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SHIELDING FOR CYLINDRICAL SOURCES

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DISCLAIMER

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INTRODUCTION

Predictions of external dose rates for casks of irradiated fuel are often highly conservative, partly to allow for uncertainties in shielding data and calculative procedures. A factor of 4 or 5 between calculation and observation is considered by many designers to be within the normal estimating range, although it may represent about an inch of lead or equivalent shielding. Sometimes this is a significant economic penalty for a new cask. Also, when an existing cask is to be used for a purpose not originally intended, calculations may lead to unnecessary modification or to an incorrect conclusion that the cask is unsuitable. On the other hand, it may turn out that calculated values are low, with adverse consequences.

For neutron shielding, the two basic methods of calculation in common use are solution of the transport equation by discrete ordinates S_n codes (e.g., ANISN[1] in one dimension, and TWOTRAN[2] or DOT[3] in two dimensions) and application of statistical procedures with Monte Carlo codes (e.g., MORSE[4]). ANISN is most frequently used because changes to input data are fairly easy to make, and for ordinary problems the computer CPU time is less than a minute. TWOTRAN and MORSE are much more difficult to set up, and are reported to require 20 to 40 times as much computer time; these codes were not used in the present study.

For gamma shielding, the same methods as for neutron shielding may be used, but in addition the point kernel integration procedure with application of buildup factors is available and often advantageous. The point kernel technique can be applied to a wide range of geometries, is amenable to manual calculation in many cases, and has been embodied in fast-running computer codes such as QAD[5] and SDC.[6]

The principal purposes of the investigation are to provide some guidance to the cask designer in the use of ANISN and SDC for a cylindrical source, and to present data on the effectiveness of cylindrical shields (customary for casks) compared with flat slab shields (assumed in SDC and in usual manual calculations). For illustration, calculations with ANISN and SDC were made for a homogeneous cylindrical source of 40 cm radius. Also included are examples of the effectiveness of water and uranium for neutron shielding.

* Work done under Contract No. AT(07-2)-1 with the U. S. Department of Energy.

MODEL FOR CALCULATIONS

An array of fuel elements can often be satisfactorily approximated by a homogeneous cylindrical source. The hypothetical cask for illustrating the effect of varying certain input parameters and for comparing different methods of calculation was as follows:

Source:	Radius:	40 cm
	Length:	Infinite
	Density:	2.15 gm/cm^3
	Material:	Lead (to approximate the gamma self-absorption properties of 78% UO_2, 22% stainless steel)
	Neutron energy spectrum:	^{252}Cf
	Photon energy:	Various
Shield:	Fe:	2 cm
	U or Pb:	Variable thickness
	Fe:	3 cm
	H_2O:	Variable thickness
	Fe:	1 cm

Dose points were selected at various distances from the cask surface.

ANISN Calculations

ANISN is limited to one-dimensional calculations in which source and shield are infinitely long cylinders (or cylindrical shells), spheres (or spherical shells), or infinite flat slabs. Different versions are in use; more recent ones use improved techniques for interpolation and convergence.

For this study, all neutron shielding calculations were made assuming no primary photons and using a 22-group set of coupled n,γ cross sections with 13 groups assigned to neutrons. Gamma shielding calculations were made using only the 21 photon groups of a 58-group set. The source and the materials of the shield were taken each to be one zone; thus, there were six zones from the centerline to the surface of the cask.

One item of input data is the number of nodes or mesh points in each zone. The effect of varying the spacing is shown in Figures 1 and 2. Too small a number of mesh points results in a gross overestimate of dose rate. It appears that a spacing of about 1 cm in water and 0.5 cm in uranium (or inversely as density for other metals) is suitable; there is not a significant gain in accuracy for closer spacing.

If the dose conversion factors used in the program take account of absorption in tissue (as is done in NCRP Report No. 38[7] and ANSI/ANS 6.1.1 1977[8]), the receptor should be taken as a point in air. Otherwise, a person as a receptor is sometimes represented by 30 cm of water (or a material of "standard man" composition). In the cylindrical geometry, this is an annulus 30 cm thick. For points close to a cask, such representation may be appropriate; at distances of a meter or more, it results in a large overestimate of the thermal neutron dose rate. The increase with distance indicated by the results plotted in Figure 3 is unrealistic.

Often the cask designer has need for a rule-of-thumb on the thickness of water or equivalent to reduce the dose rate due to neutrons by a certain amount. Table 1 summarizes results for different conditions, with dose rates being at 82 cm from the centerline of the source for Cases 1-12 and 87 cm for Cases 13-15.

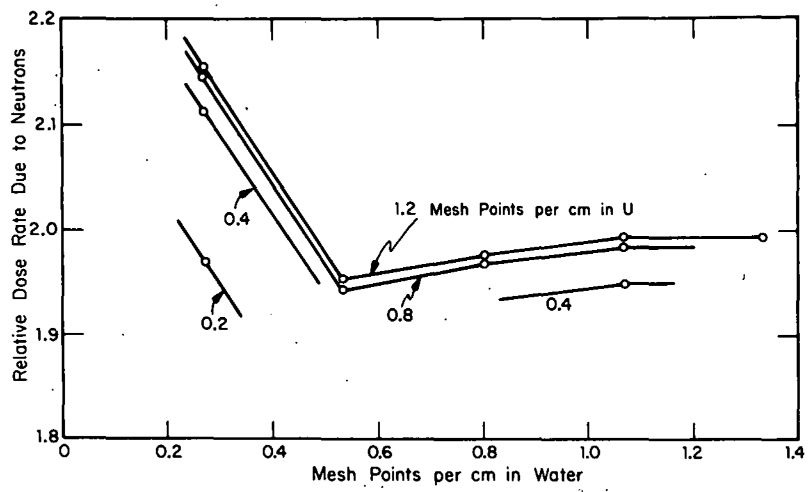

FIGURE 1. Effect of Mesh Point Spacing in Water and in Uranium for the Neutron Shielding Calculation with ANISN

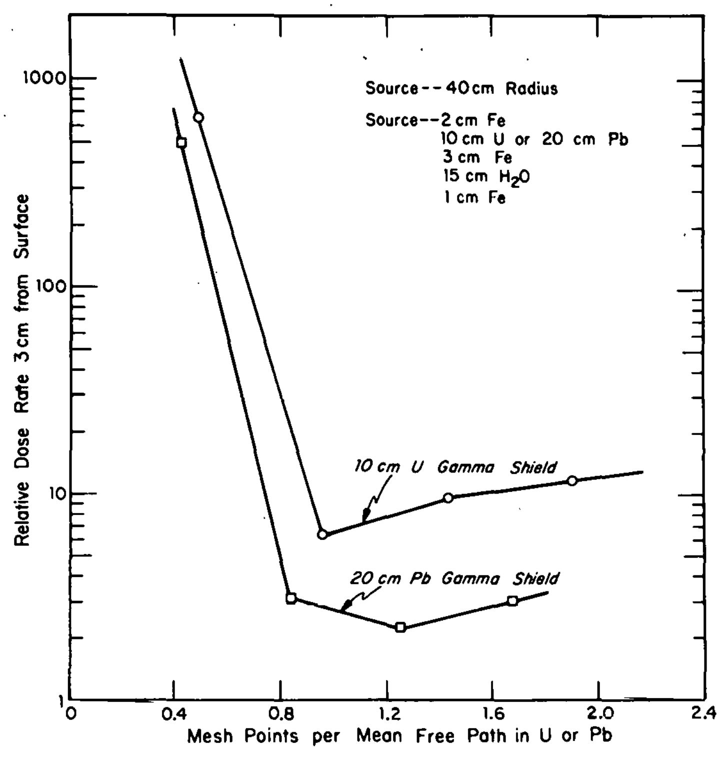

FIGURE 2. Effect of Mesh Point Spacing in Gamma Shield for Calculation by ANISN with 2.0-2.5 MeV Source

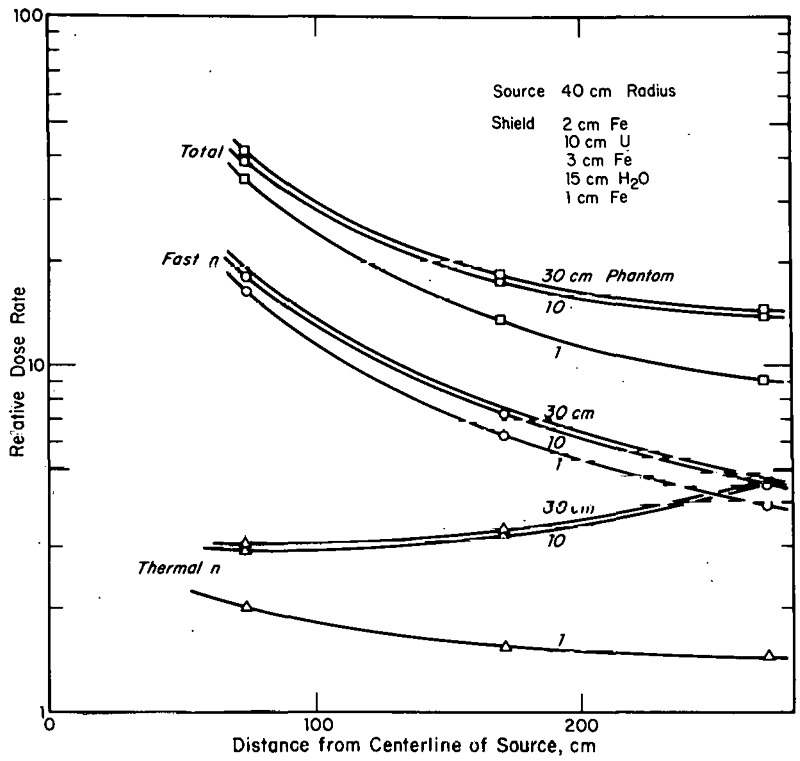

FIGURE 3. Dose Rate Due to Neutrons vs. Distance from Source for Different Thicknesses of Phantom Man

TABLE 1

Effectiveness of Water as Neutron Shield

			Cm H_2O		
Case	*Cm U*	*Cm Pb*	*Inside*[a]	*Outside*[a]	*Relative Dose Rate*[b]
1	5	0	0	0.1	168
2	5	0	0	7	27
3	5	0	0	15	6.4
4	5	0	0	23	2.7
5	10	0	0	0.1	107
6	10	0	0	7	15
7	10	0	0	15	3.5
8	10	0	0	23	1.6
9	0	10	0	0.1	219
10	0	10	0	15	7.5
11	0	20	0	0.1	211
12	0	20	0	15	5.9
13	0	20	0.1	19.9	1.6
14	0	20	10	10	0.9
15	0	20	19.9	0.1	2.2

a. Inside or outside the gamma shield.
b. Total dose rate due to fast neutrons, thermal neutrons, and secondary gammas.

The shell of the external water jacket was assumed to be 1 cm of iron. For 15 cm or more of water, the secondary gammas were the principal contributor to dose rate. In an actual situation, the relative amounts and locations of neutron and gamma shielding would normally be adjusted to achieve more nearly optimum results. However, as a rough guide, it was found that 10 cm of water reduced dose rates due to neutrons by a factor of about 10, that 20 cm of water reduced dose rates by a factor of about 50, and that 10 cm of uranium reduced dose rates by a factor of about 3.

An advantage of ANISN for both neutron and gamma shielding calculations is that (unlike SDC for gamma shielding) it takes account of the cylindrical shape of the shield. Disadvantages are that it assumes infinite length and that the input may have some photon groups covering too wide an energy range.

For gamma shielding calculations, adjustment for the finite length of source can be made by using factors derived from SDC or TID-7004.[9] A similar adjustment may be feasible for neutron calculations by determining the number of "equivalent mean free paths," although this has not been confirmed by recourse to codes such as TWOTRAN or MORSE.

The 22-group set of cross sections in use at SRL has one photon group for the energy range 1.0 to 2.0 MeV. The 58-group set used for gamma shielding calculations in this study includes groups of energy ranges 0.7 to 1.0, 1.0 to 1.5, 1.5 to 2.0, and 2.0 to 2.5 MeV. For a point source of a given number of photons per second and 10 cm of uranium shielding, the relative dose rates are approximately in the ratio 500:100:1 for 2.0, 1.5, and 1.0 MeV, respectively. The need for attention to group boundaries is evident. The "effective average energy" for each ANISN group has not been determined, but may be close to the midpoint of the range.

Results of gamma shielding calculations by ANISN are compared with results of other methods in Table 2.

TABLE 2

Relative Dose Rates for Different Methods of Gamma Shielding Calculation

Principal γ-Shield	*Distance from Centerline, cm*	*Source, MeV*	*Relative Dose Rate* A *ANISN*	B *SDC*	C *TID-7004*	*Point Kernel Integral*
5 cm U	74	0.7	41	0.08	1.0	1.0
		1.0	3800	320	210	180
		1.5	29000	3100	6300	5900
		2.0	77000	40000	25000	24000
		2.5		74000	55000	53000
	271	0.7		0.01*	0.22*	0.29
		1.0		43*	72*	53
		1.5	7400	440*	1600*	1600
		2.0		6100*	6700*	6400
		2.5		11500*	15000*	14000
10 cm U	74	0.7	0.004	$2x10^{-7}$*	$2x10^{-5}$*	$2x10^{-5}$
		1.0	11	0.30	0.17	0.15
		1.5	250	11	35	33
		2.0	1040	500	270	260
		2.5		1160	810	760
	271	0.7	0.001	$3x10^{-8}$*	$5x10^{-6}$*	$7x10^{-8}$
		1.0	2.8	0.032*	0.036*	0.040
		1.5	62	1.4*	8.7*	9.3
		2.0	260	70*	70*	74
		2.5		160*	210*	210

* These values were off-scale on the charts and required extrapolation of certain functions.

Point Kernel Techniques

Integration of the expression $\exp(-\mu t)/\rho^2$ over the volume of the source yields a reasonably accurate estimate of the uncollided photon flux. A buildup factor to account for scattered photons may be incorporated in the kernel before integration, or applied later. The buildup factors commonly used are based on infinite media and either a point source or a collimated source. These approximations introduce significant uncertainty for heavily shielded containers, where as much as 80% of the dose rate may be due to scattered photons.

Other things being constant, when a shield is moved away from the source and toward the receptor, the dose rate increases. This phenomenon is related to the angular distribution of scattered photons. Similarly, as the receptor moves away from a cask, the dose rate decreases more rapidly than would be predicted by the point kernel method. The assumption of an infinite medium is more appropriate for points on the surface of the cask than for points far removed from the cask.[10] An empirical fit to observed data for both cylindrical and rectangular casks, showing effect of distance, has been generalized in charts in Section 7 of the Cask Designers Guide, ORNL-NSIC-68.[11]

The cylindrical shield of a cask interposes a greater average thickness between source and receptor than the flat slab shield assumed by SDC and by the charts of TID-7004. Taking account of the cylindrical shape results in a lower calculated value for uncollided flux, by a factor as much as three in some cases. Computer programs were formulated at SRL to calculate the integrals for cylindrical sources. It was found that the ratio of uncollided fluxes for cylindrical vs. flat slab shield is almost invariant with length of source. Results for practical application are consolidated in two nomographs (Figures 4 and 5).

The calculated values for a flat slab shield were compared with values derived from the curves of TID-7004 (used also in the SDC program).[9] The latter, which are recognized to be only approximations, were found to be in error by as much as -40% to +30%.

Point kernel techniques have the advantage of being applicable to many different geometries, and in particular can take account of the finite length of a cylindrical source. Also, they permit precise specification of source energy, rather than an energy group. A disadvantage is the uncertainty in buildup factors, particularly with composite shields. SDC requires assuming a shield of a single material; this is generally taken to be of thickness corresponding to the total number of mean free paths through the actual composite shield.

Table 2 lists results of gamma shielding calculations for the hypothetical cask by four different methods:

A. ANISN
B. SDC
C. TID-7004 charts and factors[9]
D. Computer values of point kernel integrals, and buildup factors from SDC library

For B, C, and D, adjustment for the cylindrical shape of shield was made by means of Figure 4. ANISN automatically takes this into account.

Values marked with an asterisk were off-scale on the charts and required extrapolation of certain functions. For C, this was done by a procedure which should be fairly accurate. The manner in which SDC performs such extrapolation was not determined, but it may account for the rather large discrepancies for the 0.7 and 1.5 MeV cases.

Of the point kernel methods, D should be the most reliable. ANISN results, which are offset in the table to indicate that the MeV energies immediately above and below are the group boundaries, are sometimes inside and sometimes outside the group limits calculated in D. Direct comparison of ANISN results with the others is not possible, but the tabulation emphasizes the need for a finer group structure, particularly in the 1 to 2 MeV range which is important for shielding of irradiated fuel.

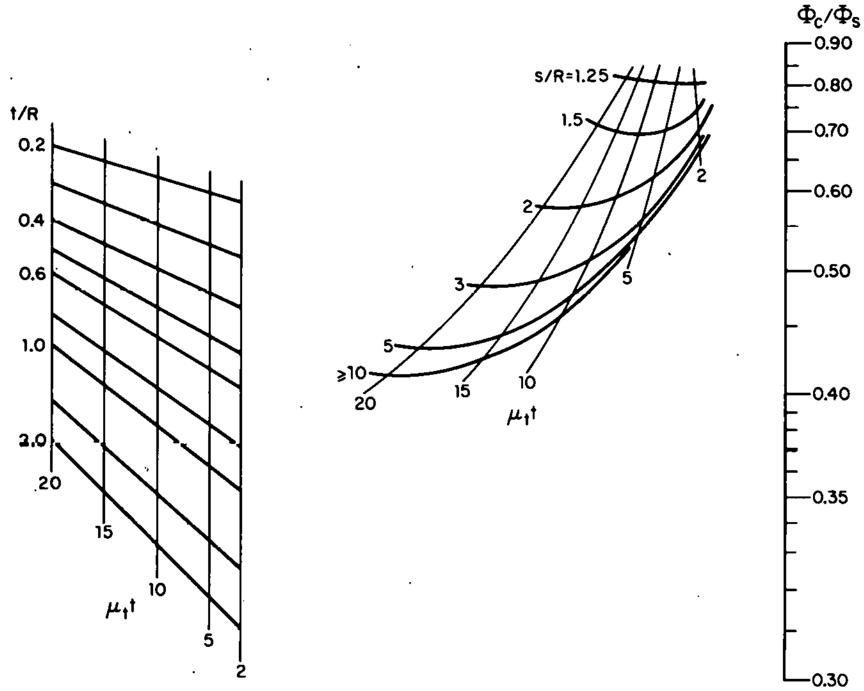

FIGURE 4. Cylindrical vs. Flat Slab Shield for $H/R \geq 1$ and $\mu_r R \geq 4$ (For $\mu_r R < 4$, multiply this factor by factor from Figure 5.)

Φ_c/Φ_s is ratio of uncollided flux for cylindrical shield to that for flat slab shield of thickness t:

Homogeneous cylindrical source of radius R and half-length H. s is distance from axis. For cylindrical shield, negligible gap between source and shield is assumed.

Key: From point in ($\mu_t t$, t/R) grid, project straight line through point in ($\mu_t t$, s/R) grid to Φ_c/Φ_s scale.

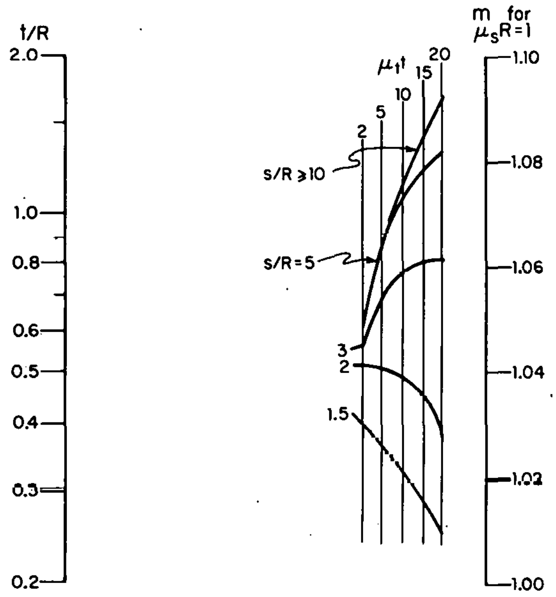

FIGURE 5. Cylindrical vs. Flat Slab Shield: Additional Factor for $\mu_r R < 4$

When $\mu_r R < 4$, value of Φ_C/Φ_S from Figure 4 should be multiplied by m, which for $\mu_r R = 1$ is given by Figure 5.

For $\mu_r R = 2$, m is about midway between unity and value for $\mu_r R = 1$.

Key: From t/R scale, project straight line through proper point in ($\mu_t t$, s/R) grid to scale for m.

CONCLUSIONS

Several precautions and guidelines have been pointed out for using ANISN and point kernel methods of shielding calculation for casks. Subjects suggested for further investigation include:

- Comparing calculated and experimental results, for the methods considered in this paper as well as for other methods.
- Determining adjustment factors to apply to ANISN for finite length of source.
- Determining the effect of narrower limits for photon groups in ANISN.
- Determining dose rate as a function of distance from cask surface.
- Preparing a guide for cask shielding calculations.

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