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Criticality Calculations for The IAEA 10 MW Research Reactor using Nodal Diffusion Methods

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ABSTRACT

In this work Nodal diffusion methods will be used for research reactor calculations. The unified nodal method was developed for plate type research reactor calculations. It was validated for PWR benchmark problems and then applied for the IAEA MTR benchmark problem for static calculations to check the validity and accuracy of the method. A 10 MW research reactor core is considered with three calculation cases for low enriched uranium fuel depending on the core burnup status of fresh, beginning-of-life, and end-of-life cores. The validation work included criticality calculations, flux distribution, and power distribution. The homogenized cross sections were generated using the TRITON–NEWT system. The results were compared with a reference, which was taken from IAEA-TECDOC-233 and they are in a good agreement all-rod out case.

1. Introduction

Nodal diffusion methods are rarely used for research reactor analyses owing to the complex shape and small size of the core, which has a high leakage potential. Nodal diffusion methods have been shown to be computationally efficient and are used in practical routine core calculations which need excessive computing times. The wide application of neutron diffusion codes has been continuously conducted for criticality calculations, calculations of the neutron flux and power distribution, and in-core fuel management [1].

In addition, important safety parameters such as excess reactivity, control rod worth, and shutdown margin can be calculated. The use of nodal diffusion theory for a small core such as in material testing reactors (MTR) maybe over optimistic, but if reasonable results can be obtained, the nodal method can be used for a fast core design and fuel management which cannot be achieved using Monte Carlo codes [2].

In this study, the unified nodal method UNM was used for IAEA 10 MW research reactor calculations; criticality calculations, a flux distribution, and power distribution to validate the UNM for MTR. The results of the static part of the safety-related benchmark calculation proposed by the IAEA are reported and discussed. The TRITON–NEWT system was considered to obtain group constants.

2. Unified nodal method

In unified nodal method UNM [3,4] the ANM solution to two-group diffusion equations can be reformulated in exactly the same way as the NEM solution, and thereby, the two most popular transverse integrated nodal method formulations can be integrated into a unified nodal method UNM formulation. It was demonstrated that the numerical instabilities at the near critical nodes can naturally be resolved by the UNM formulation itself without introducing any approximate stabilization schemes.

A 3-D UNM code was written to perform the calculation. Each assembly is considered one node in the radial direction, and in the axial direction, the node was divided into multiple nodes depending on the size of the assembly in the radial direction. This code uses two group constants generated by the NEWT code. A reflective, zero flux, and zero incoming current (vacuum) can be applied to the problem boundaries. The eigenvalue k_{eff}, thermal and fast flux distribution, and power distribution can be obtained from this code.

To validate this code, two benchmark problems were applied: the IAEA 2-D and 3-D PWR problem. The IAEA 2-D and 3-D PWR problem is a very important standard benchmark problem for measuring the performance of neutronics calculation methods. The results are shown in Table 1.

Table (1) Accuracy of UNM for eigenvalue benchmark problems.

Benchmark	Reference k_{eff}	Calculated k_{eff} with UNM (diff pcm)	Maximum power error (%)	Root mean square error (%)
IAEA 2D	1.02956	1.02965 (-9.6)	0.45	0.19
IAEA 3D	1.02903	1.02913 (-10.6)	1.46	0.59

3. IAEA 10 MW core description

The 10MW core is considered. The core configuration consists of a 6 x 5 grid containing 21 standard fuel elements, in which there are four control elements (fuel elements with absorber blades). The fuel material is UAlx-Al LEU fuel with an enrichment of 20 w/o ^{235}U. The fuel density is 6.108 g/cc, and 72 w/o of uranium in the UAlx-Al. For the beginning of life (BOL), the burnup distribution of standard fuel elements (SFEs) and control fuel elements (CFEs) in percent loss of ^{235}U is shown in Fig. 1. The core is reflected by graphite on two opposite sides and surrounded by light water. Axially, all fuel and graphite elements are reflected in their edges by a 15.0 cm Al–H2O reflector containing 20% Al and 80% H2O as volume fractions. 20 cm thick water was added above the axial end boxes. The SFEs contain 23 plates with dimensions of 77 mm x 81 mm, whereas the CFEs contain 17 standard plates with a special region for the four fork-type absorber blades having the same dimensions of the SFE, four plates of pure aluminum, each 1.27 mm thick at the first, third, twenty-first, and twenty-third standard plate positions. Detailed specifications of this benchmark are described in (IAEATECDOC- 233) reference [6].

Water	Graphite	Graphite	Graphite	Graphite	Water
Water	FA 5% 10%	FA 25% 30%	FA 25% 30%	FA 5% 10%	Water
FA 5% 10%	CFA 25% 30%	FA 45% 50%	4.48 45% 50%	CFA 25% 30%	FA 5% 10%
FA 25% 30%	FA 45% 50%	FA 45% 50%	Water+Al FA 45% 50%	FA 45% 50%	FA 25% 30%
FA 5% 10%	CFA 25% 30%	FA 45% 50%	FA 45% 50%	CFA 25% 30%	FA 5% 10%
Water	FA 5% 10%	FA 25% 30%	FA 25% 30%	FA BOL 5% EOL 10%	Water
Water	Graphite	Graphite	Graphite	Graphite	Water

Figure (1) Cross-sectional view of the 10 MW MTR IAEA problem for BOL and EOL cases.

4. Two-group constant generation

NEWT is a multi-group discrete-ordinates transport code with flexible meshing capabilities that allow 2-D neutron transport calculations using complex geometric models. NEWT can be used to prepare a collapsed weighted cross section. The primary function of NEWT is to calculate the spatial flux distributions within a nuclear system and collapse the cross sections into multiple (or single) energy groups as specified by the user. Used as a part of the TRITON depletion sequence, NEWT provides spatial fluxes, weighted multi-group cross sections, and power distributions used for depletion calculations and coupled depletion and branch calculations needed for lattice physics analysis [7].

The 238-group ENDF/B-VII neutron cross section library and NEWT were used to generate the two-group macroscopic cross sections needed for the UNM code input. The 238-group cross sections and fluxes were collapsed into a two-group form: fast flux (group 1) consisting of neutrons with energies between 3 eV and 20 MeV and thermal flux (group 2) consisting of neutrons with energies between 10 5 eV and 3 eV. A cutoff energy of 3 eV was chosen such that upscattering could be neglected. Using the NEWT model, the cross sections for the standard fuel were generated with a three-region unit cell consisting of meat (fuel), aluminum clad (Al clad) and light water. The aluminum side plate and extra water region on the edge of the side plate were treated as individual cells. The fuel assembly modeled in NEWT is shown in Fig. 2.

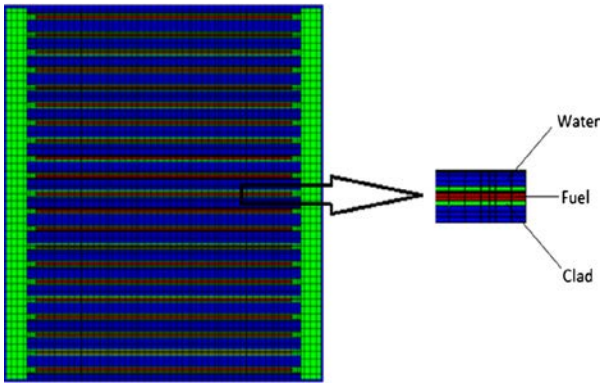


Figure (2) Standard fuel assembly model in NEWT code

5. Results

5.1. UNM model

For whole-core calculations, a 3-D diffusion model of the core was made. Each assembly was divided into 4 nodes in the radial direction and 12 nodes in the axial direction, that is, a total of 48 nodes per assembly, to model the exact shape of the core. Each node has a dimension of 3.85 cm x 4.05 cm x 5.0 cm in the x, y, and z directions, respectively. The core is reflected in the radial direction with three layers of light water reflector each having the size of one node. In the axial direction, the core was reflected with 15 cm of water and aluminum, and 20 cm of light water. The vacuum boundary condition is applied at the outer reflector periphery. A 3-D view of the core is shown in Fig. 3.

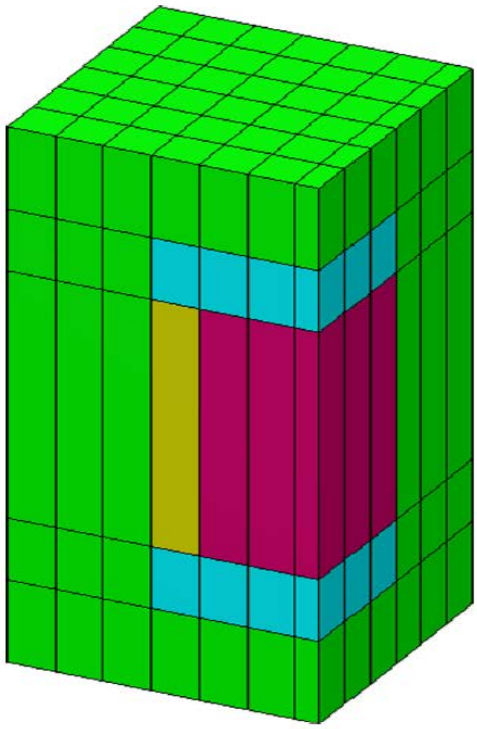


Figure (3) 3-D view for aquarter of the UNM core model.

5.2. Criticality results

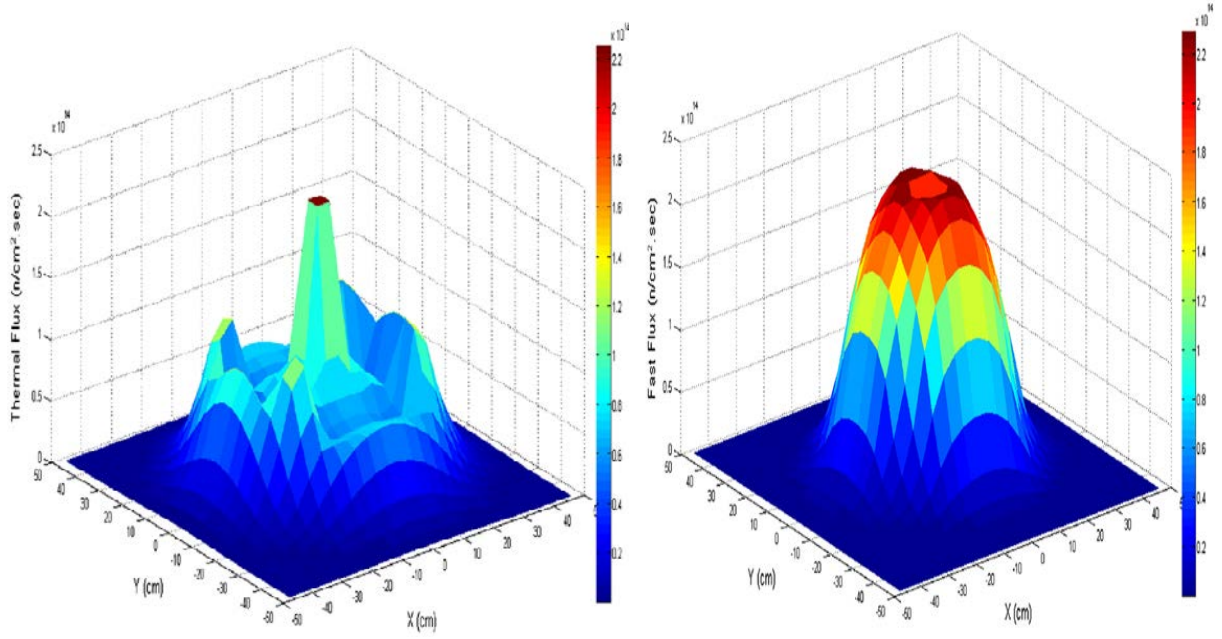
The calculated values for k_{eff} using the UNM code for the fresh, BOL, and EOL cases with all rods out (ARO) are given in Table 2. The eigenvalues calculated by ANL and other organizations are given in the same table. All fresh-core calculations are done for xenon free cores. The results obtained by a UNM calculation were slightly higher than those obtained by the reference ANL, and varied depending on the methodologies, models, and cross-section libraries used in the calculations. The MCNP5 results were taken from reference [8] to compare UNM the results with the MCNP5 reference results.

Table (2) k_{eff} Results compared with other organizations results

Organization (code)	Fresh	BOL	EOL
Current study (UNM)	1.17313	1.06223	1.04078
MCNP5 (difference)	1.17238 ± 0.00033 (0.75 mk)	1.05617 ± 0.00031 (6.06 mk)	1.04111 ± 0.00032 (0.33 mk)
ANL (VIM) (difference)	1.168 ± 0.0033 (5.13 mk)	N/A	1.048 ± 0.0034 (7.22 mk)
ANL (DIF2D)	1.1683	N/A	1.03934
EIR (CODIFF)	1.15937	N/A	1.04981
OESGAE (EXTERMINATOR)	1.1813	N/A	1.0120
CEA (NEPTUNE)	1.187	N/A	1.05468
JAERI (ADC)	1.18339	N/A	1.04122

5.3. Flux distribution

For the flux distribution of the BOL core, Fig. 4 illustrates the x–y radial flux distributions at the middle axial layer for thermal and fast neutron fluxes. As expected for all studied cases, the maximum thermal flux occurs at the central flux trap and drops exponentially, moving away from the core. The value of the average thermal flux in the central trap for the BOL case is 2.255E+14, which was calculated by the ANL codes as 2.585E+14; The thermal flux slightly increases at the interface of the fuel/water and fuel/graphite. On the other hand, the fast flux behaves in an opposite way as the thermal flux. Fig. 5 shows the fluxes in the middle plane for the BOL stage plotted along the x-axis. In Fig. 6, the distributions of the axial fluxes in the water trap are given.



(a) (b)

Figure (4) (a) Radial thermal flux distribution for BOL core. (b) Radial fast flux distribution for BOL core.

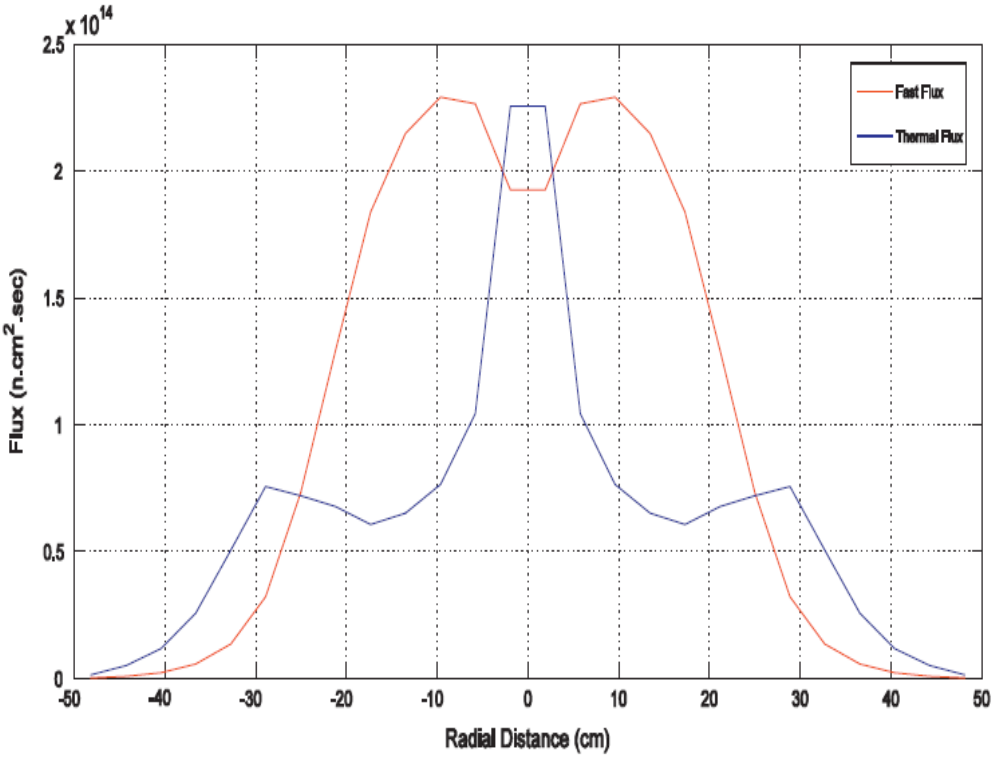


Figure (5) Neutron fluxes along the x-axis in the BOL core

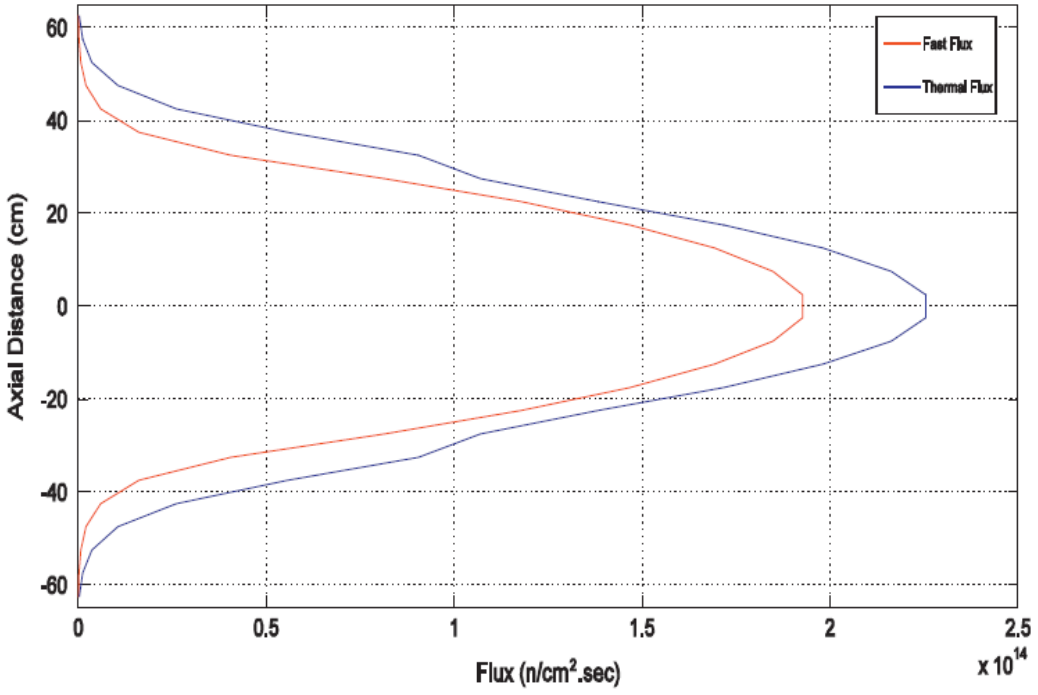


Figure (6) Axial neutron fluxes along the z-axis in the BOL core

5.4. Power distribution

The relative radial power distributions for each fuel element are illustrated in Fig. 7 for the BOL stage in comparison with the MCNP5 calculations which were taken from reference [8]. As expected, the maximum power occurs in the trap channel localized at the center of the core, and the highest difference from the MCNP5 result was 7% for the BOL and the EOL cases.

Water	Graphite	Graphite	Graphite	Graphite	Water
Water	4.32 (5.12) 4.11	4.04 (3.33) 3.91	4.04 (4.40) 3.87	4.32 (3.86) 4.16	Water
3.97 (1.06) 3.93	3.11 (5.33) 3.29	4.54 (2.66) 4.66	4.54 (2.24) 4.64	3.11 (7.03) 3.35	3.97 (1.06) 3.93
3.44 (2.18) 3.37	3.91 (2.62) 3.81	2.68 (2.88) 2.76	UNM (difference) MCNP5	2.68 (3.23) 2.77	3.91 (0.51) 3.93	3.44 (0.10) 3.44
3.97 (2.89) 3.86	3.11 (5.62) 3.30	4.54 (2.87) 4.67	4.54 (4.30) 4.74	3.11 (6.75) 3.34	3.97 (0.21) 3.98
Water	4.32 (4.36) 4.14	4.04 (3.87) 3.89	4.04 (1.26) 3.99	4.32 (4.11) 4.15	Water
Water	Graphite	Graphite	Graphite	Graphite	Water

Figure (7) Power fraction distribution for each fuel element in the BOL case.

6. Conclusion

The two-group unified nodal methods were applied to the IAEA 10MW MTR after validation for PWR benchmark problems. The homogenized group parameters and depletion cross sections were obtained from the TRITON–NEWT system. A number of calculations were conducted using the 3-D UNM model, and the results were compared with the reference values given in IAEA-TECDOC-233 and MCNP5 results. For the cases of ARO, the values obtained from UNM were in a good agreement with the reference. For the flux and power calculations, the results from UNM were almost the same as the reference, and the highest value of the thermal flux was in the central flux trap. In addition, the power distribution was as expected, and the highest power fraction was in the central flux trap assembly.

The differences between UNM and reference results are that due to different models, different methods of solution and also different cross section libraries are used. In UNM model we use the two-group diffusion approximation with homogenized group constants while in MCNP5 which uses different solution method, continues cross section library and exact geometry description.

7. References

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