MONTE CARLO SIMULATION OF THE GREEK RESEARCH REACTOR NEUTRON IRRADIATION POSITIONS USING MCNP

I.E. STAMATELATOS, F. TZIKA

Institute of Nuclear Technology and Radiation Protection, NCSR "Demokritos",

Aghia Paraskevi, Greece

Email: ion@ipta.demokritos.gr

Abstract Prediction of neutron flux at the irradiation devices of a research reactor facility is essential for the design and evaluation of experiments involving material irradiations. A computational model of the Greek Research Reactor (GRR-1) was developed using the Monte Carlo code MCNP with continuous energy neutron cross-section data evaluations from ENDF/B-VI library. The model included detailed geometrical representation of the fuel and control assemblies, beryllium reflectors, irradiation devices and the graphite pile. The MCNP model was applied to predict neutron flux at the in-pool irradiation positions and the graphite pile. The MCNP estimated neutron fluxes were compared with measurements using activation foils and a good agreement between calculated and experimental results was observed.

1. INTRODUCTION

Greek Research Reactor (GRR-1) is an open pool type light water moderated and cooled reactor, operating at a thermal power of 5 MW. It is fueled by Material Test Reactor (MTR) type fuel elements and uses beryllium reflectors at the two sides of the core. The reactor is now converted to Low Enrichment Uranium (LEU) fuel. However, the reactor was operated with a mixed core containing both LEU and HEU fuel, of an enrichment of 19.75% and 93%, respectively, for a transition period of a few years. The reactor offers several irradiation positions, including in-pool irradiation devices, beam tubes and a graphite column. Modeling of the irradiation devices is essential for the design and evaluation of experiments involving material irradiations. For this purpose, a computational model of GRR-1 was developed using Monte Carlo code MCNP. The MCNP model was compared against calculations performed using the deterministic code CITATION and experimental measurements [1].

In this work, the MCNP model was applied to predict thermal neutron flux at the reactors in-pool irradiation positions and at the graphite pile. The results of the calculations were verified against experimental measurements performed using gold foils.

2. SIMULATIONS

A three dimensional model of the Greek Research Reactor (GRR-1) core configuration was developed using the Monte Carlo code MCNP (version 4C2) with continuous energy neutron cross-section data evaluations from ENDF/B-VI library [2]. Code and cross-section libraries package was obtained from NEA Data Bank. The developed model includes geometrical representation of the fuel and control assemblies, beryllium reflectors, irradiation traps and the graphite thermal neutron column. Figure 1 shows the core configuration in x and y coordinates (letters and numbers respectively) used. Irradiation devices are located at core lattice positions D4, A7 and F7 and in the beryllium reflector as well. A pneumatic sample transfer system for short-time neutron activation analysis is positioned at F9.

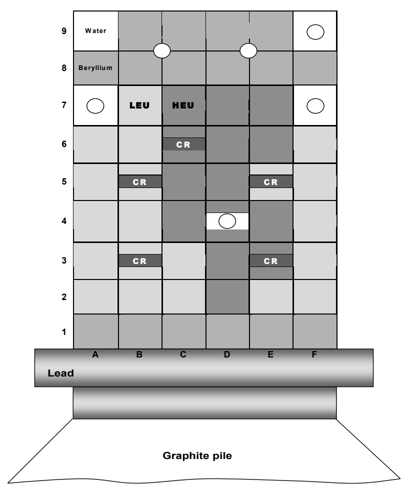

Fig.1 Core configuration (horizontal cross-section)

2.1 Reactor core simulation

The reactor core was modeled as a 9 × 6 lattice. The repeated structures capability was utilized for filling the lattice with standard and control fuel (HEU or LEU), beryllium, water and special irradiation assemblies (Figure 2). Control rods were model partially inserted. Preliminary calculations were performed to make the fission source converge from an initial guess distribution with an arbitrary but uniform set of points in the fuel regions. The final runs involved typically 50 settle cycles followed by 300 cycles of 5000 histories. The free gas scattering kernel model was chosen, except for hydrogen in light water beryllium and graphite, for which the appropriate S(a, b) data at 300^o K were employed to account for molecular binding effects below 4 eV. Each GRR-1 loading contains fuel assemblies of a different burn-up time. An average homogeneous burn-up of 105 days for all the fuel elements was considered. Fuel composition and burn-up was obtained from previous calculations performed using WIMS-REBUS codes. Track length estimates of neutron flux in a cell (F4) tallies were used. Cell tallies were positioned within the fuel, water channels, beryllium reflectors and graphite (pile).

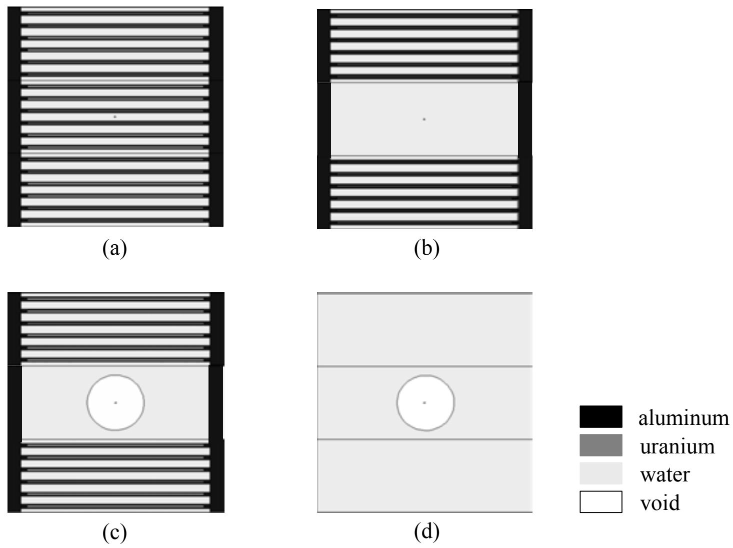

Fig. 2: Geometric configuration of (a) fuel assembly, (b) control assembly, (c) irradiation assembly and (d) water irradiation trap

2.2 Graphite pile simulation

The simulation geometry of the graphite thermal neutron column configuration is shown in Fig. 3. The thermal neutron column is a stacking of graphite blocks 130 cm in height, 130 cm in width and 282 cm in length. A 1 cm thick Boral layer and a stainless steel layer of 1.2 cm surrounds the graphite stack. The rest of the shielding consists of barytes concrete. A 60 cm-long thermal column extension is placed between the core and the thermal column entrance surface. This consists of stacked graphite blocks within an aluminum container. Two lead blocks of 20 cm total thickness are placed between the core and the thermal column extension for gamma-ray shielding purposes.

A significant gain in computation time was achieved by employing the Surface Source Write (SSW) option. Surface source was written at the front face of the graphite thermal column by recording the positions and the velocities of neutrons crossing the surface and entering the graphite column. The original number of histories used to write the SSW file, was 1×10^9 resulting in 1.6×10^6

tracks registered on the surface source. This file was then used as a source for subsequent MCNP simulations.

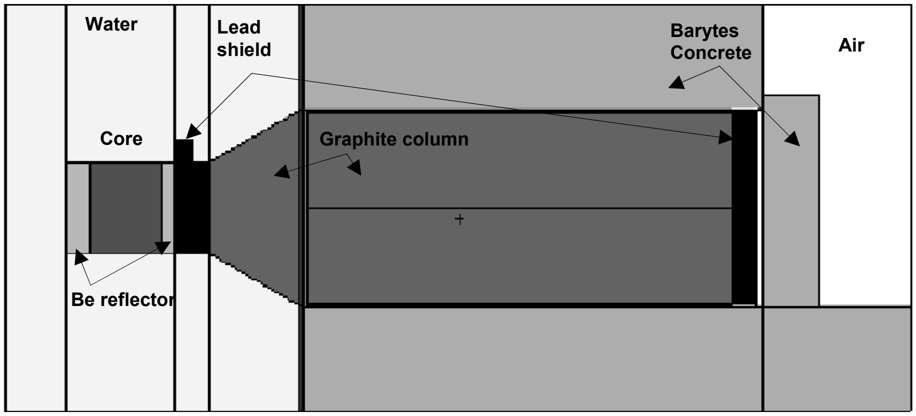

Fig. 3: Simulation geometry of the GRR-1 core and graphite column (vertical cross-section)

3. ACTIVATION FOIL MEASUREMENTS

Neutron flux verification measurements were performed using gold foils with and without cadmium covers to obtain the thermal and epithermal neutron flux distribution in several reactor positions. The foils were mounted on aluminum stringers placed at the positions of measurement. For the core neutron flux measurements the reactor was operated at 20 W for about 15 min. For the graphite pile neutron flux measurements the reactor was operated for 5 h at 1 MW, since neutron flux in the pile was several orders of magnitude lower. The stringers were removed and each foil was counted at a calibrated high-purity germanium semiconductor detector of 20% efficiency. The activity of the foils was determined and corrected for absolute detector efficiency and decay.

4. RESULTS AND DISCUSSION

4.1 In-pool irradiation positions

Table 1 shows a comparison of the “sub-cadmium” ($E < 0.5$ eV) neutron flux as evaluated by MCNP and measured using gold foils at the reactor in-pool irradiation positions. The maximum deviation between calculated and experimentally determined thermal neutron flux was observed at position Z-9 and was of 10.8 %. Other workers have also reported agreement between MCNP neutron flux calculations and experimental measurements within 10-20% [3-11].

TABLE 1. COMPARISON OF CALCULATED AND EXPERIMENTALLY MEASURED "SUB-CADMIUM" NEUTRON FLUX AT GRR-1 IRRADIATION POSITIONS

Irradiation Position	Calculated * $\times 10^{13}$ (cm^{-2} s^{-1})	Experimental ** $\times 10^{13}$ (cm^{-2} s^{-1})	$\left(\frac{(Calc - Exper)}{Exper} \right) \times 100$
D-4	10.26	10.28	- 0.19
A-7	6.49	6.30	+ 3.02
Z-7	6.61	6.01	+ 9.98
Z-9	1.23	1.11	+ 10.81
Be-1	9.15	8.33	+ 9.84
Be-2	7.70	7.30	+ 5.48

* Statistical error less than 5%

** Estimated experimental error less than 10%

4.2 Special irradiation assembly at D4 lattice position

Moreover, a comparison of the "sub-cadmium" (E < 0.5 eV) neutron flux distribution along the vertical water channel of the special irradiation assembly, at D4, as evaluated by MCNP and measured using gold foils is shown in figure 4. A very good agreement between calculated and experimentally determined thermal neutron flux is observed. It is noted that neutron flux distribution is skewed towards the lower level of the core. This result is explained by the perturbation of the thermal neutron flux due to the partial insertion of the control rods from the upper level of the core.

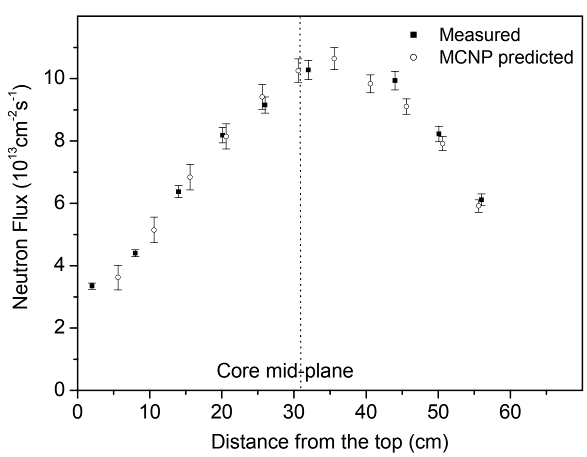

FIG. 4 Comparison of MCNP calculated and measured neutron flux (E<0.5 eV) along the vertical irradiation tube at D-4 lattice position.

4.3 Graphite pile

Fig. 5 shows a comparison of the "sub-cadmium" (E < 0.5 eV) neutron flux distribution along the graphite pile as evaluated by MCNP and measured using gold foils. As it can be observed calculations performed for detailed core geometry in criticality mode and a simplified homogenous core assembly with a Maxwell fission neutron source spectrum showed no significant difference in the thermal neutron flux distribution.

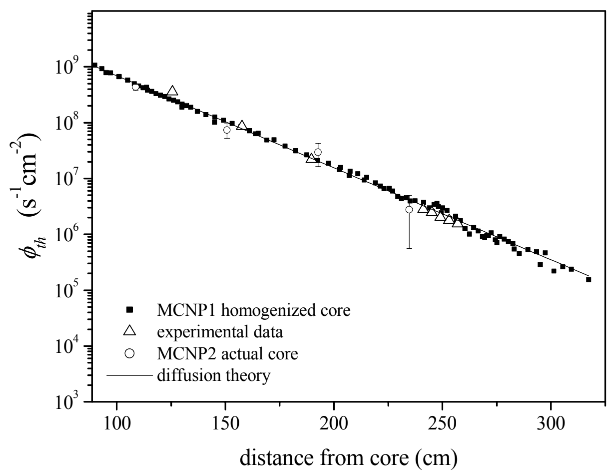

Fig. 5 Thermal neutron flux distribution along graphite pile.

5. GENERAL DISCUSSION AND CONCLUSIONS

Modeling of the reactor irradiation devices is essential for the evaluation and assessment of experiments involving material irradiations [12, 13]. MCNP code was used to model GRR-1 reactor and irradiation devices. An agreement of better than 11% between the predictions and neutron flux measurements performed using gold foils was observed. Future work will be aimed to improve the model by taking into account the spatial burn-up distribution [14, 15].

Since experimental determination of the flux characteristics at the reactor irradiation devices is time and labor consuming, the developed model is a useful tool in predicting neutron flux in irradiated samples. It was proven particularly useful in the time period of gradual conversion of GRR-1 core from HEU to LEU fuel since it enabled estimation of the neutron field characteristics at the irradiation devices under the actual mixed core configuration conditions.

ACKNOWLEDGEMENTS

The useful discussions with Dr. S. Day of McMaster Nuclear Reactor, Dr. O. Calzetta and Dr. F. Leszczynski of Bariloche Atomic Centre are most gratefully acknowledged.

REFERENCES

[1] STAMATELATOS I.E., VARVAYANNI M., TZIKA F., ALE A.B.F., CATSAROS N. Nucl. Instr. Meth. Phys. Res. B, 263 (2007) 136-139

[2] J.F. BRIESMEISTER (Ed.), MCNP – A General Monte Carlo N-particle Transport Code (Version 4C), RSIC CCC-700, Oak Ridge National Laboratory (1997).

[3] FERNANDES A.C., GONCLAVES I.C., BARRADAS N.P., Nucl. Technol. 143 (2003) 358.

[4] PERLADO J.M., MARIAN J., SANZ J.C., Nucl. Technol. 129 (2000) 285.

[5] REDMOND E.L., YANCH J.C. , HARLING O.K. , Nucl. Technol. 106 (1994) 1.

[6] KIM C-H , JANG S., REECE W.D. , Nucl. Technol., 145 (2004) 1.

[7] KITAMURA A., MATSUMOTO J., FURUYAMA Y., TANIKE A., KUBOTA N., OHSAWA T., HASHIMOTO K., HORIGUCHI T., TSURUTA T., J. Nucl. Sci. Technol., 40 (2003) 349.

[8] MOHAPATRA D.K, SUNIL SUNNY C., MOHANAKRISHNAN P., SUBBAIAH K.V., Ann. Nucl. Ener., 31 (2004) 2185-2194.

[9] AZIZ M., Kerntechnik, 70 (2005) 286-290.

[10] KUZMINOV V., KALCHEVA S., KOONEN E., Proceedings Monte Carlo 2005 Topical Meeting, (2005) 1405-1422.

[11] HUDA M.Q., RAHMAN M., SARKER M.M., BHUIYAN S.I., Ann. Nucl. Ener. 31 (2004) 1299-1313.

[12] TZIKA F, STAMATELATOS I.E., Nucl. Instr. Meth. Phys. Res. B 213 (2004) 177-181.

[13]TZIKA F., STAMATELATOS I.E., KALEF-EZRA J., BODE P., Nukleonika, 49 (2004) 115-121.

[14] HANAN N.A., POND R.B., WOODRUFF W.L., BRETCHER M.M., MATOS J.E., The Use of WIMS-ANL Lumped Fission Product Cross Sections for Burned Core Analysis with the MCNP Monte Carlo Code, Proceedings 1998 International Meeting on Reduced Enrichment for Research and Test Reactors, Sao Paolo, Brazil, October 18-23, 1998.

[15] HAECK W., VERBOOMEN B., Nucl. Sci. Eng., 156 (2007) 180-196.