MULTIGROUP VERSUS CONTINUOUS ENERGY MCNP FOR PWR FLUENCE CALCULATIONS

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INTRODUCTION

The life of a reactor and its possible extension are directly dependent on the embrittlement of the reactor pressure vessel (RPV) under neutron irradiation. Therefore, in order to ensure RPV integrity and, possibly, provide a basis for plant life extension, the neutron fluence at the RPV must be accurately determined. The S_n transport method[1] is currently used to determine a synthesized 3-D flux distribution based on 1-D and 2-D transport calculations.[2] This calculational methodology, however, contains uncertainties caused by several factors such as the multigroup library, multi-dimensionality effects, the "ray-effect", and geometric approximations. In order to determine the uncertainties, a more accurate calculational methodology has to be used. The Monte Carlo method offers explicit, 3-D geometric representation and continuous energy and angular simulation, and is therefore, well suited for this task.

In previous studies,[3,4] a reference Monte Carlo model for determining RPV neutron fluence for Three Mile Island unit 1 (TMI-1) cycle 7 was developed, and good agreement with experimental data was achieved. In comparison to the multigroup S_n transport calculations, the continuous energy Monte Carlo method predicted a larger number of high energy neutrons within the RPV. Further, the differences between the Monte Carlo and Sn results tend to increase as the RPV is traversed from front to back. This result may have significant impact on the RPV embrittlement estimations, hence, it is necessary to investigate its cause.

We believe the effect of geometric approximations in the deterministic model (i.e., approximations in modeling and 3-D synthesis) are negligible. Also, a recent study on the effect of quadrature order[5] indicates that the use of a S_8 quadrature set is adequate for these calculations (i.e., the "ray-effect" is negligible). Therefore, we expect that the differences are caused mainly by the P_3 truncation of the Legendre expansion and/or the energy group structure of the multigroup library.

This paper attempts to resolve these differences by comparing multigroup and continuous energy Monte Carlo calculations with multigroup deterministic calculations.

METHODOLOGY

For the discrete ordinates S_n calculations, the DORT code[6] is used to simulate R-θ, R-Z, and R reactor models. The resultant flux distributions are synthesized to obtain an equivalent R-

θ-Z flux distribution. For these calculations, the SAILOR 47-group cross-section library[7] and a symmetric S_8 quadrature set with convergence criterion of 0.01% are used.

For the Monte Carlo calculations, the MCNP code, version 4A,[8] is used to model one octant of the TMI-1 reactor, from the core through the concrete wall. This model explicitly represents the rectangular and cylindrical regions in three dimensions, while the deterministic model uses cylindrical geometries to represent rectangular regions in only two dimensions. For the axial dimension, both models extend from the bottom of the lower grid plate to the top of the upper grid plate. Reflective boundary conditions are used for the left, front, and back surfaces, and a vacuum boundary condition is prescribed on the top, bottom, and right surfaces. For the reference Monte Carlo calculation, a continuous form of the ENDF/B-V cross-section library is used.

In order to determine the effect of the multigroup cross sections, MCNP calculations have been performed with three multigroup libraries commonly used for RPV fluence calculations.[9] These cross-section libraries are: (1) 47-group SAILOR, (2) 47-group BUGLE-80,[10] and (3) 56-group ELXSIR[11]. (The libraries were processed into a form suitable to MCNP with the CRSRD code[12].) SAILOR and BUGLE-80 were derived from the ENDF/B-IV library for the same group structure but with different collapsing procedures. ELXSIR was created based on a combination of ENDF/B-IV and ENDF/B-V libraries for a finer group structure (with more fast groups) with a collapsing procedure similar to SAILOR's. Note that SAILOR/BUGLE-80 group boundaries are a subset of the ELXSIR group boundaries.

MCNP AND DORT COMPARISONS

Transport calculations are performed for each of the following five cases: (1) MCNP with continuous ENDF/B-V, (2) DORT with SAILOR, (3) MCNP with SAILOR, (4) MCNP with BUGLE-80, and (5) MCNP with ELXSIR. For tallying, the SAILOR/BUGLE-80 group boundaries are used, and for comparison Case 1 is considered as the reference.

Figs. 1 compare relative differences in radial flux distributions at the core midplane and 30° extending from the core periphery to the cavity dosimeter for three energy groups including group 5 (7.408-8.607 MeV), group 11 (2.466-2.725 MeV), and group 16 (1.653-1.920 MeV). For all energy groups, the multigroup cases (Monte Carlo or deterministic) predict generally lower flux values (up to ~30%). Also, differences for energy groups 11 and 16 tend to increase beyond the radial position of ~217 cm (i.e., front of the RPV). We believe that this may be due to the inadequacy of the multigroup structure which possibly cannot represent an accurate particle transfer to these energy groups. Group 5 is not affected by this phenomenon because fewer neutrons are transferred into it. The three multigroup libraries yield similar results as were obtained in a previous study via 1-D Sn transport calculations[9]. Further, the S_n results are within 5-10% of the multigroup Monte Carlo results. It is important to note that the 1σ statistical uncertainties associated with the Monte Carlo results are ~5%, ~10%, and ~5% for the three energy groups, respectively.

For benchmarking purposes, measured reaction rates from the ex-vessel cavity dosimeter are compared to calculated values. Table 1 lists ratios of calculated-to-experimental (C/E) reaction rates, corresponding to all five cases, for the following reactions: ^{63}Cu(n,α), ^{54}Fe(n,p), and ^{58}Ni(n,p). For these calculations, the response cross sections were obtained from the SAILOR library. Ratios calculated by continuous energy MCNP are ~15% higher than those calculated by

DORT and ~10-15% higher than those calculated with multigroup MCNP. (Note, no attempt has been made to correct for the new ENDF/B-VI iron cross sections. This issue is addressed in reference 4.)

Table 1. Calculated-to-Experimental (C/E) Ratios at Cavity Dosimeter for TMI-1

Reaction	DORT	MCNP	Multigroup MCNP		
	SAILOR	ENDF/B-V	SAILOR	BUGLE-80	ELXSIR
^{63}Cu(n,α)	0.780	0.894 (.018)[a]	0.798 (.052)	0.756 (.068)	0.791 (.043)
^{54}Fe(n,p)	0.813	0.949 (.016)	0.874 (.041)	0.861 (.049)	0.866 (.031)
^{58}Ni(n,p)	0.789	0.936 (.015)	0.843 (.037)	0.833 (.045)	0.838 (.028)

[a]Numbers in parentheses refer to 1σ statistical uncertainties.

CONCLUSIONS

It is concluded that the flux distributions and cavity dosimeter activities predicted via the 3-D synthesized multigroup Sn transport calculations are within 10% of the multigroup MCNP results. While these calculations clearly demonstrate the effect of the multigroup libraries, they do not, however, conclusively reveal the origin of the effect (i.e., P_3 truncation, group structure, or both). Also, it is demonstrated that different multigroup libraries including SAILOR, BUGLE-80, and ELXSIR show close agreement.

Future work will examine the effects of the P_3 truncation and group structure.

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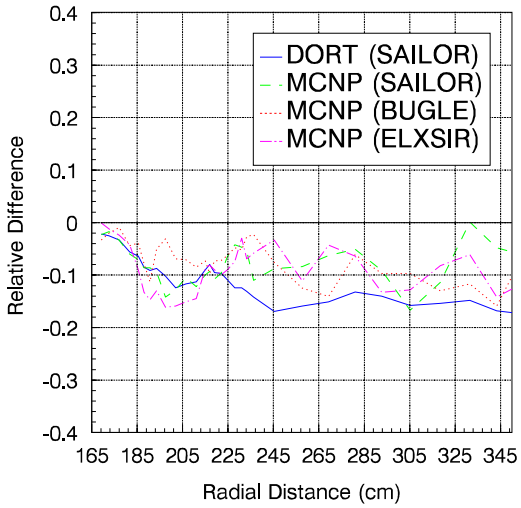
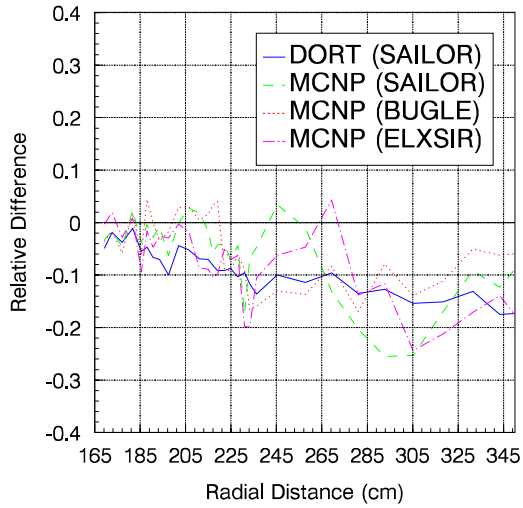
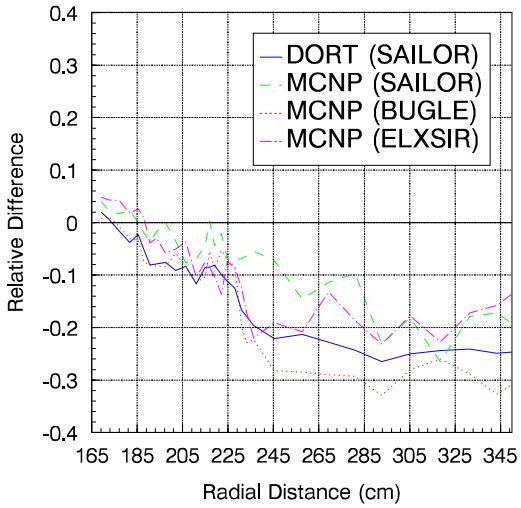


(a) Group 5 (7.408-8.607 MeV)



(b) Group 11 (2.466-2.725 MeV)



(c) Group 16 (1.653-1.920 MeV)

Fig. 1. Relative Differences in Radial Flux Distributions