POINT KERNEL VERSUS RADIATION TRANSPORT FOR IRON DEEP PENETRATION PROBLEMS

B. L. Broadhead and C. V. Parks
Oak Ridge National Laboratory*
Oak Ridge, Tennessee 37831

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The purpose of this study was to determine the relative merits and adequacy of the QAD-CG[1] and QAD-CGGP[2] point kernel codes for spent fuel shielding problems by comparing gamma dose results with those of several radiation transport codes - DOT IV[3], MORSE-SGC[4], XSDRNPM-S[5], and MCNP.[6] An intercomparison of results from the radiation transport codes was also of secondary interest. The problem considered was an R-Z spent fuel cask model consisting of a homogenized source region (UO_2 fuel + basket) surrounded by a 38 cm thick cast iron body. The total cask height is 530 cm. Photon source strengths for eight different discrete energy lines (0.6-2.8 Mev) were specified.

The two QAD versions are very similar to each other with differences occurring in three areas:

1) The energy range for the QAD-CGGP cross sections is 0.01-30 Mev while for QAD-CG the range is only 0.05-10 Mev,
2) QAD-CGGP uses a log-log interpolation to evaluate the cross section value at the mean energy for each group while QAD-CG uses a linear interpolation, and
3) QAD-CG uses the standard exponential build-up factor method, whereas QAD-CGGP optionally uses the standard or the new geometric progression (GP) build-up factor method.[7]

Since the energy range for this problem eliminated the importance of the first difference in the codes, the QAD-CGGP option for choosing the build-up factor method allowed the latter two differences to be evaluated for this problem.

The ENDF/B-IV 18 gamma-group library of the SCALE system[8] was utilized for the DOT, MORSE, and XSDRNPM analyses while the MCNP analyses used the point data library based on ENDF/B-IV. The QAD analyses were performed using the built-in point library for the cross sections and the 18 group gamma energy structure for the source.

Table 1 gives the photon doses along the axial centerline of the cask model at the surface and 2, 10, 50 and 100 meters from the surface. Doses in the external void were calculated for DOT IV using the FALSTF code which uses a last-flight estimator from collision sites to the detector. For XSDRNPM-S, the XSDOSE code evaluated the detector dose by numerically integrating the contributing angular leakage from a finite portion of the shield. The table provides XSDOSE results using the actual cask height and an arbitrarily large (5000 cm) height. The MORSE-SGC and XSDRNPM/XSDOSE calculations were actually done with the SAS4 and SAS1 shielding control modules available in the SCALE system.[8]

Calculations using the standard build-up option in QAD-CGGP were virtually identical to those using the GP option, hence they were omitted from Table 1. Thus, the differences seen in the QAD-CG and QAD-CGGP results appear to be due to the cross section interpolation differences between the two codes. The effect of varying the interpolation method from linear to log-log was seen to be around 2% in the iron cross section which is sufficient to cause the variations seen in the results.

Comparison of the QAD results with the transport results indicates good agreement at the surface of the cask. However, as the distance from the cask surface increases and dimensional effects are more important, the QAD and 1-D transport results become increasingly conservative. Unexpectedly, as the detector distance increases the QAD results continue to increase relative to the 1-D SAS1 results. The apparent explanation is that the build-up factors in the point kernel codes assume radiative scatter from an infinite shield instead of the finite cask shield. Thus, the conservatism in the build-up factors becomes more evident as the entire cask becomes more like a point source. In contrast, the SAS1 results perform the radiation transport through an axially infinite source and shield region, but then use the angular flux from only a finite portion (cask height) of the shield surface to obtain the external dose. The case labeled SAS1-inf has a cask height of 50 m and shows the excessive overprediction expected when an "infinite" cask height (relative to the actual cask height) is assumed.

Dose values at 50-100 m are important to the ALARA purpose in spent fuel/transport problems, and many times these calculations are made with point kernel techniques. From the results thus far, the QAD values appear to be conservative, however, the SAS1 values using the finite cask height more accurately reproduce the multidimensional transport values. Neither the QAD-CG nor the QAD-CGGP results are clearly better than the other, however, the QAD-CGGP results agree more closely with the multigroup transport results (DOT, XSDRNPM and MORSE) at the cask surface, where dimensional effects are small.

REFERENCES

1. V. R. Cain, "A User's Manual for QAD-CG, the Combinatorial Geometry Version of QAD-P5A Point Kernel Shielding Code," NE007, Bechtel Power Corp. (July 1977).

2. "QAD-CGGP: A Combinatorial Geometry Version of QAD-P5A, A Point Kernel Code System for Neutron and Gamma-Ray Shielding Calculations Using the GP Buildup Factor," Radiation Shielding Information Center Code Package CCC-493/QAD-CGGP (1986).

3. W. A. Rhoades and R. L. Childs, "An Updated Version of the DOT 4 One- and Two-Dimensional Neutron/Photon Transport Code," ORNL-5851, Oak Ridge Natl. Lab. (April 1982).

4. J. T. West, T. J. Hoffman, and M. B. Emmett, "MORSE/SGC/S for the SCALE System," Vol. 2, Sect. F9 of Ref. 8.

5. N. M. Greene and L. M. Petrie, "XSDRNPM-S: A One-Dimensional Discrete-Ordinates Code for Transport Analysis," Vol. 2, Sect. F3 of Ref. 8.

6. Los Alamos Monte Carlo Group, "MCNP--A General Monte Carlo Code for Neutron and Photon Transport," Version 2B, LA-7396-M, Los Alamos Natl. Lab. (April 1981).

7. D. K. Trubey and Y. Harima, "New Buildup Factor Data for Point Kernel Calculations," Trans. Am. Nucl. Soc., 53, 416 (1986).

8. "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," Vols. 1-3, NUREG/CR-0200, U. S. Nuclear Regulatory Commission (Rev. 3, 1984).

Table 1. Comparison of QADOR and QADCGGP Doses Rates (mrem/hr) with Various Radiation Transport Results

Code	Distance from cask surface: Surface	2 meters	10 meters	50 meters	100 meters
QAD-CG	33.1	9.1	1.3	0.063	0.016
QAD-CGGP	35.8	9.9	1.4	0.068	0.017
SAS1	41.7	9.7	1.17	0.054	0.014
SAS1-inf[a]	41.7	10.0	1.90	0.279	0.103
DOT IV	37.4	8.7	0.95	-	-
DOT IV-avg[b]	36.3	-	-	-	-
SAS4-avg	37.0(.04)[c]	6.2(.04)	-	-	-
MCNP	-	7.7(.07)	0.85(.06)	-	-
MCNP-avg	34.1(.07)	-	-	-	-

[a] "inf" signifies that the results for this case are for an infinite cask case. This was performed by specifying a height of 5000 cm for the XSDOSE portion of the SAS1 case.

[b] "avg" indicates dose is average over cavity height for the sidewall doses. Otherwise, point detectors are used and located at the axial midplane.

[c] Number in parentheses indicates value of one fractional standard deviation.