SKYSHINE DOSE FROM ON-SITE LOW-LEVEL WASTE STORAGE MODULES

M. E. Crotzer
Westinghouse Waste Technology Services Division
Madison, PA 15663

ABSTRACT

Preliminary analytical results are presented which address the skyshine radiation dose from selected radionuclides in radioactive waste stored in concrete on-site storage modules. Methods are described extending these results to modules of different size and thickness, and to other important radionuclides. The methods are applied to some on-site storage scenarios.

INTRODUCTION

Several operators of light water reactors (LWR's) have implemented or are considering plans for the on-site storage of their low-level radioactive waste (LLW). These plans have been motivated to a large extent by uncertainties associated with the LLW Policy Act of 1980, which could have denied operators access to existing disposal facilities in 1986. The recently enacted Compact Consent Bill extends access to existing disposal facilities until 1993. In return, however, waste generators must deal with additional surcharges, volume limitations, and milestones. Faced with additional costs and the possibility of missing a milestone, LWR operators are unlikely to alter their on-site storage plans.

The storage plans of LWR operators include reinforced concrete buildings, prefabricated buildings, and concrete and steel storage modules. The advantages and disadvantages of each option are summarized in Ref. 1. Whatever storage plan is used, the operator must perform and document a safety evaluation as required by 10CFR50.59, including consideration of on-site and off-site radiation doses. The Nuclear Regulatory Commission (NRC) has issued guidelines and requirements for the temporary on-site storage of LLW.[2] According to this position, off-site doses from direct radiation and effluent release must be limited to a small fraction of the 40CFR190[3] limit of 25 millirem (mrem) per year. On-site doses are to be controlled in accordance with 10CFR20,[4] including provisions to maintain doses as low as reasonably achievable (ALARA).

This paper presents preliminary results of analyses which can be used to answer certain questions during a 10CFR 50.59 safety evaluation of on-site storage modules. One likely question is how far from a site boundary or a non-restricted, on-site area should storage modules be placed? Alternatively, given a possible location, what are the requirements for shielding, module arrangement, or radionuclide concentration? Because the dose limits discussed above are quite low (25 mrem/yr. $\simeq 3 \times 10^{-3}$ mrem/hr.) and not readily detectable, an analytical approach is needed to address these questions during the planning stages. The approach must be able to predict dose rates for different distances, module thicknesses and sizes, and radionuclide concentrations.

This analysis focuses on the skyshine gamma radiation emitted by LLW contained in three types of Westinghouse concrete storage modules, called SUREPAK's. A typical SUREPAK is shown in Fig. 1. Relevant dimensions are listed in Table I. Skyshine radiation is distinguished from the radiation directly transmitted through the sides of storage modules. Skyshine results from radiation emitted upward through the top of the module that is scattered in the air volume above, and then directed back down toward the ground. It is lower in intensity and energy than the direct radiation. However, it can cover a large area and be the predominant source of radiation when the sides of the modules are well shielded. This situation would occur when buildings, landscape, or modules containing less radioactive material exist between the "hot" modules and the point of interest.

Fig. 1. A Typical SUREPAK.

The presentation of the analysis will proceed as follows: First the analytical method will be discussed, including a comparison of preliminary calculations with benchmark data. Graphs of SUREPAK

TABLE I

SUREPAK Dimensions

Module	Cavity Diameter	Cavity Height	Lid Thickness	Minimum Wall Thickness
SP-1	77 in (196 cm)	81 (206)	6 (15.2)	3 (7.6)
SP-2	77 (196)	81 (206)	17 (43.2)	15 (38.1)
SP-3	62 (157)	75 (191)	24.5 (62.2)	22.5 (57.2)

skyshine dose rate versus distance are then shown for some key radionuclides. Methods are presented which can extend these results to other radionuclides and to modules of different size and thickness. Finally, these results are applied to some examples, using waste volumes and radionuclide content from recent publications.

ANALYTICAL METHOD

The analysis was performed using the Discrete Ordinates Transport (DOT) code, DOT III W. DOT is a two-dimensional transport code which computes the space and energy-dependent particle flux for a given source distribution and geometry[5]. Physically, the geometry is divided radially and axially into small volumes. For each volume, DOT balances the incoming and outgoing gamma rays by energy and direction, thus accounting for both direct and scattered gammas. This balancing procedure continues iteratively until the problem converges. A DOT problem can be solved to various degrees of approximation and for different numbers of direction, depending on the accuracy desired (or the cost that can be incurred). The resulting dose rates from DOT are computed for each volume. This method is thus well-suited for analyzing movable storage modules because one computer run can generate dose rates for all points of interest. Other possible methods, such as Monte Carlo or point kernal techniques generally produce results only for a single location.

Because of the large density difference when going from concrete to air, the skyshine problem had to be performed in 3 steps. First, DOT was used to obtain the vertically-directed gamma flux from a single SUREPAK with a unit concentration of a major radionuclide (e.g. 1μCi/cc of Co-60). Next, the detailed flux from the first step was converted to an equivalent anisotropic point source. This source was then used as input to a second DOT run to compute the dose rates in air over large distances. The same degree of approximation was used for both DOT runs (P-3, 48 angles, 15 energy groups). This procedure was repeated for the key gamma-emitting radionuclides, (Co-60, Co-58, and Cs-137) and for SP-1's and SP-2's. At this point, simplified methods were investigated to address other important radionuclides and other module sizes.

Comparison to Data

Before performing the SUREPAK analyses, the procedure described above was tested, using results from a benchmark experiment[6]. This experiment was conducted with point sources of Co-60 in concrete silos. The silo walls were thick enough to shield against the direct, horizontal radiation. One test was conducted without any shielding on the silo top. Two more were conducted with concrete thicknesses of 21 cm and 43 cm.

In Fig. 2, the DOT results are compared to the NaI detector measurements described in Ref. 6. The measured exposures are plotted as micro-Roentgen per hr, normalized to one Curie of Co-60, multiplied by the square of the source-detector distance. This eliminates the geometry effect and condenses the data. For the horizontal axis, density x radius (ρR gm/cm^2) is used to account for variations in air density. The variable ρR is referred to as the areal density. For the DOT analysis, an air density of 0.00129 gm/cm^3 was used (0°C, 1 atm).

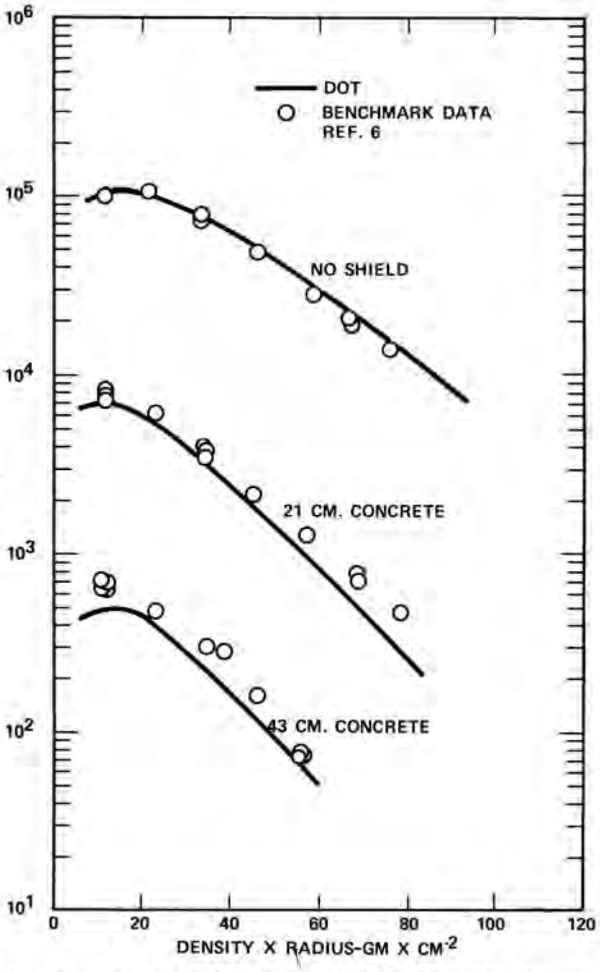

Fig. 2 Comparison of DOT Results to Benchmark Data.

Without any concrete, the DOT results compare well with the benchmark data. For 21 cm of concrete, the data is, on the average, a factor of 1.3 higher than the DOT results. For 43 cm of concrete, the data is a factor of 1.5 higher than the DOT results. This underprediction was also found by the authors of Ref. 6. The fact that the underprediction increases with increasing shield thickness may indicate that the problem lies in the low energy group structure used in the analysis. As the shield thickness increases, the skyshine gamma energy spectrum shifts toward lower energies because of the increased contributions from radiation that has scattered in the shield, before being directed upward. This effect would be greatest for radionuclides emitting relatively low energy gamma rays such as Cs-137. An improved low energy group structure could more adequately address this component of the skyshine radiation. Another problem may be the number of angles used in the analysis. This hypothesis is supported by the benchmark test discussed in Ref. 7, where the source was an open reactor pressure vessel. In this case, 166 angles were used to define the gamma flux above the vessel before converting to an equivalent point source. The Ref. 7 DOT calculations overpredicted the NaI measurements by a factor of 1.35. Further analysis is underway to improve the analytical method. Considering that the DOT results in Fig. 2 have the same trend as the data, a decision was made to proceed with the SUREPAK analysis using the current analytical method, except that the results would be multiplied by a factor of two to account for the underprediction.

RESULTS

Figures 3 and 4 show results of the analysis for an SP-1 and an SP-2, respectively. The results are plotted as annual dose, normalized to a unit concentration of a given radionuclide, and multiplied by the square of the source to point-of-interest distance, versus the areal density. The density of the waste material was assumed to be 1 gm/cm^3. Using simplifying assumptions, these results can be extended to other important radionuclides and to other storage modules.

Extension to Other Radionuclides

In addition to the isotopes shown in Figs. 3 and 4, Zn-65 and Mn-54 are also commonly found in LLW. Mn-54 has a yield and energy emission almost identical to that of Co-58 (835 Kev, 100% vs 810 Kev, 99%). Therefore, results for Co-58 can be used directly for Mn-54. Zn-65 emits 1115 Kev gamma rays with a yield of 51%. Comparing this to Co-60 (1173 Kev and 1332 Kev, 100%), a reasonable estimate of the dose effect for Zn-65 can be obtained by taking one-fourth of the Co-60 result.

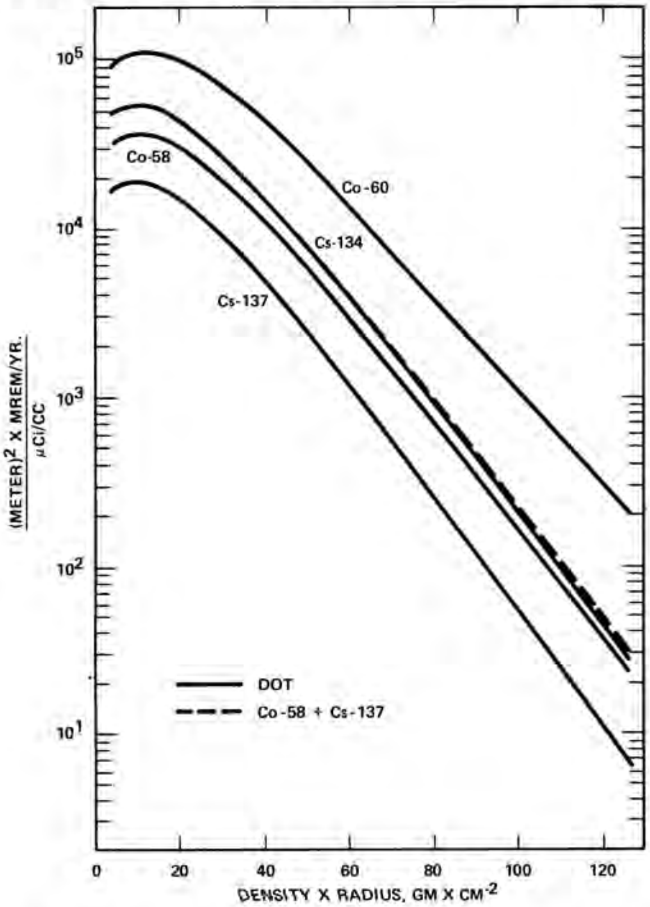

Fig. 3. Normalized Annual Skyshine Dose versus Areal Density for an SP-1.

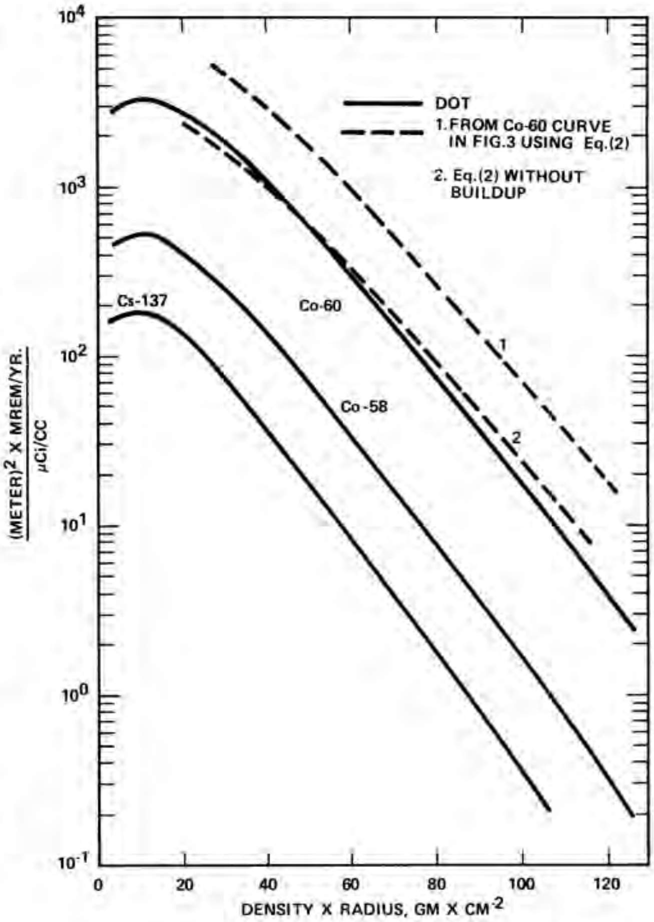

Fig. 4. Normalized Annual Skyshine Dose versus Areal Density for an SP-2.

A comparison of the energy and yield structures of Co-58, Cs-137, and Cs-134 suggests that the Cs-134 dose can be approximated by adding the Co-58 and Cs-137 results. The dotted line in Fig. 3 shows that this is a good approximation.

Extension to Other Storage Modules

The results of Figs. 3 and 4 can be extended to storage modules of other sizes and thicknesses by first noting that the scattered skyshine radiation is proportional to the radiation transmitted vertically through the top of the module. At a distance a, far above a storage module of thickness t and area A, the gamma flux, F can be estimated for a given radionuclide by the following relationship:

$$F = K \frac{CA}{\mu_s} \frac{Be^{-\Sigma\mu_i t_i}}{4\pi a^2} \qquad (1)$$

where C is the radionuclide concentration of the module contents and μ_s is the linear attenuation coefficient of the contents. K is a constant that converts the radionuclide concentration to a gamma emission rate. The expression $\Sigma\mu_i t_i$ accounts for the attenuation of the radiation through the module lid and intervening air space. The buildup factor B accounts for the additional radiation that results from scattering in the module lid and the intervening air space. This latter radiation, although lower in energy than the unattenuated radiation, can still contribute to skyshine, especially at distances close to the source.

Given a skyshine dose, D_1, for module 1, the above development suggests that, for the same source material and radionuclide concentration, the skyshine dose for module 2 can be estimated as follows:

$$D_2 = D_1 (A_2/A_1) (B_2/B_1) e^{-\mu (t_2-t_1)} \tag{2}$$

where μ is the linear attenuation coefficient for the lid material.

The above expression was used to estimate the SP-2 Co-60 dose response from the SP-1 response. The results are shown by the dotted curves in Fig. 4. Curve 2 shows the results without the buildip factor effect. This suggests that the buildup radiation does not contribute significantly to the skyshine dose, especially at large distances from the source. Similar results are obtained for the other key radionuclides. Although buildup appears to be unimportant, it should be included in applications for conservatism, especially for thick lids and low-energy gamma-emitting radionuclides.

APPLICATIONS

Dose Criteria

Several designers have limited the site boundary dose rate to 10% or less of the Ref. 3 limit of 25 mrem/yr. (Ref. 8 for example). For the current application, a dose limit of 2 mrem/yr is assumed. For on-site dose considerations, the low radiation zone suggested in Ref. 9 is assumed (500 mrem/yr).

Storage Scenarios

Two basic storage scenarios are presented. The first addresses multiple SUREPAK's containing typical BWR LLW. According to Ref. 10, an average BWR generates about 269 m^3/yr (9510 ft^3/yr) of resins, sludges, and concentrates. The volume-weighted industry average radionuclide distribution for this component of BWR waste is listed in Table II. This quantity of waste can be handled by 70 SP-1's or SP-2's. The second scenario addresses a single SP-2 containing resins with a high loading of fission products. The radionuclide distribution for this case is taken from Ref. 11 and listed in Table II. Note that this second secenario is not a typical case. This example is indicative of a nuclear reactor with significant fuel defects.

Scenario Results

The results of these applications, including some variations, are summarized in Table III. Figure 5 shows dose versus distance curves for these scenarios. Figure 5 was constructed from the information in Figs. 3 and 4 and Tables I and II, along with the methods described earlier. An air density equivalent to 20 C was used to calculate the distances (0.00120 gm/cm^3).

TABLE II

Radionuclide Concentrations for Two Cases (μCi/cc)

Radionuclide	Typical BWR LLW(1)	High Resin Loading of Fission Products(2)
CO-60	1.7	30.1
CO-58	-	-
Cs-134	0.5	546.
Cs-137	0.9	757.
Zn-65	0.9	128.
Mn-54	0.3	-

(1) Reference 10

(2) Reference 11

TABLE III

Distances Corresponding to Selected Dose Limits for Various Storage Scenarios

	Case	Distance at 2 mrem/yr	Distance at 500 mrem/yr
1.	70 SP-1's with typical BWR radwaste	630 meters	190 meters
1A	Case 1 - SP-1's stacked 2-high	570	140
1B	Case 1 - Lid thickness increased by 5 cm	570	140
1C	Case 1 - with SP-2's	310	<30
2.	Resins with high loading of fission products in one SP-2	330	40
2A	Case 2 with an SP-3	180	<30

For case 1, it is assumed that the 70 SP-1's are arranged in a roughly circular area of 10 meters radius, surrounded by a ring of SP-1's containing low-activity dry, active waste. This effectively shields the horizontally-directed radiation, making skyshine the predominant source. The SP-1's are presumed not to be stacked. Stacking would reduce the skyshine dose, but could complicate handling and inspection procedures. Figure 5, *curve* 1 and Table III show that the off-site criteria would be met for distances greater than 630 meters from the site boundary. The on-site criteria could be met for distances greater than 190 meters. Stacking the SP-1's 2-high, (an effective source of 35 SP-1's) or adding 5 cm (2 inches) of additional concrete marginally reduces the distances. Using SP-2's would significantly reduce these distances. However, as discussed in Ref. 1, SP-1's and their contents could be shipped over the road with suitable overpacks to a disposal facility. Additional on-site handling of the waste is thus avoided. This option is lost with

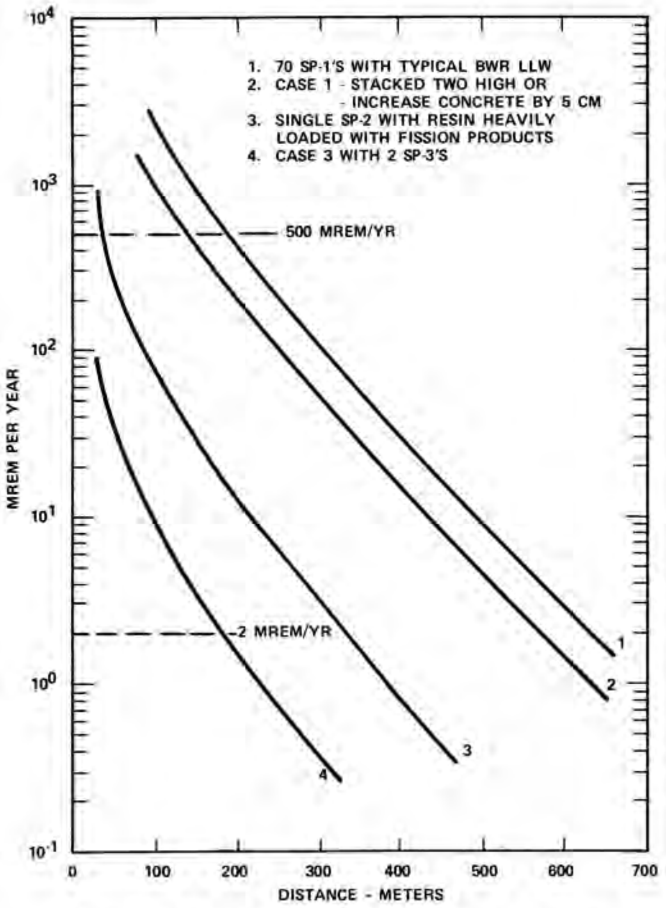

Fig. 5. Annual Skyskine Dose vs Distance for Selected Storage Scenarios (20C).

SP-2's. However, the additional shielding afforded by the SP-2, especially in the vicinity of the modules, may be an overriding benefit.

Results for the SUREPAK's containing highly radioactive resins are also listed in Table II. For case 2, it is noteworthy that Co-60, which comprises only 2% of the total activity, contributes 18% to the total skyshine dose at 400 meters for an SP-2 and 23% of the total for an SP-3 at 300 meters. Also, Zn-65 contributes about the same percent as Co-60. This illustrates the highly penetrating and far-reaching effect of Co-60 relative to Cs-134 and Cs-137.

SUMMARY

This paper has presented preliminary results addressing the skyshine radiation emitted by LLW contained in on-site storage modules. The figures and methods presented address modules of different size and thickness as well as LLW having different compositions of gamma-emitting radionuclides. They can be used to help an LWR operator locate storage modules on his site consistent with meeting regulations. The results could be equally useful to operators of LLW disposal sites in planning for the disposition of waste containers.

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