CONF- 891103--20

EVALUATION OF SHIELDING ANALYSIS METHODS FOR SPENT FUEL CASKS

C. V. Parks

CONF-891103--20

Computing and Telecommunications Division
Oak Ridge National Laboratory*
P.O. Box 2008
Oak Ridge, Tennessee 37831-6370

DE89 015893

To be presented at the 1989 Winter Meeting of the American Nuclear Society, San Francisco, California, November 26–30, 1989; to be published in the Proceedings.

*Operated by Martin Marietta Energy Systems, Inc., under contract DE-AC05-84OR21400 with the U.S. Department of Energy.

Evaluation of Shielding Analysis
Methods for Spent Fuel Casks

C. V. Parks

Computing and Telecommunications Division
Oak Ridge National Laboratory *
P. O. Box 2008
Oak Ridge, TN 37831

Introduction

Accurate results from shielding analyses of spent fuel casks are increasingly important as the desire for optimized designs increases. ALARA concerns also contribute to the need for accurate dose evaluations for casks. Three areas require the attention of cask shielding analysts – radiation source generation, utilization of cross-section data, and the radiation transport and dose evaluation. This paper reviews recent efforts carried out at Oak Ridge National Laboratory (ORNL) to evaluate the impact of various codes, data, and analysis assumptions on the calculation of radiation doses from spent fuel casks.

Radiation Sources

Point depletion codes are typically used to provide the radiation source strength and spectra based on the average burnup of the fuel. A recent comparison study[1-2] using three popular point depletion codes indicates the variation in radiation sources that can be obtained. For 33-GWd/MTU PWR fuel cooled for five years, neglecting bremsstrahlung radiation reduced the photon dose 10% in the analysis of a cast-iron cask and the absence of ^{144}Pr spectral data reduced the dose another 20%. The impact of burnup-dependent cross-section data on the production of ^{244}Cm contributes to 10–30% differences in the neutron production rate.

*Operated by Martin Marietta Energy Systems, Inc. under contract DE-AC05-84OR21400 with the U.S. Department of Energy.

Dose rates are also affected by the radiation spectrum and the manner of implementation into the radiation transport code. A depleted uranium/solid neutron shielded cask model was loaded with 35-GWd/MTU PWR fuel cooled ten years and an analysis performed using a neutron spectrum based on measured isotopic spectral data[3] and a generic fission spectrum. The fission spectrum source yields a 7% lower neutron dose.[4] Point Monte Carlo and point kernel calculations with discrete line sources verify the importance of conserving energy to obtain group-wise photon sources. Dose variations of 10 to 20% have been seen when photons are grouped without regard to energy conservation.[4] Codes that generate the source spectrum in an energy group structure consistent with the available cross sections are particularly helpful to the shielding analyst.

Cross Section Data

The thick shielding and, hence, deep penetration characteristics found in spent fuel casks make calculated dose rates particularly sensitive to the cross-section data. Several standard multigroup libraries were selected and/or altered to illustrate dose sensitivity due to resonance self-shielding, energy-collapsing spectrum, and parent fine group data. A one-dimensional cylindrical cask model with a homogenized spent fuel source region and 38-cm-thick cast iron body was used. Comparison of dose results[5] using the different libraries showed (1) absence of resonance self-shielding for iron decreased the neutron dose 33%, (2) the neutron dose decreases 8-10% in going from ENDF/B-IV to ENDF/B-V data, (3) an inappropriate weighting spectrum for the neutron data can easily alter the neutron dose by a factor of two, and (4) a fine-group (45–60) photon library needs to be utilized and/or careful attention paid to the collapsing spectrum for more accurate results. Improper weighting (e.g., concrete flux rather than iron flux), and use of broad groups in the important energies (1–3 MeV) yielded 30–50% changes in the photon dose. Unfortunately, most cask shielding analysts are limited to available "off-the-shelf" libraries and are unable to do these types of assessments because they do not have the resources to do their own cross-section processing.

Cross section interpolation techniques used in point kernel and point Monte Carlo codes can also lead to small cross section changes that have a larger impact on the dose. For the iron cask above, varying the interpolation method from linear (QAD-CG) to log-log (QAD-CGGP) yielded a 2% change in the cross section, which was sufficient to cause an 8 to 10% variation in the dose results.

Radiation Transport Codes

Time and available funds often dictate the selection of radiation transport techniques used. Point kernel (for photons only) and one-dimensional discrete ordinates codes are most often used by cask designers because they are fast, inexpensive, and relatively easy-to-use. Monte Carlo and two-dimensional discrete ordinates codes are typically reserved for specific problems and/or to validate the simpler methods.

Table 1 shows good dose agreement between several codes[6−9] used to analyze the cast iron cask described above. All of the multigroup codes utilized an ENDF/B-V library (27n-18g) developed as part of the cross section study noted above. MCNP results were obtained with the point library based on ENDF/B-V. The results obtained from the SAS1/XSDRNPM sequence indicate that one-dimensional discrete ordinates codes can be used to obtain acceptable sidewall doses. However, axially, the large height/diameter (H/D) ratio of most spent fuel casks makes the one-dimensional approximation a poor one. For this cask (H/D = 3.4) the dose was drastically overpredicted (factor of 10 for neutrons and factor of 3 for photons). Specification of "buckling" parameters (based on the cask diameter) can be used to account for the particle leakage out the side and give reasonable (although slightly underpredicted) results from SAS1/XSDRNPM. The dependence of the results on the buckling means some verification (multidimensional analysis) must be performed to assure reasonable results are being obtained.

The results of Table 1 are for a model with a homogenized (fuel assemblies and basket) source region. Modeling the assemblies (five) as an array of homogenized source zones within a 1-cm-thick steel basket changes the Monte Carlo results from the SAS4/MORSE sequence to 50.7 mrem/hr for neutrons and 20.5 mrem/hr for photons. The small increase in neutron dose (6%) is of only slight concern; however, the significant decrease (26%) in the photon dose indicates the improved accuracy that can be obtained with more detailed modeling of the source region. Earlier work with explicit pin-by-pin models of the assemblies showed no significant difference from results obtained with a model using homogenized individual assemblies placed heterogeneously within the source region.[7]

Summary

The codes, data, and user must interact effectively to ensure an accurate and complete shielding analysis of a cask. The available radiation transport codes are reliable tools when handled by the knowledgeable user. However, more work is needed to reduce and/or quantify the uncertainty in the dose arising from specific source characterization methods and cross-section libraries.

References

1. O. W. Hermann et al., "Multicode Comparison of Selected Source- Term Computer Codes," ORNL/CSD/TM-251, Oak Ridge National Laboratory (April 1989).

2. M. C. Brady, O. W. Hermann, and W. B. Wilson, "Comparison of Radiation Spectra from Selected Source-Term Computer Codes," ORNL/CSD/TM-259, Oak Ridge National Laboratory (April 1989).

3. O. W. Hermann and R. M. Westfall, "ORIGEN-S: Scale System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," as described in Sect. F7 of *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, Vols. 1-3*, NUREG/CR-0200, U.S. Nuclear Regulatory Commission (originally issued July 1980, reissued Jan-

uary 1982, Revision 1 issued July 1982, Revision 2 issued June 1983, Revision 3 issued December 1984).

4. C. V. Parks et al. "Assessment of Shielding Analysis Methods, Codes, and Data for Spent Fuel Transport/Storage Applications," ORNL/CSD/TM-246, Oak Ridge National Laboratory (July 1988).

5. C. V. Parks et al., "Intercomparison of Cross-Section Libraries Used for Spent Fuel Cask Shielding Analyses," Proc. Theory and Practices in Radiation Protection and Shielding, Vol. 2 of 2, Knoxville, TN, April 22-24, 1987, p. 559 (ISBN: 0-89448-132-0).

6. 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. Code Package CCC-493/QAD-CGGP, Radiation Shielding Information Center, 1986.

7. J. S. Tang, C. V. Parks, and O. W. Hermann, "Automated Shielding Analysis Sequences for Spent Fuel Casks," Proc. Theory and Practices in Radiation Protection and Shielding, Vol. 2 of 2, Knoxville, TN, Apr. 22-24, 1987, p. 549. (ISBN: 0-89448-132-0).

8. W. A. Rhoades and R. L. Childs. "DORT - A Two-Dimensional Discrete Ordinates Transport Code. Code Package CCC-484/DORT, Radiation Shielding Information Center, 1988.

9. MCNP - A General Monte Carlo Code for Neutron and Photon Transport, Version 3A, LA-7396-M, Rev. 2, Los Alamos National Laboratory, September 1986.

Table 1. Cask surface dose (mrem/hr) results from various analysis tools

Code	Neutron dose		Primary gamma dose	
	Sidewall	Bottom	Sidewall	Bottom
QAD-CGGP	-	-	29.6	38.2
SAS1/XSDRNPM	51.7	-	26.8	-
DORT	51.8	54.1	24.0	40.3
DORT-avg[a]	48.5	41.0	23.3	29.1
SAS4/MORSE-avg	47.6(4)[b]	35.2(7)	26.0(3)	32.5(10)
MCNP-avg	58.9(2)	42.2(7)	27.9(6)	31.6(11)

[a]"avg" indicates dose is averaged over the cask cavity height (sidewall) or cavity diameter (bottom); other results are for dose locations on the axial midplane (sidewall) or radial centerline (bottom).

[b]Number in parentheses indicates percent standard deviation.