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Civilian Radioactive Waste Management System
Management & Operating Contractor

Technical Justification to Support the PRM by the DOE to Exempt HLW Canisters from 10CFR71.63(b)

Rev. 0

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1. INTRODUCTION

The Department of Energy (DOE) requests that the regulations in 10 CFR 71.63(b) be amended to exempt canisters containing vitrified high-level waste (HLW) from the double containment requirement for the shipment of plutonium with an activity greater than 20 curies per package. Both spent fuel and vitrified HLW -- which will be shipped in the Civilian Radioactive Waste Management System (CRWMS) to a final disposal site in a geologic repository -- contain plutonium in quantities greater than the threshold limit of 20 curies per package. Spent fuel was exempted from the double containment requirement in the original rulemaking because of its physical integrity and low potential for inhalation of plutonium particles. The fuel pellet itself and the surrounding metal fuel rod cladding were determined to provide adequate protection against the possible dispersion of plutonium particles under both normal conditions of transport and during hypothetical accident conditions. Vitrified HLW contained in stainless steel canisters provides a comparable level of safety protection to that provided by fuel elements, and therefore should also be exempted from the double containment requirement.

The following sections provide a background on this issue, and provide a detailed description of a representative HLW canister and its vitrified waste content (bounding worst-case radionuclide concentrations) at the Savannah River Site in Aiken, SC. The Savannah River Site will be the first of the 3 HLW sites (the others are Hanford, WA and West Valley, NY) to vitrify its HLW into a borosilicate glass waste form suitable for transport and final disposal. In addition, the waste acceptance specifications, analyses, and supporting documentation which must be provided to DOE's Office of Civilian Radioactive Waste Management (OCRWM) is described. This documentation assures that the waste glass is prepared and packaged in accordance with approved standards to meet CRWMS system-level requirements and to verify the quality and consistency of the waste form. A summary of supporting technical analyses, including the results of numerous impact tests of simulated HLW canisters, which confirms the physical integrity and low dispersion characteristics of this waste form is presented in Sections 6 and 7. Finally, the DOE's programmatic justification for the request for exemption from double containment is provided.

2. BACKGROUND

As provided by the Nuclear Waste Policy Act of 1982 (NWPA), OCRWM has been given the responsibility to develop a geologic repository for the disposal of high-level radioactive waste and spent fuel. The transportation of these waste forms is addressed in the 1987 amendments to the NWPA. All packages used to transport spent fuel and high-level waste must be certified by the Nuclear Regulatory Commission (NRC).

The NRC's regulations governing the packaging and transport of radioactive materials are given in the Code of Federal Regulations - Section 10, Part 71. Specifically, 10 CFR 71.63 imposes special requirements for the shipment of plutonium with an activity greater than 20 curies per package. Since all spent fuel and high-level waste transported in the CRWMS will exceed this threshold radioactivity level, 10 CFR 71.63 is applicable. The waste must be shipped as a solid per Part 71.63(a), and it "must be packaged in a separate inner container placed within outer packaging that meets the requirements of Subparts E and F..." (i.e., double containment) in accordance with Part 71.63(b). In addition, if the entire package is subjected to the tests specified in Part 71.71 (Normal Conditions of Transport), no release of plutonium from the inner-most container is allowed. When subjected to the more severe tests specified in Part 71.73 (Hypothetical Accident Conditions), then the separate inner container must restrict plutonium release to specified limits. An exemption is provided, however, in 10 CFR 71.63(b) for spent reactor fuel elements from the double containment requirement of this paragraph.

The double containment requirement was originally implemented in 10 CFR 71 through a final rule in 1974 (39 FR 20960, June 17, 1974). The Statement of Consideration accompanying this rulemaking action discusses the basis for the exemption of certain solid plutonium forms from the double containment requirement. The text of the relevant section of the Federal Register notice states: "...solid forms of plutonium that are essentially nonrespirable should be exempted from the double containment requirement." Spent reactor fuel elements, and plutonium-bearing metal or metal alloys were exempted from the double containment requirement because these solid forms were deemed to be "essentially nonrespirable". The evaluation of the respirability potential of various waste forms is based primarily on the results of impact tests which measure the quantity of respirable fines produced. Respirable particles are considered to be those having an aerodynamic diameter of less than 10 um. Physical integrity of the inner container and low dispersability of the contained plutonium are also key considerations in the evaluation of the transportation package.

In this Technical Justification, the vitrified high-level waste will be shown to compare very favorably with the respirability characteristics of spent fuel. The canister containing the vitrified HLW in borosilicate glass will also be compared with the physical integrity of the metal cladding surrounding the spent fuel pellets in reactor fuel assemblies.

3. PROPOSED RULE

The DOE requests that the regulations in 10 CFR 71.63(b) be amended to exempt high-level waste canisters from the double containment requirement, as currently provided for spent fuel. The high-level wastes are currently stored at three federal DOE defense-related sites and one former commercial reprocessing site. The waste is stored in various interim forms such as liquids, slurries, sludge, calcine, etc., at each of the sites. The high-level waste will be heated and processed with material components of glass to form a homogeneous mixture which will be poured into a stainless steel canister, sealed and allowed to cool to form a solidified glass waste form. Because of the high degree of confinement provided by the waste canister and the physical characteristics of the solid plutonium-bearing waste glass (i.e., negligible respirable fines), a separate container inside the outer packaging is not considered to be necessary.

DOE therefore proposes that the last sentence of 10 CFR 71.63(b) be reworded to read as follows:

> "Solid plutonium in the following forms is exempt from the requirements of this paragraph:
> (1) Reactor fuel elements;
> (2) Metal or metal alloy;
> * (3) Canisters containing vitrified high-level waste; and
> (4) Other plutonium-bearing solids that the Commission determines should be exempt from the requirements of this section."

* Added

4. DESCRIPTION OF HLW CANISTER AND CONTENTS

During the vitrification process, molten borosilicate glass containing high-level radioactive waste will be poured into special stainless steel canisters which are cooled and sealed for eventual shipment of the solidified waste to a geologic repository. Such canisters of high-level waste will be produced at the Savannah River Site (SRS), Hanford (HANF), and at the West Valley Demonstration Project (WVDP). In addition, high-level waste from the reprocessing of DOE and Navy fuels at the Idaho National Engineering Laboratory (INEL) will eventually be vitrified and transported to the repository. Since the first canisters from INEL are not expected to be produced until 2014, the final decision on the waste form to be used has not been made at this time. This waste form will, however, be at least comparable in quality to those described herein. The vitrification process and final waste form characteristics have been defined for the SRS, HANF, and WVDP sites. A detailed physical description of the high-level wastes from each of these sites is provided in a U.S. DOE (OCRWM) publication issued in July 1992: "Characteristics of Potential Repository Wastes", DOE/RW-0184-R1.[1] Supporting data from this document is referenced extensively in the technical details which follow.

Cylindrical stainless steel canisters will be used at each of the three sites to enclose the borosilicate glass waste form. The canisters will be fabricated from 304 or 304L stainless steel which conforms to American Society for Testing and Materials (ASTM) specifications. Depending on the component, the base material may be pipe, plate stock or another form. The canisters will be 24 inches (61 cm.) in diameter and 118 inches (300 cm.) high, filled with borosilicate glass to about 85% of the total canister volume (to minimize the potential for overfilling the canister). "The canister designs for SRS and HANF are identical. The WVDP canister has the same outside diameter and length but has a smaller wall thickness and a wider filler neck."[1] All canisters will meet the requirements defined in the Waste Acceptance System Requirements Document (WASRD) - Ref. 2. A summary comparison of some of the relevant physical and radiological characteristics of the HLW glass waste forms and canisters from each of the 3 HLW sites is given in Table 1. The estimates of maximum radioactivity and thermal power are indicated as of the time of filling of the canister; these numbers are based on the most highly radioactive immobilized waste composition currently planned for these sites. The maximum values of these parameters are of prime importance in the design of the repository and transportation system.[1]

The latest schedule for the commencement of vitrification operations at each of the three HLW sites is as follows: (1) 1993 for SRS, (2) 1996 for WVDP, and (3) 2000 for HANF. Since the limiting in terms of the maximum plutonium content per canister, the SRS canister and its HLW contents have been selected as the representative waste form package for this supporting study. A detailed description of the SRS HLW canister and its contents is given in the narrative which follows. Similar information for the HLW at Hanford and West Valley can be found in Reference 1.

Table 1. Summary of Canister Characteristics

	West Valley Demonstration Project (WVDP)	Savannah River Site (SRS)	Hanford Site (HANF)
Nominal wall thickness			
cm	0.34	0.95	95
in	0.134	0.375	0.375
Weights, kg			
Canister	252	500	500
Glass	1,900	1,682	1,650
Total	2,152	2,182	2,150
Plutonium content per canister*			
Weight, g	0.130	0.352	.025
Activity, curies	361	3176	29
Total curies per canister*	114,700 (1.147E+05)	234,400 (2.344E+05)	298,000 (2.980E+05)
Watts per canister*	342	709	869

*These are estimated maximum values from ORIGEN2 calculations based on radionuclide compositions supplied by the sites. Curies and watts shown are at time of filling the canister, except for WVDP where the values shown are for the end of year 1991. For WVDP, maximum values are assumed to be 110% of average values. Maximum values for SRS and HANF do not necessarily represent initial operations.

SOURCE: Ref. 1.

4.1 HLW CONTENTS

Interim forms of HLW have been produced and stored onsite at Savannah River since 1954 as a result of the reprocessing of defense reactor fuels. These wastes are stored in large underground tanks where they have been allowed to settle and have been neutralized, resulting in the formation of a bottom layer of heavy sludge and a top layer of supernatant liquid. Subsequent evaporation of the lighter top layer has reduced the total volume of HLW produced by almost 60%, to a total volume of approximately 122,000 m^3 as of the end of 1989.[1]

The evaporation of the supernatant liquid, which contains almost all of the Cs-137 activity, has produced a saturated salt solution and a salt cake consisting of the salts crystallized out of the saturated solution. The major components of the salt solution and salt cake are sodium nitrate, sodium aluminate, and sodium hydroxide, together with most of the Cs-137. " Almost all of the radioactivity in the salt solution and salt cake is due to Cs-137 and its short-lived daughter Ba-137m."[1]

The sludge represents approximately 11% of the total waste volume in the storage tanks. It "is composed largely of the precipitated hydroxides of iron, aluminum, manganese, and other metals; it contains about 60-65% of the total radioactivity, including most of the Sr-90 and small amounts of actinides (principally isotopes of uranium, plutonium, and curium) that were not recovered from the fuel during reprocessing. The largest portion of the actinide radioactivity is due to the plutonium isotopes Pu-238 and Pu-241. The sludge is kept essentially separate from the salt solution and salt cake by storage tank selection and transfer operations."[1]

"Starting in 1993, the sludge and most of the radioactivity in the salt solution and salt cake will be processed at the Defense Waste Processing Facility (DWPF) complex at the SRS to produce canisters of borosilicate glass in which the HLW is dispersed and immobilized. The glass to be produced at this site is referred to as sludge-precipitate glass. It will consist of a blend of (1) washed sludge, (2) washed precipitate made by treating the salt solution"[1] in order to separate the cesium and smaller quantities of other radionuclides, and (3) glass frit. Consistent glass quality will be achieved through the careful monitoring and control of glass waste pour temperature, pour rate, and chemical composition. The borosilicate glass waste form will contain approximately 28 wt. % sludge oxides. After the HLW canisters have been filled, cooled, sealed and decontaminated, they will be transferred to interim storage buildings until final shipment to the geologic repository.

The radionuclide composition estimated by SRS to represent the most highly radioactive glass likely to be made at this site is shown in Table 2; this is the best current estimate of maximum activity per canister. The data given in this table is based on sludge aged an average of 5 years and a cesium-containing precipitate derived from the supernatant liquid which has aged an average of 15 years. The total activity and decay heat at the time of filling of the canister (based on the maximum limiting radioactivity values given in Table 2) are 234,400 Ci and 709 W per canister. SRS has made a forecast of the radionuclide content of the glass produced during each year of vitrification plant operation. The calculated average radioactivity of canisters produced through the year 2020 is 65,900 Ci per canister, considerably less than the maximum. Firm estimates of the detailed radionuclide compositions of individual feed batches are not expected to be available until about one year before the start of vitrification of each batch.[1]

Seven reference compositions of HLW borosilicate glasses which span the range of compositions expected to be produced at the SRS are given in Table 3. Four of these compositions, denoted as Batches 1 - 4, have been projected from ' : existing HLW inventory. These compositions are representative of the actual waste mater· ·ich will be vitrified during the first 10 years of operation of the DWPF. In addition, three hypothetical glass compositions have been projected. The first, denoted as Blend, is a mixture of Batches 1 - 4 and represents the glass composition used as the design basis for the DWPF. The Purex and HM glasses are hypothetical extreme ranges of possible glass compositions. The Purex composition represents the lower design limit on glass viscosity. This is a "worst-case" bounding composition in terms of glass durability. The HM waste glass composition contains a high level of aluminum waste and is representative of the upper design limit on glass viscosity.

4.2 CANISTER DESIGN

Design specifications and dimensions of the SRS canister are given in Figures 1 and 2. "The main body of the canister is made of schedule 20 type 304L stainless steel pipe with an outside diameter of 61 cm. and a nominal wall thickness of about 0.95 cm. The overall length of the canister is 300 cm. (118 in.). The weight of the empty canister is about 500 kg. (1,100 lbs.). Each canister will contain 0.626 m^3 of glass, or about 1,680 kg. (3,710 lbs.), when loaded to about 85% of its total volume. The density of the reference glass is about 2.73g/cm^3 at a temperature of 25 deg. C. The total weight of a loaded canister is therefore about 2,180 kg. (4,810 lbs.)."[1]

The canister is fabricated and inspected according to the American Society of Mechanical Engineers (ASME) Code (Sections VIII and IX). Procurement specifications and inspection procedures will be documented. All welding operations will be performed in accordance with ASME Section IX - Welding and Brazing Qualifications. After fabrication, but prior to filling and sealing operations, the canister integrity is verified by test to show that leakage is less than 10^{-7} atm-cc/sec. Pressure testing of the canister to 225 psi is also performed.[2] Immediately after the canister is filled, a temporary seal plug is shrunk-fit into a sleeve in the neck of the canister. Shrinkage of the canister nozzle and sleeve occurs during subsequent cooling operations such that the closure plug becomes tightly sealed. After the canister cools, a helium pressure leakage test is performed to verify that the canister temporary seal leakage rate is less than 2 x 10^{-4} atm-cc/sec of helium to prevent moisture infiltration into the canister. If the seal fails this test, it is removed and replaced with another temporary seal plug. The outer surface of the canister is then decontaminated by blasting with glass frit, and the temporary seal plug and sleeve are pushed down further into the canister neck. The final seal is made by upset resistance welding a weld plug (5-in. diameter, 0.5 in. thick, 304L stainless steel material) into the canister nozzle to complete the sealing of the closure. "A force of 75,000 lb, a current of 225,000 amps, and a voltage of approximately 10 volts is used to make the 1.5-sec weld. The technique was chosen after consideration of seven alternative processes, including gas tungsten arc, gas metal arc, plasma arc, Thermit, electron beam, laser beam, and friction welding, because of the high weld quality and relatively simple equipment required."[4] Tests performed on the seal weld indicate that it is capable of withstanding at least 4,000 psi internal pressure while maintaining a leak tightness of 10^{-4} atm/cc/sec.

Table 2. Radionuclide Content of Most Highly Radioactive Glass Expected to Be Produced at SRS (Ref.1).

Radionuclide	Mass (g/canister)	Radioactivity (Ci/canister)	Thermal Power (W/canister)
Cr-51	1.008E-21	9.312E-17	1.996E-20
Co-60	1.502E-01	1.699E+02	2.619E+00
Ni-59	3.163E-01	2.397E-02	9.519E-07
Ni-63	4.824E-02	2.975E+00	3.000E-04
Se-79	2.439E+00	1.699E-01	4.232E-05
Rb-87	9.961E+00	8.719E-07	7.278E-10
Sr-89	1.470E-09	4.267E-05	1.473E-07
Sr-90	3.426E+02	4.675E+04	5.426E+01
Y-90	8.795E-02	4.786E+04	2.653E+02
Y-91	3.085E-08	7.568E-04	2.715E-06
Zr-93	4.443E+02	1.117E+00	1.298E-04
Zr-95	4.680E-07	1.005E-02	5.084E-05
Nb-94	5.147E-04	9.646E-05	9.830E-07
Nb-95	5.407E-07	2.115E-02	1.013E-04
Nb-95m	3.272E-10	1.247E-04	1.730E-07
Tc-99	1.816E+02	3.079E+00	1.545E-03
Ru-103	5.217E-13	1.684E-08	5.827E-11
Ru-106	6.729E-01	2.252E+03	1.339E-01
Rh-103m	5.028E-16	1.636E-08	3.761E-12
Rh-106	6.346E-07	2.259E+03	2.167E+01
Pd-107	2.863E+01	1.473E-02	8.732E-07
Ag-110m	2.647E-05	1.258E-01	2.098E-03
Cd-113	1.472E-01	5.009E-14	8.420E-17
Cd-115m	4.763E-14	1.213E-09	4.518E-12
Sn-121m	1.336E-03	7.902E-02	1.581E-04
Sn-123	3.101E-05	2.549E-01	7.951E-04
Sn-126	1.556E+01	4.415E-01	5.508E-04
Sb-124	4.071E-12	7.123E-08	9.445E-10
Sb-125	8.226E-01	8.496E+02	2.656E+00
Sb-126	7.365E-07	6.159E-02	1.138E-03
Sb-126m	5.619E-09	4.415E-01	5.622E-03
Te-126m	1.532E-02	2.760E+02	2.320E-01
Te-127	4.555E-08	1.202E-01	1.622E-04

Table 2. (continued)

Radionuclide	Mass (g/canister)	Radioactivity (Ci/canister)	Thermal Power (W/canister)
Te-128m	1.302E-05	1.228E-01	6.597E-05
Te-129	1.457E-19	3.053E-12	1.089E-14
Te-129m	1.576E-16	4.749E-12	8.316E-15
Cs-134	2.606E-01	3.372E+02	3.433E+00
Cs-135	8.633E+01	9.943E-02	3.319E-05
Cs-136	1.068E-44	7.838E-40	1.066E-42
Cs-137	4.989E+02	4.341E+04	4.802E+01
Ba-136m	3.195E-50	8.607E-39	1.040E-41
Ba-137m	7.724E-05	4.155E+04	1.632E+02
Ba-140	1.404E-41	1.024E-36	2.853E-39
La-140	7.734E-43	4.304E-37	7.205E-39
Ce-141	1.260E-15	3.591E-11	5.250E-14
Ce-142	4.005E+02	9.609E-06	0.0000E+00
Ce-144	3.093E+00	9.869E+03	6.547E+00
Pr-143	1.780E-39	1.198E-34	2.291E-38
Pr-144	1.306E-04	9.869E+03	7.255E+01
Pr-144m	6.545E-07	1.187E+02	4.063E-02
Nd-144	4.110E+02	4.860E-10	0.0000E+00
Nd-147	1.570E-49	1.261E-44	3.038E-47
Pm-147	2.609E+01	2.419E+04	8.679E+00
Pm-148	4.243E-16	6.975E-11	5.364E-13
Pm-148m	4.722E-14	1.009E-09	1.277E-11
Sm-147	8.796E+01	2.000E-06	2.738E-08
Sm-148	1.916E+01	5.788E-12	6.901E-14
Sm-149	7.420E+00	1.781E-12	0.0000E+00
Sm-151	9.418E+00	2.478E+02	2.906E-02
Eu-152	2.132E-02	3.688E+00	2.790E-02
Eu-154	2.295E+00	6.196E+02	5.543E+00
Eu-155	1.021E+00	4.749E+02	3.455E-01
Eu-156	9.489E-37	5.231E-32	5.392E-34
Tb-160	9.923E-11	1.120E-06	9.110E-09
Tl-208	3.829E-12	1.128E-03	2.645E-05

Table 2. (continued)

Radionuclide	Mass (g/canister)	Radioactivity (Ci/canister)	Thermal Power (W/canister)
U-232	6.256E-04	1.339E-02	4.301E-04
U-233	1.636E-04	1.584E-06	4.605E-08
U-234	5.485E+00	3.428E-02	9.875E-04
U-235	7.278E+01	1.573E-04	4.122E-06
U-236	1.742E+01	1.128E-03	3.054E-05
U-238	3.122E+04	1.050E-02	2.663E-04
Np-236	1.323E-06	1.744E-08	3.514E-11
Np-237	1.263E+01	8.904E-03	2.722E-04
Pu-236	2.297E-04	1.221E-01	4.249E-03
Pu-237	7.401E-16	8.941E-12	3.292E-15
Pu-238	8.667E+01	1.484E+03	4.919E+01
Pu-239	2.076E+02	1.291E+01	3.979E-01
Pu-240	3.809E+01	8.681E+00	2.704E-01
Pu-241	1.620E+01	1.670E+03	5.176E-02
Pu-242	3.206E+00	1.224E-02	3.616E-04
Am-241	3.210E+00	1.102E+01	3.661E-01
Am-242	1.776E-08	1.436E-02	1.628E-05
Am-242m	1.488E-03	1.447E-02	5.709E-06
Am-243	2.902E-02	5.788E-03	1.860E-04
Cm-242	1.057E-05	3.495E-02	1.288E-03
Cm-243	1.078E-04	5.565E-03	2.039E-04
Cm-244	1.329E+00	1.076E+02	3.763E+00
Cm-245	3.910E-05	6.715E-06	2.225E-07
Cm-246	1.739E-06	5.342E-07	1.747E-08
Cm-247	7.116E-09	6.604E-13	2.107E-14
Cm-248	1.614E-10	6.864E-13	8.533E-14
Totals	3.427E+04	2.344E+05	7.093E+02

Table 3. Projected DWPF Waste Glass Compositions
Source: Ref 1.

Constituent Sludge Type (Wt. %)							
Major Glass Components	Blend	Batch 1	Batch 2	Batch 3	Batch 4	HM	Purex
Al_2O_3	3.98	4.87	4.46	3.25	3.32	7.08	2.89
B_2O_3	8.01	7.69	7.70	7.69	8.11	6.94	10.21
$BaSO_4$	0.27	0.22	0.24	0.26	0.38	0.18	0.29
CaO	0.97	1.17	1.00	0.93	0.83	1.00	1.02
$CaSO_4$	0.08	0.12	0.11	0.10	Trace	Trace	0.12
Cr_2O_3	0.12	0.10	0.12	0.13	0.14	0.09	0.14
CuO	0.44	0.40	0.41	0.40	0.46	0.25	0.42
Fe_2O_3	6.95	8.39	7.[illegible]1	7.48	7.59	4.95	8.54
FeO	3.11	3.72	3.15	3.31	3.36	2.19	3.78
Group A[a]	0.14	0.10	0.14	0.10	0.20	0.20	0.08
Group B[b]	0.36	0.22	0.44	0.25	0.60	0.89	0.08
K_2O	3.86	3.49	3.50	3.47	3.99	2.14	3.58
Li_2O	4.40	4.42	4.42	4.42	4.32	4.62	3.12
MgO	1.35	1.36	1.35	1.35	1.38	1.45	1.33
MnO	2.03	2.06	1.62	1.81	3.08	2.07	1.99
Na_2O	8.73	8.62	8.61	8.51	8.88	8.17	12.14
Na_2SO_4	0.10	0.10	0.12	0.10	0.13	0.14	0.12
NaCl	0.19	0.31	0.23	0.22	0.09	0.09	0.26
NiO	0.89	0.75	0.90	1.07	1.09	0.40	1.21
SiO_2	50.20	49.81	50.17	49.98	49.29	54.39	44.56
ThO_2	0.19	0.36	0.63	0.77	0.24	0.55	0.01
TiO_2	0.90	0.66	0.67	0.66	1.02	0.55	0.65
U_3O_8	2.14	0.53	2.30	3.16	0.79	1.01	2.89

[a]Group A: radionuclides of Tc, Se, Te, Rb, and Mo.
[b]Group B: radionuclides of Ag, Cd, Cr, Pd, Tl, La, Ce, Pr, Pm, Nd, Sm, Tb, Sn, Sb, Co, Zr, Nb, Eu, Np, Am, and Cm.

ORNL DWG 90-418

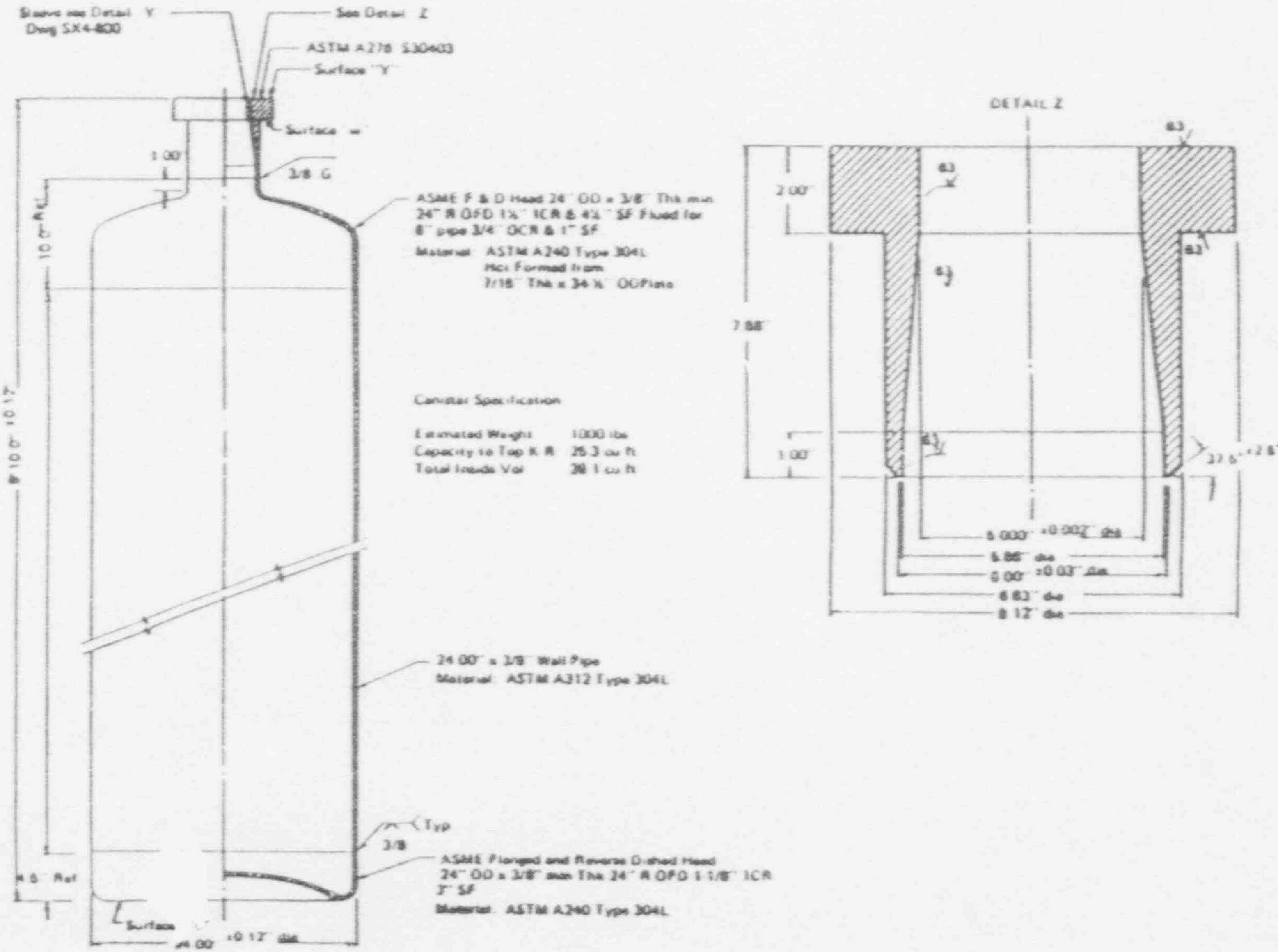

Figure 1. Savannah River Site HLW canister. Source: Ref. 1.

ORNL DWG 90-419

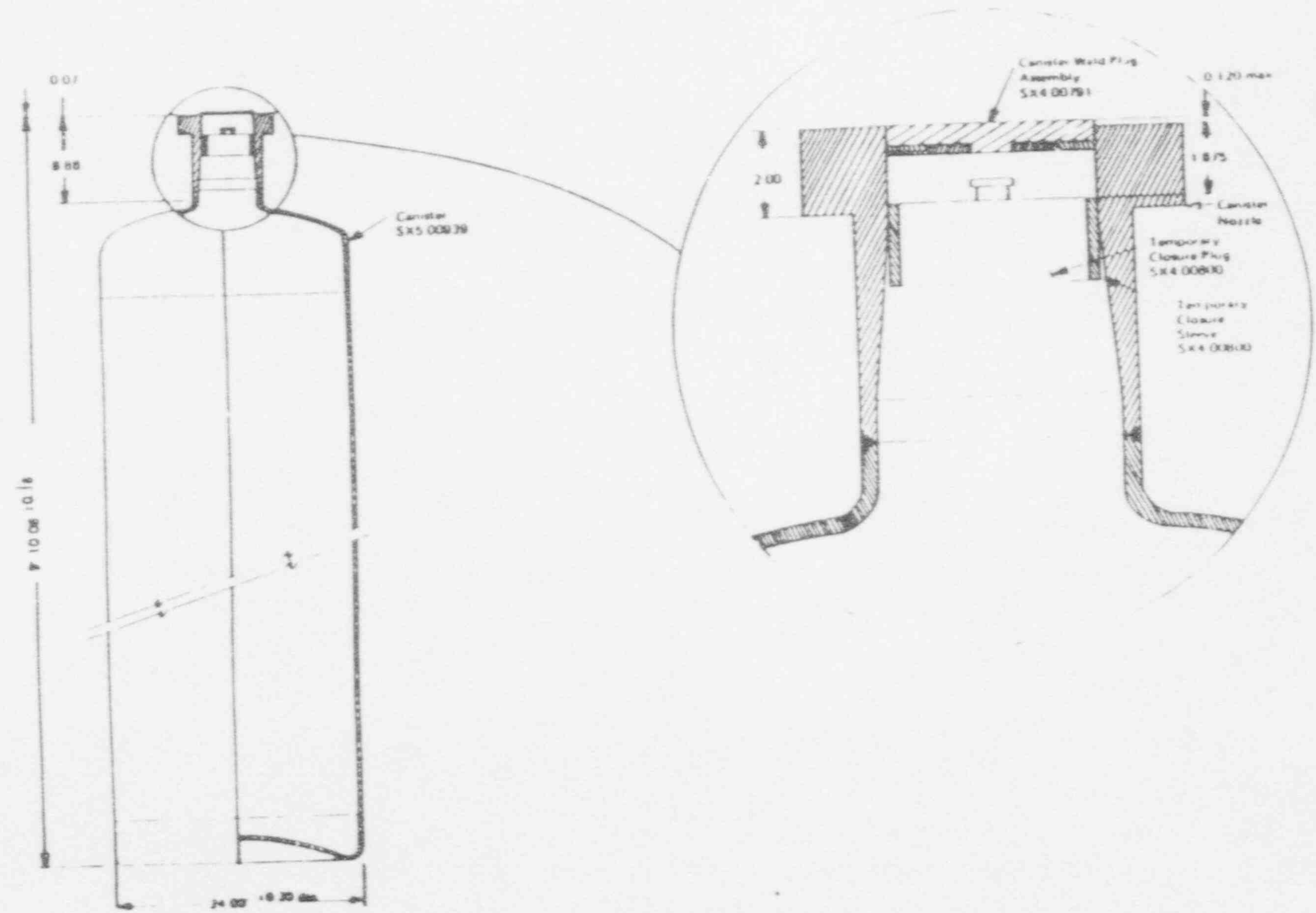

Figure 2. Savannah River Site HLW Canister Closure. Source: Ref. 1.

5. WASTE ACCEPTANCE SPECIFICATIONS

The specifications which HLW forms from each of the sites must meet in order to be acceptable for transportation to and disposal in the geologic repository have been defined in the Waste Acceptance System Requirements Document. The form and content of the documentation which each site must submit to DOE/OCRWM to demonstrate compliance with the WASRD will be defined by DOE's Office of Environmental Restoration and Waste Management (EM) since they are the responsible organization within DOE for HLW form production. As a minimum, the documentation must include a Waste Form Compliance Plan (WCP), a Waste Form Qualification Report (WQR), Production Records, and Storage and Shipping Records.[2] The WCP is a detailed description of the methods, analyses and programs which will be put in place in order to demonstrate compliance with each of the specifications defined in the WASRD. The results of the waste form testing and analyses defined in the WCP will be given in the WQR. The Production Records will describe the individual HLW canisters and their contents based on samples to be drawn during waste vitrification operations and other production documentation (e.g., welding records). The Storage and Shipping Records will describe the physical characteristics of each of the HLW canisters and contents. In addition, these records will identify any unusual events which have occurred during either interim storage or transportation.

The WASRD requires the HLW producers to establish, maintain, and execute a quality assurance program which satisfies each of the applicable criteria of the DOE OCRWM Quality Assurance Requirements and Description (QARD) document - Ref. 5. This QA program for the HLW sites will cover all activities from the time of waste form production through waste acceptance. As such, this program governs the HLW glass and canister production processes. The collection of QA records which includes the WCP, WQR, Production Records, and Storage and Shipping Records will form the basis for the acceptability of the HLW canisters for disposal in the repository.

A summary of the major specifications defined in Ref. 2 which must be adhered to in order to ensure a high quality, consistent HLW glass product and canister is given below.

5.1 BOROSILICATE GLASS WASTE FORM

Before production processing of the HLW into borosilicate glass commences, each site must make projections of the chemical composition and radionuclide inventory of the finished glass product. During actual production operations, each site must report the chemical composition and crystalline phase stability for the waste form. In addition, the oxide composition of the waste form must be reported for the oxides of elements present in concentrations greater than 0.5 wt.% based on chemical analyses of samples. The estimated total and individual canister radionuclide inventory of the glass must be defined for all radionuclides which constitute more than 0.05% of the total activity of the glass and have half-lives greater than 10 years. The material composition of the waste form must be compatible with that of the canister such that no internal corrosion of the canister takes place which would adversely affect normal or abnormal handling, storage and transport operations. One of the most important aspects of the HLW vitrification operations which must be demonstrated is adequate control over the consistency of the waste product. Each site must demonstrate control of waste form production by comparing melter batch production samples against the Environmental Assessment (EA) benchmark glass using the Product Consistency Test (Ref. 6) or equivalent. In addition, the concentrations of lithium, sodium, and boron in the leachate (after normalization for the concentrations in the glass) must be less than those of the benchmark glass. Finally, in order to preclude a nuclear criticality incident, the waste form shall be designed for criticality safety under both normal and postulated accident conditions such that the calculated effective multiplication factor after applying all uncertainties is below 0.95.[2]

5.2 CANISTER CHARACTERISTICS

The canister enclosing the vitrified HLW glass waste form must be made of austenitic stainless steel with a concentric neck and a lifting flange. Canister dimensions, weight, and glass filling height are based on the requirements of the WASRD. The canisters described in Section 4 of this document conform with these specifications, and will meet all other requirements identified in the WASRD.

The specific material composition of the canister and its components must be reported to DOE/OCRWM. This shall include the ASTM alloy specification and composition of the fill canister material, canister label material, any filler material used for welding, and the method of fabrication of the canister. Each canister is also required to have a unique alphanumeric identifier which must be clearly visible from the top and side of the canister at least until the end of the retrievability phase at the repository. The cover gases used to provide protection against corrosive processes will be helium, argon, or other inert gases. The leak rate of these gases from the outermost closure of the canister shall be less than 10^{-4} atm-cm^3/sec.[2]

5.3 FINISHED PRODUCT CHARACTERISTICS

The finished product specifications detail a wide range of requirements for the sealed canister, ranging from limits on surface dose rates and internal heat generation to drop tests. The canistered waste form shall not contain explosive, pyrophoric, or chemically reactive materials in an amount that could compromise the repository's waste isolation capability. After closure, the canister shall not contain: (1) free gases other than air, cover, and radiogenic gases; and (2) detectable amounts of organic materials. The canistered waste form must be capable of remote handling using a grapple design specified by each site, and must maintain its dimensions throughout normal handling operations encountered during storage, transportation and repository disposal. At the time of shipment, the canistered HLW form must be capable of withstanding a 7-meter drop onto a flat, essentially unyielding surface without a release of its contents. The results from this canister impact test shall include information on the measured canister leak rates and deformation.[2]

The requirement of a 7-meter drop test applies only to the canistered HLW. It should not be confused with the hypothetical accident condition tests which are specified in 10 CFR 71.73. The 9-meter drop test "onto a flat, essentially unyielding, horizontal surface" in 10 CFR 71.73(c)(1) applies to the entire package. The cask which is used to transport the HLW canisters must be certified by the NRC to withstand the 9-meter drop test.

6. COMPARISON TO REACTOR FUEL ELEMENTS

The canisters containing vitrified HLW are plutonium-bearing solids of the type that are suitable for exemption from the double containment requirement based on the justification given by the Commission in the Statement of Considerations for the exemption of reactor fuel elements. Reactor fuel elements were exempted because they are considered to be "essentially nonrespirable." The plutonium contained in reactor fuel elements is encased in a solid ceramic fuel pellet matrix surrounded by a sealed and sturdy metal cladding material which inhibits any possible dispersion of the hazardous fission products.

6.1 HLW GLASS VERSUS FUEL PELLETS

The canisters of vitrified HLW are at least as good as spent reactor fuel elements with respect to their low dispersability characteristics and their barriers to radionuclide release. The HLW glass form serves as the initial barrier to the release of plutonium and other immobilized radionuclides. Production of potentially respirable particles within the glass matrix could result from these sources: (1) cooldown processes after being poured into the HLW canister; (2) normal handling and transport operations; and (3) hypothetical accident conditions. The sources causing the production of potentially respirable particles for reactor fuel elements are identical with the exception of the first source. For reactor fuel elements, irradiation of the fuel rods causes cracking of the 1.0 - 1.5 cm. long UO_2 pellets and other dimensional changes due to the high operating temperature and pressures. Post-irradiation examination of typical fuel rods have indicated that cracks form in both the radial and longitudinal direction so that a previously whole pellet consists of 20 - 40 interlocking pieces. The sharp corners of the dished fuel pellets are often crushed into many small fragments.[20]

The durability and fracture resistance of the HLW glass form are desirable characteristics since they limit the production of fine particles potentially available for inhalation and dispersion as a result of mechanical stresses induced during postulated transport conditions. Impact studies of Savannah River simulated waste glass (SRL131 formulation) have shown comparable levels of fracture resistance and similar fractions of respirable particles when compared to unirradiated UO_2 pellets and other solid potential waste form materials. An experimental laboratory-scale brittle fracture study was conducted by Argonne National Laboratory in the early 1980's (References 7 and 8) to measure the size distribution of the impact fragments for several different simulated glass waste forms and ceramics. The impact test consisted of placing a cylindrical specimen of potential waste form on its side between two hardened tool steel plates inside a sealed chamber. Each specimen was impacted by a standard weight from a preselected height so that the available impact energy per unit volume of the test specimen was identical for comparative purposes. The size and number of particles produced by the impact were measured using a variety of techniques which were determined to give the most accurate results. The fraction of respirable particles produced by equivalent impacts (at an energy density of 1.2 J/cm^3) was determined to be 0.02 wt. % for the UO_2 fuel pellet specimens and 0.016 wt. % for the SRL131 simulated waste glass. Sensitivity studies performed at varying energy densities indicates that the respirable fraction increases linearly with increasing energy densities (drop heights). Although the extrapolation of these laboratory-scale results to larger sizes cannot be justified due to the lack of proven scaling laws, the results are useful in comparing the relative properties of the unirradiated materials. The fracture resistance of simulated HLW glass is therefore comparable to that of UO_2 fuel pellets.

Another factor which should be considered in evaluating the potential for respirability of plutonium particles is the total quantity of plutonium present in each of the waste forms. As noted in Table 1, a maximum quantity of approximately 350 grams of plutonium will be contained in the Savannah River HLW canister. Since the nominal volume of HLW glass is 6.26 x 10^5 cm^3, the concentration of plutonium in the HLW glass form is approximately 5.6 x 10^{-4} grams of plutonium per cm^3 of waste. Based on ORIGEN2 analyses of the nuclide composition expected for average burnup PWR fuel which has been allowed to decay for 5-years prior to shipment, the total quantity of plutonium per assembly is on the order of 4000 grams. This yields a plutonium concentration of approximately 7.8 x 10^{-2} grams per cm^3 of spent fuel. Thus, the concentration of plutonium in an individual fuel assembly is more than 100 times greater than that in a HLW canister. This result should be intuitively expected since the HLW was produced as a byproduct of the reprocessing of commercial and military fuels to extract useful plutonium and uranium. In addition, the maximum quantity of plutonium projected for the Hanford and West Valley HLW canisters is much less than that of the Savannah River HLW canister.

6.2 HLW CANISTER VERSUS FUEL ROD CLADDING

The HLW metal canister serves as an additional barrier to the potential release of radionuclides, including plutonium, into the interior of a HLW transport cask. The canister is similar in this respect to the reactor fuel cladding. The structural integrity of the HLW canisters has been demonstrated through numerous impact tests at Pacific Northwest Laboratories (PNL) and Sandia National Laboratories (SNL). The results of these tests are described in the following section.

Canisters of vitrified HLW are required to be capable of withstanding a drop of 7-meters onto a flat, essentially unyielding surface, as specified in the WASRD. The drop test analyses will be documented in the WCP and WQR for each of the HLW sites. Final closure weld controls and procedures for the HLW canisters will also be described in detail in the WCP, WQR, and Production Records which are required to be submitted to DOE/OCRWM (per the WASRD - Ref. 2) by each interim HLW storage site.

Another factor contibuting to the integrity of the HLW canisters, as compared to the spent reactor fuel cladding, is that the metal canisters have not been exposed to the high levels of radiation existing in a commercial power reactor. High levels of radiation have been shown to cause slight molecular-level changes in material properties, including increased embrittledness. The protection that the canister provides should be at least comparable to that provided by spent fuel cladding

7. CANISTER DROP TESTS

A number of drop tests using canisters containing simulated HLW glass forms have been performed over the past 20 years. These tests have yielded important information on the durability of the canisters and the fracture resistance of the simulated HLW glass form. This provides further support for the position that canisters containing vitrified HLW are a waste form that qualifies for exemption from the double containment requirement. A summary of the results of drop tests of bare scale-model or full-size canisters containing simulated HLW onto an unyielding surface is given in Table 4.[3] The testing program included a number of different canister and glass combinations with variations in the drop orientation and the angle of impact. There was no release of the glass material from any of the canisters as a result of the impact. Helium leak tests and dye penetrant tests conducted following the impacts have shown the ability of these canisters to withstand an impact (9-meter drop) greater than that required by the WASRD (7-meter drop) with no penetration of the canister shell.

7.1 CANISTER INTEGRITY

A total of at least 13 different canisters made of the reference Type 304L stainless steel have been drop-tested from a height of 9 meters (~ 30 feet) in various studies sponsored by the Department of Energy (References 9 - 13). None of the canisters showed any observable evidence of rupture or cracking. Ten of the canisters of the reference dimensions identified in the WASRD were leak-tested with helium following the drop tests with no leakage detected greater than the instrument sensitivity (2.4×10^{-10} std-m^3/sec/scale division). In addition, nine of these ten canisters were subjected to a dye penetrant examination in the area of the damage zones to detect for any non-visible cracks or flaws. No defects were observed based on these tests. Two of the studies measured strain levels by the use of strain circles etched on the canisters and measured before and after the impacts. Maximum strain levels of 12 - 16% were observed, well below the 55% strain level at which Type 304L stainless steel begins to exhibit signs of failure.[6] Detailed descriptions of the testing methodology and results can be found in References 9 - 13.

7.2 HLW GLASS PARTICLES

Of critical importance in the scientific/technical basis for the exemption of the HLW canister from the double containment requirement is the evidence that the waste form is "essentially nonrespirable". Several studies have performed detailed analyses on the particle size distribution generated as a result of the maximum mechanical forces that would be imposed on the HLW canister during shipment (i.e., following the canister drop tests). Despite the mechanical impact forces exerted upon the HLW canister during both normal transport and postulated accident conditions, nearly all the glass waste form remains in a solid form of large shards with limited generation of small particles.

7.2.1 MCC-15 Test And Analyses

The Materials Characterization Center (MCC) at PNL has been developing standard tests to characterize the performance of nuclear waste forms under both normal and accident conditions. As a part of this effort, the MCC has developed "MCC-15: Waste/Canister Accident Testing and Analysis" (Ref. 11). This test method was developed in order to provide data on HLW canister integrity, deformation, and waste form particle size distribution following a free drop impact under standard accident conditions. As a part of the verification of the test method, two separate tests were performed using one full size prototypic Savannah River DWPF canister. The canister was filled according to the DWPF reference process (Ref. 15). The canister was dropped from a height of 9 meters onto its bottom corner at an angle that placed the canister's center of gravity over its corner. Following the impact and subsequent evaluation of canister integrity, the canister was disassembled and the waste form removed to determine the particle size distribution. The results of this screening and sieving process (described in Ref. 11) yielded about 50 g. of particles of respirable size (smaller than 10 μm). Since the DWPF canister contains approximately 1680 kg. of waste glass with a maximum plutonium activity of 3176 Ci., the 50 g. of respirable fines corresponds to a plutonium activity of less than 0.10 Ci. The quantity of respirable particles contained within the intact canister as a result of the impact corresponds to 0.003 wt. % of the total canister waste form mass; this fraction is considerable less than that observed in the laboratory-scale impact tests (0.016 wt. %).

7.2.2 Other PNL Impact Tests and Analyses

The MCC at PNL performed additional impact testing and analyses which were completed in December 1988 (Reference 12). Two full-scale DWPF canisters filled with reference simulated borosilicate glass waste were impacted under either normal (0.3 m. vertical drop) or accident conditions (9.1 m. vertical drop). The goals of these series of tests were: (1) to generate data on waste fine generation as a result of the impacts; and (2) to measure particle release through holes which were intentionally drilled into the canisters to estimate the effect of an artificially imposed and arbitrary worst case event.[12]

Four holes with a diameter of 0.28 cm. were intentionally drilled into the shell of the canister which had been dropped from a height of 0.3 m and the canister was subsequently transported over 2000 miles to simulate actual transport conditions. Three of the holes were located in the canister impact area (canister bottom) to provide conservative (maximum) estimates of the mass and size distribution of fines released from potential canister flaws. The quantity and size of the fines released through these manufactured flaws was measured carefully using filter assemblies attached to each hole. The canister was placed in a horizontal orientation inside a wooden box which was placed on the bed of a tractor trailer for the highway round trip from the Hanford Site to Cheyenne, Wyoming and back. During this transportation flaw leak testing, between 0.1 and 260 mg. of glass particles exited each hole. However, only 0.044 to 12.3 mg of these glass particles were in the respirable size range (i.e, with a diameter smaller than 10 μm).

Following the transportation flaw leak testing of the normal condition canister (0.3-m. drop), pressurized flaw leak testing was performed on both impacted canisters. The purpose of this test was to determine the mass and size distribution of glass fines that would exit a design flaw in a pressurized canister following the free drop impact test. For the canister which had been subjected to the 0.3-m. drop, the four holes that had been created in the transportation flaw leak test were welded closed. For both DWPF canisters, four holes were again drilled into the wall of each canister with all holes located in the bottom 7.5 cm. of the impacted canisters. The holes were then closed and the internal plenum of the canister was pressurized to a predetermined level. The holes were then opened and the canister was allowed to depressurize. During depressurization, the glass waste particles exiting each of the holes was collected and measured in terms of quantity and size. The variations in testing conditions and results are as follows:

(1) For the normal condition canister, the diameter of the four holes was 0.28 cm., and the canister was pressurized to 2.0 psig. All particles exiting the artificially created canister flaws were collected and the particle size distribution determined. Only 1.2 to 2.9 mg of glass fines were detected exiting these holes, with only 0.78 to 1.9 mg being less than 10 μm in diameter.

(2) For the accident condition canister (9.1-m. drop), the diameter of the four holes was 0.95 cm. and the canister was pressurized to 3.0 psig. An analysis of the quantity and sizes of particles exiting these flaws during depressurization yielded a total quantity of 79 to 333 mg. per hole, with the maximum quantity of respirable particles released through any one hole being 3.04 mg.

Upon completing all the leak testing, each of the canisters was disassembled and the particle size distributions of the entire canister were measured. The quantity of particles in the respirable category (< 10 μm) was 61 g. (0.004 wt.%) for the normal conditions canister dropped from 0.3 m., and 239 g. (0.014 wt.%) for the accident condition canister. The total activity of plutonium for the largest quantity of particles produced (i.e., 9. m. free drop and fine sizes < 10 μm) corresponding to 239 g. is 0.45 Ci.[3] Since the total quantity of respirable fines was approximately 4 times greater for the canister dropped from 9.1 m. and these fines were concentrated in the bottom (impacted) area of the canister, the majority of the respirable fines for the 9.1 m. impacted canister were a result of the impact.

A summary of the particle size distributions for both the PNL normal conditions canister drop and the 9.1 m. (30-ft.) drop test is given in Fig. 3, along with similar results from the MCC-15 test described in 7.2.1 and other past impact test data. The similarity of the results of these different tests confirms the consistency of the particle size results.

The quantity of respirable fines produced is minute and provides full support for the classification of the HLW canister waste form as "essentially nonrespirable", analogous to reactor fuel elements. An evaluation of the significance of the total quantity of respirable particles contained within the canister as a result of the 9.1 m drop can be performed by comparison to the methodology used for a corresponding safety analysis for spent fuel transport. For shipping cask safety assessments, it is assumed that 0.003% of the spent fuel is considered to be released as a result of cladding failure of a single fuel rod. Furthermore, the portion of the fuel particles ejected from the fuel which are potentially respirable as a result of cladding failure is approximately 10%.[16] These assumptions are based on experiments conducted at Battelle and Oak Ridge National Laboratory. If similar conservative assumptions are made for the HLW canister, then 0.007 g. of respirable HLW glass particles (0.00003 * 239 g.) are assumed to escape from the canister into the interior of a transportation cask. This amount corresponds to a plutonium activity of 1.4×10^{-5} curies, or 5×10^{-3} A_2 values. The assumption that 0.003% of the available HLW glass particles escape into the interior of the shipping cask cavity during a postulated breach is borne out by similar magnitude results from the PNL drop tests. A total of 0.0013% of the respirable size glass particles escaped from the holes which were intentionally drilled into the canister body (3.04×10^{-3}g/239 g). Since the HLW canisters are robust, it is reasonable to conclude that any HLW glass particles that are produced in the glass matrix will be retained within the canister. A number of independent drop tests have confirmed the structural integrity of the canisters following extra-regulatory impacts. During actual transport conditions, the HLW canister will be enclosed within a shipping cask -- resulting in reduced canister damage and minimal production of HLW glass fines.

Table 4. Summary of DWPF HLW Canister Drop Tests

TEST REPORT	CANISTER NUMBER	SCALE	TEST NUMBER	DROP WEIGHT AND ORIENTATION	GLASS PROPERTIES	TEST RESULTS: DYE PENETRANT	TEST RESULTS: HELIUM LEAK
Peterson and Alzheimer PNL-5250	1-4 (18-21) Each of 4 canisters tested 3 times.	1/1	1	9 M. Bottom Corner	Ref. Borosilicate Glass Waste (1983)		
			2	1 M. Side Puncture			
			3	9 M. Nozzle Corner		No Cracks	No Leaks
Peterson, Alzheimer, Slate PNL-5251	1 (4)	1/1	1	9 M. Nozzle Corner	Ref. Glass	---	No Leaks
	2 (5)		1	9 M. Bottom Corner			
			2	1 M. Side Puncture		No Cracks	No Leaks
Slete, Pulsipher, Scott PNL (Waste Management 1985 Paper)	1	1/1	1	9 M. Bottom Corner	Ref. Glass	No Cracks	No Leaks
Farnsworth and Mishima PNL-6379	1 (A27)	1/1	1	0.3 M. Bottom	Frit 165 (Ref.) @ Pour Rate of 240 lb/hr.	No Cracks	No Leaks
	2 (A10)	1/1	1	9 M. Bottom		No Cracks	No Leaks
Uncapher, Madsen, Stenberg (SAND87-2516)							
· Tested Bare	1	1/2	1	9 M. Bottom	Frit 165 @ Ref. Pour Rate	Visual Inspection Revealed No Flaws	
	2	1/2	1	9 M. Top at -20°F		Visual Inspection Revealed No Flaws	
· Tested Inside 1/2 Scale Cask	0	1/2	1	9 M. Bottom		Visual Inspection Revealed No Flaws	
	1	1/2	2, 3	9 M. Top Puncture		Visual Inspection Revealed No Flaws	
	2	1/2	4, 5	9 M. Side Puncture		Visual Inspection Revealed No Flaws	
	3	1/2	6, 7	9 M. C. G. Bottom, Side		Visual Inspection Revealed No Flaws	

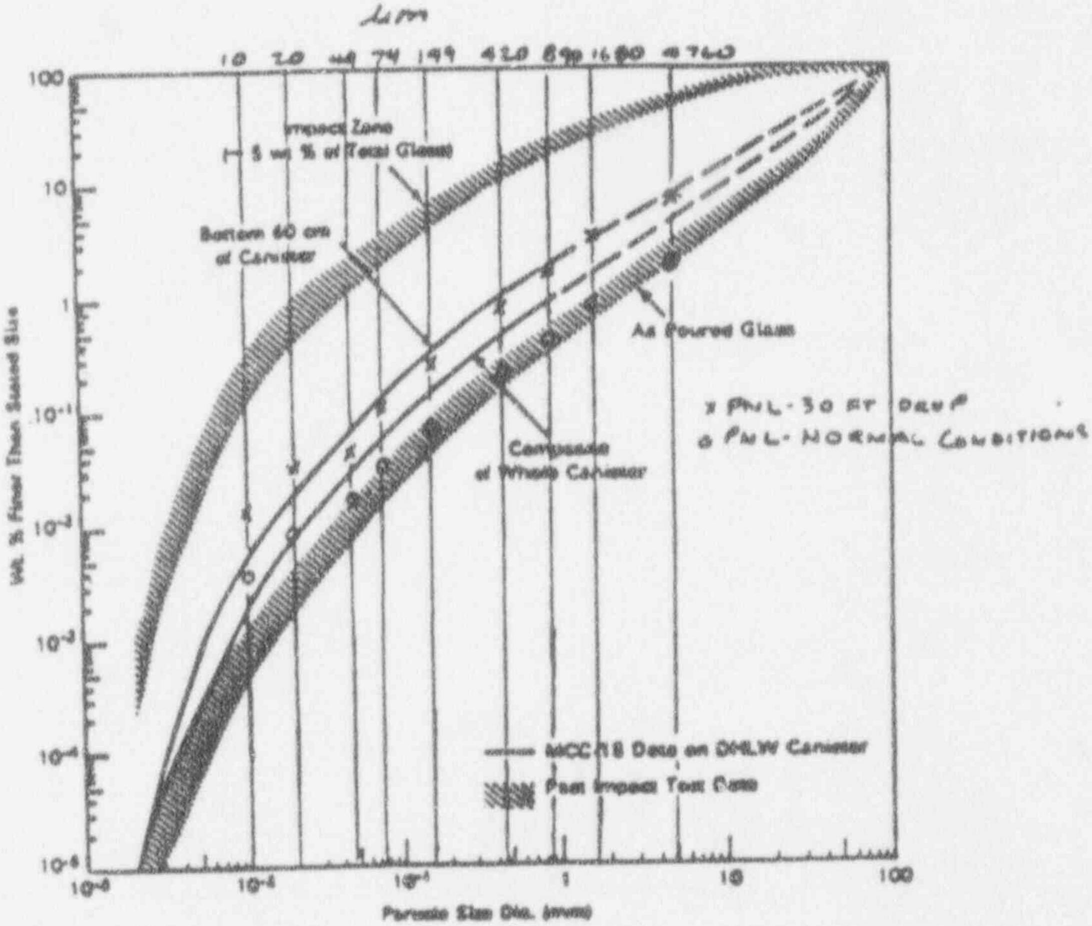

Figure 3. Summary of Impact Particle-Size Distribution Results

8. PROGRAMMATIC JUSTIFICATION

8.1 ALTERNATIVES

The only alternative would be to initiate efforts to design a double containment transportation cask. Technically, this could be accomplished. However, it is unnecessary from a safety perspective since the plutonium embedded in the solid glass waste form is essentially nonrespirable and a level of containment is provided comparable to that provided for spent fuel elements (which only require single containment). The technical basis for a double containment requirement for the borosilicate glass HLW logs during transport from the HLW producer sites to the geologic repository for permanent disposal is unsupported. A double containment requirement for HLW in the form of borosilicate glass contained within sturdy metal canisters would impose an unnecessary and onerous burden on the DOE as discussed below.

8.2 OPERATIONAL CONSIDERATIONS

The design of a cask for double containment would add additional handling steps to the loading and unloading of the HLW canister, increasing the time (and resultant expense) required to return the cask for the next shipment. The operating efficiency of the CRWMS transportation program is reduced by these unnecessary handling steps. In addition, this would increase the total radiation dose received by workers loading and unloading the cask. One of the goals of the CRWMS is to keep radiation exposure to a level that is ALARA (As-Low-As-Reasonably-Achievable). The addition of a second level of containment would also create more activated metal hardware which must be disposed of.

Another major factor which must be considered is the potential reduction in payload capacity caused by the extra volume and weight of a second containment level. If more than one HLW canister is to be transported in a single shipping cask (as is likely for a rail cask), then the payload of double containment HLW canisters would likely be reduced. This, in turn, would negatively impact the operating efficiency of the CRWMS by creating the need for more shipments and a corresponding increase in risk to affected populations along the transportation corridor.

8.3 ECONOMIC CONSIDERATIONS - Life Cycle Cost Impact

If the double containment requirement is enforced for this waste form, the total system life cycle cost of transporting these HLW canisters will be affected due to the added cost of: (1) the material comprising the added containment barrier; (2) the labor resulting from the extra handling steps in the loading and unloading of the canister from the transportation cask; and (3) the added number of shipments caused by a potential decrease in payload capacity of the cask. The added costs incurred by the inclusion of an additional level of containment can not be justified in terms of any incremental benefits to public health and safety.

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