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Original Article

An investigative study of enrichment reduction impact on the neutron flux in the in-core flux-trap facility of MTR research reactors

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ABSTRACT

Research reactors in-core experimental facilities are designed to provide the highest steady state flux for user's irradiation requirements. However, fuel conversion from highly enriched uranium (HEU) to low enriched uranium (LEU) driven by the ongoing effort to diminish proliferation risk, will impact reactor physics parameters. Preserving the reactor capability to produce the needed flux to perform its intended research functions, determines the conversion feasibility. This study investigates the neutron flux in the central experimental facility of two material test reactors (MTR), the IAEA generic10 MW benchmark reactor and the 22 MW s Egyptian Test and Research Reactor (ETRR-2). A 3D full core model with three uranium enrichment of 93%, 45%, and 20% was constructed utilizing the OpenMC particle transport Monte Carlo code. Neutronics calculations were performed for fresh fuel, the beginning of life cycle (BOL) and end of life cycle (EOL) for each of the three enrichments for both the IAEA 10 MW generic reactor and core 1/98 of the ETRR-2 reactor. Criticality calculations of the effective multiplication factor (K_{eff}) were executed for each of the twelve cases; results show a reasonable agreement with published benchmark values for both reactors. The thermal, epithermal and fast neutron fluxes were tallied across the core, utilizing the mesh tally capability of the code and are presented here. The axial flux in the central experimental facility was tallied at 1 cm intervals, for each of the cases; results for IAEA 10 MW show a maximum reduction of 14.32% in the thermal flux of LEU to that of the HEU, at EOL. The reduction of the thermal flux for fresh fuel was between 5.81% and 9.62%, with an average drop of 8.1%. At the BOL the thermal flux showed a larger reduction range of 6.92%–13.58% with an average drop of 10.73%. Furthermore, the fission reaction rate was calculated, results showed an increase in the peak fission rate of the LEU case compared to the HEU case. Results for the ETRR-2 reactor show an average increase of 62.31% in the thermal flux of LEU to that of the HEU due to the effect of spectrum hardening. The fission rate density increased with enrichment, resulting in 34% maximum increase in the HEU case compared to the LEU case at the assemblies surrounding the flux trap.

1. Introduction

In the international atomic energy agency (IAEA) consultant meeting on “preparation of a program on research reactor core configurations to use LEU instead of HEU”, in Vienna, Austria [1] the calculation of benchmark problems were specified with the intention to compare the consistency of the reactor physics calculations by different research centers. The IAEA 10 MW MTR benchmark was created to be representative of an ideal pool-type reactor for material testing, and safely related calculations with fuel management rather than a lifetime core. In recent years, several studies analyzed this generic reactor to validate reactor physics and advanced neutronics using different code structure and nuclear data libraries. The studied coding platforms include COMSOL [2], OpenMC [3], MCNP5 [4], and Serpent [5]. The benchmark of these MTR calculations included the neutronic, thermal-hydraulics, different transient and accidental scenarios. Criticality experiments and calculations data for the ETRR-2 are well documented for core 1/98 with a complete core description for LEU, which make it suitable for validating coding platform [11]. ETRR-2 provided calculations were performed using WIMSD4 for cell calculations and CITATION code for core calculation. The calculation and experiments include excess reactivity calculations through control rods differential worth experiments [11].

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OpenMC Monte Carlo simulation platform was developed by the Computational Reactor Physics Group (CRPG) members at the MIT in 2011. The initial development was focused on criticality calculation that is necessary for nuclear reactor simulation [6]. Continuous development of the platform brings much more functionality for the user like the tally (flux, reaction rate, fission, etc.) capability with a rudimentary plotting to track geometry error and the data post-processing. And this development is subjected to improve even more, where more complex geometry will be available to design with advance plotting/post-processing. Although a few studies have been performed utilizing the OpenMC Monte Carlo simulation platform for particle transport in research reactor physics calculation, OpenMC is a powerful simulation platform for neutron criticality calculation and particle transport simulation based on a 3D simulated model of solid geometry construction. The particle interaction data used in OpenMC is on HDF5 native format, which can be generated from the ACE format used in the MCNP and Serpent Monte Carlo platform.

In recent years several studies have investigated enrichment reduction in research reactors. A study on the enrichment reduction of High Flux Isotope Reactor (HFIR), from HEU (93.1%) to LEU (20%) shows that for BOL fuel configuration, the reactivity is reduced by 7.7% [7]. Another study on the use of LEU (less than 20%) instead of HEU (90%) in the MNSR reactor showed that the K_{eff} variance was less than 0.00007 for LEU than HEU. Likewise, in the case of flux, the variance was less than 10% in the inner radiation sites and less than 8% in the outer irradiation sites [8]. Another study on the use of LEU (12.6%) instead of HEU (90.2%) in Pakistan research reactor shows that the thermal fluxes in the inner radiation sites drop 8%, 8.5% in the fission chambers, 6% in the small outer irradiation sites and 6% in the big outer irradiation sites. Also, the fast neutron flux reduction is one-third of that thermal fluxes in LEU [9]. The Oregon State TRIGA reactor (OSTR) was converted from HEU (70%) to LEU (19.75%) in 2008. The flux was calculated in its 6 irradiation sites for both HEU and LEU. The result shows that the flux spectrum was somewhat similar in shape to each other despite a small degree of spectrum hardening in the LEU case due to the increased loading of ^{238}U [10]. The aim of this study is to calculate reactor physics parameters by utilizing the MTR benchmark IAEA 10 MW and ETRR-2 reactors using OpenMC Monte Carlo simulation platform. In this work, we calculated the K_{eff}, flux distribution and the fission reaction rate of the reactor, and compare the results with other previous studies on the same benchmark done with other computational platforms. The study is focused on investigating the change of the neutron flux in the reactor central flux trap facility when configured with different fuel enrichment. It should be noted that, this study replaced fuel compositions directly, with no change to fuel management or other conversion design considerations. As noted in the case of the OSTR conversion, recent conversions have applied design changes to the fuel, fuel management, or experimental configuration to assure that capabilities are not unduly impacted by conversion.

2. Reactors description

The IAEA 10 MW is the standard benchmark for material testing and research purposes set by the IAEA [1]. This reactor is a pool type, light water cooled, and moderated by forced upward circulation of light water. Its core consists 5 × 6 grid matrix of 7.6 cm × 8.1 cm square array, loaded with 24 fuel elements (7.6 cm × 8.05 cm) having 23 fuel plates. The two central half fuel assemblies have a dimension of (7.6 cm × 3.9615 cm) and 12 fuel plates, leaving one flux trap central position (3D) for in core experimental facility, as shown in Fig. 1(a). There are 8 graphite assemblies as the radial reflector with the same dimension of the fuel assemblies. The core height is considered in the z-axis (up and down), the width is in the x-axis (left and right) and the thickness is in the y-axis (front and back). Each Standard Fuel Assemblies (SFA) consists of 23 fuel plates.

There are 4 Controlled Fuel Assemblies (CFA) consisting of 17 fuel plates and 4 dummy graphite plates. Each fuel assembly has a height of 60 cm with 0.475 cm of Al cladding on top and bottom. There is 0.475 cm of side cladding (left and right) for each assembly. The dimension of each assembly is 7.6 × 8.05 × 60. Details of the dimensions are shown in Table 1. There is a 0.1 × 0.05 water gap in between each assembly. There are in total 22 grids (same dimension as the fuel assembly) of water surrounding the fuel and the graphite reflector assemblies. Two of the middle fuel assembly constructs the flux trap facility containing water. Each of has 12 fuel plates like SFA, without having the rest of fuel plates, they constructing the 7.6 × 7.923 × 59.05 flux trap facility with water. There are axial reflectors as 20%Al and 80% H_2O material covering the whole reactor core with its surrounding water on top and bottom. The height of the axial reflector is 15 cm, the design specifications data are listed in Table 1.

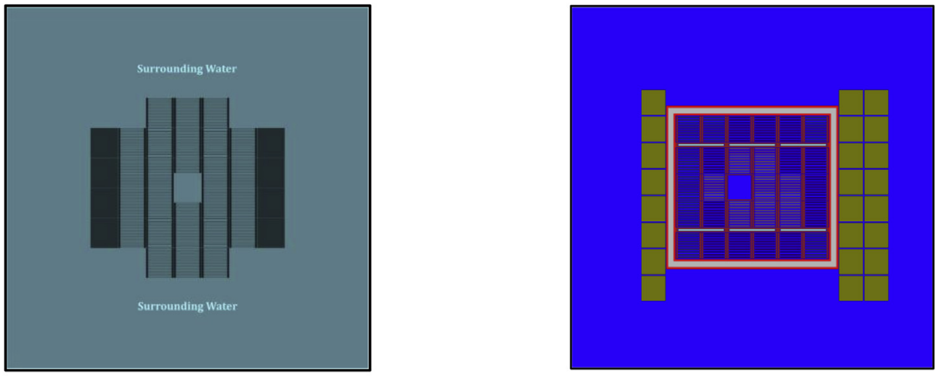
ETRR-2 is a 22 MW_{th} reactor consisting of 29 fuel assemblies surrounded by beryllium reflector and a central in-core-flux-trap facility as shown in Fig. 1(b). There are three types of fuel assemblies with three different loadings of 8.1 gm/cm^3 U_3O_8 with enrichment 19.7% ^{235}U, which are 404, 209 and 148 indicating the grams of uranium in each assembly. Each fuel assembly is of an active height of 80 cm and contains 19 fuel plates; the active meat dimension in each plate is 80.0 × 6.4 × 0.070. The reactor design specifications are listed in Table 1, for a detailed description of the reactor 1/98 core the reader is referred to as reference [11,12]. The in-core-flux-trap is preserved for cobalt irradiation device. Loading of the cobalt device started from core 2/98, while the position contained just light water in core 1/98. In core 1/98, excess reactivity was measured by calibrating control rods differential worth by the period method.

3. Methodology

In this work, IAEA generic 10 MW and ETRR-2 benchmark problems specifications provided in the guide [1,11] has been used to analyze and the simulation was generated to verify the result with other platforms. Three-dimensional full core model was built using the design specifications data of the IAEA 10 MW and ETRR-2 MTR reactors. Reactors analysis were done on three fuel enrichments 20%, 45%, and 93%. In addition, for the IAEA, 10 MW three states of the fuel life configuration; fresh, beginning of lifecycle (BOL) and end of lifecycle (EOL) was also taken into account for this enrichment. Fresh fuel was taken for ETRR-2 reactor to be consistent with the benchmark. Each of the configurations was designed by direct fuel replacement, with specific fuel material composition [1].

It should be noted that, while fuel replacement allows the results of this study to be compared to other evaluations, it does not represent the actual conversion process. Typically fuel composition, density and chemical form, maybe changed to optimize the HEU to LEU conversion [13]. Furthermore, all calculations were performed with a supercritical core, which does not represent an actual reactor state.

The simulation platform used in this work was OpenMC Monte Carlo simulation platform and the Python programming language. OpenMC is a particle transport Monte Carlo simulation code that allows nuclear reactor simulation mainly for neutron criticality calculation. The Computational Reactor Physics Group (CRPG) team in the Massachusetts Institute of Technology (MIT) developed OpenMC in 2011. OpenMC is a powerful simulation platform that



(a) (b)

Fig. 1. OpenMC model x-y cross-section view of (a) the IAEA generic reactor (b) the ETRR2 reactor.

Table 1
Design specifications data of the IAEA generic 10 MW benchmark and ETRR-2 Core 1/98.

	IAEA 10 MW	ETRR-2 Core 1/98
Thermal power	10 MW	22 MW
Core active height	60 cm	80 cm
Core grid matrix	5 × 6	5 × 6
Nuclear fuel	MTR fuel type plate	MTR fuel type plate
Fuel material	UAlx-Al	U_3O_8
Uranium density 93/45/20% (g/cm^3)	0.679/1.604/4.398	1.105–2.565/same/same
Clad material	Al	Al-6061
Coolant	Light water	Light water
Moderator	Graphite-light water	Light water
Total number of standard fuel assembly	22	29
Total number of control fuel assembly	4	6
Total number of fuel plates per standers fuel assembly	23	19
Total number of fuel plates per control fuel assembly	17	1
Number of Al plates per control fuel assembly	4	NA
Radial reflector	Graphite	Beryllium
Graphite assembly dimensions (cm)	7.6 × 8.05 × 60	NA
Axial reflector	20% Al − 80% H_2O	H_2O
Axial reflector height (cm)	15	20
Total number of graphite assembly	8	NA
Grid plate dimensions (cm)	7.7 × 8.1	8.1 × 8.1
Fuel assembly dimensions (cm)	7.6 × 8.05 × 60	119.5 × 8.0 × 8.0
Fuel plate dimensions (cm)	6.65 × 0.127 × 59.05	84.0 × 7.0 × 0.150
Fuel meat dimensions (cm)	6.3 × 0.051 × 58.7	80.0 × 6.4 × 0.070
Fuel front and back cladding thickness (cm)	0.038	0.04
Fuel top and bottom cladding thickness (cm)	0.175	2
Coolant channel between fuel plates dimension (cm)	0.223	0.270

can simulate a 3D object designed with solid geometry [6,14]. The geometry defined for the program is built with second-order surfaces in depth. OpenMC can design almost all geometry surfaces available in the nuclear reactor design. The coding language used by OpenMC is Extensible Markup Language (XML). The cross-sectional data for the material of nuclide OpenMC support both continuous-energy and multi-group energy mode. In continuous energy mode, OpenMC uses a native HDF5 file formed of the material data than an ACE file format date used by the MCNP and Serpent. The ACE files can also be converted into HDF5 files with OpenMC to use. The National Nuclear Data Center (NNDC) provided nuclear material cross-sectional data version ENDF/B-VII.1 was used in this work for analysis. The OpenMC version used in this work was 0.10.0. A 3D model of the whole IAEA 10 MW and ETRR-2 reactors including its water pool was designed with OpenMC to perform the calculations mentioned below.

Python programming language was used in this work to generate plots and in data post-processing. The output data files from OpenMC are in HDF5 file format, which can be farther analyze using Python API. The mesh flux plots shown in Figs. 2–-5, was generated with python API using the output data file "statepoint.h5" generated from the OpenMC simulation of the reactor. OpenMC uses Python scripts for data visualizations and post-processing.

The IAEA 10 MW benchmark was modeled exactly as described by the specifications stated [1,2]. While the ETRR-2 core 1/98 was modeled exactly as described by the specifications stated [11] for 20% fuel enrichment and two other hypothetical states for 45% and 93% enrichment. Since the benchmark was a pool-type reactor, the model included not only the core itself but also the surrounding water on the sides, above and below. In benchmark criticality calculation the reading was taken for 550 batches with 50 inactive batches and 20000 particles per batch. The space of source was considered a box of dimensions −37 cm−37 cm in X-axis, −35 cm−35 cm in Y-axis and −45 to 45 in Z-axis direction for IAEA, 10 MW. While, the space of source was considered a box of dimensions −44 cm−44 cm in X-axis, −35 cm−35 cm in Y-axis and −70 to 55 in Z-axis direction for ETRR-2 reactor.

In OpenMC the flux tally is normalized per source neutron as like in MCNP. The default unit of the flux tally in OpenMC is

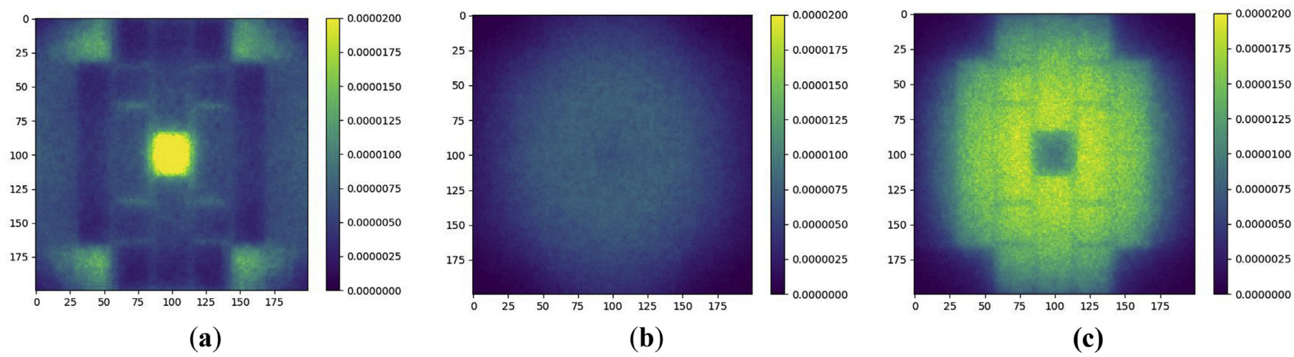


Fig. 2. Radial flux disribution at the midplane of 20% enriched core of the IAEA generic reactor at BOL (a) Thermal flux (b) Epithermal flux (c) Fast flux.

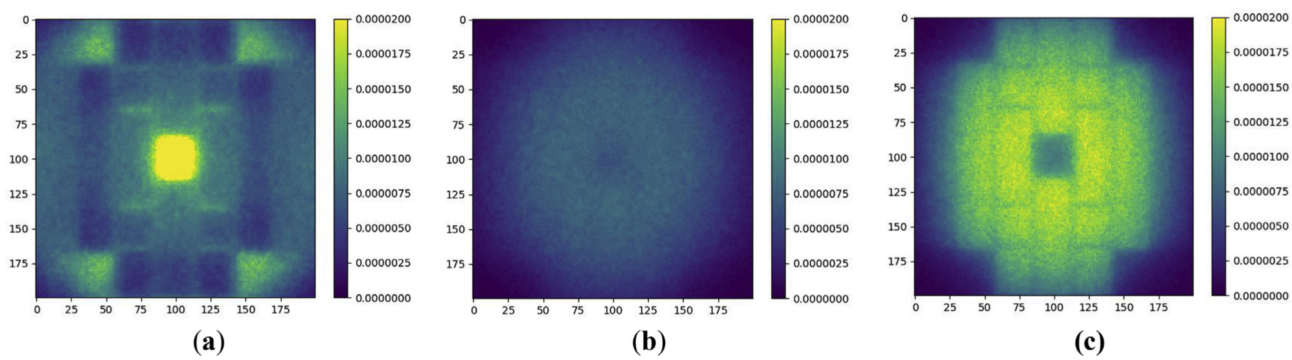


Fig. 3. Radial flux distribution at the midplane of 93% enriched core of the IAEA generic reactor at BOL (a) Thermal flux (b) Epithermal flux (c) Fast flux.

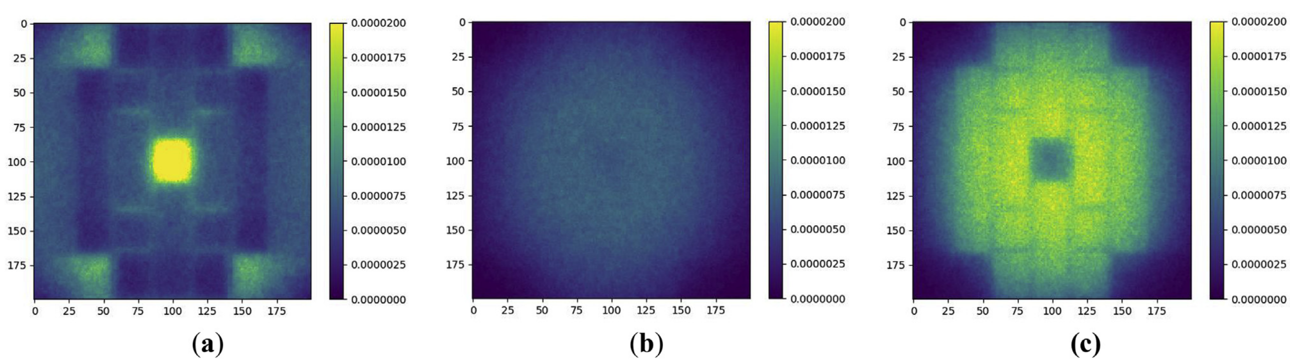


Fig. 4. End of life (EOL) radial flux distribution at the midplane of 20% enriched core of the IAEA generic reactor (a) Thermal flux (b) Epithermal flux (c) Fast flux.

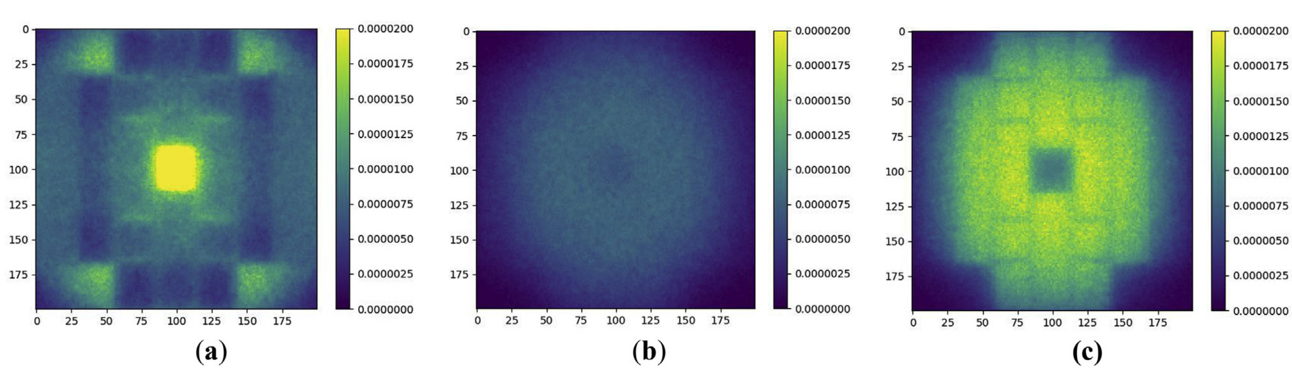


Fig. 5. End of life (EOL) radial flux distribution at the midplane of 93% enriched core of the IAEA generic reactor (a) Thermal flux (b) Epithermal flux (c) Fast flux.

Table 2
Criticality calculations of K_{eff} for the IAEA reactor core with three different enrichments, for fresh fuel and at the beginning (BOL) and end (EOL) of cycle.

Code Platform	93%			45%			20%		
	Fresh	BOL	EOL	Fresh	BOL	EOL	Fresh	BOL	EOL
This work	1.19127 ± .00026	1.06356 ± .00025	1.04560 ± .00024	1.18355 ± .00026	1.06276 ± .00025	1.04661 ± .00025	1.17551 ± .00026	1.06305 ± .00025	1.04802 ± .00024
JAERI [1]	1.18104	1.04199	1.02195	1.18107	1.04893	1.03058	1.18339	1.05782	1.04122
ANL [1]	1.18343	1.02333	1.00038	1.17817	1.02471	1.00331	1.16830	1.02127	1.00142
CEA [1]	1.202	1.04041	1.01703	1.195	1.04077	1.01896	1.187	1.0394	1.01913
OpenMC [3]	1.19382	1.04190	1.02003	–	–	–	1.15494	1.03750	1.01925
COMSOL [2]	1.19097	1.03984	1.02474	–	–	–	1.17241	1.03542	1.02406
MCNP5 [4]	1.18962	1.05768	1.03959	–	–	–	1.17238	1.05617	1.04111
Serpent [5]	1.18410	1.02381	1.00037	–	–	–	1.16636	1.02003	0.99916
CNEA [1]	1.20018	1.03620	1.01278	–	–	–	1.18150	1.03334	1.01348

(neutrons-cm/source). In a collision estimator tally (our case), OpenMC acquire $\frac{1}{\sigma_t}$, which has units of cm, and in the tracklength tally, OpenMC accumulates the tracklength in cm. First, these tally value**s** were divided by the volume of each mash taken into consideration, to make the unit as (neutrons/cm2-source). The normalization factor P*nu/(Q*k) has unit of [J/sec*neutrons/fission/(J/fission*neutrons/source)] = [source/sec], which was then multiplied with the flux to get a flux unit of (neutrons/cm^2-sec) [15]. The fission reaction rate tally acquired from the OpenMC simulation also has a unit of [reactions/source]. The tally values for each assembly was divided by the summation total fission reaction rate of all the assemblies and multiplied with 100 to get the percentage value [15].

4. Results and discussion

In this work, OpenMC three-dimensional full core calculations were performed to calculate reactor physics parameters. The OpenMC simulation runs were executed for 10 million particles in 550 batches with 50 inactive batches and 20000 particles per batch. First reactor criticality K_{eff} was calculated for each of the three cores and compared with published results. Second, the radial flux was calculated utilizing mesh tallies for the neutrons three energy groups. The flux distribution in the central flux-trap facility was tallied and compared for the three different enrichments. Furthermore, the fission reaction rate for each fuel assembly was calculated and the mesh plots were generated.

4.1. Criticality benchmark calculations

Results of the 10 MW MTR benchmark reactor criticality (K_{eff}) calculations at three different enrichments were presented in Table 2, the results are compared with publishing studies and benchmark results performed by a different international organization [1]. The results are in general agreements with other work as shown in Table 2 nonetheless they appear to lie on the higher side of published estimates. The difference in results could be attributed to the use of different nuclear data libraries and also to the difference in computational method. In this study, ENDF/B-VII.1 nuclear cross-sectional data was used for analysis with OpenMC.

The previous study with COMSOLE used WIMDS-D4 code to generate the macroscopic cross-sections using 69 neutron-energy groups. The MCNP5 study used the ENDF/B-VI nuclear cross-sectional data. The Serpent study used the ENDF-VII nuclear cross-sectional data. The study in ANL was performed with DIF2D and EPRI-CELL code to generate the cross sections as a function of burn up. The JAERI used its own code of ADC, and the CEA used the APOLLO code structure with a combination of UKNDL and ENDF/BIV nuclear cross-section data library.

The CNEA used the EXTERMINATORII code and WIMS-D code to generate 69 groups cross-sections library. It should be mentioned that JAERI, ANL, CEA, and CNEA designed a 2D model of the benchmark, whereas recent computational tools like OpenMC, COMSOLE, MCNP5, and Serpent was used to design a 3D model of the said benchmark. Furthermore, it should also be mentioned that the material composition used in this work for different ^{235}U enrichments in different stages of the burnup cycle had 9 isotopes. The material composition data used in the work was documented in Appendix F-6 [1].

The data in Table 2 were taken from the published results in Appendices F-1, 5, 6, and 7 from different organizations on the 10 MW benchmark. For the fresh fuel configuration with different enrichment, the Keff OpenMC results are close to the results of the previous studies as shown in Table 2.

A large difference of 2000 pcm at EOL between our model and

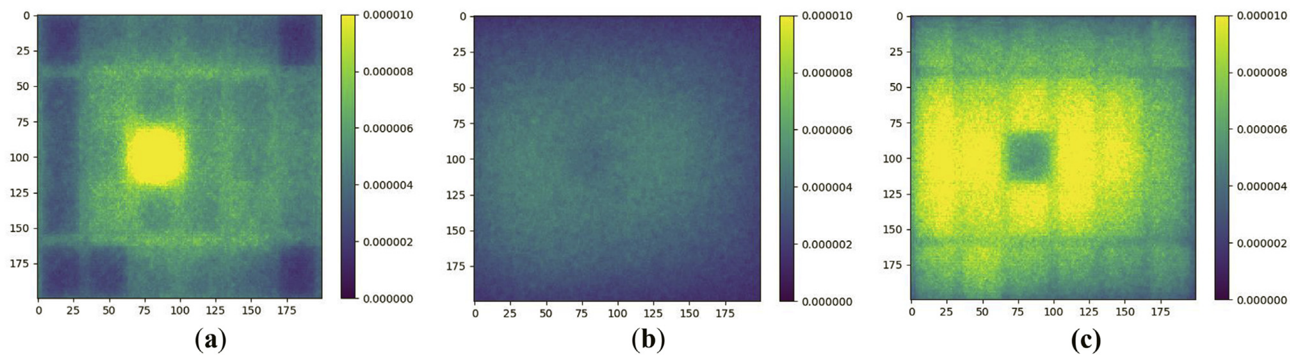


Fig. 6. Radial flux disribution at the ETRR2 midplane of 19.7% enriched core (a) Thermal flux (b) Epithermal flux (c) Fast flux.

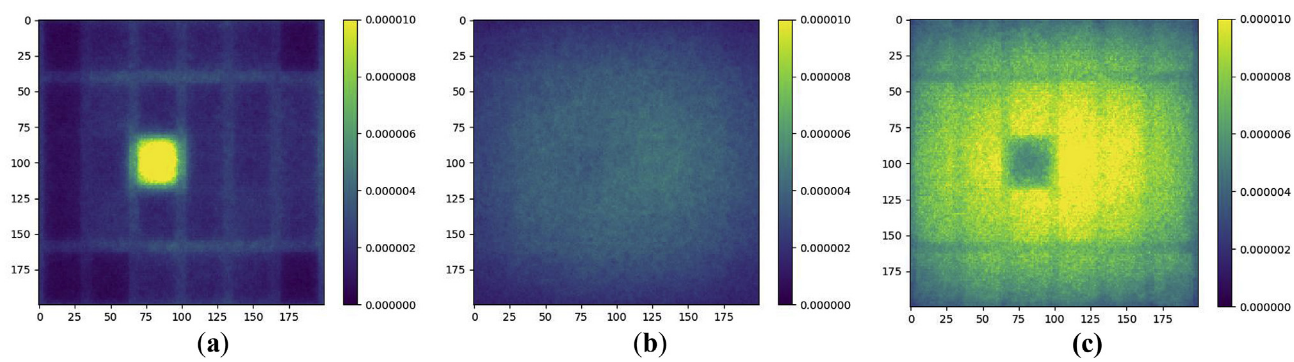


Fig. 7. Radial flux distribution at the ETRR2 midplane of 93% enriched core (a) Thermal flux (b) Epithermal flux (c) Fast flux.

that of other OpenMC study [3] could be attributed to the use of different material and cross-section data. In this study, 9-isotopes material composition for the BOL and EOL by the JAERI was used, where 16-isotopes material composition generated by WIMS was used in the other OpenMC study [3].

Criticality calculations of the ETRR2 core 1/98 benchmark with 19.7% enrichment and all control rods up resulted in K_{eff} of 1.07119 $\pm$ 0.00013. This results in an excess reactivity of $8.86, which is in agreement with the benchmark measured value of $9.1 and calculated value of $8.8 [11].

Criticality calculations results of the MEU and HEU are 1.26662 $\pm$ 0.00013 and 1.39167 $\pm$ 0.00013 respectively, which correspond to excess reactivity of $28.07 and $37.52.

4.2. Radial flux distribution

The result of the neutron flux distribution for the HEU and LEU cores are illustrated in Figs. 2–-7, the radial flux is tallied at the reactor mid-plane utilizing 200 by 200 meshes across the core that are 1 cm thick, resulting in 40000 meshes with the dimension of 0.2695 $\times$ 0.243 $\times$ 1.0 cm for each mesh. The thermal, epithermal and fast flux distribution for the IAEA generic reactor at the beginning of life are shown in Figs. 2 and 3 for the low enriched (LEU) and highly enriched (HEU) cores. Figs. 4 and 5 show the radial flux distribution at the end of life of the IAEA generic reactor. The thermal, epithermal and fast flux distribution for the ETRR2 reactor is shown in Figs. 6 and 7 for the low enriched (LEU) and highly enriched (HEU) cores.

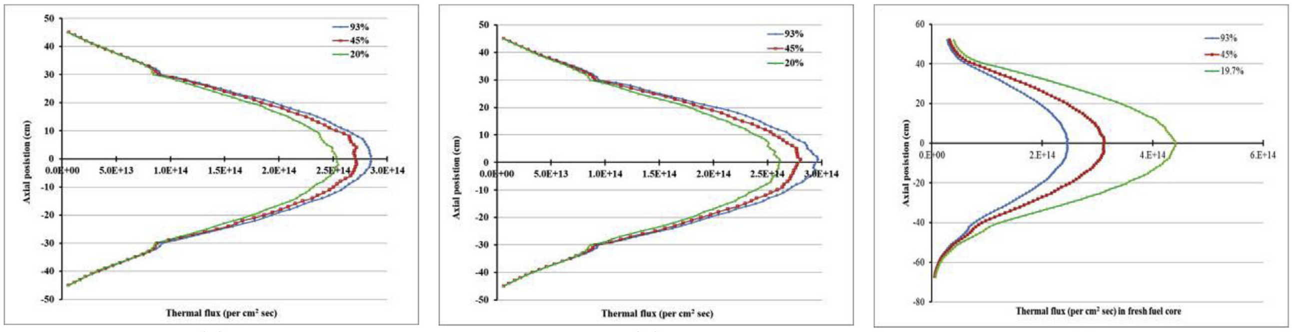
4.3. In-core flux-trap flux distribution

The axial flux in the central experimental facility was tallied at 1 cm intervals, for each of the three core enrichments. The area taken into account was only the dimension that covers the central flux trap for the 10 MW MTR, being from −3.8635 to 3.8635 in the x-axis, and −3.95 to 3.95 in the y-axis, extending along the core centerline from −45 to 45 in the z-axis, hence producing a mesh of 7.727 $\times$ 7.9 $\times$ 1 cm. For the ETRR-2, a dimension that covers the central flux trap, being from −40 to 40 for the z-axis, −7.75 to −0.35 for the x-axis, and −3.7 to 3.7 for the y-axis, producing a mesh of 7.4 $\times$ 7.4 $\times$ 1 cm. Fig. 8 shows the thermal flux distribution for the 10 MW MTR at both the beginning (a) of life and at the end of life (b) for each of the enrichment, and for the ETRR-2 (c); the results show a slight increase in the thermal flux for the 10 MW MTR at the end of life.

Table 3 shows the average thermal flux in the central trap for the 9 cases of the IAEA benchmark obtained in this research in comparison with results reported by previous studies. Average flux values are in agreement with those reported by Chaudri and Mirza [3], and those published by ANL [1].

For the HEU core, the maximum thermal flux increases from 2.85053E+14 n/cm^2 sec at the BOL to 2.96471E+14 n/cm^2 sec at the EOL. For the LEU core, the thermal flux is reduced to 2.53854E+14 n/cm^2 sec at the BOL and to 2.61319E+14 n/cm^2 sec at the EOL. It should be noted that the maximum thermal flux value, obtained in this research is calculated as an average over a mesh of 61 cm^3 for the MTR IAEA reactor and 54.76 cm^3 for the ETRR2 reactor. Consequently producing a lower value than that calculated as a point or over a smaller mesh, and reported by other studies [3,12,16,17].

Results show an increase in the thermal flux for the ETRR-2 due to enrichment conversion from HEU (93%) to LEU (20%) as illustrated in Fig. 8-c. For the HEU core, the maximum thermal flux was 2.4614E+14 n/cm^2 sec, and is increased for the LEU to a maximum



(a) (b) (c)

Fig. 8. The thermal flux in the central experimental facility of HEU, MEU and LEU cores of (a) IAEA generic reactor Beginning of life (b) IAEA generic reactor End of life (c) ETRR2 core 1/98.

Table 3
Average thermal flux (E+14 n/cm2 s) in the central flux trap over 60 cm height.

Code Platform	93%			45%			20%		
	Fresh	BOL	EOL	Fresh	BOL	EOL	Fresh	BOL	EOL
This work	1.86	2.15	2.22	1.80	2.06	2.11	1.71	1.92	1.96
MCNP5 [4]	2.78	2.85	2.78	–	–	–	2.56	2.57	2.58
OpenMC [3]	1.87	2.17	2.23	–	–	–	1.82	1.93	1.97
CNEA (DIF2D [1]	1.98	2.57	2.64	–	–	–	1.72	2.37	2.42
ANL (DIF2D) [1]	–	2.13	2.19	–	2.03	2.09	–	1.90	1.95
EIR (CODIFF) [1]	2.22	2.22	2.29	–	2.13	2.19	2.03	2.03	2.07

value of 4.4102E+14 n/cm^2 sec. The increase of thermal neutron flux is due to spectrum hardening as the amount of ^{238}U is decreased in the HEU which cause the spectrum to shift from lower energies toward higher energies.

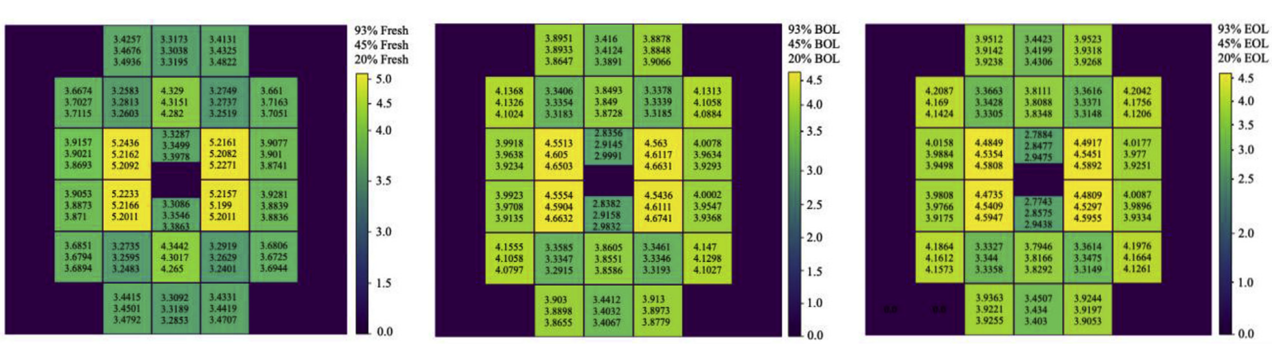
For the 10 MW MTR, results show a decrease in the thermal flux due to enrichment conversion from HEU (93%) to LEU (20%). Results show a maximum reduction of 14.32% in the thermal flux of LEU to that of the HEU, at the end of life. The reduction of the thermal flux for fresh fuel was between 5.81% and 9.62%, with an average drop of 8.1%. At the BOL the thermal flux showed a larger reduction range of 6.92%–13.58% with an average drop of 10.73%.

4.4. Fission rate distributions

The fission reaction rate per assembly was calculated for the 10 MW MTR core three enrichments at BOL, EOL, and fresh fuel as shown in Fig. 9, and for the ETRR-2 as shown in Fig. 10. For the 10 MW MTR, the highest fission rate occurs in the assemblies surrounding the flux trap, in particular, the highest fission density occurs in the half assemblies on the top and bottom of the flux trap, as can be seen in Fig. 9. For the fresh fuel cases the fission rate, the highest fission density increases with the reduction of enrichment as can be seen in the half-assemblies in Fig. 9 (a). The maximum fission rate density per assembly increases from 2.83% to 2.99% in the top half-assembly, at the beginning of life (BOL), as shown in Fig. 9 (b). For the end of life (EOL) case the maximum fission rate density per assembly increases from 2.79% to 2.95% in the top half-assembly, as shown in Fig. 9 (c). For the ETRR-2, the fission rate density per assembly increased between 19.39% and 34.18% at the assemblies surrounding the flux trap when converting from LEU to HEU. The assemblies to the right have the maximum increase of fission rate density per assembly due to the asymmetry of the ETRR-2 core results from the asymmetry of beryllium reflector and the total amount of uranium in each assembly. Current results show that the enrichment reduction from HEU to LEU increases the peak fission reaction rate in the LEU case; such increase in peak power would reduce the reactor safety margin.

5. Conclusions

This work was motivated by the necessity to examine neutron flux variations in the in-core experimental facilities, in light of the international drive to convert research reactors fuel from HEU to LEU. The IAEA generic 10 MW and ETRR-2 benchmark problems were selected for this study, based on well-defined standardized specification and several reactor physics calculations and neutronics analysis, by international organizations. OpenMC Monte Carlo simulation platform was used for the calculations; the code is a powerful simulation platform for analyzing neutronics behavior



(a) (b) (c)

Fig. 9. Assembly base fission reaction rate (%) of the IAEA generic reactor HEU, MEU and LEU at (a) Fresh fuel configuration (b) BOL configuration (c) EOL configuration.

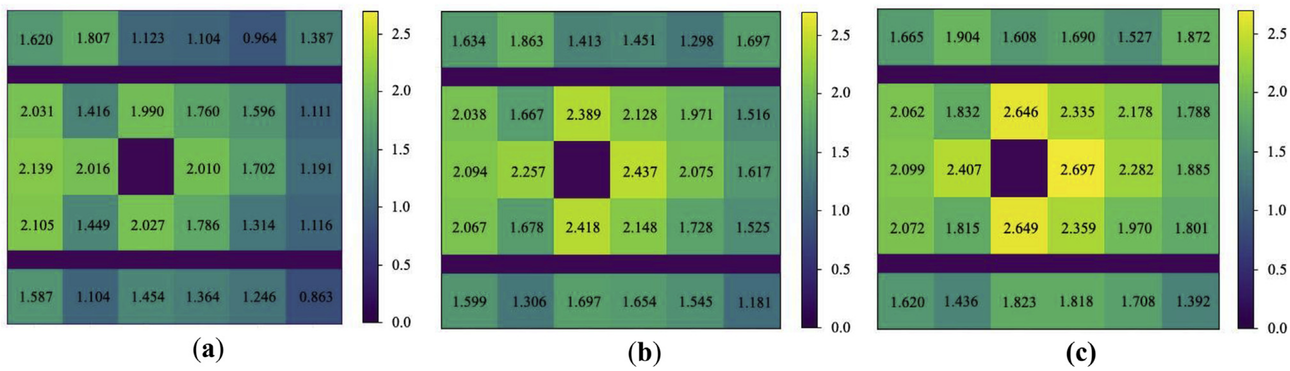


Fig. 10. Assembly base fission reaction rate (%) of the ETRR2 research reactor core 1/98 (a) 19.7% enrichment (b) 45% enrichment (c) 93% enrichment.

of nuclear reactors using continuous energy cross-section data provided by the NNDC.

The four aspects of the study in this work with different enrichments of the fuel, core assembly configuration, and their subparts were first documented in this work and later compared and verified with the previous studies. Findings of this study for the IAEA generic 10 MW MTR show that the impact of enrichment reduction on the neutron flux is minor. In LEU case it was observed that the reactivity drops 1.3% and 0.05% from the HEU reactivity value for fresh and BOL fuel configuration. The results of the multiplication factor calculated in this OpenMC simulation of the IAEA reactor showed reasonable agreement with prior work, but differences with respect to the Chaudri and Mirza [3] OpenMC model should be explored further. For the ETRR-2 reactor, the calculated excess reactivity is in a good agreement with the calculated and measured values of the ETRR-2 benchmark [11,12].

Second, the mesh plot of the flux tally on the x-y plane in the middle of the reactor was generated for three energy group thermal, epithermal and fast. The comparison was made on two fuel configuration BOL and EOL for HEU (93%) and LEU (20%) enrichment.

For the 10 MW MTR in the central flux trap facility, calculations indicated that the maximum reduction in the thermal flux is 14.32% when converting from HEU to LEU, observed at the end of life. The reduction of the thermal flux for fresh fuel was between 5.81% and 9.62%, with an average drop of 8.1%. At the BOL the thermal flux showed a larger reduction range of 6.92%–13.58% with an average drop of 10.73%. With LEU, fission density increases with the reduction of enrichment in the half-assemblies of the core central region. The maximum fission rate density per assembly increases from 2.83% to 2.99% in the top half-assembly, at the beginning of life (BOL). For the end of life (EOL) case the maximum fission rate density per assembly increases from 2.79% to 2.95% in the top half-assembly.

For the ETRR-2 in the central flux trap facility, calculations indicated that the average increase of thermal flux is 62.31% when converting from HEU to LEU due to spectrum hardening. The fission density per assembly is increased between 19.39% and 34.18% in the assemblies surrounding the flux tarp.

Finally, comparing the results for 10 MW MTR and ETRR-2 show that converting the core from HEU to LEU may result in decrease (10 MW MTR case) or increase (ETRR-2 case) in the central flux trap, which indicates the importance of such calculations when considering the conversion from HEU to LEU in research reactor. Provided that, for both benchmarks, OpenMC is proved to be a reliable platform for neutronic calculations in MTR research reactors.

Appendix A. Supplementary data

Supplementary data to this article can be found online at https://doi.org/10.1016/j.net.2019.08.008.

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