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Estimation of the Dose Effects of Gamma-Ray Streaming Through Cracks in the MVST Storage/Disposal Cask

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SUMMARY
This paper describes the source term and radiation shielding analyses performed to quantify the expected dose rate increase due to cracking from a tip-over test for the MVST storage/disposal cask. These calculations were performed on idealized cracks to predict the generic behavior of various straight line and jagged cask penetrations. The resulting procedure is used to estimate the number of bends or jags in the cask-cracking pattern in order to achieve the desired shielding properties. The results indicate that the observed cracks require only about 3 bends of 20 degrees to ensure no significant increases in surface dose rates over the as-built cask shield.

I. BACKGROUND

The Melton Valley Storage Tank (MVST) Storage and Disposal Cask is a large, concrete package that was originally designed to store solid radioactive waste on site. The contents consist of waste material thoroughly mixed with cement and solidified in a 20,000-lb monolith, contained in a steel shell. These monoliths are loaded into 40,000-lb outer concrete shields with 10,000-lb concrete lids. Currently, there are approximately 180 MVST casks on the Oak Ridge Reservation, which contain these monoliths.

Previously, a number of these casks have been shipped to the Nevada Test Site (NTS) for disposal. The current procedure is to unload the monoliths and place them in a lead-shielded cask for shipment to NTS. This activity has resulted in increased radiation exposure, both when the monoliths were transferred to the lead-shielded cask and when they were unloaded and buried at the NTS, and high shipping and handling costs for the program. Under this procedure, the empty MVST casks are then shipped to NTS for subsequent burial. As a result, DOE Oak Ridge has been exploring ways to ship the MVST cask with its monolith to the NTS for disposal as a unit. To do this, the MVST cask would have to be self-certified as meeting Industrial Package (IP-2) requirements as described in 49 CFR-Part 411.

The MVST cask, which weighs over 70,000 lb when loaded, was not designed for off-site shipment. The concrete lid currently rests on the top of the cask body and is not fastened. Thus, to meet IP-2 requirements, a lid-retaining device (LRD) was designed and shown to retain the lid on the cask and retain the monolith within the shield under the test conditions specified for an IP-2. Analyses indicated that the new LRD design would retain the cask contents in a 1-ft drop. This study reports the calculations performed to ensure that no significant increase (assumed to be 20%) in radiation levels would occur as a result of a tip-over test, which was used to simulate the drop test.

II. APPROACH

The tip-over test produced a series of hairline cracks in the concrete monolith. The worst observed crack was approximately one-fourth inch in width and penetrated about one-half of the thickness of the concrete shield (see Fig. 1). Many of the observed cracks (not pictured) spanned the entire height of the cask body, though typically not in straight vertical lines. The vertical cracks, which were jagged and in a helical pattern, appeared to be due to torsion loads. Two tasks were initiated to estimate streaming through the jagged cracks in the shield. The first task was to estimate the increase in dose due to streaming down right-circular cylindrical holes of various sizes. The second set of calculations modeled the reduction in doses due to various bends in these cylindrical-shaped holes. The bend studies consisted of the determination of dose rates at the outlet of a hole consisting of two links with turns of 5, 10, and 20 degrees at a distance halfway through the shield. The simplified geometries for these two cases are shown in Fig. 2. In each case, the inner and outer radius of the concrete shield was 101.6 and 132.08 cm with a height of 274 cm. Both the monolith and the concrete shield were modeled as concrete at a density of 2.3 g/cc. The specific compositions for the concrete were those corresponding to the ORCONCRETE material given in Ref. 1.

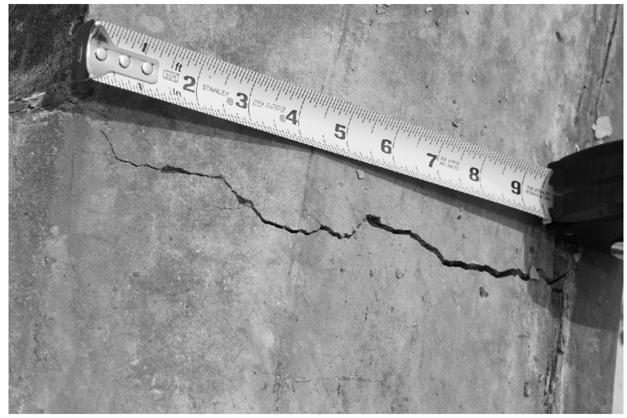

Figure 1. Worst case crack in the concrete monolith shield.

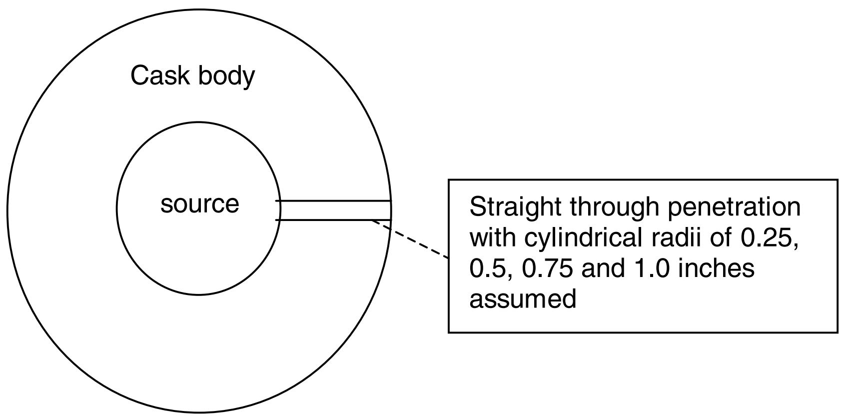

Figure 2a. Straight through penetration in shield body.

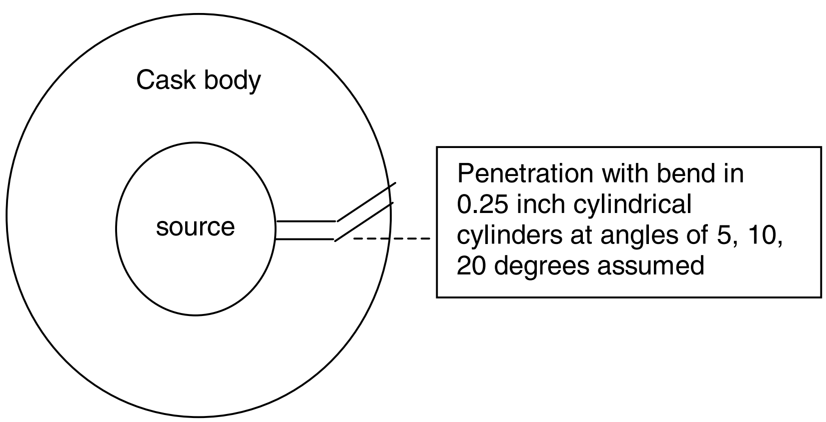

Figure 2b. Bend in penetration through shield body.

These scenarios were calculated using the SCALE SAS4 code system.[2] The SAS4 module of SCALE uses the MORSE three-dimensional Monte Carlo shielding code as a tool for solving a limited set of problems, primarily corresponding to cylindrical source and shield bodies. In this instance, the code was used to quantify the dose due to particle streaming through holes or cracks in the shield materials. Special techniques are necessary to use SAS4 for these studies, since the automated biasing present in the code is based on a one-dimensional approximation that does not allow for irregularities in the geometry. The technique used consisted of two modifications to a standard analysis. The first is the shortening of the cask height, in this case by a factor of 5, with a corresponding reduction of the source by the same factor. This makes the tracking of the particles through the penetration hole much more efficient. Secondly, the one-dimensional model that forms the basis for the automated bias generation is modified to simulate the material geometry *inside* the cask penetration. The source material followed by concrete shield material is replaced by source material followed by void material.

These calculations used the SCALE 27-neutron group and 18-gamma group library[3] distributed with the code system and based on ENDF/B-IV data. The flux-to-dose conversion factors are the default values for the SCALE system and are based on the ANSI/ANS-6.1.1-1977 standard.

These monoliths contain a large number of radioactive isotopes, however, the specific quantities are not known for each case. Measurements taken at the surface of the loaded casks indicate dose rates in the 1-3 mrem/h range, due mostly to ^{137}Cs. The assumed source for the multi-dimensional calculations was therefore estimated by executing a one-dimensional calculation using the SCALE module SAS1[4] to calculate the dose rate per ^{137}Cs source particle, and then scaling the source up to 3 mrem/h.

III. RESULTS

The results of these calculations are shown in Figs. 3 and 4 for the straight penetrations and bent penetrations, respectively. These results correspond to a point detector placed at the outer surface of the shield, centered on the penetration. The results in Fig. 3 can be used to establish that any straight penetration with a radius of less than 0.36 inch should meet the criterion of less than a 20% increase in the dose rate due to the crack. However, as noted from the series of pictures, the cracks have very rough edges and are not straight as assumed in the calculations. The results in Fig. 4 can be used to establish the shielding value of a bend in the penetration.

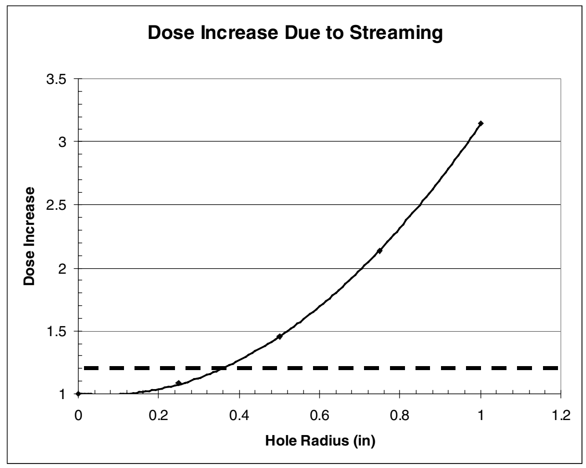

Figure 3. Dose increase as a function of the radius of a cylindrical void penetration.

The increases in dose due to the various penetration sizes shown in Fig. 3 are useful even if the penetrations are not cylindrical in nature. The literature[5] has shown that the results of studies with cylindrical penetrations can be used for non-cylindrical penetrations if the *areas* of the penetrations are equal. Thus, a rectangular penetration with an equivalent area to the cylindrical penetrations shown can be expected to have similar dose characteristics. This was verified by an additional calculation for a 0.25 x 12 inch rectangular penetration. These results were consistent with the results shown for a one-inch-radius cylindrical penetration.

Using this data, the generalizations shown in Table 1 can be made with respect to the dose increases for actual cracks.

Table 1. Scenarios to Meet 20% Dose Increase Criterion

Average crack thickness (inch)	Crack propagation length (inch)	Equivalent penetration radius (inch)	Dose increase factor due to penetration	Number of 20 degree bends necessary to meet 20% dose increase criterion
1/16	6[a]	0.35	1.18	0[b]
1/8	6	0.49	1.40	1
1/4	6	0.69	1.90	3

[a] Crack length of 6 inches based on observations where h airline cracks have at most 6 inches without large direction changes.
[b] Based on dose reduction factor of 0.85 for each 20 degree bend.

These results indicate that for the largest crack thickness seen, only three bends in the crack are necessary to limit the dose increase at the cask surface to 20%. These conclusions are based on only single bend calculations; however, the multiplicative nature of multiple bends is assumed due to the multiplicative nature of bend angles seen in Fig. 4. From the photo in Fig. 1 it is clear that the *average* crack thickness is less than ¼ inch and also that at least six large bends are present. Thus, the criterion of less than a 20% increase in dose due to cracks in the concrete shield is easily met.

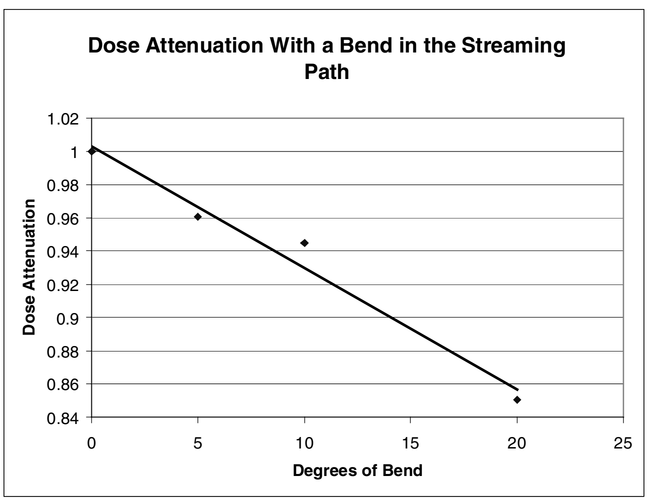

Figure 4. Dose attenuation as a function of bend angle for cylindrical void penetration (see Fig. 2a for geometry assumed).

REFERENCES

1. L.M. Petrie, P.B. Fox, and K. Lucius, "Standard Composition Library," Vol. III, Sect. M8 of *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation,* NUREG/CR-0200, Rev. 6 (ORNL/NUREG/CSD-2/R6), Vols. I, II, and III, May 2000. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-545.
2. J.S. Tang and M.B. Emmett, "SAS4: A Monte Carlo Cask Shielding Analysis Module Using An Automated Biasing Procedure," Vol. I, Sect. S4 of *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, NUREG/CR-0200, Rev. 6 (ORNL/NUREG/CSD-2/R6), Vols. I, II, and III, May 2000. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-545.
3. W.C. Jordan and S.M. Bowman, "SCALE Cross-Section Libraries," Vol. III, Sect. M4 of *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation,* NUREG/CR-0200, Rev. 6 (ORNL/NUREG/CSD-2/R6), Vols. I, II, and III, May 2000. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-545.
4. J.R. Knight, C.V. Parks, S.M. Bowman, L.M. Petire, and J. A. Bucholz, "SAS1: A One-Dimensional Shielding Analysis Module," Vol. I, Sect. S1 of *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation,* NUREG/CR-0200, Rev. 6 (ORNL/NUREG/CSD-2/R6), Vols. I, II, and III, May 2000. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-545.
5. B. T. Price, C. C. Horton, and K. T. Spinney, *Radiation Shielding,* Pergamon Press, 1957.