Annals of Nuclear Energy 36 (2009) 995–997

Contents lists available at ScienceDirect

Annals of Nuclear Energy

journal homepage: www.elsevier.com/locate/anucene

Technical Note

A comparative study of ZPR-6/7 with MCNP/5 and MC^2-2/REBUS

Masood Iqbal *, Hany Abdel-Khalik, Paul Turinsky

Department of Nuclear Engineering, North Carolina State University, USA

A R T I C L E I N F O

Article history:
Received 11 December 2008
Received in revised form 2 March 2009
Accepted 3 March 2009
Available online 2 April 2009

A B S T R A C T

This work models the INL ZPR-6/7 assembly employing two different approaches: a probabilistic approach using MCNP/5 and a deterministic one using MC^2-2/REBUS. With MCNP/5, each drawer of the assembly is modeled in detail with regard to geometry and fuel loading. In the deterministic approach, the MC^2-2 collapses cross-sections in energy and space into a 15 few group structure homogenized spatially over each drawer for the REBUS 3D model. Various reactivity coefficients and reaction rates at different locations inside the core were evaluated and compared for both approaches and contrasted to published experimental data and were found to be in good agreement.

1. Introduction

This study aims to develop both deterministic and probabilistic models for a Zero Power assembly (ZPR) to calculate reactivity values and reaction rates at different locations inside the core. The ZPR-6/7 assembly was selected as a basis for this study. These assemblies are being used as a benchmark for the validation of different cross-section libraries and reactor physics codes. These assemblies are also being modeled to study Generation IV reactor design. Computer codes MCNP/5 and MC^2-2/REBUS were used for the detailed probabilistic and deterministic 3D analyses, respectively. Varuttamaseni and Lee (2006) modeled ZPR-6/7 assembly using MCNP and calculated the reaction rate ratios and eigenvalue for the assembly. In Argonne National Laboratory (2003) detailed description of ZPR-6/7 assembly is given. Results for eigenvalue calculations with KENO, VIM, MONK and MCNP/4C computer codes are also presented. A brief description of the ZPR-6/7 assembly design and the computer codes utilized is given here.

1.1. ZPR-6/7 assembly

The ZPR-6/7 fast critical facility consists of a large, cast-steel bed supporting two tables, of which one is stationary and the other is movable. It uses a horizontal split-table type machine. Each table is 12 ft wide and 8 ft long. The square lattice is made of stainless steel having a thickness of 0.040 in.; side of 2.175 in. and is 4 ft long. The combination of these lattice tubes in 5-tube by 5-tube bundles stacked horizontally on both tables forms a 45-row and 45-column square ‘‘honeycomb” matrix.

The ZPR-6/7 loading began in July 1970 and experiments on this assembly were continued through October 1971. This assembly was loaded with a fissile loading of 15.4 kg of ^{235}U and 1118.1 kg of ^{239}Pu and ^{241}Pu. Given the voluminous design details of the ZPR-6/7 facility (Argonne National Laboratory, 2003), it is not computationally practical to obtain a detailed neutronic description in energy and space of the assembly behavior. Therefore, a very accurate transformation to a simplified model is needed to render any of the ZPR-6/7 assemblies a practical criticality-safety benchmark experiment.

1.2. Computer codes

In this study REBUS, MC^2-2 and MCNP/5 are used to model ZPR-6/7 assembly.

REBUS is a system of codes which solves for the steady state flux and then solves the depletion equation to calculate the change in isotopes (Argonne National Laboratory report, 2001). REBUS can accommodate different geometries including triangular and hexagonal mesh. In this problem rectangular geometry is used. No depletion calculations were done during ZPR-6/7 core analysis. Main concern was criticality calculations, nodal fluxes and power densities.

MC^2-2 is a program for solving the neutron slowing down problem to determine a detailed spectrum for use in deriving multigroup cross-sections (Argonne National Laboratory report, 2000). P_1B_1, consistent P_1 and consistent B_1 extended transport theory algorithms, and Greuling–Goertzel continuous slowing down theory solutions, and the fine group structure can be used. In this case consistent P_1 is used. ENDF/B-V library was used to generate cross-sections. Cross-sections were generated at room temperature and assumed no structural heating.

MCNP/5 is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/pho-

* Corresponding author. Address: Nuclear Engineering Division, Pakistan Institute of Nuclear Science and Technology, P.O. Nilore, Islamabad, Pakistan. Tel.: +92 51 2207264; fax: +92 51 9290275.
E-mail address: masiqbal@hotmail.com (I. Masood).

0306-4549/$ - see front matter © 2009 Elsevier Ltd. All rights reserved.
doi:10.1016/j.anucene.2009.03.005

ton/electron transport, including the capability to calculate eigen values for critical systems. The code treats an arbitrary 3D configuration of materials in geometric cells bounded by first- and second-degree surfaces and fourth-degree elliptical tori. Pointwise cross-section data are used. For neutrons, all reactions given in a particular cross-section evaluation (such as ENDF/B-VI) are accounted for. Each neutron history is tracked from the point of its birth until it dies, i.e. absorbed. Important standard features that make MCNP/5 very versatile and easy to use include a powerful general source, criticality source, and surface source; both geometry and output tally plotters; a rich collection of variance reduction techniques; a flexible tally structure; and an extensive collection of cross-section data (Los Alamos National Laboratory (2005)).

2. Computational modeling

Two different MCNP/5 models were considered: (i) a detailed core model taking into account all geometrical and compositional details of the assembly design and (ii) a coarser model homogenizing each drawer with all constituents present initially in full core analysis. The MCNP/5 simulation of the ZPR-6/7 assembly was performed with 100,000 neutrons per cycle for a total of 130 cycles, with the first 30 cycles skipped. The cross-sectional views from top and side of the core are shown in Figs. 1 and 2. The typical core drawer configuration is depicted in Fig. 3.

The deterministic model employed the MC^2-2 code in two different approaches to calculate the homogenized drawer's cross-sections in 15 group energy structure. The first approach, hereinafter called the heterogeneous approach, considered all the geometrical and compositional details of each plate in 1D geometry. The second approach, hereinafter called the homogenous approach, homogenized the constituents of each drawer in a single cell. Both approaches were based on the ENDF/B-V data (November 2000 Version).

Based on both homogenous and heterogeneous approaches, the cross-section libraries were used in REBUS calculations separately. In REBUS, each drawer was modeled separately. Two or more consecutive drawers of the same dimensions and compositions in one row were merged together.

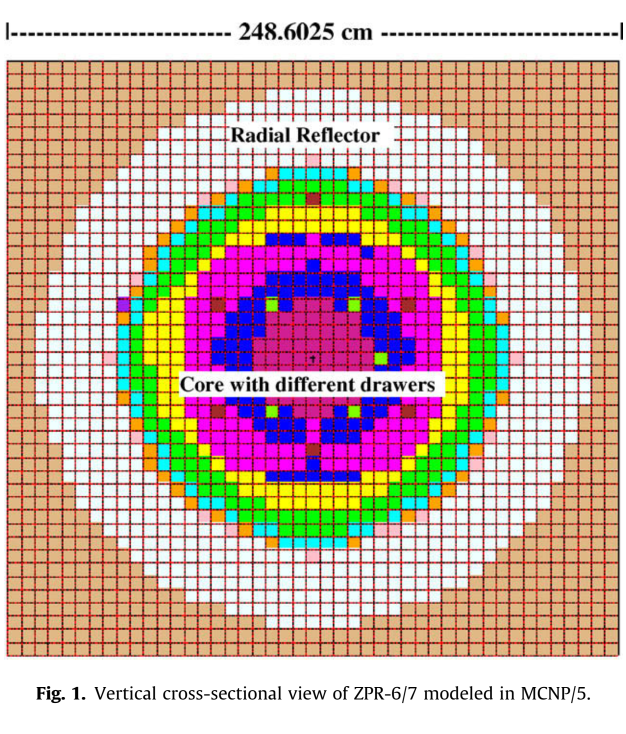

Fig. 1. Vertical cross-sectional view of ZPR-6/7 modeled in MCNP/5.

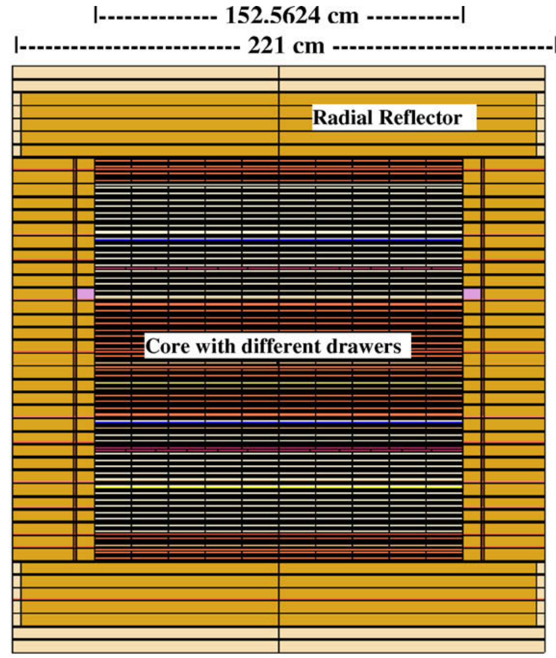

Fig. 2. Side cross-section view of ZPR-6/7 assembly modeled in MCNP/5.

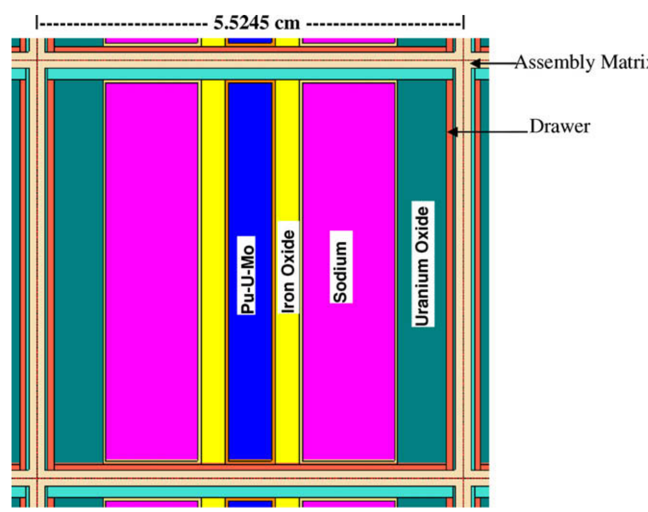

Fig. 3. Typical modeling of ZPR-6/7 drawer in MCNP5.

3. Results and discussion

Detailed modeling of the ZPR-6/7 assembly was completed using both probabilistic and deterministic models. The probabilistic approach employed the MCNP/5 code and the deterministic approach used the MC^2-2/REBUS code system. In MC^2-2/REBUS 15 few group energy structure while in MCNP/5 continuous energy group structure was used. Both approaches employed a heterogeneous and a homogenous models for the various assembly drawers. The MCNP/5 heterogeneous model accounted for all drawers compositional and geometrical details and the homogeneous model is based on homogenization of compositional and geometrical detail of a drawer in one cell. In MC^2-2/REBUS, cross-sections were calculated with homogenized materials in drawers with 1D analysis. The detailed geometry in REBUS was considered. The keff and reaction rates calculated are listed in Table 1. The experimental values (Argonne National Laboratory, 2003) are also listed for

Table 1
Comparison of reaction rates ratio and k_{eff} for ZPR-6/7 calculated from MCNP/5 and MC^2-2/REBUS with experimental values and published data.

Parameter	MCNP (as it is)	MCNP homogenized	MC^2-2/REBUS (hetro.)	MC^2-2/REBUS (homo.)	Experimental	Varuttamaseni (2006)
k_{eff}	0.99931 ± 0.00017	0.98644 ± 0.00014	0.99066	0.99883	1.0005 ± 0.00003	1.00663 ± 0.00051
$(\sigma_f\varphi)_{28}/(\sigma_f\varphi)_{49}$	0.02736 ± 2.08%	0.02492 ± 1.29%	0.0239	0.0177	0.02517 ± 3%	0.02703 ± 3.2%
$(\sigma_c\varphi)_{28}/(\sigma_f\varphi)_{49}$	0.1447 ± 1.36%	0.1493 ± 0.86%	0.139	0.143	0.144 ± 4%	0.1464 ± 2.2%
$(\sigma_c\varphi)_{28}/(\sigma_f\varphi)_{28}$	5.289 ± 2.27%	5.995 ± 1.39%	5.81	8.190	5.626 ± 1%	5.425 ± 3.7%

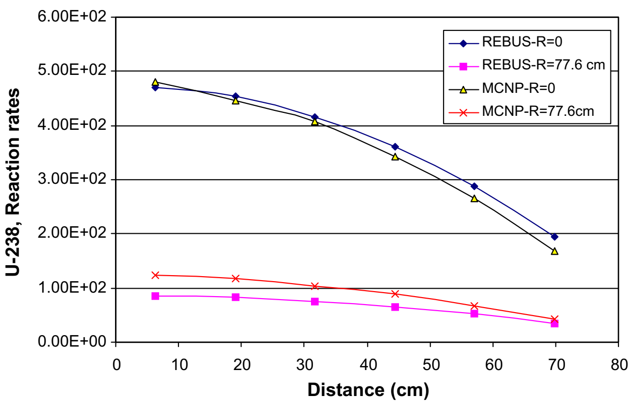

Fig. 4. Axial behavior of U-238 reaction rates calculated with MCNP/5 and MC^2-2/RERBUS atn centre (R = 0 cm) and periphery (R = 77.6 cm) of the core.

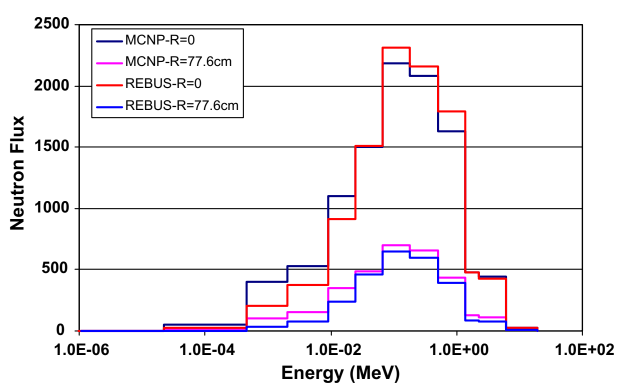

Fig. 5. Neutron flux spectra calculated with MCNP/5 and MC^2-2/RERBUS at centre (R = 0 cm) and periphery (R = 77.6 cm) of the core.

comparison. It can be seen that the calculated keff by the MCNP/5 heterogeneous model compares well with the experimental data with a difference of 119 pcm. This provides a better comparison than previously published results (Varuttamaseni and Lee, 2006) which showed a discrepancy of 697 pcm between calculations and measurements. This good comparison is due to the very precise modeling of the ZPR-6/7 assembly. Same behavior is also seen with the reaction rate ratio results. With other methodologies, errors are more than MCNP/5 heterogeneous approach. It concludes that our MCNP/5 detailed modeling provides good results for further analysis. The axial behavior of fission reaction rates at the centre and at the periphery of the core for 238U using MCNP/5 and MC^2-2/REBUS calculations are shown in Fig. 4. The variation in axial reaction rates with MCNP/5 and MC^2-2/REBUS is within 10%. It is more pronounced at the periphery (R = 77.6 cm) of the core. The neutron energy spectra in the middle and near the blanket of the core are shown in Fig. 5. The neutron spectra calculated from MCNP/5 and MC^2-2/REBUS also compare well and have discrepancies less than 7%.

Acknowledgement

This work supports an ongoing research project sponsored by a United States Department of Energy Nuclear Energy Research Initiative (NERI) grant. Dr. Masood Iqbal wishes to thank Higher Education Commission (HEC) of Pakistan for providing the financial support required to complete this work at North Carolina State University.

References

Argonne National Laboratory, Illinois, 2000. MC2-2; Code System for Calculating Fast Neutron Spectra and Multigroup Cross-sections, PSR-350.

Argonne National Laboratory, Illinois, 2001. REBUS-3/VARIANT8.0 Code System for Analysis of Fast Reactor Fuel Cycles, CCC-653.

Argonne National Laboratory, 2003. ZPR-6 Assembly 7: A Cylindrical Assembly with Mixed (Pu, U)-Oxide Fuel and Sodium with a Thick Depleted-Uranium Reflector, International Handbook of Evaluated Criticality Safety Benchmark Experiments, MIX-COMP-FAST-001, NEA/NSC/DOC(95)03/VI.

Los Alamos National Laboratory, 2005. MCNP – A General Monte Carlo N-Particle Transport Code, Version 5, vols. I–II, LA-UR-03-1987 and LA-CP-03-0245.

Varuttamaseni, A., Lee, J.C., 2006. Simulation of ZPR-6 assembly 7 with MCNP/5. Trans. Am. Nucl. Soc. 95, 734.