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BENCHMARKING OF DOSE RATE CALCULATIONS FOR SPENT FUEL TRANSPORTATION CASKS

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INTRODUCTION

The Nuclear Energy Agency—Committee on Reactor Physics (NEACRP) established in 1985 a working group on shielding assessment of transportation packages. Initially a set of six problems was drawn up and distributed to the working group participants. Following preliminary computations by the various participating countries, the initial working group meeting was held at OECD Headquarters in Paris in June/July 1986. Subsequent meetings were held in May 1987, May 1988, and February 1990 to distribute and discuss solutions to the initial set of six problems.

The United States (U.S.) contribution to the working group was provided by the authors, and this paper summarizes the U.S. work performed to date.

PROBLEM SET DESCRIPTION

The original six-problem-set specifications were described in the document NEACRP-L-290 issued initially to the program participants. Several amendments have been issued to the original release.

The concept behind the definition of problems 1–4 was to begin with a simple cask geometry and add additional details for each subsequent problem. Problem 1 consists of three subcases: case 1a is a 38-cm-thick cylindrical-shaped cast-iron cask body with a dry homogeneous fuel region of 40-cm radius; case 1b adds a 6-cm-thick polyethylene neutron shield recessed within the outer cast-iron shield body, and case 1c is identical to case 1a except the cavity is water filled.

Problem 2 consists of two subcases: case 2a adds to the case 1a shield a total of 60 polyethylene cylindrical rods placed vertically within the shield body; and case 2b places 35 5-cm-thick by 8-cm-thick epoxy case resin rings axially along the outer shield body. In both problem 2 subcases the cavity region is dry.

Problems 3a and 3b are identical to problem 2 with the addition of 35 radial cooling fins placed along the outer axial shell. Case 3a was not analyzed due to the difficulty in modeling both cylindrical rods within the shield and radial cooling fins external to the shield.

Problems 1–3 have a common approximation that the fuel, structural materials, and water/void are homogeneously mixed and smeared over the entire cavity region. Problem 4 examines the effect of cavity heterogeneity by including the basket and explicit assembly model in the calculations. The two subcases for problem 4 consist of a dry (4a) and a wet (4b) cask cavity. The shield for both subcases is the same as case 1a (this is a change from the original specification).

Radial dose results at a point are requested along the cask axial midplane at the cask surface and 1, 2, and 10 m away. Similarly, doses averaged over the cavity height at the same locations radially were desired for comparison. For axial calculations, point and surface-averaged doses were required for both the top and bottom along the cavity centerline at the cask surface and 1, 2, and 10 m away.

In all problems described above, the neutron and fission product gamma sources are fixed to values specified in the problem definition. The total neutron source rate is 1.0×10^9 n/s, the total fission product gamma source rate is 5.0×10^{16} g/s.

Model problems 5 and 6 are unique in that they require the user to calculate the neutron and photon source strengths and spectra. This phase of the shielding benchmark problem introduces an additional level of complexity and should serve to identify problem areas concerning the calculation of sources.

SOLUTION METHODOLOGY AND ASSUMPTIONS

Analyses were performed with several different computer programs and cross-section libraries. In particular, the tools utilized were QAD-CGGP[1] (primary gamma only), DORT,[2] MCNP Version 3,[3] and two shielding analytic sequences developed for the SCALE system[4]—SAS1 and SAS4. The SAS1 and SAS4 sequences are new additions to the SCALE system and have only recently (February 1990) become available via the release of SCALE-4. Basically, SAS1 is a convenient module for problem-dependent cross-section preparation and subsequent 1-D discrete ordinates shielding analysis via the XSDRNPM-S code. For detector points exterior to the cask, SAS1 employs the XSDOSE module to integrate the outgoing angular flux over the finite cask height and obtain the scalar flux at the detector. The SAS4 sequence prepares the cross sections and uses 1-D adjoint results from XSDRNPM-S to automatically prepare biasing parameters for a subsequent Monte Carlo cask analysis with MORSE-SGC/S.

The analyses were performed with a variety of different cross-section libraries. The DORT, SAS1, and SAS4 analyses were performed with a 27n-18g group library based on ENDF/B-V cross sections and collapsed from the SCALE 227-group library[5] using a cast-iron spectrum weighting. The MCNP calculations were done with the point (continuous energy) cross-section library provided with MCNP Version 3b and based on ENDF/B-V. The QAD-CGGP analyses were done using the available built-in attenuation coefficients and log-log interpolation methods.

RESULTS SUMMARY

The results from this effort allow for evaluation of a number of approximations and effects that must be considered in a typical shielding analysis of a transportation cask. Among the effects reported are the performance of various cross-section sets, the comparison of several source generation codes, and multidimensional vs 1-D solutions.

A summary of cross-section findings includes:

1. ENDF/B-V neutron dose results were some 8% lower than those of ENDF/B-IV for a 38-cm-thick iron shield.

2. Underprediction of the neutron doses by nearly 50% by omission of resonance processing.

3. For broad-group analysis of deep penetration through iron shields, the method of fine-to-broad-group collapse is important to both neutron and gamma dose results.

4. A trend of 10–20% underprediction is seen in the neutron doses from multigroup vs point cross sections.

The major findings of the source generation study included the importance of incorporating bremsstrahlung in the analysis and the near equivalence of results from ORIGEN-S and ORIGEN2, two popular source generation codes.

Finally, the following observations were made on multidimensional vs one-dimensional (1-D) models:

1. Excellent 1-D to multidimensional agreement was seen for the sidewall surface doses for simple shields. At 10 m away from the sidewall cask surface, the 1-D dose results were 20% higher than the multidimensional results.

2. Point kernel methods performed quite well for the sidewall, top, and bottom surface doses, but were up to a factor of 2 conservative at 10 m away from the surface.

3. A common approximation of smearing the basket and accompanying source over the cask cavity produced higher (conservative) dose values in both wet and dry cavities for gammas but only in the wet case for neutrons.

4. The smearing of cylindrical polyethylene rods into rings within the shield produced nonconservative results when compared to multidimensional results.

5. Energy vs particle conservation in converting gamma line data to multigroup sources produced 9–15% dose differences.

The detailed calculations leading to these conclusions will be given in the paper and are also reported in ref. 6.

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