Evaluation of Gamma Radiation Shielding for Nuclear Waste Shipping Casks*

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SUMMARY

A method has been developed for evaluating gamma radiation shielding of shipping casks that are used to transport nuclear waste with ill-defined radionuclide contents. The method is based on calculations that establish individual limits for a comprehensive list of radionuclides in the waste, assuming that each radionuclide is uniformly distributed in a volumetric source in the cask. For multiple radionuclide mixtures, a linear fraction rule is used to restrict the total amount of radionuclides such that the sum of the fractions does not exceed 1. As long as the radionuclide limits and the linear fraction rule are followed, it can be shown that the regulatory dose rate requirements for a cask will be satisfied under normal conditions of transport and in a hypothetical accident during which the shielding thickness of the cask has been reduced by 40%.

INTRODUCTION

In evaluating the gamma radiation shielding of nuclear waste shipping casks, one issue that is frequently encountered is the specification of the radionuclide contents in the nuclear waste. While precise information on the radionuclide contents is highly desirable in order to accurately define the radiation source terms, this precise information is often difficult to obtain due to practical or economic reasons. Under normal conditions of transport (NCT), the lack of knowledge of the radionuclide contents does not cause serious problems because a radiological survey of the shipping cask before shipment can be used to ensure that the measured dose rates would remain below the regulatory limits, e.g., ≤2 mSv/h at any point on the external surface of the cask and ≤0.1 mSv/h at any point 2 m from the surface of the cask. For hypothetical accident conditions (HAC), additional considerations may be necessary, depending on whether the effectiveness of the shielding of the cask has been reduced in the hypothetical accident tests. If the tests show no reduction of shielding effectiveness, the dose rate at 1 m from the surface of the cask should be less than the regulatory limit of 10 mSv/h because the dose rate at the surface of the cask should remain ≤ 2 mSv/h. However, difficulty arises when the cask sustains damage in hypothetical accident tests (such as during a drop of the cask onto a puncture bar) that reduces the shielding thickness. This situation requires analysis, which necessitates specification of the radionuclide source terms that are generally ill-defined for nuclear waste. The method by which such an analytical evaluation may be performed is the subject of this paper.

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METHOD OF ANALYSIS

To illustrate the method of analysis, a Type 304 stainless steel cylindrical cask containing an inner vessel and a carbon steel payload canister is used. A cutaway view of the cylindrical cask (Fig. 1) shows the dimensions of the cask and the other components, which include a Type 304 stainless steel thermal shield, lead shielding, inner vessel radial spacer, and air gaps. Several packaging configurations are possible for the nuclear waste by using cans and drums inside the payload canister, or by direct packing of the waste into the canister. For simplicity, the latter configuration is chosen for analysis and the radionuclides in the waste are assumed to be uniformly distributed inside the volume of the canister, which has a diameter of 65 cm and a height of 306 cm.

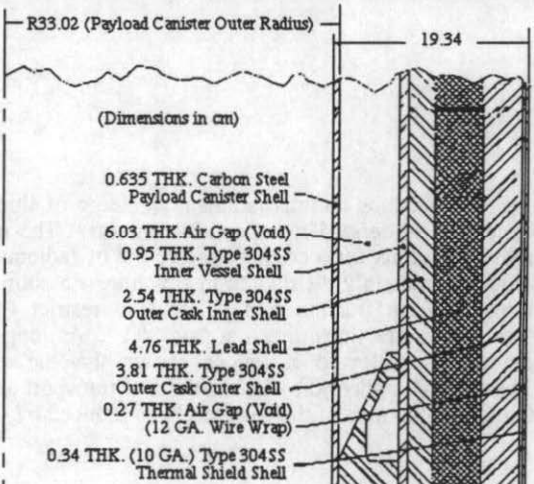

Figure 1 Radial shielding configuration for nuclear waste shipping cask

The calculational tool used in the analysis is a PC version of MicroShield 5.1 (User's Manual), which is one of the widely used point-kernel software programs for gamma radiation shielding analysis. There are many built-in geometrical models in MicroShield 5.1 for various combinations of source and shield geometry. The model that closely represents the shipping cask and that has been selected for this work is the Cylinder Volume - Side Shields, with which the entire radial shielding configuration depicted in Fig. 1, except for the inner vessel radial spacer, has been modeled as concentric cylinders. (The radial spacer can be conservatively ignored and treated as part of the air gap.) Inputs on the material densities for the concentric cylinders have used default values in the code library, as well as user-defined "custom" material for Type 304 stainless steel. The source region is assumed to be an iron matrix with a density of 1.6 g/cm^3 (100 lb/ft^3), which is typical of the gross density of waste drums.

In a point-kernel-based method, the energy-dependent gamma radiation dose rate D(E) at a distance R from a source of activity S(E) is calculated as

$$D(E) = \frac{S(E)B(E,x)K(E)}{4\pi R^2} e^{-x(E)}, \tag{1}$$

where x(E) is the number of mean free paths of the scattering events, B(E, x) is the buildup factor that accounts for photon scattering in the shield materials, and K(E) is the integral of unit response of point kernels representing differential volumes of the source. In all of the MicroShield 5.1 calculations performed for this work, K(E) has been evaluated by numerical integration using Gauss quadrature with default parameters, and the lead shield has been selected as the reference material for calculating the buildup factor B(E, x). The lead shield is chosen because it is the most dominant shield of the cask and has the largest number of mean free paths in photon scattering.

Unlike conventional dose rate calculations for which the source term is known, a reverse problem is solved for the maximum activity limit of S_i (E) of a gamma-emitting radionuclide i, i.e.,

$$S_i(E) = \frac{4\pi R^2 D(E)}{B(E,x)K(E)} e^{x(E)} \tag{2}$$

where the dose rate D(E) is set at the regulatory limits under normal conditions of transport, e.g., 2 mSv/h at the surface of the cask and 0.1 mSv/h at 2 m from the surface of the cask. Figure 2 shows the dose points at which the regulatory limits are assessed at the axial midplane of the cask.

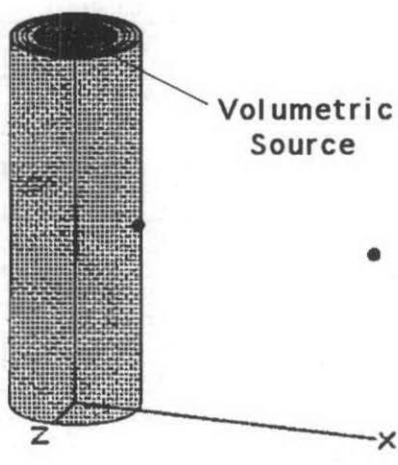

Figure 2 Concentric cylinder model for the nuclear waste shipping cask. Dose point or detector locations (dots) are shown on the surface and at 2 m from the surface, both at the axial midplane of the cask.

Table 1 lists the radionuclides for which the source activity limits have been determined in the MicroShield 5.1 calculations for an undamaged cask under normal conditions of transport (NCT). The list of radionuclides is generally compiled from process knowledge of the nuclear waste, which should be comprehensive and cover all of the important radionuclides in the waste. The activity limits (TBq) in the second column of Table 1 are determined on the basis of the 2 mSv/h dose rate at the surface of the cask, whereas the activity limits (TBq) in the third column of Table 1 are determined on the basis of the 0.1 mSv/h dose rate at 2 m from the surface of the cask. For each radionuclide, the activity limit determined by the 2 m dose rate requirement is always lower than the corresponding limit determined by the surface dose rate requirement. Because the cask must satisfy both regulatory dose rate requirements, the lower NCT activity limits based on the 2 m dose rate requirement thus establish the overall activity limits for each of the radionuclides in the cask.

Table 1 NCT activity limits and calculated HAC dose rates for individual radionuclides in the shipping cask

Radionuclide	NCT Limit Based on 2 mSv/h at Surface (TBq)	NCT Limit Based on 0.1 mSv/h at 2 m (TBq)	HAC Dose Rate at 1 m Based on 2 m NCT Limit (mSv/h)
Co-60	1.459e+1	3.952e+0	0.538
Kr-85	5.021e+7	1.259e+7	2.978
Sr-89	1.879e+6	4.921e+5	0.760
Sr-90		No gammas	
Y-90m	5.261e+5	1.326e+4	0.300
Y-91	1.278e+4	3.443e+3	0.561
Nb-95m	2.928e+8	7.436e+7	1.532
Nb-95	1.309e+3	3.379e+2	0.995
Zr-95	1.766e+3	4.547e+2	1.047
Ru-103	1.750e+5	4.413e+4	2.383
Rh-103m		Unlimited	
Ru-106		No gammas	
Rh-106	1.378e+3	3.728e+2	0.566
Sn-119m		Unlimited	
Sb-125	3.842e+4	9.752e+3	1.586
Te-125m		Unlimited	
Te-127m	7.014e+7	1.786e+7	1.416
Te-127	1.653e+9	4.130e+8	8.841
Cs-137		No gammas	
Ba-137m	6.942e+3	1.769e+3	1.349
Ce-141		Unlimited	
Ce-144		Unlimited	
Pr-144	3.463e+2	9.790e+1	0.436
Pm-147		Unlimited	
Sm-151		Unlimited	
Eu-155		Unlimited	

The last column in Table 1 contains the dose rates calculated for a cask that has been damaged in a puncture bar test prescribed for the hypothetical accident conditions (HAC). The damage, shown schematically in Fig. 3, consists of a local reduction of the thickness of the lead shield by 40% in a circular area with the diameter of the puncture bar (15.24 cm). In the MicroShield 5.1 calculations performed for the HAC dose rates at 1 m from the surface of the cask, the local puncture bar damage is replaced by a uniform reduction of the lead thickness by 40%; this greatly simplifies the modeling while also being conservative, because significantly less lead shield material is considered in the calculations. The resulting dose rates obtained for such a damaged cask, with source activities given by Column 3 of Table 1, are all considerably less than 10 mSv/h, thus satisfying the regulatory dose rate requirement for HAC.

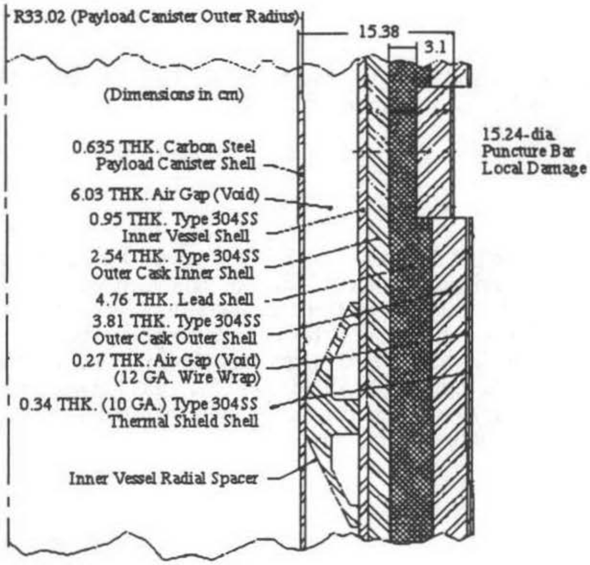

Figure 3 Radial shielding configuration for puncture-bar-damaged shipping cask

MULTIPLE RADIONUCLIDE MIXTURES

The results presented so far have shown that both NCT and HAC regulatory dose rate requirements are satisfied when the activities of the individual radionuclides are restricted to levels below the limits given in the third column of Table 1. For multiple radionuclide mixtures in the cask, a linear fraction rule

$$\sum_{i=1}^{26} \frac{s_i}{S_i} \leq 1 \tag{3}$$

can be imposed on the number of radionuclides and their activities (s_i) such that the sum of the fractions s_i/S_i, where S_i is the activity limit for radionuclide i, does not exceed 1. (There are 26 radionuclides listed in Table 1.) Equation 3 may seem to represent too many possible radionuclide combinations. However, it is important to realize that the choice of radionuclide is not arbitrary, but is subject to a constraint as each radionuclide is added to the mix. For example, when there are two radionuclides and after the first one is chosen with an activity of s_1, the activity s_2 of the second radionuclide must obey

$$\frac{s_1}{S_1} + \frac{s_2}{S_2} \leq 1 \rightarrow s_2 \leq S_2\left(1 - \frac{s_1}{S_1}\right). \tag{4}$$

When there are three radionuclides in the mix and after the first two are chosen, the activity s_3 of the third radionuclide must similarly obey

$$\frac{s_1}{S_1}+\frac{s_2}{S_2}+\frac{s_3}{S_3}\le 1 \rightarrow s_3 \le S_3\left[1-\left(\frac{s_1}{S_1}+\frac{s_2}{S_2}\right)\right]. \quad (5)$$

For n radionuclides in the mix and after s_{n-1} has been specified for the $(n-1)_{th}$ radionuclide, the activity s_n of the n_{th} radionuclide is then limited by

$$s_n \le S_n\left(1-\sum_{i=1}^{n-1}\frac{s_i}{S_i}\right), \quad (6)$$

which is a recursive relationship for n from 2 to 26. Equations 4-6 clearly demonstrate that the linear fraction rule imposes restrictions on the selection of radionuclides and their activities. It has been shown in MicroShield 5.1 calculations that for arbitrarily selected mixtures of radionuclides with their S_i's given in Table 1 and their s_i's chosen according to Eqs. 3-6, the calculated dose rates have satisfied both the NCT and the HAC regulatory dose rate requirements for the shipping cask. The calculational models used in the MicroShield 5.1 calculations for the undamaged and damaged casks are the same as those used before for individual radionuclides; only the source terms were modified for the radionuclide mixtures.

DISCUSSION

As mentioned at the beginning of this paper, the lack of knowledge of the contents (composition, activity, and distribution of radionuclides) in the nuclear waste shipment does not cause serious problems because a preshipment radiological survey of the cask can be used to ensure that the regulatory dose rate requirements for NCT will be met. The difficulty has been the HAC evaluation, which could be performed experimentally with a mockup damaged cask, or analytically as shown in this paper. A key assumption made in this work is the volumetric geometry for the source terms, which is judged to be much more realistic than an alternative assumption of a point source being placed on the inner surface of the payload canister. Such a point-source configuration has been known to produce activity limits for individual radionuclides that are far lower than those determined on the basis of the volumetric source geometry. Also, in such a point-source configuration, the overall NCT activity limits would have been those determined on the basis of the surface dose rate requirement, not the 2 m dose rate requirement as in the case of the volumetric source geometry.

One serious problem with the above point-source configuration is that there can be only one spot on the entire inner surface of the payload canister for the radionuclide point source, and that spot must be directly across the wall of the cask from the dose point detector, whose location is assumed to be on the outer surface at the axial midplane of the cask. Any change of the relative position of the point source and the detector can drastically lower the dose rate, especially when the distance between the point source and detector becomes large. The assumption of such a point-source geometry, therefore, is extremely restrictive and bears little relationship to the preshipment radiological survey of the cask.

With the volumetric source geometry in a "low-density" iron matrix, there is no restriction on the location of radionuclides in the payload canister other than the assumption of uniform distribution. Because of axisymmetry in the source geometry (see Fig. 2), the dose rates are constant anywhere along the circumference of the cask and also maximum at the axial midplane of the cask. The spatial distribution of the sources and dose rates from a volumetric source, while still somewhat idealistic, is a more realistic representation of the radiological survey during which the detector is being moved around in the entire vicinity of the cask.

In the HAC shielding evaluation for a damaged cask, an "incredible" worst case may be construed such that the entire volumetric source would somehow consolidate into a point source during the puncture event, and that this consolidated point source would happen to be located on the inner surface of the cask directly across the remaining wall from the center of the puncture hole. A quick calculation using 3.95 TBq of Co-60 for the consolidated point source obtained a total dose rate of 26.6 mSv/h at a distance of 1 m from the bottom of the puncture hole, which exceeds the HAC limit of 10 mSv/h. Thus, this worst-case consolidated Co-60 point source has failed to satisfy the HAC requirement of ≤ 10 mSv/h. (Note: This worst case is not considered a credible event; it is included here for illustration only.)

CONCLUSIONS

A method has been developed to evaluate the gamma radiation shielding of shipping casks that are used to transport nuclear waste with ill-defined radionuclide contents. The method is based on calculations performed to establish individual limits for a comprehensive list of radionuclides in the waste, assuming that each radionuclide is uniformly distributed in a volumetric source in the cask. For multiple radionuclide mixtures, a linear fraction rule is used to restrict the total amount of radionuclides such that the sum of the fractions does not exceed 1. Using the activity limits and the linear fraction rule, we have demonstrated that as long as a set of radionuclides satisfies the NCT regulatory dose rate requirements, that same set of radionuclides would also satisfy the HAC regulatory dose rate requirement for a damaged cask with a 40% reduction in the thickness of its lead shield. The use of volumetric geometry for the source terms provides a realistic linkage between the method of analysis and the preshipment radiological survey of the cask.

REFERENCE

MicroShield Version 5.1, User's Manual, Grove Engineering, 1700 Rockville Pike, Suite 525, Rockville, MD 20852, 1997