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SHIELDING METHODS FOR EVALUATING THE VERSATILITY OF PROPOSED SHIPPING CASKS

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After a shipping cask has been designed for a certain type of spent fuel, the number of assemblies it can carry is fixed, as are the thicknesses of the various steel shells, the neutron shield, and the gamma shield. The question then becomes "What other types of spent fuel may be shipped in the cask?"

Using the same neutron and gamma source terms, miscellaneous correlations, and one-group cross sections found in the CAPSIZE program,[1] a fast new interactive shielding program called KWIKDOSE has been written for the IBM-PC which computes and displays a 2-D table showing the total dose rate ten feet from the centerline of a cask, as a function of the spent fuel's burnup and cooling time. Table 1 shows the results corresponding to a hypothetical cask containing 21 PWR assemblies in a 143.6-cm-diameter cavity, surrounded by a 3.81-cm-thick steel shell, a 10.08-cm Pb gamma shield, a 5.08-cm steel shell, a 10.49-cm neutron shield, and a 1.91-cm-thick outer steel barrel. Although approximate and subject to all of the caveats described in ref. 1, this type of information is useful in estimating what types of fuel may or may not be shipped in a particular cask. The major limitation of the KWIKDOSE program is that the data in the cross section library always assumes the neutron shield to be 1/3 water and 2/3 ethylene glycol, containing a total of 1% natural boron by weight.

Given the actual neutron and gamma source terms and the corresponding spectra for various types of fuel, one could perform a similar, more rigorous, set of 1-D multigroup discrete ordinates shielding calculations to generate a similar set of results for any particular cask of interest. While the results would be very useful in evaluating the versatility of the cask, the cost associated with a large number of cases may be prohibitive if done in the normal "forward" mode. Fortunately, the same field of information may be generated inexpensively with no loss of rigor by using the results of a single multigroup adjoint calculation. In this type of calculation, the actual source in the homogenized fuel zone is set to zero, the multigroup adjoint source terms (S^{g*}) in a thin mesh interval at the point of interest (10 ft from the centerline) are defined to be proportional to the flux-to-dose conversion factors for the respective energy groups, and the total adjoint source per cm of length (axially) in this mesh interval is set equal to the sum of the multigroup flux-to-dose conversion factors. The volume-integrated adjoint flux (Φ^{g*}) in what is normally the homogenized fuel zone is then the result of interest. [This volume-integrated quantity is explicitly listed as the "total flux" in the fine group summary table for zone 1 in the XSDRNPM adjoint calculation.[2]] Using the results of this single adjoint calculation, the dose rate at the point of interest may be easily obtained for spent fuel at any burnup and/or cooling time. If S^g represents the homogenized neutron and gamma source density (n/s/cc and p/s/cc) in zone 1 for any arbitrary burnup and cooling time, then the dose rate at the point of interest is given by

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$$D = \sum_{g} \Phi^{g*} S^{g} \qquad (1)$$

Moreover, the different dose rates obtained by substituting different source terms in Eq. 1 will be identical to those that would be obtained if the normal (forward mode) shielding calculations were actually performed for each set of spent fuel source terms.

The attractive feature of this approach is that the volume-integrated multigroup adjoint flux inside the cask cavity may be tabulated on a single datasheet and treated as a characteristic function of the cask itself. Likewise, multigroup source terms for spent fuel at various burnups, decay times, initial enrichments, power histories, etc., may be tabulated for future reference by other independent groups or organizations. (One such tabulation for standard PWR fuel at 33000 MWD/MTIHM may be found in Appendix C of ref. 3.) DOE personnel trying to evaluate the versatility of a proposed cask design, vendors trying to market a given cask for different applications, or regulatory personnel trying to evaluate the adequacy of an existing cask for somewhat different conditions, could then fold the volume-integrated multigroup adjoint flux with the accepted standardized source terms to quickly and easily generate a table of external dose rates for a wide variety of spent fuels.

Table 1. Neutron plus gamma dose rate (mrem/hr), 10 ft from the centerline of the cask, as a function of the burnup and cooling time of the spent fuel

Cooling Time	Burnup (GWD/MT)											
	5	10	15	20	25	30	35	40	45	50	55	60
1 YRS	19.7	39.5	63.2	87.1	114.4	142.0	172.6	203.7	237.4	271.7	308.0	345.4
2 YRS	8.3	16.7	27.4	38.3	51.2	64.5	79.6	95.1	112.3	130.2	149.4	169.5
3 YRS	5.4	10.7	17.8	24.9	33.5	42.4	52.7	63.4	75.4	88.0	101.7	116.1
4 YRS	3.6	7.2	11.8	16.6	22.4	28.4	35.4	42.8	51.2	60.2	70.0	80.6
5 YRS	2.6	5.3	8.6	12.0	16.1	20.4	25.4	30.9	37.1	43.9	51.4	59.6
6 YRS	2.2	4.3	6.9	9.6	12.8	16.1	20.1	24.5	29.4	34.9	41.1	47.9
7 YRS	1.8	3.6	5.7	7.9	10.5	13.2	16.4	20.0	24.2	28.8	34.0	39.9
8 YRS	1.6	3.2	5.0	6.8	9.0	11.3	14.1	17.2	20.7	24.8	29.4	34.6
9 YRS	1.4	2.8	4.4	6.0	7.9	10.0	12.3	15.1	18.2	21.9	26.0	30.8
10 YRS	1.3	2.5	3.9	5.3	7.0	8.8	10.9	13.3	16.1	19.5	23.2	27.5
12 YRS	1.2	2.4	3.7	5.0	6.4	8.1	10.0	12.2	14.7	17.7	21.0	24.9
14 YRS	1.1	2.3	3.5	4.7	6.1	7.5	9.3	11.3	13.6	16.3	19.4	22.9
16 YRS	1.1	2.2	3.3	4.5	5.7	7.1	8.7	10.6	12.7	15.2	18.0	21.3
18 YRS	1.0	2.1	3.2	4.3	5.4	6.7	8.2	10.0	11.9	14.3	16.9	19.9
20 YRS	1.0	2.0	3.0	4.1	5.2	6.4	7.8	9.4	11.3	13.4	15.9	18.7
25 YRS	.9	1.8	2.7	3.6	4.5	5.6	6.8	8.2	9.7	11.6	13.6	16.0
30 YRS	.8	1.6	2.4	3.2	4.0	4.9	5.9	7.1	8.4	10.0	11.7	13.7

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