. The first, named SUMQ_3D, was designed to determine the temperature at any point and at any time within an arbitrary array of

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Figure H-2. Temperature history for 1-D conduction.

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Figure H-3. Boundary heat loss history for 1-D conduction.

decaying thermal line sources of finite length. The code is based on the superposition of a large number of decaying point source solutions in an infinite medium. This code permits various canister array designs to be studied easily and very rapidly.

The second, COND_1D, is an explicit finite-difference code that was designed to study one-dimensional heat conduction in spherical or cylindrical coordinates. Different material properties, time-varying heat-flux boundary conditions, and internal thermal sources are permitted. This code is useful in determining the temperature field within and adjacent to a canister.

The governing equations for the SUMQ_3D code are based upon analytical work discussed in Carslaw and Jaeger (1959). The temperature change at any point in a field away from a constant thermal source and at any time assuming constant conductivity and diffusivity is governed by the equation,

$$\Delta T = \frac{\bar{q}}{4\pi Kr} \operatorname{erfc} \left[\frac{r}{\sqrt{4kt}} \right], \quad (\text{H-3})$$

where

- ΔT = the temperature change,
- $\bar{q}$ = source strength,
- K = thermal conductivity,
- r = radius from source,
- k = thermal diffusivity,
- t = time.

If the source decays with time, the governing equation must be written in the integral form as

$$\Delta T = \frac{k}{4K(\pi k)^{\frac{3}{2}}} \int_0^{\infty} \bar{q} \left(t - \frac{l}{\tau^2} \right) \exp \left(-\frac{r^2 \tau^2}{4k} \right) d\tau \quad (H-4)$$

For a constant thermal source, the solutions to an array of point sources (Equation H-3) configured to approximate a field of canisters of finite length is summed to determine the temperature change at any point and at any time. Typically, a canister can be represented by 50 point sources; the SUMQ_3D code will model up to 100 canisters although fewer can be used to determine the temperature change at a particular point. Thus, for a 100-canister array, Equation H-3 would be solved 5000 times, and the results summed to obtain the temperature at a single point. If thermal decay is taken into account, a similar process is used with Equation H-4. The latter equation must be integrated numerically.

The COND_1D code is based on the solution to the thermal conduction equation (Scipio, 1967)

$$\nabla \cdot (K \nabla T) + Q = \rho c \frac{\partial T}{\partial t} \quad (H-5)$$

where

Q = heat rate within medium,
 c = specific heat,
 ρ = mass density,
 ∇T = the gradient of the temperature.

Equation H-5 is solved explicitly using finite-difference approximations to the derivatives along a nonuniform one-dimensional grid. Decay heat sources are also permitted.

H.2.3 Verification

The two codes described above were verified partially by comparing results for similar input configurations. The far-field solutions of Equations H-3 and H-4 in the SUMQ_3D code have been shown to accurately correspond to solutions obtained for spherical heat-conduction solutions utilizing COND_1D. Similarly, when a canister is modeled using SUMQ_3D, the far-field results match well with comparable results from COND_1D when run in cylindrical geometry. Verification of COND_1D has also been accomplished for the far field by comparing results.

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1335	6305	S. A. Goldstein
1326	6312	H. A. Dockery
1328	6342	D. R. Anderson
1328	6342	J. W. Berglund
1328	6342	W. E. Beyeler
1328	6342	K. F. Brinster
1328	6342	M. M. Gruebel
1328	6342	M. E. Lord
1328	6342	H. J. Iuzzolino
1328	6342	A. Johnson
1328	6342	M. G. Marietta
1328	6342	J. L. Ramsey
1328	6342	J. S. Rath
1328	6342	R. P. Rechard (5)
1328	6342	D. K. Rudeen
1328	6342	C. T. Stockman
1328	6342	M. S. Tierney
1328	6342	K. M. Trauth
1328	6342	P. Vaughn

1	1328	6342	Sandia INEL Central Files (50)
2	1341	6345	R. C. Lincoln
3	1343	6351	R. E. Thompson
4	0727	6622	M. S. Y. Chu
5	0899	7141	Technical Library (5)
6	0619	7151	Technical Publications
7	0100	7613-2	Document Processing for
8			DOE/OSTI (10)
9			
10	9018	8523-3	Central Technical Files
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