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MCNP BENCHMARK ANALYSES OF CRITICAL EXPERIMENTS FOR THE SPACE NUCLEAR THERMAL PROPULSION PROGRAM*

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Abstract

Benchmark analyses have been performed of Particle Bed Reactor (PBR) critical experiments (CX) using the MCNP radiation transport code. The experiments have been conducted at the Sandia National Laboratory reactor facility in support of the Space Nuclear Thermal Propulsion (SNTP) program. The test reactor is a nineteen element water moderated and reflected thermal system. A series of integral experiments have been carried out to test the capabilities of the radiation transport codes to predict the performance of PBR systems. MCNP was selected as the preferred radiation analysis tool for the benchmark experiments. Comparison between experimental and calculational results indicate very good agreement. This paper describes the analyses of benchmark experiments designed to quantify the accuracy of the MCNP radiation transport code for predicting the performance characteristics of PBR reactors.

INTRODUCTION

The particle bed reactor system is characterized by a highly heterogeneous, compact configuration with many streaming pathways. The neutronics analyses performed for this system must be able to accurately predict reactor criticality, kinetics parameters, material worths at various temperatures, feedback coefficients, and detailed fission power and heating distributions. The latter includes coupled axial, radial, and azimuthal profiles. These responses constitute critical inputs and interfaces with the thermal hydraulics design and safety analyses of the system. The compact nature of the PBR reactors requires that many calculational approximations used for larger, more homogeneous systems cannot be implemented. The explicit three dimensional geometry capability of MCNP with the use of double differential pointwise cross sections removes uncertainties associated with modelling and with multigrouping and with the finite expansion of the scattering kernel. These approximations are imbedded in other conventional radiation transport codes, including those employing two and three dimensional discrete ordinates and Monte Carlo techniques. This is the reason why MCNP was selected as the physics analysis tool for the SNTP project.

The integral experiments to be analyzed include measurements of initial criticality and excess reactivity, prompt neutron lifetime, worths of various reactivity control concept configurations, material worth of a polyethylene cylindrical plug, temperature coefficients of reactivity for the water moderator and for cryogenic polyethylene placed in the center of the core, and fission power distributions. This paper focuses only on the analytical aspects of the CX experimental series. Several other companion papers presented in this symposium will discuss the overall description of the design and safety aspects of the reactor (Parma et al. 1993) and the experimental design, setup, operations, results, and measurement errors (Ball et al. 1993 and Hoovler et al. 1993). A more complete description of the analytic work is given in another report (Selcow et al. 1993).

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METHODOLOGY

The MCNP three dimensional radiation transport code (Briesmeister 1986) solves the Boltzmann transport equation by tracking neutrons and photons throughout an explicitly defined geometry with probability distribution functions determined by the nuclear cross sections describing the specific reactions with the materials in the system modelled. The geometry can be made as complex as necessary to accurately represent any design detail found in an actual reactor system. This modelling capability is an essential component in the analysis of a highly heterogenous compact system. The mechanism by which MCNP estimates system responses, such as criticality, flux, heating, and fission power distributions, is by sampling neutron and/or photon particles in specific three dimensional spatial locations, and with specified angular and energy distributions.

The overall accuracy of a calculation is described in terms of systematic and statistical errors. The systematic errors are associated with the accuracy of representing the physical geometric system, the component material compositions and the accuracy of the nuclear cross sections used. The analyses performed for this study utilized ENDF/B-V cross sections with scattering kernels generated at several different temperatures. The statistical errors can be minimized by carrying out a specific calculation to a large number of particle histories. The use of variance reduction may, under certain conditions, reduce the amount of computer time to achieve a desired statistical accuracy. The analyses associated with this paper did not employ any specific variance reduction techniques other than a generalized Russian roulette and splitting scheme.

The following section will discuss the calculations performed to benchmark some of the integral CX experiments. In particular, the models will be described, the statistical accuracy will be given, and any pertinent characteristics of the scattering kernel data will be mentioned if relevant. The experimental setup, operations, and errors are not discussed in this paper and may be found in the other documents and reports cited above.

ANALYSES AND RESULTS

Initial Criticality and Excess Reactivity

The nineteen fuel elements used in the water moderated and reflected CX reactor system consist of 93% enriched fuel contained in aluminum canisters. Polyethylene plugs function as the axial reflectors and the length of the fuel region is 43 cm. The reactor operates at ambient conditions and produces less than 10 watts of total power. A complete description of the reactor core is found in (Parma et al. 1993). The explicit model developed to represent the reactor system is presented in Figure 1, and contains the inner and outer aluminum canisters, fuel regions, water radial reflector, polyethylene axial reflectors, top and bottom grid plates, and the outer aluminum tank which has a radius of 36 cm. The actual fuel loading of the reactor was individually specified for each element and included fuel impurities such as U-234 and U-236. The presence of U-234 results in a reactivity penalty of approximately one-half percent in the effective multiplication factor.

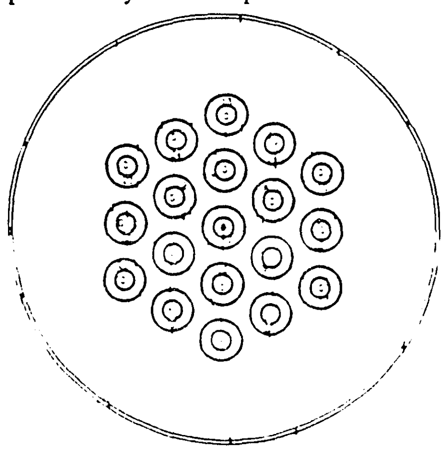

FIGURE 1. MCNP Representation of CX Geometry: Plan View.

Worth of Polyethylene Cylindrical Plug

An experiment was performed to measure the worth of a cylindrical polyethylene plug placed in the center of an element in the outer ring. The calculational benchmark was carried out to 1.2 million histories and the resultant worth was \$0.74 +/- 0.18. This is compared with the experimental value of \$0.72 shown in Table 1.

Moderator Temperature Coefficient

Several experiments were performed to measure the CX moderator temperature coefficient of reactivity for temperatures in the range of 283 to 353 K. A small positive temperature coefficient was measured and was reproduced analytically. The moderator coefficient is a combination of density and scattering kernel effects. The calculations that were performed demonstrated that there is a negligible change in reactivity due to changes in moderator and reflector density for the temperature range measured in the experiment. It was determined that the positive temperature coefficient is attributable to the temperature dependence of the scattering kernel for hydrogen bound in water. An increase in temperature will result in more upscatter in the system and hence a harder neutron spectrum. As the temperature rises, the capture reaction rate in the water moderator decreases resulting in an increase in the thermal utilization and an increase in the effective multiplication factor. Due to the blackness of the fuel, the decrease in the fission reaction rate is much smaller that the reduction in the poisonous effect of the hydrogen in the moderator and reflector. Analyses were carried out to examine the sensitivity of the temperature coefficient to variations in the scattering kernel temperature. For a temperature range of the scattering kernel of 300 to 400 K, a reactivity coefficient of +3.1 +/- 0.2 cents per K was calculated. These analyses were carried out to 400,000 histories. The average temperature used in the analysis was 350 K, whereas the average temperature in the experiment was 318 K. Mughabghab found that the temperature dependence of the scattering kernel could be represented by $T^{1.543}$ (Mughabghab 1992). Using this formulation, the temperature corrected reactivity coefficient is +3.6 +/- 0.3 cents per K. This compares with the experimental measurement of +3.5 cents per K given in Table 1.

Cryogenic Polyethylene Temperature Coefficient

A central test section region was designed in the CX core in order to measure the temperature coefficient of various candidate moderator materials at cryogenic temperatures. This test section contained a sufficient amount of Lytherm insulating material such that the rest of the moderator and the fuel elements remained at room temperature. In this experiment, the central element was removed and polyethylene beads (64% volume fraction), were placed in the test section and cooled down to 97 K with liquid nitrogen. The liquid nitrogen was pumped out and the system warmed up to 296 K. In order to compensate for the reactivity penalty incurred by the loss of the central element and the presence of the test section, solid polyethylene plugs were placed in the center of ten of the elements in the second and outer rings. In the calculation, the test section was explicitly modelled and scattering kernels for hydrogen bound in polyethylene at 100 and 300 K were utilized. The analytic prediction of the polyethylene worth for the ~ 200 degree temperature differential was \$1.62 +/- 0.17, compared with the experimental value of \$1.65 shown in Table 1. The analyses were carried out to one million neutron histories. Figure 2 shows a MCNP representation of the geometric model.

Fission Power Distributions

A series of experiments were performed to measure the axial, radial, and azimuthal fission power distributions for four elements in the CX core. The experimental technique employed uranium-aluminum compacts and uranium-aluminum wires. A complete experimental description may be found in several reports (Ball 1992). An example of an axial power distribution calculation for the central element is shown in Figure 3. The error bars on the curve are two standard deviations which represents a ninety-five percent confidence factor in the data. The analysis was carried out to three million neutron histories. The power peaks at the top and bottom positions are due to neutron reflection from the polyethylene plugs at the axial endpoints of the fuel element. This axial profile compares well with similar profiles measured in the experiment.

The effective multiplication factor for the reactor system was calculated to be 1.022 with a standard deviation of 0.001. This analysis was carried out to 700,000 neutron histories and corresponds to an unborated moderator system with the control, safety, and shim rods withdrawn. Several calculations using varying boric acid concentrations were performed to determine the critical boron concentration. The value of 93 ppm was determined using a linear regression analysis of the data. This translates into a boric acid worth of 3 cents per ppm using the delayed neutron fraction β_{eff} of 0.0079. Comparison with the experimental data listed in Table 1 indicates that the analysis predicted the system criticality to approximately one-half percent in K_{eff} and the boric acid worth to within one percent.

TABLE 1. Benchmark Data.

	Experiment	Calculation
• Initial criticality/ excess reactivity		
▪ Δ K_{EFF}	0.017	0.022 (.001)
▪ Critical boric acid concentration, ppm	71	93
▪ Boric acid worth, $/ppm	0.03	0.03
• Prompt generation time, μsec	33.3	34.9 (0.1)
• Worth of polyethylene plug, $	0.72	0.74 (0.18)
• Moderator temperature coefficient of reactivity, ¢/ K (283 to 353 K)	+3.5	+3.6 (0.3)
• Worth of cryogenic polyethylene, $ (97 to 296 K)	1.65	1.62 (0.17)

Prompt neutron lifetime

The prompt generation time is defined as the average time between successive neutron generations in a system. The calculation of this lifetime considers the number of neutrons per generation, the total number of generations, the fission weight and the elapsed time, defined as the distance a neutron travels multiplied by 1/v, where v is the neutron velocity. However, the lifetime calculated in MCNP differs from this in that it represents the average life span of a neutron in the system until it escapes from the system or is absorbed, either by fission or parasitic absorption. There are several methods of estimating the generation lifetime in a reactor system using MCNP. The first technique is to implement a modification to the code to explicitly calculate the generation time. Another method involves perturbing the system by adding a small amount of a 1/v absorber, such as boron-10, uniformly throughout the reactor system. The lifetime is the difference in the multiplication factor between the perturbed and unperturbed systems. A third method for estimating the lifetime involves analyzing an equivalent system that has the same fission weight and eigenvalue. This is accomplished by replacing the 12 cm radial water reflector, which has a large parasitic absorption cross section, with a 3 cm beryllium reflector for the same reactor core. The same total amount of neutrons effectively lost to the system is preserved; the difference is that the total absorption weight is reduced by minimizing the fraction that are parasitically absorbed in the reflector. Any significant spectral difference between the two systems is masked in the integral calculation. The lifetime calculated using MCNP with this method is 34.9 +/- 0.1 microseconds. Results of estimations of the lifetime using the other two techniques described above are to be discussed in a more comprehensive report (Selcow et al. 1993). The experimental value obtained with the pulsed neutron technique was 33.3 microseconds, as shown in Table 1.

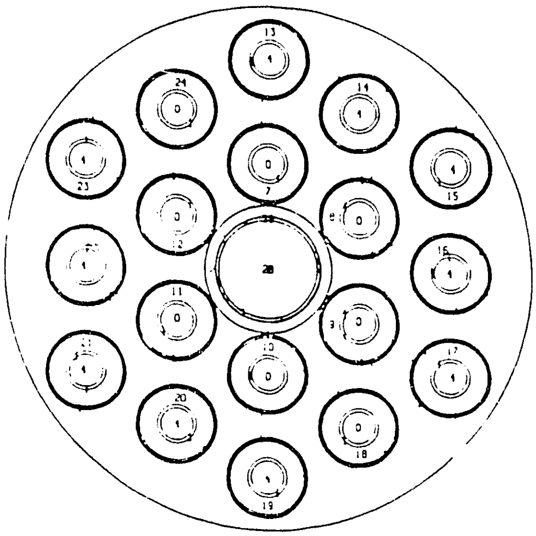

FIGURE 2. MCNP Representation of CX Central Test Section.

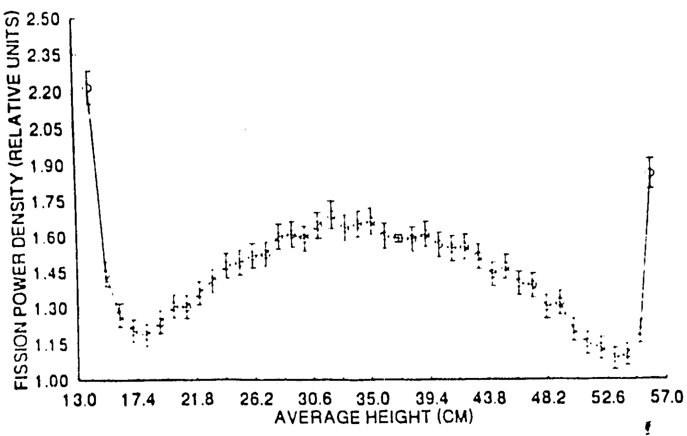

FIGURE 3. CX: Axial Fission Power Distribution - Central Element.

DISCUSSION AND CONCLUSIONS

The major results of the benchmark analyses are given in Table 1. Comparison between experimental and calculational results indicate very good agreement. These integral reactor experiments have validated the MCNP physics design methods in combination with the nuclear data in the thermal region. The benchmark calculations have shown that a high level of statistical accuracy is required. These results have demonstrated that the explicit three dimensional geometry capability is essential for precise modelling of complex reactor configurations. Other methods, such as Sn codes, were found to yield results which had much larger discrepancies from the experimental results. Calculations performed by Cerbone one year before the experiment showed that the two-dimensional Sn analysis resulted in a four percent higher value in the system eigenvalue than the MCNP predictions (Cerbone 1988). The major impact for the designs developed in the SNTP program is the reduction in estimated calculational uncertainties for the neutronics inputs into thermal hydraulics and safety analyses, effecting the design and safety margins of candidate propulsion systems.

Acknowledgments

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