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COMPUTATIONAL MODEL FOR THE OAK RIDGE NATIONAL LABORATORY (ORNL) BULK SHIELDING REACTOR (BSR)*

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In the past several years, there has been a growing concern in the industry to predict the radiation damage to the pressure vessel and core support structure. With respect to these goals, several experiments are being performed. Along with these experiments, computational models are being formed using the experiments as a guide to the accuracy of the model. Once an acceptable model has been validated, important parameters, such as fluences and reaction rates can be accurately calculated for similar systems. These calculated parameters can be compared with radiation damage measurements to obtain points on a trend curve. This will facilitate better prediction of the radiation damage to the reactor pressure vessel and support structure for specific configurations of interest. This work, however, is limited to the determination of an accurate neutronics model of the Oak Ridge National Laboratory Bulk Shielding Reactor (ORNL-BSR).

The procedure for formulating the computational model of the ORNL-BSR is as follows:

1. Obtain physical data pertaining to the core and experiment configuration.
2. Generate a 40-group cross section library starting from the 218-group master library using AMPX[1] (A Modular Code System for Generating Coupled Multigroup Neutron-Gamma-Ray Cross sections from data in ENDF format).

3. Perform a cell-weighting calculation to obtain self-shielded multi-group cross sections with the 40-group cross section library, and then perform a zone-weighting calculation over the whole core to obtain 40, 27, 15, and 7 group multi-group libraries.
4. Run one-dimensional XSDRNPM and VENTURE[2] calculations with 40, 27, 15, and 7 groups. Calculate the relative fluxes above 0.1, 0.5, and 1.0 MV. Also calculate the activities for the ^{54}Fe and ^{58}Ni (n, p) reactions and the average displacement cross section.
5. Run two-dimensional DOT 4.2 and VENTURE calculations with seven groups; also, run a 15-group two-dimensional VENTURE. Obtain the relative fluxes, activities, and cross sections specified by item No. 4.
6. Run a three-dimensional VENTURE calculation with seven groups. Obtain relative fluxes, activities, and cross sections specified in item No. 4.
7. Report the results and analysis of the one-, two-, and three-dimensional calculations.

In following the procedure outlined above, the results to date have been good. Table 1 shows a comparison of normalized flux spectra obtained from the one-dimensional XSDRN and VENTURE calculations and from the two-dimensional VENTURE calculations. As one can see from Table 1, the spectra are in very close agreement. This illustrates that the diffusion theory calculations (VENTURE) are applicable to this core. Table 2 shows a comparison of the experimental data and the calculated results. Because of the method in which the experimental results were reported, only the ratios of the ^{54}Fe and ^{58}Ni activities can be directly compared. Again as one can see, the computation and experimental results are within 10%. An intercomparison of computational results for several neutronics models is also shown in Table 2. Note that fairly good agreement is obtained between all of the models.

The results indicate that the selected parameters can be accurately calculated for the ORNL-BSR. Thus it is assumed that these parameters can be accurately calculated for similar systems with the data base and computational methodology employed for this work. If these parameters provide an accurate measure of irradiation damage, lifetimes of pressure vessels and core-support structure can be determined from applicable trend curves.

REFERENCES

1. N. M. Greene et al., *AMPX-II: A Modular Code System for Generating Coupled Multigroup Neutron-Gamma-Ray Cross-Section Libraries from Data in ENDF Format*, ORNL-3706 (November 1978).
2. D. R. Vondy, T. B. Fowler, and G. W. Cunningham, *VENTURE: A Code Block for Solving Multigroup Neutronics Problems Applying the Finite-Difference Diffusion-Theory Approximation to Neutron Transport*, ORNL-5062 (October 1975).

Table 1. Comparison of Normalized Flux Spectra (flux/unit lethargy)

Group	1-D XSDRNPM	1-D VENTURE	2-D VENTURE	LETHARGY
1	4.8780E-03	4.9787E-03	5.9149E-03	1.1348E+00
2	3.3137E-02	3.4834E-02	3.6532E-02	2.9236E-01
3	6.7787E-02	7.1869E-02	7.2194E-02	4.7000E-01
4	1.1655E-01	1.2243E-01	1.2600E-01	2.3995E-01
5	1.2502E-01	1.3133E-01	1.3982E-01	5.5118E-01
6	1.2134E-01	1.2706E-01	1.4331E-01	2.9753E-01
7	1.1589E-01	1.2002E-01	1.3983E-01	7.0510E-01
8	8.4729E-02	8.7394E-02	1.0212E-01	4.1351E-01
9	7.1701E-02	7.3317E-02	8.5795E-02	7.1150E-01
10	4.3921E-02	4.4707E-02	5.3474E-02	4.8243E-01
11	3.4403E-02	3.4753E-02	3.6524E-02	2.0402E+00
12	2.9782E-02	2.9540E-02	3.2203E-02	4.6687E+00
13	2.9532 E-02	2.8805E-02	3.0440E-02	4.1834E+00
14	3.1016E-02	3.0053E-02	2.7832E-02	1.2221E+00
15	2.3856E-02	2.2937E-02	1.6942E-02	1.0911E+01

Table 2. Comparison of the Experimental Data and Calculated Results

Ratio	Experimental	1-D XSDRNPM	1-D VENTURE	2-D Venture
$\frac{\phi\ (Tot)}{\phi\ (Tot)}$		1.0	1.0	1.0
$\frac{\phi\ (0.1\ MeV)}{\phi\ (Tot)}$		0.3700	0.3839	0.4314
$\frac{\phi\ (0.5\ MeV)}{\phi\ (Tot)}$		0.2623	0.2739	0.3017
$\frac{\phi\ (1.0\ MeV)}{\phi\ (Tot)}$		0.1803	0.1892	0.2025
$\frac{^{58}Ni\ (n,\ p)}{^{54}Fe\ (n,\ p)}$	1.2470	1.3222	1.3207	1.3184