CONF-960415--33

SHIELDING ANALYSIS OF DEPLETED URANIUM SILICATE FILLER CONCEPT FOR SPENT FUEL CANISTER DESIGNS

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To be Presented at
ANS 1996 Radiation Protection and Shielding Division Topical Meeting
Advancements and Applications in Radiation Protection and Shielding
for the High-Level Waste Packaging and Site Design Issues Session
April 21-25, 1996
Cape Cod, Massachusetts

*Managed by Lockheed Martin Energy Research Corp. for the U.S. Department of Energy under contract DE-AC05-96OR22464.

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ABSTRACT

A Depleted Uranium Silicate COntainer Backfill System (DUSCOBS) has been proposed at Oak Ridge National Laboratory. This concept suggests the use of small, depleted-uranium silicate glass beads as a backfill material inside storage, transportation, and repository waste packages containing spent nuclear fuel. Use of this backfill material would substantially reduce external dose rates from a waste canister, allowing a reduction of the amount of external shielding required. This paper summarizes the results of scoping studies to estimate the dose reduction from the use of DUSCOBS in a conceptual canister design, and to determine what design modifications are required to offset the increased mass of the system, while simutaneously maintaining sufficient shielding to meet external dose rate limits.

I. INTRODUCTION

An innovative concept to enhance the performance characteristics of spent fuel storage container designs has recently been proposed and investigated[1] at the Oak Ridge National Laboratory (ORNL). This idea proposes the use of depleted uranium (DU) in the form of a uranium silicate glass matrix as a backfill material in a loaded spent fuel canister. The use of a Depleted Uranium Silicate COntainer Backfill System (DUSCOBS) may improve waste package design performance by (1) reducing long-term radionuclide releases from packages in a repository, (2) reducing the potential for a nuclear criticality event in long-term disposal, (3) enhancing the structural integrity of a package in a repository environment, (4) providing a means to dispose of significant quantities of excess depleted uranium stockpiles, and (5) reducing the radiation dose external to the package (thereby reducing design shielding requirements).

*Managed by Lockheed Martin Energy Research Corp. for the U.S. Department of Energy under contract DE-AC05-96OR22464.

Scoping computational studies were completed to determine the effects of DUSCOBS on criticality, heat transfer, mass, and shielding requirements for models based on the Multipurpose Canister (MPC) design.[2] This paper describes work performed at ORNL to assess the shielding and collateral mass implications of the use of DU glass backfill.

II. ANALYSIS APPROACH

All calculations described in this paper were performed using the SAS2H[3] and SAS4[4] sequences of the SCALE-4 code system.[5] Source term calculations were performed using ORIGEN-S within the SAS2H sequence; all shielding calculations were performed using the Monte Carlo code MORSE-SGC via SAS4. Gamma-ray and neutron source terms were calculated based on 3.75 wt % enriched fuel with an assumed burnup of 40 GWd/MTU and a 10-year cooling time.[2]

Design-basis shielding limits are determined under dry conditions. Thus calculations were performed assuming full canister loading with and without DU silicate (DUS) backfill. Assembly properties were determined assuming 100% void in interstitial regions of the assembly for empty (no DUS) conditions, and 65% DUS, 35% void for backfilled conditions.

Calculations were performed on an IBM RS/6000 workstation using SCALE-4.3 and the SCALE coupled 27-group neutron/18-group gamma-ray cross-section library. Dose response functions were obtained from the ANSI-6.1.1-1977 standard.[6] Doses were calculated at a radial distance of 2 m from the canister wall, at the axial centerline of the fuel length, for azimuthal angles of zero and 30.96°. Figure 1 illustrates the orientation of the canister relative to the reference coordinate system. The 0° rotation angle represents the location that sees the largest source area, while the 30.96° rotation represents the closest

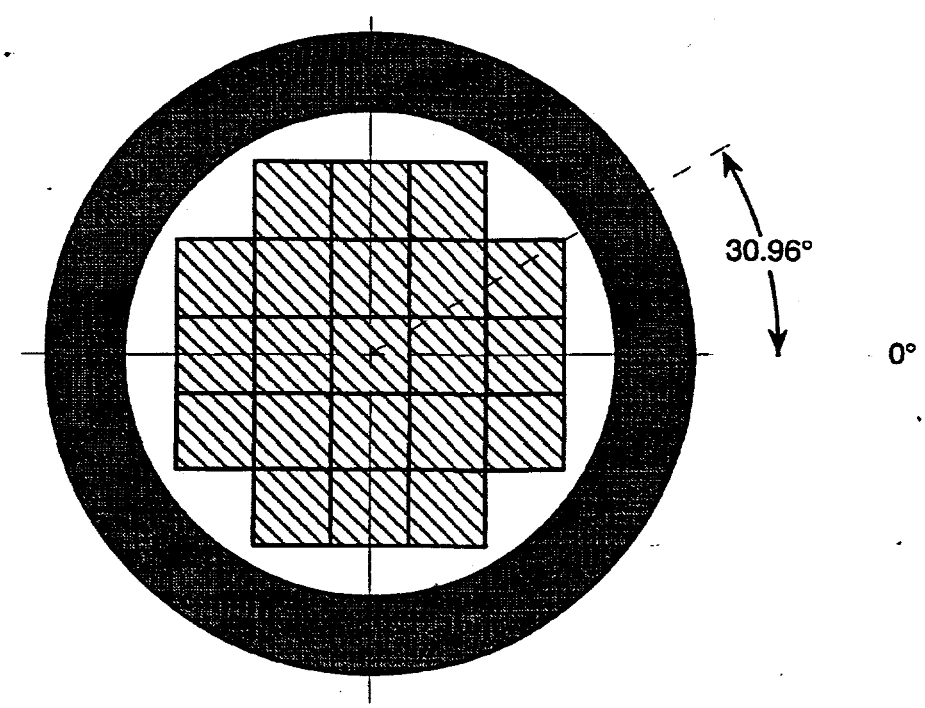

Figure 1. Orientation of cask relative to coordinate system for MORSE-SGC calculations.

approach of the source region to the dose point. Calculations were performed to determine point dose rates (mrem/h) due to neutrons, primary gamma rays (emitted as the result of fission product and activation product decay), and secondary gamma rays (produced as the result of n-γ reactions).

III. LWR SPENT FUEL CANISTER MODEL

Shielding calculations were nominally based on the 125-ton MPC design.[2] This conceptual design contains 21 basket positions for the storage of pressurized-water reactor (PWR) spent fuel assemblies. Calculations were performed with (1) the nominal design, (2) variations in the shield design, and (3) a 17-basket arrangement to assess other possible design configurations. Assembly compositions were based on a Westinghouse 17 × 17 assembly design; however, material properties and source terms were homogenized over the volume of the assembly since the geometry of the fuel pin lattice has little effect on canister shielding calculations. Each assembly was assumed to be identical in terms of composition and source terms. Canister and basket specifications were modeled explicitly. The radial dimensions and composition of the nominal canister design are given in Figure 2. The canister was assumed to be the length of the fuel assemblies (366 cm); radial calculations were performed for a point located 183 cm from either axial end. End regions of the canister were not modeled.

DUS composition was assumed based on a modified soda-lime glass with a 25 wt % content of 0.2 wt % enriched DU. The composition of the DUS is provided in Table 1. Based on this composition, the glass density was determined to be 4.1 g/cm^3. The sand was assumed to occupy all void space within a fully loaded (i.e., 21 assemblies) MPC shell, with a packing density of 65% sand, 35% void (by volume).

Shielding calculations were performed to assess two considerations. First, calculations were used to determine the reduction in radiation dose due to the use of a DUS backfill relative to a non-backfilled canister. Second, calculations were performed to assess the external dose which would result from various changes in the nominal canister design. Because of the increased shielding due to DUS backfill, it would be possible to remove other canister shield components and maintain the same external dose; removal of shield components, especially the high-density lead and DU shield materials, would serve to offset the increased mass resulting from the added DUS. The following modified shield configurations were studied: (1) no lead shield; (2) no lead shield and a 15% reduction in the DU shield thickness; and (3) no lead shield and a 25% reduction in the DU shield thickness.

In addition, a potential 17-element canister design was considered; removal of four assemblies closest to the canister outer walls and repositioning of remaining outer perimeter assemblies results in a configuration in which a greater thickness of DUS can be placed between the outermost assembly position and the inner shield wall. This configuration allows an additional reduction in required shielding material and its associated mass. The 17-element canister configuration has no lead shield and a 50% reduction in the thickness of the DU shield relative to the nominal design, as illustrated in Figure 3.

IV. RESULTS AND DISCUSSION

Calculated doses for each configuration are provided in Tables 2 through 4 for neutron, primary gamma-ray, and secondary gamma-ray doses, respectively. The calculated dose is given together with its fractional standard deviation expressed in percent. The estimated total dose from all sources is given in Table 5.

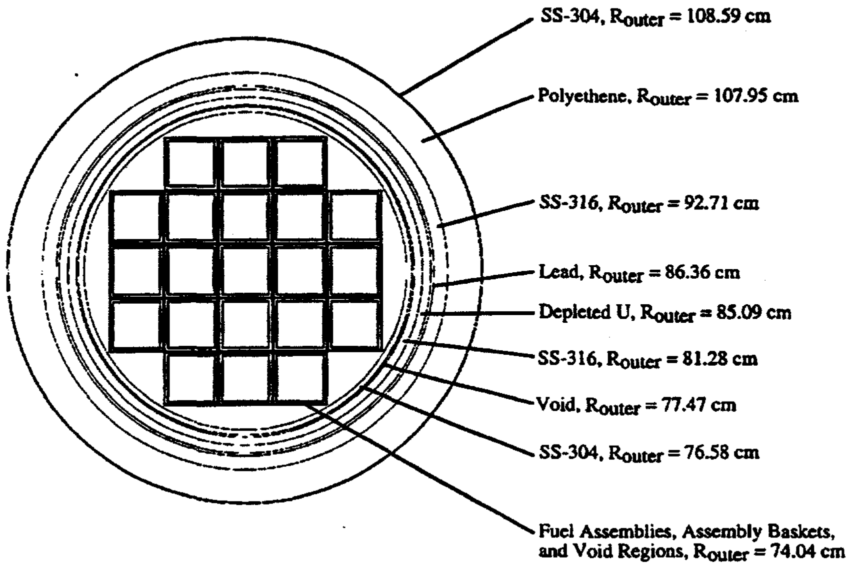

Figure 2. Nominal MPC cask design with transportation overpack.

Table 1. Composition of depleted uranium silicate

Material	Wt % in DUS
Component Data:	
U_3O_8	29.48
CaO	11.17
Na_2O	7.40
SiO_2	51.95
Isotopic Data	
U	25.00
^{234}U	0.00035 (0.0014 wt % in U)
^{235}U	0.05000 (0.2000 wt % in U)
^{236}U	0.00023 (0.0009 wt % in U)
^{238}U	24.9492 (99.7977 wt % in U)
Ca	7.98
Na	5.49
Si	24.28
O	37.25

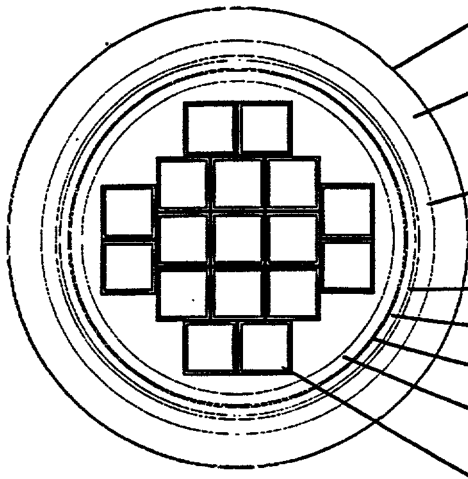

Figure 3. Modified 17-element cask design with reduced shielding.

Table 2. Results of canister shielding calculations of neutron dose

Canister model	Dose @ 0° (mrem/h)	Dose @ 30.96° (mrem/h)
21-element canister, no backfill, nominal shield design	0.5107 ± 6.49%	0.5743 ± 10.52%
21-element canister, DUS backfill, nominal shield design	0.2310 ± 14.75%	0.2982 ± 26.8%
21-element canister, DUS backfill, no lead shield	0.2832 ± 28.8%	0.2873 ± 5.76%
21-element canister, DUS backfill, no lead shield, 85% of nominal DU shield	0.2142 ± 5.38%	0.2727 ± 22.0%
21-element canister, DUS backfill, no lead shield, 75% of nominal DU shield	0.2876 ± 17.0%	0.2874 ± 11.1%
17-element canister, DUS backfill, no lead shield, 50% of nominal DU shield	0.1578 ± 4.9%	0.1533 ± 10.3%

Table 3. Results of canister shielding calculations of primary gamma-ray dose

Canister model	Dose @ 0° (mrem/h)	Dose @ 30.96° (mrem/h)
21-element canister, no backfill, nominal shield design	5.0572 ± 6.01%	4.1154 ± 2.65%
21-element canister, DUS backfill, nominal shield design	0.92514 ± 5.12%	0.99892 ± 8.17%
21-element canister, DUS backfill, no lead shield	2.2909 ± 9.68%	2.6116 ± 14.1%
21-element canister, DUS backfill, no lead shield, 85% of nominal DU shield	4.5765 ± 9.13%	5.4230 ± 10.8%
21-element canister, DUS backfill, no lead shield, 75% of nominal DU shield	6.3896 ± 8.53%	6.9364 ± 11.2%
17-element canister, DUS backfill, no lead shield, 50% of nominal DU shield	6.2786 ± 4.01%	4.1154 ± 2.65%

Table 4. Results of canister shielding calculations of secondary gamma-ray dose

Canister model	Dose @ 0° (mrem/h)	Dose @ 30.96° (mrem/h)
21-element canister, no backfill, nominal shield design	1.6857 ± 5.14%	1.6495 ± 2.65%
21-element canister, DUS backfill, nominal shield design	0.7034 ± 2.65%	0.7288 ± 3.42%
21-element canister, DUS backfill, no lead shield	0.7219 ± 3.48%	0.7097 ± 3.13%
21-element canister, DUS backfill, no lead shield, 85% of nominal DU shield	0.8467 ± 5.69%	0.8385 ± 5.81%
21-element canister, DUS backfill, no lead shield, 75% of nominal DU shield	0.9878 ± 11.7%	0.8268 ± 1.80%
17-element canister, DUS backfill, no lead shield, 50% of nominal DU shield	0.5931 ± 2.89%	0.5883 ± 2.72%

Table 5. Total dose estimated for various canister configurations

Canister model	Dose @ 0° (mrem/h)	Dose @ 30.96° (mrem/h)
21-element canister, no backfill, nominal shield design	7.3	6.3
21-element canister, DUS backfill, nominal shield design	1.9	2.0
21-element canister, DUS backfill, no lead shield	3.3	3.6
21-element canister, DUS backfill, no lead shield, 85% of nominal DU shield	5.6	6.5
21-element canister, DUS backfill, no lead shield, 75% of nominal DU shield	7.7	8.1
17-element canister, DUS backfill, no lead shield, 50% of nominal DU shield	7.0	5.1

It is clear that the simple addition of DUS in the nominal MPC design results in a significant reduction in dose rates. Neutron doses are reduced by roughly 50%, primary gamma-ray doses are roughly 20 to 25% of nominal doses, and secondary gamma-ray dose rates are more than halved. Note that secondary gamma rays are produced primarily by neutron absorption in the outer neutron shield, thus a similar reduction is expected in both neutron and secondary gamma-ray does rates. Estimated total doses given in Table 5 show a reduction in total dose of 65% to 75% by the addition of a DUS filler.

These results also show that some of the shielding may be removed from the original design to offset the mass of the added backfill. Calculations in which heavier components of the MPC shield design were removed confirm that these components have the most significant effect on primary gamma rays. Neutron and, therefore, secondary gamma ray does rates are relatively unaffected by the removal of the shielding material. This is as would be expected since these dense materials are intended primarily as gamma-ray shields and act primarily as scattering media for neutrons. However, because the largest component of the total dose comes from primary gamma rays, the total dose is sensitive to the presence of the heavy materials. The shielding calculations indicate that the 1.27-cm lead shield can be completely removed from the MPC design, together with roughly 15% of the 3.81-cm (1.5-in.) depleted uranium shield, while maintaining a dose comparable to that of the air-filled canister. Calculations for the modified 17-element canister also show a significant reduction in neutron and secondary gamma-ray dose rates, due both to the increased neutron absorption in the outer DUS region and the reduced total source (17 assemblies rather than 21 assemblies). The primary gamma-ray dose rate is also reduced by the increased "thickness" of DUS to the point that the DU shield can be reduced to one-half of the nominal thickness and the lead shield removed while retaining essentially the same external dose rates as a nominal canister with no backfill.

The effects of shield design changes on total system mass are estimated in Tables 6 through 8. Material volumes are estimated given material radii and an assumed axial height of 488.95 cm (nominal canister length less upper and lower end regions). Table 6 gives the masses of the various shield components for the nominal canister design. Table 7 provides the reduced masses resulting from the removal of the lead shield and a 15% reduction in the radius of the depleted uranium shield. Table 8 gives the reduced shield masses associated with the 17-element canister design (i.e., no lead shield and a 50% reduction in the radius of the depleted uranium shield). (Note that this table reflects only the change in mass in the shield, and does not account for the mass difference between the four removed assemblies and the increased DUS loading.) These results indicate that the redesigned 21-element canister shield could result in a total mass reduction of 7016 kg over the nominal design. The total volume of sand/air mixture in a backfilled MPC is approximately 1.06 10^6 cm^3. For an effective density of 2.66 g/cm^3, this represents a total added mass of 13,100 kg of DUS. Thus a 21-element MPC with DUS backfill and a modified shield would have an increased mass of roughly 6,100 kg. However, the shield associated with the proposed 17-element canister design results in a 13,801kg decrease in mass, which completely offsets the increased DUS mass. Since the density of DUS is less than that of a fuel assembly, accounting for the removal of four assemblies and replacement with backfill material would result in a further decrease in mass.

V. CONCLUSIONS

The addition of a fine depleted-uranium silicate glass to a spent fuel canister has numerous potential benefits for both transportation and storage applications. The simple addition of this glass backfill would reduce external doses significantly. This fact itself would not justify the use of a canister backfill since dose rates in the current design meet dose rate limits, and the addition of a DUS backfill could potentially exceed canister handling weight limits. However, there are clear advantages for a DUS backfill in terms of short-term and long-term criticality control and the minimization of effects from long-term radionuclide release

Table 6. Nominal MPC shield component masses

Material	Density (g/cm^3)	r_{inner} (cm)	r_{outer} (cm)	Volume (cm^3)	Mass (kg)
SS-316	7.75	77.47	81.28	929,080	7,200
Depleted U	19.05	81.28	85.09	973,676	18,549
Pb	11.344	85.09	86.36	334,469	3,794
SS-316	7.75	86.36	92.71	1,746,670	13,537
Neutron Shield	0.91	92.71	107.95	4,697,428	4,275
SS-304	7.92	107.95	108.59	211,211	1,673
Total					49,027

Table 7. Modified 21-element MPC shield component masses

Material	Density (g/cm^3)	r_{inner} (cm)	r_{outer} (cm)	Volume (cm^3)	Mass (kg)
SS-316	7.75	77.47	81.28	929,080	7,200
Depleted U	19.05	81.28	84.52	824,781	15,712
Pb	11.344	84.52	84.52	0	0
SS-316	7.75	84.52	90.87	1,710,746	13,258
Neutron Shield	0.91	90.87	106.11	4,611,209	4,196
SS-304	7.92	106.11	106.74	207,618	1,644
Total					42,011

Table 8. Modified 17-element MPC shield component masses

Material	Density (g/cm^3)	r_{inner} (cm)	r_{outer} (cm)	Volume (cm^3)	Mass (kg)
SS-316	7.75	77.47	81.28	929,080	7,200
Depleted U	19.05	81.28	83.19	482,541	9,192
Pb	11.344	83.19	83.19	0	0
SS-316	7.75	83.19	89.54	1,684,829	13,057
Neutron Shield	0.91	89.54	104.78	4,549,090	4,140
SS-304	7.92	104.78	105.42	206,646	1,637
Total					35,226

to the environment. Thus it is important to demonstrate that a DUS filled canister is feasible in terms of shielding and total mass limits.

Based on the 21-element design it may not be possible to wholly offset the mass of DUS in a fully loaded MPC by corresponding reductions in shielding masses. The design thicknesses of stainless steel components are most likely required to meet structural requirements; hence no attempt was made to assess the shielding worth of such components. If the increased mass of the filled system exceeds design constraints, it may be necessary to redesign the canister. However, other options are available. Removal of the depleted uranium plug added to the lid of the MPC inner shell would reduce the MPC mass by about 1600 kg. Replacement of DUS in the top 50 cm of the inner shell with a lighter material would save more than 2000 kg; this region of the MPC does not contain fuel and thus does not necessarily require a DUS backfill. However, the effect of such modifications has not been evaluated from a shielding (or criticality) perspective.

If redesign is necessary, the scoping calculations presented here indicate that a 17-element design derived from the original 21-element MPC configuration would meet both external dose and total mass requirements when backfilled with DUS. No attempt has been made to analyze other design parameters (e.g., heat transfer, criticality, and structural integrity). However, these results indicate that much of the outer shielding mass can be removed from a shield design because of the shielding effect of DUS located outside the outer perimeter of assemblies appears to be the most important factor in minimizing canister external dose rates.

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