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Preliminary High Level Waste Conceptual Cask Designs and an Assessment of a Clad Waste Cask

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PRELIMINARY HIGH LEVEL WASTE CONCEPTUAL CASK DESIGNS AND AN ASSESSMENT OF A CLAD WASTE CASK

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ABSTRACT

Neutron and gamma-ray transport calculations in one- and two-dimensional cylindrical geometry have been performed for a number of high-level waste and cladding waste conceptual cask shield designs. Stainless-steel-lined depleted uranium and lead were the primary gamma-shield materials considered in this study. The thicknesses of these shield-material regions were varied until the total dose rate, at 91.44 cm (3 ft) from the cask surface, was less than 1000 mrem/hr for accident environments. Neutron-shield materials including borated beechwood, water, serpentine, and calcium borate were then added to the cask surface until the total dose rate, at 1.83 m (6 ft) from the cask surface, was less than 10 mrem/hr for normal operating conditions. It was determined that the best combination of these shield materials, based on overall cask dimensions was depleted uranium and borated beechwood.

This work was funded by the Office of Waste Isolation (OWI).

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PRELIMINARY HIGH LEVEL WASTE CONCEPTUAL CASK DESIGNS AND AN ASSESSMENT OF A CLAD WASTE CASK

1. Introduction

In the event recycling of nuclear power plant spent fuel is initiated, shipping containers will be needed to transport the byproduct waste from a reprocessing facility to a permanent storage site. As the conceptual design work on such a facility proceeds, it is important that concurrent conceptual high level waste (HLW) shipping cask design be conducted to insure the availability of a fleet of casks of adequate capacity which interface efficiently with the reprocessing plant as well as the disposal facility. Previous design work on shipping containers[1,2] has identified a variety of problems and provided extensive data on selected candidate cask materials.

The objective of the present study is to use the concepts from related cask designs along with updated estimates of radiation and thermal source strengths and waste concentrations as a basis for establishing new HLW shipping cask conceptual designs. These casks would be used to transport solidified HLW and clad waste expected from nuclear-fuel reprocessing plants. The new conceptual designs provide needed detail to bridge the gap from scoping analyses to the engineering design of shipping containers. The conceptual designs are presently based only on radiation-shielding calculations and therefore do not include any structural or thermal analysis.

Funding for this work was provided by the Office of Waste Isolation (OWI).

2. Description of High-Level Waste and Cladding Waste

High-level waste and cladding waste are generated as byproducts in the reprocessing of reactor spent fuel. Following Kee et al,[3] the reprocessing procedure entails chopping the spent fuel rods and separating the remaining fuel and fission products from the cladding hulls and fuel assembly structure by acid leaching. Chemical-separation techniques are then used to remove most of the fissile material from the acid solution. The resulting waste material contains nearly all the fission products, roughly 0.5% of the U and Pu originally present in the spent fuel, and ~99.5% other actinides generated during the irradiation. Federal regulations require that these wastes be solidified within a period of 5 years from the time the spent fuel is reprocessed and then shipped to a government repository within 10 years after reprocessing. In either its liquid or solidified form, these wastes are termed high-level waste. A current plan for the disposition of HLW includes solidification in a borosilicate glass.

Until a current generation reprocessing plant is in operation, accurate data concerning the concentration of the waste following solidification are not available. Therefore, for present purposes, the predictions of Reference 3 have been used to define the radiation and thermal characteristics of HLW. Under these assumptions, the waste resulting from reprocessing one metric ton of heavy metal (MTHM) charged to a light-water reactor (LWR) is assumed to be solidified in 0.08 m^3 (3 ft^3) of glass, the upper limit specified in Reference 1. Calculation of the heat source in LWR spent fuel is reasonably reliable using current methods and data; however, Schmittroth[4] has estimated the uncertainty in decay heat predictions for nominal LWR spent fuel to be approximately ±7%. Unfortunately, the accuracy of neutron and gamma-ray source predictions for LWR spent fuel is neither as good nor as quantified as the thermal source.

Figure 2-1 illustrates a potential container design for the solidified HLW.[5] The steel pipe container would provide some protection for the glass during handling and transport. Conceptual container designs are 3-m-long cylinders with diameters ranging from 30.48 to 50.8 cm filled with 2.4 m of glass.

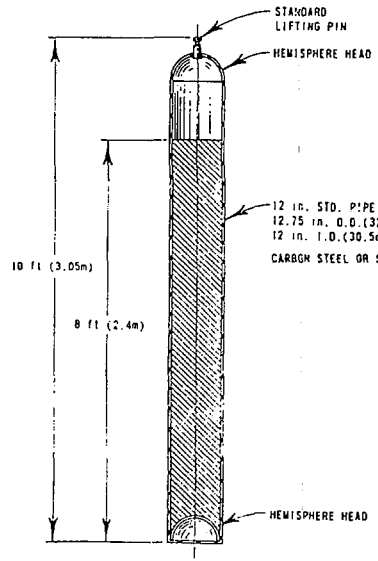

Figure 2-1. Potential Design for High-Level-Waste Canister

The current lifting fixture arrangement has some potential weaknesses. In an accident during shipment, it is conceivable the HLW canister could travel forward and impact against the cask lid. Since the waste cylinder weighs in excess of 680 kg (1500 lbs), there is a potential for damaging the lifting fixture, the cask lid, or the waste canister. Various means of protecting the lifting pin from direct contact with the cask lid have been considered. One possible method of reducing potential damage is by inserting individual collars (that would fit inside the cask lid) on top of each canister (as shown in Figure 2-2). The disadvantage of this concept is that there would be additional pieces of hardware to handle during loading and unloading of the cask. A preferable alternative, at least from the shipping viewpoint, is to make the collar an integral part of the waste canister. This concept, which is illustrated in Figure 2-3, provides protection for the lifting fixture, not only during transport, but in all handling operations. If the collar were slotted, it would provide an additional means of lifting the canister. Although the exact configuration of the waste canister is not essential in this cask-shielding design work, it is important to consider problems associated with the present canister design.

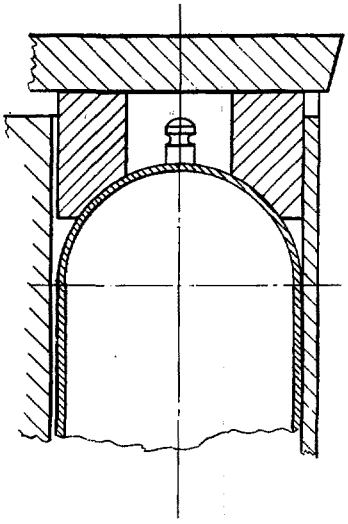

Figure 2-2. High-Level-Waste Canister with Separable Collar

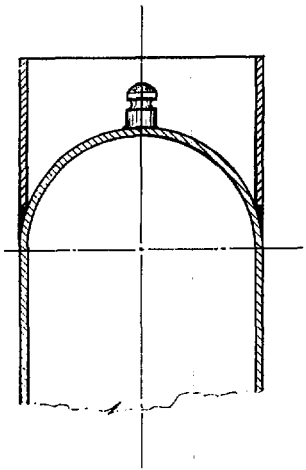

Figure 2-3. High-Level-Waste Canister with Attached Collar

Since HLW generates appreciable quantities of heat, there is a limitation on the maximum diameter suitable for the waste cylinder for a given dilution of HLW in glass. If the diameter is too large, the glass on the canister centerline will melt. Analysis indicates the maximum allowable diameter for cylinders of borosilicate glass with HLW from 1 MTHM/0.085 m^3(3 ft^3) glass is 50.8 cm (20 in.) unless cooling fins or other heat transfer techniques are used in the canister interior.[6]

The diameter of the waste cylinder also affects the overall size of the HLW shipping cask. Hexagonal packing is the optimal packing configuration for waste in a cylindrical cask cavity. If waste radius dimensions are limited to a range 30.48 cm (12 in.) to 50.8 cm (20 in.), the best canister arrangement consists of seven individual containers (six outer canisters surrounding a central canister). When a minimum spacing of 3.8 cm (1.5 in.) is maintained between individual cylinders in the cask, and 1.3 cm (0.5 in.) between the cavity wall and the nearest waste cylinder, seven 30.48-cm-diameter canisters fit in a 108-cm-diam cavity; seven 40.64-cm cylinders in a 132-cm cavity; and seven 50.8-cm-diam cans in a 163-cm cavity. These canister dimensions conform to standard steel pipe sizes. Table 2-I identifies the pipe sizes considered and lists the weights of the HLW and steel involved for each size. In order to determine shielding requirements, the cask design capable of transporting seven 30.48-cm-diam canisters in a 107.4-cm-diam cavity was examined in detail. An aluminum basket for supporting the HLW canisters in the cask cavity was also considered in the design.

TABLE 2-I

High-Level Waste Canister Weights

	30.5 cm SCH 40 pipe	40.6 cm SCH 30 pipe	50.8 cm SCH 30 pipe
	Weight (kg/canister)		
Steel	225	284	472
Waste	570	920	1430
Total	795	1204	1902

The neutron and gamma-ray sources associated with the HLW are presented in Table 2-II. The present generation sources result from analytical work performed at Oak Ridge and made available to Sandia by the Office of Waste Isolation. The uranium and plutonium recycle sources were generated at Sandia utilizing the ORIGEN code.[15] The energy spectra used were chosen to correspond to energy groups in existing neutron and gamma-ray cross-section libraries. The source strengths are for waste resulting from reprocessing of spent fuel 1 year after removal from the reactor and allowing the resulting waste to age for 4 years. Present generation HLW refers to waste obtained by reprocessing enriched U fuel which has been used for only one cycle in the reactor. Uranium recycle HLW is the waste obtained from spent fuel containing uranium previously extracted for recycle. Similarly, plutonium recycle corresponds to waste from spent fuel which contains both uranium and plutonium previously recycled from LWR spent fuel. While the majority of this work considers only present generation waste, uranium and plutonium recycle HLW sources are examined to determine if shipment of these materials poses any unique problems.

The source strengths in the table actually represent a 2:1 mixture of waste from the reprocessing of pressurized water reactor (PWR) and boiling water reactor (BWR) spent fuel to approximate the current proportion of power reactor types in this country.

Cladding waste (CW), another material generated in reprocessing spent fuel elements, is also a radiation source. CW contains the fuel-element structural materials comprised of Inconel, stainless steel, and Zircaloy which remain in the fuel dissolver after the removal of fuel and HLW. Cladding waste is assumed to contain approximately 0.05% of the actinides and fission products originally in the spent fuel.[3] A significant portion of the source is also due to induced radioactivity. Federal regulations require the storage of CW at a federal repository within 10 years after its generation. Since the radiation source of CW is large enough to require biological shielding, some preliminary work was performed to determine the shielding required for a basic shipping-cask design necessary for transportation from the reprocessing plant to a federal repository. Prior to shipment, the cladding waste would be canistered in a steel pipe after being compressed to approximately 70% of its theoretical density. Using this compression, the reprocessing of 1 MTHM would result in about 0.07 m^3 (2.5 ft^3) of CW. Table 2-III indicates the radiation source associated with this waste.

TABLE 2-II

Five-Year-Old HLW Gamma and Neutron Source per One-Metric-Ton Heavy Metal Charged to a Light-Water Reactor

Gamma-Ray Group Average Energy (MeV)	Photon Source/MTHM by Group: Present Generation (gammas/s)	Uranium Recycle (gammas/s)	Plutonium Recycle (gammas/s)
3.25	1.25+9	2.42+9	3.93+9
2.75	3.97+10	7.65+10	1.24+11
2.38	5.09+11	9.85+11	1.58+12
1.99	2.14+12	4.84+12	5.18+12
1.55	3.95+13	5.55+13	5.56+13
1.10	2.53+14	2.88+14	3.87+14
0.63	6.46+15	7.66+15	7.84+15
0.30	2.54+14	3.57+14	3.65+14

Neutrons Energy Range (MeV)	Neutron Source/MTHM by Group: Present Generation (neutrons/s)	Uranium Recycle (neutrons/s)	Plutonium Recycle (neutrons/s)
3.01-10	4.47+7	3.79+7	1.23+9
1.83-3.01	5.12+7	4.39+7	1.43+9
1.11-1.83	4.82+7	4.09+7	1.33+9
0.55-1.11	4.29+7	3.65+7	1.18+9
0.111-0.55	2.69+7	2.29+7	7.43+8
0.053-0.111	4.60+6	3.91+6	1.27+8

TABLE 2-III

Five-Year-Old Cladding Waste Gamma and Neutron Source per One-Metric-Ton *Heavy Metal Charged to a Light-Water Reactor*

Gamma-Ray Group Average Energy (MeV)	Photon Source/MTHM by Group: Present Generation (gammas/s)	Uranium Recycle (gammas/s)	Plutonium Recycle (gammas/s)
3.25	1.25+6	2.41+6	1.95+6
2.75	3.94+7	7.59+7	6.20+7
2.38	5.07+8	9.84+8	7.92+8
1.99	2.43+9	5.50+9	2.59+9
1.55	2.86+10	4.01+10	2.83+10
1.10	1.02+14	1.16+14	9.68+13
0.63	5.01+12	5.94+12	4.94+12
0.30	2.03+11	2.74+11	1.95+11

Neutrons Energy Range (MeV)	Neutron Source/MTHM by Group: Present Generation (neutrons/s)	Uranium Recycle (neutrons/s)	Plutonium Recycle (neutrons/s)
3.01-10	2.35+4	1.99+4	7.32+5
1.83-3.01	2.71+4	2.30+4	8.47+5
1.11-1.83	2.53+4	2.14+4	7.90+5
0.55-1.11	2.25+4	1.91+4	7.04+5
0.111-0.55	1.41+4	1.20+4	4.42+5
0.053-0.111	2.42+3	2.05+3	7.54+4

3. Radiation Shields and Materials

HLW is a concentrated source of neutron and gamma radiation; shielding is required to reduce the dose rate to less than specified values at given distances from the containers carrying the waste.[7] These limits for the total dose rates are 10 mrem/hr at 1.83 m (6 ft) from the accessible surface of the cask under normal operating conditions; and 1000 mrem/hr at 91.4 cm (3 ft) from the cask under accident conditions. Since the interaction of neutrons with matter is an entirely different physical phenomenon than that of gamma rays, it is apparent that materials used for shielding one type of radiation will be of marginal value for the other type of radiation.

In general, the best practical materials for shielding gamma rays are dense and contain significant proportions of high-Z elements. This is evident since gamma rays in the energy range of interest (hundreds of keV to a few MeV) primarily interact by Compton scattering with the electrons rather than directly with the nucleus of an atom. Within this photon energy range, pair production will also be important for some elements. Examples of excellent gamma-ray shield materials are uranium, lead, and (to a lesser extent) iron. When the most effective shield is defined as the thinnest shield needed to accomplish attenuation, the best shield material is uranium which is approximately 60% more effective than lead of equal thickness and more than twice as effective as iron. However, the higher cost and fabrication problems of uranium compared with alternative shields may offset this size advantage in some applications.

Shielding of neutrons is best achieved by utilizing low-Z materials to moderate the neutrons and neutron absorbing materials to eliminate them. Physically, neutrons lose energy by elastic and inelastic collisions with atomic nuclei. The amount of energy transferred in each collision is a function of the scattering nucleus mass with the greatest fraction of energy being exchanged when the nucleus and the neutron have similar atomic masses. It is apparent that the interaction of neutrons with hydrogen is the most effective means of slowing epithermal neutrons. Any material containing significant amounts of hydrogen is an excellent neutron-shield candidate. These materials include water, wood, organic resins, and metal hydrides, all of which are about equally efficient in slowing neutrons. To remove moderated neutrons, a neutron absorber such as boron may be included in the hydrogenous material. A difficulty associated with this is that the absorption of the neutron in some cases results in the production of secondary gamma rays as the excited nuclei decay. These gamma rays must also be considered in the overall shielding scheme.

A serious drawback of hydrogenous shields is that nearly all of them tend to lose significant fractions of their hydrogen content at elevated temperatures. None, with the exception of some hydrides, retains more than a few percent of its original hydrogen content above 1000°F. As the hydrogen leaves the materials, the neutron-shielding capabilities of the remaining material is significantly reduced.

Another material which is reasonably effective as a neutron shield is boron carbided (B_4C). This material contains a great deal of the fairly low-Z elements: boron and carbon. Although not as effective as hydrogen in neutron moderation, these elements are present at a high particle number density which tends to offset their moderating deficiencies. The shielding efficiency of B_4C is also unaffected by even extremely high temperatures. Finally, the material has a large effective removal cross-section for fast neutrons as well as high thermal and epithermal neutron absorption cross-sections with low secondary gamma emission.[1]

The most effective shield design for neutrons and gammas together consists of a number of layers of shield material with each intended for a specific purpose (compound shield). The layer closest to the radiation source in a compound shield may contain high-Z material to attenuate most of the primary gamma rays and to aid in moderating neutrons by elastic and inelastic scattering. Uranium is an excellent choice since it possesses such a high atomic number and has a reasonably large neutron scattering cross section. The second layer could contain a low-Z material to slow the neutrons and a high neutron-absorption cross-section material to remove them. A third shield layer of high-Z material would reduce the dose rate from secondary gammas. In practice, a two-component shield is sufficient to meet basic shielding requirements.

Candidate materials for the gamma shield considered in this study were depleted uranium (0.3% ^{235}U by weight), lead, and iron. Neutron shields considered included borated beechwood, water, boron carbide, serpentine, and calcium borate. Borated beechwood with the trade name Permali JN is a commercially available product[8] which has been used extensively in the nuclear industry. Boron carbide has also been used in the nuclear industry, but its proposed use as a shield in shipping casks is a new application.[2] Serpentine rock[9] is a naturally occurring asbestos mineral which retains its hydration to temperatures as high as 950°F. Calcium borate[9] is the commercial name of a number of borated calcium minerals pressed into an asbestos matrix.

4. Calculation Description

One- and two-dimensional radiation transport calculations were performed on a series of candidate cask-shield designs to determine the shield requirements of HLW and CW. ANISN[10] and DOT[11] were the computer codes used. ANISN is a one-dimensional, discrete ordinates, transport code with anisotropic scattering capabilities, suitable for one-dimensional slab, cylindrical, and spherical geometries. DOT is a two-dimensional S_n code with applications in x-y, r-z, and r-θ geometries. Both codes solve the Boltzmann Transport Equation by making discrete the energy, space, and angle variables.

The analysis was performed in two parts: a calculation for the primary gamma-ray shielding problem and a calculation for the coupled neutron-secondary gamma problem. The primary gamma calculation was performed using eleven energy-group P1 cross sections generated by the computer code GAMLEG.[12] The coupled neutron/gamma calculation was done using 19 neutron energy groups and 13 gamma groups. The P1 cross-section library in this case was collapsed from the

DNA Few-Group Library.[13] Tables 4-I and 4-II list the energy structure used in each analysis. Included are the tissue dose rate conversion factors employed for each energy group.

TABLE 4-I

Coupled Neutron/Gamma Cross-Section Library

Group Structure		Dose Rate Conversion Factor
Group	Upper Energy (MeV)	mrem/hr/particle/cm^2/s
	Neutrons	
1	10	0.1447
2	3.01	0.1268
3	1.83	0.1201
4	1.11	0.1162
5	0.55	0.0568
6	0.111	0.01787
7	5.25 (-2)	9.893 (-3)
8	2.48 (-2)	6.912 (-3)
9	2.19 (-2)	4.969 (-3)
10	1.03 (-2)	3.601 (-3)
11	3.35 (-3)	3.694 (-3)
12	1.23 (-3)	3.790 (-3)
13	5.83 (-4)	4.015 (-3)
14	1.01 (-4)	4.275 (-3)
15	2.90 (-5)	4.451 (-3)
16	1.07 (-5)	4.522 (-3)
17	3.06 (-6)	4.482 (-3)
18	1.13 (-6)	4.317 (-3)
19	4.14 (-7)	3.724 (-3)
	Gamma Rays	
1	14	8.991 (-3)
2	8.0	7.534 (-3)
3	7.0	6.898 (-3)
4	6.0	6.155 (-3)
5	5.0	5.388 (-3)
6	4.0	4.578 (-3)
7	3.0	3.625 (-3)
8	2.0	2.920 (-3)
9	1.5	2.308 (-3)
10	1.0	1.736 (-3)
11	0.7	1.125 (-3)
12	0.3	5.297 (-4)
13	0.1	5.822 (-4)

TABLE 4-II

Primary Gamma-Ray Cross-Section Library

Group Structure		Dose Rate Conversion Factor
Group	Upper Energy (MeV)	(mrem/hr/photon/cm^2/s)
1	3.5	4.36 (-3)
2	3.0	4.00 (-3)
3	2.6	3.71 (-3)
4	2.2	3.24 (-3)
5	1.8	2.77 (-3)
6	1.35	2.30 (-3)
7	0.9	1.51 (-3)
8	0.4	0.83 (-3)
9	0.2	0.36 (-3)
10	0.1	0.37 (-3)
11	0.01	0.37 (-3)

The primary objective of these calculations was to determine the neutron and gamma-ray dose rates at certain distances from the cask surface and adjust the thicknesses of the various shields until a satisfactory dose rate resulted. Cask operation under both normal and accident conditions must meet the dose rate requirements. In this work, normal operation conditions are defined as the cask loaded to its design capacity with HLW or CW and all the shields in place. Accident conditions were modeled by assuming the cask had lost all perishable shielding but nonperishable shielding was not damaged. Water was treated as perishable and, pending further evaluation, the borated beechwood was also assumed to be perishable. Obtaining this objective entailed initially satisfying the accident cask condition by adding primary gamma-shield material to the primary gamma-shield zone until the combined neutron and gamma-ray dose rate at 91.44 cm was less than 1000 mrem/hr. Neutron shielding material was then added to the cask exterior until the 1.83-m dose-rate condition for normal cask operation was also met.

Table 4-III lists the materials other than depleted uranium and lead considered in the calculations. Included are the assumed material densities and compositions used in the work. Natural density and isotopic concentrations were assumed for lead; the depleted uranium used consisted of 0.2% ^{235}U and 99.7% ^{238}U at normal density.

TABLE 4-III

Shield Material Specifications

Material	Density (g/cm^3)	Composition (atoms/barn-cm)	
Beechwood (a)	1.35	H	0.0482
		B	0.00225
		C	0.0256
		N	0.00232
		O	0.0248
B_4C (b)	2.38	B	0.0649
		C	0.0234
		Fe	0.0004
		Cu	0.0255
Stainless Steel (c)	7.90	C	0.00032
		Si	0.00169
		Cr	0.0174
		Mn	0.00173
		Fe	0.0579
		Ni	0.0081
Borosilicate (d)	3.20	B	0.00617
		C	0.00416
		O	0.0319
		Si	0.012
Serpentine (e)	2.55	H	0.0222
		O	0.0499
		Si	0.0111
		Mg	0.0166
Calcium Borate (f)	1.12	H	0.00896
		B	0.0134
		O	0.0259
		Ca	0.00224

(a) Assumed composition by weight: 6.0% H, 3.0% B, 49.0% O, 38% C, 4% N.

(b) B_4C in a copper matrix. Assumed 70% by volume $B_{2.77}CFe_{0.015}$ in Cu.

(c) 304 stainless steel with composition by weight: 0.08% C, 2.0% Mn, 1.0% Si, 12.0% Cr, 10% Ni, 67.92% Fe.

(d) Composition from SAND77-0274, WIPP Conceptual Design Report.

(e) Serpentine-hydrous magnesium silicate. Assumed composition: $3\ MgO{:}2\ SiO_2{:}H_2O$.

(f) Assumed composition: $CaO{:}3\ B_2O_3{:}H_2O$.

Figure 4-1 illustrates the basic HLW cask design assumed for the two-dimensional multigroup neutron and gamma-ray transport calculations. Figure 4-2 is an idealized sidewall section of the cask which presents the configuration used for the one-dimensional calculations. The only variable parameter considered in these models was the thickness of the neutron- and gamma-shield regions. Dimensions, such as the overall length of the cask cavity (304.8 cm), cavity diameter (107.4 cm), and steel liner thickness (2.54 cm) about the primary gamma shield, were held constant for all analyses. The one-dimensional analyses, which were performed primarily to provide a basis for the two-dimensional work, were used only to determine the shielding requirements in the radial direction through the cask sidewalls. The two-dimensional calculations modeled the cask approximately as shown by Figure 4-1.

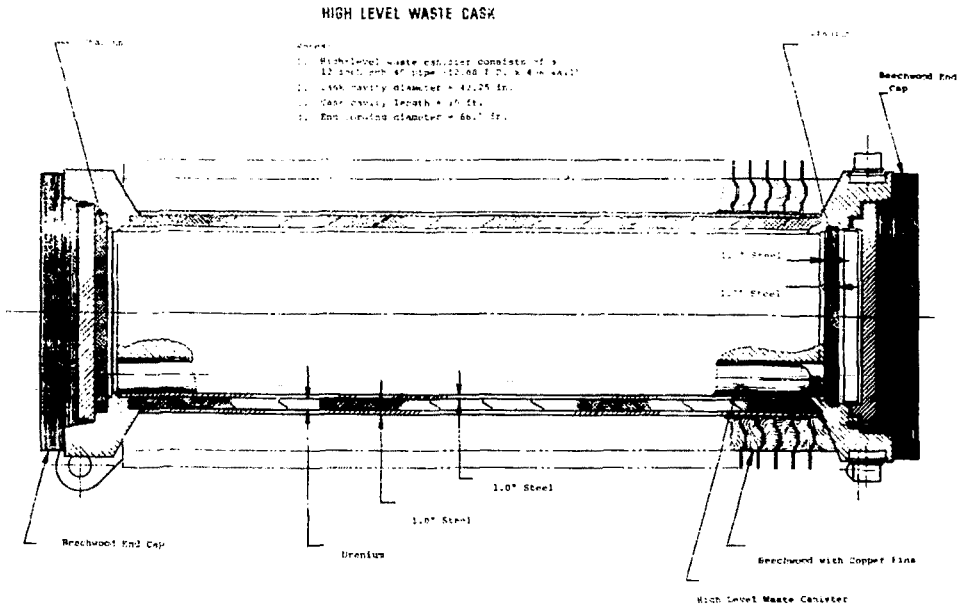

Figure 4-1. High-Level-Waste Cask

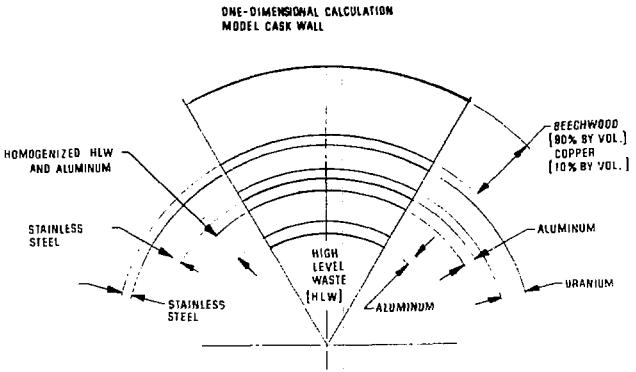

Figure 4-2. Cask Model for One-Dimensional Transport Calculations

A prime difficulty in determining a satisfactory HLW cask design was that the HLW constitutes a long radiation source. In the calculations it was assumed the HLW was located uniformly along the length of the canister to within 8 cm of the cask top and bottom. This did not affect the shielding problem in the cask wall or ends but posed a problem in the boundary between these regions. Limiting line-of-sight paths at the cask corners was a major difficulty. Two-dimensional shielding analyses indicated the inadequacy of early designs. Two possible solutions to the problem of radiation streaming at the corners were considered. The first was to increase the cask length until these line-of-sight paths were eliminated. However, this procedure unacceptably increased the total weight of the cask. The second, and preferable, solution was to incorporate additional shielding material in the end forgings or cover. Figure 4-3 illustrates a means for doing this. Since it was deemed desirable to maximize the structural integrity of the forging, the insert scheme utilizing additional shielding as an integral part of the lid was chosen. This had the added advantage of locating the shield material close to the waste, thereby decreasing the amount of shield material required. Such a shield design, however, would require some modification of the HLW canister in the event a canister design similar to that shown in Figure 2-3 was used.

The HLW generates significant quantities of decay heat, and, since most neutron shield materials are poor thermal conductors, steps must be taken to prevent the cask from overheating and the borosilicate glass from melting. Towards this end, copper heat-conduction fins extending through the neutron shield were employed as shown in Figure 4-1. Similar fins are employed in the conceptual casks designs described in Reference 2. The shape of the fins was selected to remove line-of-sight paths between the HLW and the cask exterior through the copper metal. For computational purposes, the presence of the copper fins was approximated by assuming that 10% by volume of the material forming the neutron shield was copper.

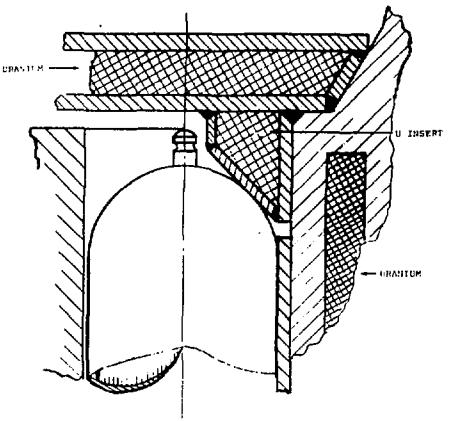

Figure 4-3. Ring of U in Cask End to Eliminate Radiation Streaming at Cask Corners

The cask design considered for the CW shipping cask would be identical to the HLW cask except the CW cask design would not include a neutron shield and the cask would have a capacity of nine 30.48-cm-diam CW canisters (cask cavity diameter of 124.5 cm). The canister arrangement in this case was one central canister surrounded by a ring of the other eight canisters. The major shielding problem posed by CW is the primary gamma-ray radiation source. The neutron source strength is relatively insignificant and the dose rate due to neutrons is sufficiently reduced by the shielding necessary to attenuate the gamma rays to safe dose-rate levels.

One-dimensional calculations were performed to determine the shield requirements for cladding waste. Depleted uranium and lead were considered as shielding materials. Table 4-IV presents the composition assumed for the cladding waste. Included in the waste were Zircaloy, 302 stainless steel, 304 stainless steel, and Inconel. The composition reflects a 2:1 mixture of structural material from PWR and BWR spent fuel, respectively.

TABLE 4-IV

Cladding Waste Material Specifications

Element	Composition (atoms/barn-cm)	Element	Composition (atoms/barn-cm)
C	0.000039	Mn	0.000189
Si	0.000186	Fe	0.00628
Sn	0.000311	Ni	0.00246
Cr	0.00221	Zr	0.0265

5. Results

Table 5-I presents the results of the one-dimensional calculations for a number of cask shield designs which meet 10 CFR 71 requirements using the present-generation HLW radiation source of Table 2-II. The first two sets of entries in the table provide the dose rates for the assumed cask accident condition of total loss of the neutron shield but retention of the primary gamma shield. As previously mentioned, the accident dose-rate limitation is a total of 1000 mrem/hr at 91.44 cm from the cask surface. However to assure that a conservative design had been obtained, an overall safety factor of 2 was employed thus requiring the accident dose rate condition to be around 500 mrem/hr. Utilizing these results, the effect of adding various neutron shields to the cask was examined until the total dose rate at 1.83 m from the cask was less than 10 mrem/hr (normal operation conditions for the cask). Actually the safety factor mentioned above was utilized again and the thicknesses of the neutron shield zone were varied to obtain dose rates in the neighborhood of 5 mrem/hr at 1.83 m. The remainder of Table 5-I lists combinations of neutron- and gamma-ray shielding materials which were determined to result in acceptable 1.8-m dose rates for normal cask conditions.

TABLE 5-I

Shielding Requirements for Nonrecycle HLW
(Estimated by one-dimensional calculations corrected for geometry)

Gamma Shield (cm)	Neutron Shield (cm)	Radiation	Dose Rate (mrem/hr)	Total Dose Rate (mrem/hr)
7 depleted U	None	Neutron (postaccident)	95 at 91.44 cm	326 at 91.44 cm
		Gamma (postaccident)	231 at 91.44 cm	
15 lead	None	Neutron (postaccident)	290 at 91.44 cm	510 at 91.44 cm
		Gamma (postaccident)	220 at 91.44 cm	
7 depleted U	25 beechwood	Neutron	1.2 at 1.8 m	4.6 at 1.8 m
		Gamma	3.4 at 1.8 m	
7 depleted U	30 H_2O	Neutron	1.1 at 1.8 m	3.1 at 1.8 m
		Gamma	2.0 at 1.8 m	
7 depleted U	25 serpentine	Neutron	3.4 at 1.8 m	5.4 at 1.8 m
		Gamma	2.0 at 1.8 m	
7 depleted U	30 calcium borate	Neutron	6.1 at 1.8 m	8.2 at 1.8 m
		Gamma	2.1 at 1.8 m	
15 lead	25 beechwood	Neutron	1.4 at 1.8 m	5.4 at 1.8 m
		Gamma	4.0 at 1.8 m	
15 lead	30 H_2O	Neutron	1.2 at 1.8 m	3.5 at 1.8 m
		Gamma	2.3 at 1.8 m	
15 lead	25 serpentine	Neutron	3.6 at 1.8 m	5.7 at 1.8 m
		Gamma	2.1 at 1.8 m	
15 lead	33 calcium borate	Neutron	4.2 at 1.8 m	7.1 at 1.8 m
		Gamma	2.9 at 1.8 m	

The results of the one-dimensional calculations have been corrected approximately for geometric effects. In solving the one-dimensional shielding problem in cylindrical geometry, the ANISN code considers the height of the cylinder to be infinite. Since the HLW canisters are only 3 m long, the results produced by ANISN overestimate the dose rates at 91.44 cm and 1.83 m from the cask surface. To compensate for this geometry, correction factors of 1.5 and 2.0 were used for the 91.44 cm and 1.83 m results, respectively. The entries in Table 5-I were obtained by dividing the dose rate results calculated by ANISN by the appropriate geometric correction factor.

To evaluate the impact of uranium recycle on a HLW shipping-cask shield design, a representative number of the present-generation shield designs from Table 5-I were reexamined using the uranium-recycle neutron and gamma sources of Table 2-II. The results of this analysis appear in Table 5-II. The conclusion is that it is possible to obtain HLW shipping-cask shield designs suitable for the transportation of both present-generation and uranium-recycle HLW. This is understandable since the total primary gamma source and neutron source from uranium-recycle waste are approximately 19% greater and 18% less, respectively, than that of present-generation waste. These results are within the uncertainty factor used in designing the shields.

TABLE 5-II

Comparison of Present-Generation and Uranium Recycle HLW Shielding Requirements*

Waste Type	Neutron Shield	Radiation	Dose Rate (mrem/hr)	Total Dose Rate (mrem/hr)
Uranium Recycle	None	Neutron (postaccident) Gamma (postaccident)	81 at 91.4 cm 275 at 91.4 cm	356 at 91.4 cm
Uranium Recycle	25-cm beech-wood	Neutron Gamma	1.0 at 1.8 m 4.0 at 1.8 m	5.0 at 1.8 m
Uranium Recycle	30-cm water	Neutron Gamma	0.9 at 1.8 m 2.3 at 1.8 m	3.2 at 1.8 m
Present-Generation	None	Neutron (postaccident) Gamma (postaccident)	95 at 91.4 cm 231 at 91.4 cm	326 at 91.4 cm
Present-Generation	25-cm beech-wood	Neutron Gamma	1.2 at 1.8 m 3.4 at 1.8 m	4.6 at 1.8 m
Present-Generation	30-cm water	Neutron Gamma	1.1 at 1.8 m 2.0 at 1.8 m	3.1 at 1.8 m

*In each case a 7-cm depleted uranium gamma shield was used.

The situation is much different when plutonium-recycle HLW is considered. Although the primary gamma source from plutonium-recycle waste is only twice the primary gamma source of present generation waste, the neutron source is about 25-30 times as great. The difficulty of satisfying the postaccident dose-rate limitation is illustrated by the results presented in Table 5-III. In this case, the utilization of B_4C was considered. The dose rates in the table are only for the postaccident case. In order to obtain normal operation dose-rate limits using these primary gamma-shield dimensions, approximately the same thicknesses of neutron shield materials as previously given in Table 5-I would be required. A cask capable of safely transporting plutonium-incorporated into the cask ends and the radial thickness of the end forgings was increased. These inserts (which are shown in Figure 4-3) are rings of uranium which, when included with the recycle HLW would have an overall diameter more than 30 cm greater for the same payload than one capable of handling uranium-recycle or present generation waste.

TABLE 5-III

Plutonium Recycle Shielding Requirements - Postaccident Condition Only
(Estimated by one-dimensional calculations corrected for geometry)

Radiation	Shield Materials (in cm)	Dose Rate at 91.44 cm (mrem/hr)
Neutron	28 depleted uranium	585
Neutron	5 depleted uranium + 15 B_4C	470
Neutron	7 depleted uranium + 20 copper	884
Neutron	23 lead + 10 B_4C	915

The uncertainties involved in these results include contributions from errors in the radiation sources, cross sections (both in their values and limitation to a P1 cross-section set), dose-rate conversion factors, and multidimensional effects. Although no detailed sensitivity studies were performed, it is estimated that the source, cross section, and dose-rate conversion factor uncertainties may yield a total uncertainty in the thickness of any layer of the shield of approximately ±2 cm. This estimate is based on assuming a 20% error in the radiation-source definition, a 20% combined uncertainty in cross-section values, and the uncertainty introduced by the finite differencing technique used in the ANISN code. The uncertainty introduced by multi-dimensional effects is diminished in the two-dimensional calculations.

A cask design which used depleted uranium for the gamma shield and borated beechwood for the neutron shield was examined using the DOT program. Uranium was chosen as the gamma shield since it results in the smallest diameter cask design. The beechwood shield has merit because of its availability and shielding efficiency. Also, in an accident involving a fire, the hydrogen content of this material may not be completely lost. Some beechwood retention may be anticipated since it would take time for the beechwood to char and a fire may be extinguished

before complete charring is effected. In the calculations, however, the beechwood was assumed to be entirely removed to simulate the worst accident condition. Therefore the results are conservative. Two-dimensional calculations which include serpentine or calcium-borate shield materials have not been performed. Using the two-dimensional beechwood results as a basis, it is possible to estimate the thickness requirements of these other materials for producing an acceptable cask shield design.

The strategy of first satisfying the postaccident cask condition was again employed in solving the two-dimensional shielding problem. Initially, only gamma-ray shielding material was included in the cask wall and ends. Early two-dimensional results indicated a severe streaming problem in the forging between the two shield regions. To alleviate this problem a uranium insert was incorporated into the cask ends and the radial thickness of the end forgings was increased. These inserts (which are shown in Figure 4.3) are rings of uranium which, when included with the uranium in the cask wall and ends, completely surround the HLW with gamma-shielding material. It was found that 7.5 cm of uranium in the cask wall, 6.5 cm in the cask ends, and 3.0 cm thick and 12.5 cm high uranium rings at each end of the cask were necessary to reduce the total dose rate at any location 91.44 cm from the cask to a value less than 1000 mrem/hr. These results account for a calculated k_{eff} value of 0.09 for the uranium in the cask and include a 25% increase[14] in the dose rate due to ground scatter. In order to satisfy the normal operation dose-rate condition at 1.83 m from the cask surface, it was determined that 35 cm of beechwood was needed on the side of the cask and 25 cm on the ends. Again, these values account for the neutron multiplicative property of the depleted uranium and a 40% increase[14] due to ground scatter effects.

A problem often associated with two-dimensional discrete ordinates transport calculations is the presence of ray effects. Ray effects are attributable to the process of differencing the continuous angular variable of the transport problem into a finite number of discrete angles. If a quadrature with too few angles is chosen, the fluxes calculated by the two-dimensional code will peak along the quadrature directions and have minima in others, a result reflecting the quadrature choice and not the true physical situation. The present two-dimensional calculations were performed using an S_6 quadrature, which, in cylindrical geometry, consists of 30 different angles. That this was a suitable quadrature for avoiding ray effects can be seen in Figures 5-1 and 5-2. The dose rate at a point is equal to the product of the particle integrated flux at that point and the dose rate conversion factor integrated over particle energies. Examination of the dose rate thus provides information about the total flux. If ray effects are not present in the results of these calculations, it would be reasonable to expect the total flux near the cask surface to be greatest on the centerline at the cask ends, and at the center of the cask side, and to decrease smoothly as the observation point moves either axially along the cask from the center of the side or radially from the centerline at the cask ends. Figure 5-1 shows the dose rate as a function of radius on the surface of the cask lid. The peak dose rate is 79 mrem/hr. The dose rate decreases smoothly with increasing radius. Figure 5-2 shows the surface dose rate along the side of the cask; the maximum value is 74 mrem/hr. Again, the dose rate decreases smoothly as a function of axial position. It is evident the dose rate remains significant at any location directly above HLW, but declines rapidly when the distance from the waste increases. The figures demonstrate that ray effects were not apparent in these calculations.

As in the one-dimensional case, the uncertainties involved in these results are contributed to by uncertainties in the cross sections, dose-rate conversion factors, and the radiation sources. Multidimensional effects are of lesser importance here than for one-dimensional results. Under the assumption of a possible ±20% error in the radiation-source definition, a ±20% uncertainty from cross-section values, and the uncertainty introduced by the discretizing procedure and use of P1 cross sections, it is believed that the individual shield-layer thicknesses required to give the calculated dose rates are accurate to within ±1 cm.

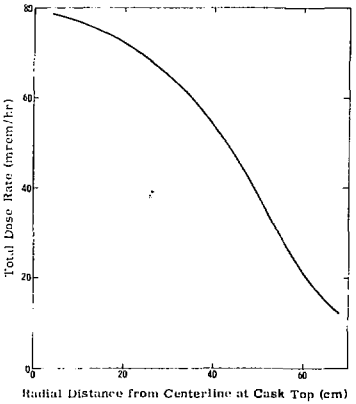

Figure 5-1
Total Dose Rate at End Surface of Cask

Figure 5-2
Total Dose Rate Along Side of Cask

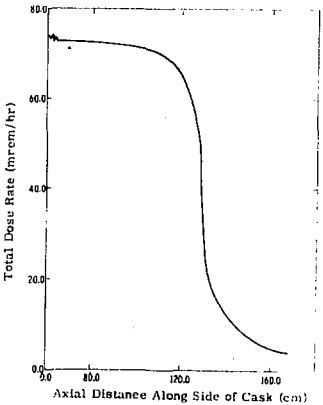

Table 5-IV provides estimated total weights of three casks capable of transporting 30.48-, 40.64-, and 50.80-cm-diam HLW cylinders, respectively, cased on the 30.48-cm-diam canister cask design determined in the two-dimensional analytical work. Implicit in the two larger cask designs is the assumption that these casks would not need thicker radiation shields even though their payloads would be greater. This assumption has been substantiated by a number of one-dimensional ANISN calculations which modeled a cask containing seven 40.64-cm-diam HLW canisters. The assumption is reasonable since the more massive aluminum baskets and greater HLW self-shielding effects in these larger casks provide additional radiation shielding.

TABLE 5-IV

Material Weight and Cost Estimates for Conceptual HLW Depleted Uranium Casks
(Seven-canister payload)

Canister Diameter (cm)	Cask Material	Material Weight (1000 kg)	Cask Total Weight (metric tonne)	Waste Capacity (metric tonne)	Ratio of Waste Weight to Cask Weight	Material Cost (1000's of dollars)
30.48	U	18.7				740
	Steel	7.5				2
	Beechwood	8.2	46.8	4.0	0.08	54
	Cu	5.1				6
	Al	3.3				3
40.64	U	23.2				920
	Steel	9.2				3
	Beechwood	9.7	58.9	6.5	0.11	64
	Cu	5.8				7
	Al	4.5				5
50.80	U	29.1				1150
	Steel	11.4				4
	Beechwood	11.6	75.1	10.0	0.13	76
	Cu	6.7				8
	Al	6.3				7

The three casks in Table 5-IV have overall diameters (exclusive of the copper fins) of 2.0 m, 2.3 m, and 2.6 m, respectively. Economy of scale is evident in these results since the payload weight divided by the cask total weight increases from 0.08 for the smaller cask design to 0.13 for the largest cask. This payload advantage, however, may be offset by such considerations as the additional cost of shipping a heavier cask, increased difficulty in handling the heavier waste canisters, and the more serious decay-heat-dissipation problem for the larger canisters.

Also shown in Table 5-IV are the estimated material costs for the various materials in the casks. The estimates are based on $39.7/kg for depleted uranium, $1.32/kg for copper, $0.35/kg for steel, $1.17/kg for aluminum, and $6.60/kg for the beechwood.

Similar weight and cost estimates have been obtained for cask designs employing lead in place of depleted uranium as the gamma shield material. These estimates are shown in Table 5-V. The cost of lead assumed was $1.76/kg. From a comparison of the cost estimates in Tables 5-IV and 5-V it is evident that a lead cask would be considerably cheaper than a corresponding cask

design which utilized depleted uranium as the gamma shield material. The lead cask would also be about 18% heavier. The cost of transporting this additional weight over the lifespan of the lead cask would tend to offset these material cost differences.

TABLE 5-V

Material Weight and Cost Estimates for Conceptual HLW Lead Casks
(Seven-canister payload)

Canister Diameter (cm)	Cask Material	Material Weight (1000 kg)	Cask Total Weight (metric tonne)	Waste Capacity (metric tonne)	Ratio of Waste Weight to Cask Weight	Material Cost (1000's of dollars)
30.48	Pb	25.1				44
	Steel	8.7				2
	Beechwood	6.8	55.5	4.0	0.07	58
	Cu	5.6				7
	Al	3.3				3
40.64	Pb	30.8				54
	Steel	11.0				4
	Beechwood	10.3	69.3	6.5	0.09	68
	Cu	6.2				8
	Al	4.5				5
50.80	Pb	38.3				67
	Steel	13.4				5
	Beechwood	12.3	87.5	10.0	0.11	80
	Cu	7.2				9
	Al	6.3				7

At the present time, only preliminary work has been completed on a CW shipping-cask-shielding design. This work involved a number of one-dimensional transport calculations in which the shielding requirements necessary for decreasing the primary gamma-ray flux to acceptable dose rates were determined. The neutron source in cladding waste is of such magnitude as to present no shielding difficulties.

Only uranium and lead were considered as candidates for the primary gamma-ray shield material. Both shield possibilities were assumed to be lined with 2.54-cm thick steel plates. Table 5-VI presents the results of these calculations. As in the HLW cask design work an effort was made to obtain CW shield designs resulting in dose rates of 5 mrem/hr at 1.83 m from the cask surface. These results are subject to the same errors involved in the one-dimensional HLW calculations and have been approximately geometry-corrected using a correction factor of 2. The experience obtained in the two-dimensional calculations performed for the HLW shipping-cask preliminary conceptual design indicates there would be no difficulty in extending these one-dimensional results to a satisfactory CW shipping cask conceptual design.

TABLE 5-VI

Cladding Waste One-Dimensional Results
(Estimated one-dimensional results corrected for geometry)

Shielding Material (cm)	Dose Rate (mrem/hr) at 1.83 m
5.5 depleted uranium	7.69
13.0 lead	3.67

6. Conclusions

Conceptual shielding designs for shipping casks capable of transporting present generation HLW and shielding designs for CW shipping casks have been developed. Primary gamma shield material candidates were restricted to depleted uranium and lead for both cask types. The neutron shield materials considered for the HLW cask were borated beechwood, water, serpentine, and calcium borate.

The results presented in the previous section are indicative of the relative merits of a number of shield materials for conceptual HLW and CW shipping casks. Cask weight, size, cost, licensability and fabricability all are of critical importance in judging the relative merits of a particular cask design. It is hoped that the information contained in this work will be of value in developing and evaluating subsequent, more detailed HLW and CW shipping cask designs.

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