Atom Indonesia

Journal homepage: http://atomindonesia.brin.go.id

A Neutronic Study of A Low-Enriched Uranium-Fueled Microreactor Cooled with A Sodium Heat Pipe System Using The OpenMC Code

A. Bourenane[1,2,*], L. Hamidatou[3], M. Dougdag[2], M. L. Yahiaoui[2]

[1]*Materials and Energies Research Laboratory (MERL), Faculty of Science and Technology, University Amine Elokkal El Hadj Moussa Eg Akhamouk, 11000, Tamanrasset, Algeria*
[2]*Nuclear Research Center of Birine, BP 180 AinOussera 17200, Djelfa, Algeria*
[3]*Nuclear Research Center of Draria, B.P 43 Sebbala, Algiers, Algeria*

ARTICLE INFO

Article history:
Received 13 April 2023
Received in revised form 9 May 2024
Accepted 10 May 2024

Keywords:

OpenMC code
Neutronics
Microreactor
Power distribution

ABSTRACT

The development of Small Modular Reactors (SMRs) represents a pivotal shift in nuclear technology, emphasizing enhanced safety, efficiency, and adaptability. This study examines Toshiba's MoveluX, an innovative micro-reactor, exemplifying advancements in reactor miniaturization suitable for limited spatial environments and hybridization with other energy sources. In this paper, the performance and safety of the MoveluX are rigorously evaluated using the OpenMC code, with an emphasis on critical parameters such as the effective multiplication coefficient and the reactivity worth of control devices. A 3D model of the given microreactor was built based on Toshiba's designs and features a solid core and a heat pipe cooling system. Preliminary results affirm the model's accuracy, and analysis of the neutron spectrum and flux indicates significant fission occurring in the U-238 isotope. Furthermore, the investigation extends to the thermal aspect within the fuel elements, uncovering a significant power density at the interfaces between fuel and moderator. Overall, this research makes a substantial contribution to the field of microreactor design and optimization.

INTRODUCTION

Micro-scale nuclear reactors known as microreactors are defined by their electricity production capacity, typically up to tens of megawatts (MWth) [1]. These reactors are engineered for compactness, modularity, and portability, contrasting sharply with conventional large-scale nuclear reactors. Their deployment is particularly advantageous in geographically isolated or inaccessible areas, including military facilities, mining operations, and remote settlements. Additionally, they serve a critical role in providing backup power during infrastructural or natural crises. Integral to these reactors is their passive safety features, which significantly bolster their security and mitigate risks.

Despite their myriad benefits, concerns regarding their safety and maintenance owing to their compact and modular design persist. Nevertheless, microreactors stand at the forefront of nuclear technology advancements, with growing interest and are expected to attract more investments [2].

A pivotal aspect of their operation is the understanding of neutronics: the behavior of neutrons encompassing their production, transportation, and interactions within the reactor. This knowledge is vital for the design and optimization of reactor cores and fuel assemblies, predicting performance under various operational scenarios (startup, normal, accidental), and evaluating safety and environmental impacts. Neutronics analyses help to identify potential design issues [3-5], such as fuel burnup and power distribution, and to ensure a safe operation, particularly in abnormal situations.

Recent advancements have shifted focus to nuclear microreactors, propelled by their lower investment costs, enhanced safety features, and the inherent safety afforded by passive regulation systems. Sodium, among other metals, is utilized either as a circulating fluid in a closed circuit or

[*]Corresponding author.
E-mail address: aissa.bourenane@crnb.dz
DOI: https://doi.org/10.55981/aij.2024.1318

within a heat pipe as a two-phase liquid/gas exchanger. The application of two-phase cooling, originally studied for passive cooling in larger reactors under exceptional conditions (water/vapor [6]), is now being adapted for use in heat pipes as the main cooling system in microreactors. Various designs of low-enriched uranium microreactors have been proposed, employing heat pipes for passive cooling. Examples include a design utilizing 19.75 % enriched uranium [7] and another incorporating consumable poisons for excess reactivity control during the reactor operation period [8]. This work specifically focuses on a solid-core nuclear microreactor utilizing low-enriched uranium.

OpenMC is a Monte Carlo code initially developed at the Massachusetts Institute of Technology (MIT) since 2013. It uses continuous energy cross-section data libraries [9].

Supported by open-source community developers, OpenMC is increasingly used in nuclear reactor analysis and neutron/photon transport simulations. Getting more popular, OpenMC could be considered a flexible open-source code since it is written in C++ and Python. Additionally, it can accurately model complex geometries and material configurations and can be customized to meet specific user needs.

Designed and built within the framework of the Large Scale Calculation project, OpenMC supports parallelized calculations, allowing for fast calculations on workstations or large clusters of computers. OpenMC is user-friendly, with an intuitive input deck format using a python interface and a visualization tool for viewing model geometry and simulation results [10,11].

The primary aim of this study is to conduct a detailed neutronic analysis of the nuclear microreactor MoveluX proposed by Toshiba and planned to be constructed in 2030’s, at the beginning of the cycle, utilizing the OpenMC open-source Monte Carlo code.

The MoveluX reactor is the result of a development process that began with a 325 kWth reactor moderated by CaH_2, cooled by heat pipes functioning with sodium and intended for space use [12,13]. Subsequently, for terrestrial use, its power was increased to the order of 10 MWth, and its structure was altered until it reached its current conceptual design [14].

This reactor was initially modeled using a different Monte Carlo code by Toshiba, its designer. Toshiba's results were used to validate our model. Post-validation, key parameters such as the two-dimensional flux distribution within the reactor core's horizontal section, axial flux distribution, heat source distribution in fuel elements, neutron spectrum, and uranium isotope heat production distribution were calculated.

DESCRIPTION OF THE MICROREACTOR MODEL

The MoveluX reactor, a recent development by Toshiba, marks a notable advancement in the field of compact nuclear reactor technology [1]. It is intended for multipurpose use: production of electricity, production of hydrogen, and urban heating. The fuel is made from Low-Enriched-Uranium (LEU) (e < 5 %) in ceramic form U_3Si_2 and ceramic CaH_2 as moderator. Figure 1 presents a schematic of the reactor, highlighting the integration of its various components.

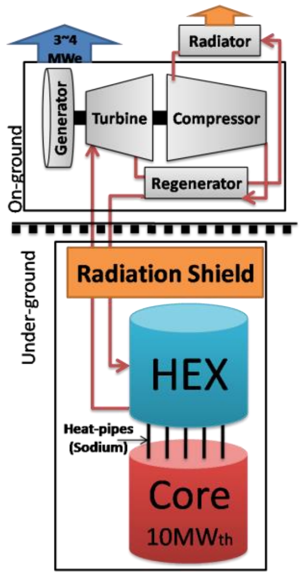

Fig. 1. Schematic view of the MoveluX reactor system.

A defining aspect of the MoveluX reactor is its passive cooling system, which integrates sodium heat pipes within the primary circuit. The heat pipes are cooled by a secondary nitrogen gas cycle, which functions according to a Brayton cycle. This cycle facilitates electricity generation through a miniaturized gas turbine situated above ground, while, the reactor core and the primary cooling cycle are located underground. This design feature not only bolsters safety measures but also elevates operational efficiency. The reactor's efficiency is underscored by its thermal/electrical capacity of 10 MWth / 3-4 MWe and an operational lifespan extending 10-20 years without refueling needs, as elaborated in Table 1 [15]. This table offers a comprehensive summary of the reactor's core specifications, encompassing dimensions, materials, and operational parameters.

The reactor's core consists of 66 fuel elements and 72 moderator elements, each meticulously crafted into hexagonal prisms with apothem of 5 cm for fuel elements and 4.3 cm for moderator elements. This geometric configuration maximizes core efficiency and compactness.

The core, with a diameter of 180 cm and a height of 220 cm, operates at a temperature of 960 K, as delineated in Table 1.

Table 1. Specifications of the MoveluX Core.

Parameters	Value
Thermal/electrical capacity,(MWth/MWe)	10/3-4
Lifetime (year)	10-20
Fuel material	U_3Si_2
Moderator	CaH_2
Neutron breeder material	BeO
Reflector material	Pb0.6-Sn0.4
Number of fuel elements	66
Number of moderator elements	72
Number of neutron breeder elements	228
Core temperature (K)	960
Core diameter/height (cm)	180/220
Fuel element width(cm)	10
Moderator element width(cm)	8.6
Clad thickness (cm)	0.1
Control device thickness (cm)	0.22
U-235 enrichment (wt%)	< 5.0

Encased in stainless steel cladding with a thickness of 1 mm, chosen for its durability, resistance to high temperatures, and corrosion, the fuel elements maintain a balance between protective integrity and heat transfer effectiveness.

Enhancements to the cooling system include the strategic placement of sodium heat pipes for optimal heat extraction.

The whole core is surrounded by a PbSn alloy neutron reflector, composed of 60 % lead, thereby augmenting overall efficiency.

For reactivity control, a metallic expansion module is used. It consists of a 1 % InGd alloy in a slender prism interposed among the fuel elements, 2.2 mm thick, facilitating fine-tuned reactivity control as detailed in Table 1. This design by Toshiba seamlessly integrates compactness, operational efficiency, and safety establishing the MoveluX as an innovative solution in nuclear reactor technology.

SIMULATION METHODOLOGY

Based on the available data on the MoveluX microreactor and to make the calculations by our means we have assumed the following: The reactor core has an angular degree of symmetry, so we only took a 30° part, and we used reflective radial boundaries to save computation time. However, axially, we took the complete reactor core. Figure 2 shows the plot of different sections of the OpenMC model. The reactor core is considered with a temperature fixed at 1000 K over the entire considered volume. The neutron-cross-section library is taken from the Evaluated Nuclear Data Libraries Archive [16]. To be more realistic, the densities of ceramics are taken to be 92-98 %, which is less than theoretical densities.

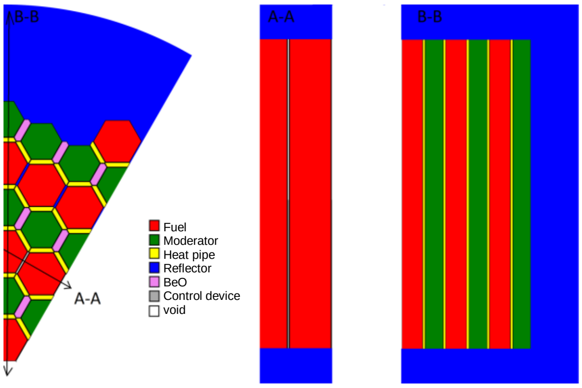

Fig. 2. Plot of the model (left) horizontal slice of the part of the core reactor; (middle) vertical cut shows two fuel elements and control device between them; (right) vertical cut shows three fuel elements, three moderator elements, and five heat pipes of the core.

In the simulation of the MoveluX core, the cross-sectional library was derived from the JEFF-3.2 dataset, which was specifically processed for a thermal condition of 1000 K, representing the closest available temperature data to the operational temperature of the MoveluX reactor core, which is 960K. This dataset was sourced from the Nuclear Energy Agency's (NEA) website [16]. To attain a variance reduction below 10 pcm, each simulation series comprised 1100 cycles excluding an initial 100 cycles allocated for source calibration. Each cycle included the computation of 100000 neutron histories.

To enhance computational efficiency, we selected only one-twelfth of the core's angular section and implemented mirror boundary conditions along the truncated edges. Additionally, vacuum boundary conditions were applied to the external surfaces.

The computational resources utilized were as follows: Processor An Intel® Core™ i7-10700 CPU @ 2.90GHz, featuring 16 cores, all of which were employed in parallel to facilitate the completion of each 1100-cycle run in approximately one hour for K_{eff} calculation. In the case of mesh tally, it takes approximately 2 hours. RAM 16 GB, which sufficiently met the demands of the simulation, Operating System: Ubuntu 20.04.6 LTS. Simulation Software: OpenMC, version 0.14.1.

RESULTS AND DISCUSSION

We will begin the presentation of the results with the data used for model validation. The excess reactivity required for this core to operate for 20 years is 1074 pcm [17]. The result obtained by Kimura [17] in the reference model for the excess reactivity is 1140 pcm, while our model yielded 1206 pcm. Subsequently, the control device was introduced, and criticality was calculated at various heights of the control device. This analysis produced the standard curve (S-shaped) of the control device shown in Fig. 3. However, the total reactivity worth of the control device in our model was 2571 pcm, compared to 2153 pcm in the reference model, resulting in a discrepancy of approximately 14 %. Additionally, the worth of the control device at 60 % insertion was 1280 pcm for the reference model and 1420 pcm for our model. The results are summarized in Table 2.

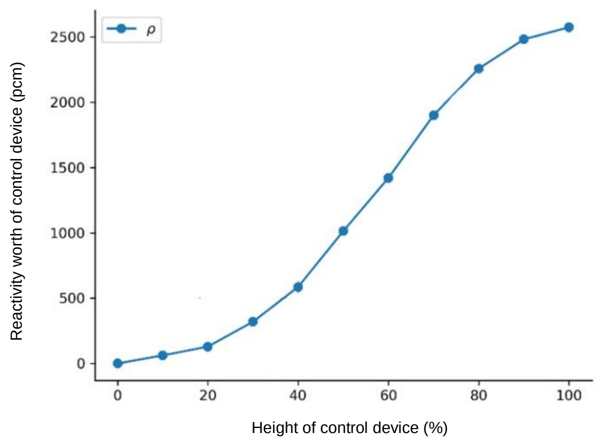

Fig. 3. Reactivity worth of control device.

Table 2. Comparison of excess reactivity results and control device reactivity worth between models.

Parameters	Our Model	Reference Model	Relative Error
Excess reactivity	1206 ± 9	1140 ± 9	5 %
ρ (60 %)	1420 ± 9	1280 ± 9	10 %
ρ (100 %)	2571 ± 9	2153 ± 9	19 %

The difference between the results is likely due to the disparities between the models, especially concerning the control device where the geometry is not very clear. Moreover, we used the generic material density of InGd, whereas the reference model may have utilized different values that could affect the outcomes. Additionally, the data on effective cross-sections may have an impact. Nonetheless, the results are close, and we consider that our model is capable of conducting more in-depth neutron studies.

We calculated the contribution of each fuel element to the total energy production, and the results are presented in Fig. 4. We found that the fuel element surrounded by more neighboring fuel elements produces more power, while those at the periphery are less productive. This is evident, but the difference varies from 0.7 % to 2 %. Interestingly, the most productive fuel element is not the central one. Because it is surrounded by the moderator in only one direction, it is less susceptible to thermal neutrons, which are the primary cause of fission and, consequently, energy production.

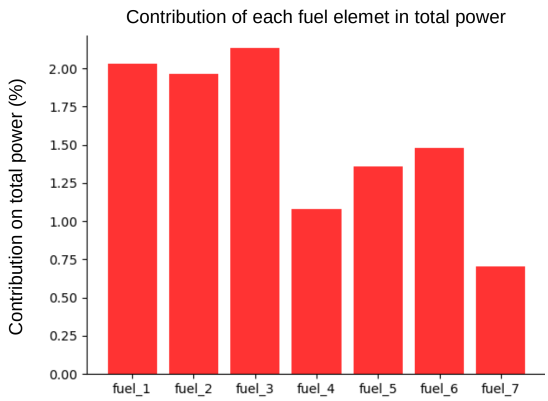

Fig. 4. Contribution of each fuel element in the total power of the reactor core.

To better understand what occurs inside, we calculated the spatial distribution within the core. We took a horizontal slice with a thickness of 60 cm at the core's midpoint and performed a tally mesh with a step of 0.1x0.1 mm² in two directions to calculate the heat generation distribution. The results are presented in Fig. 5.

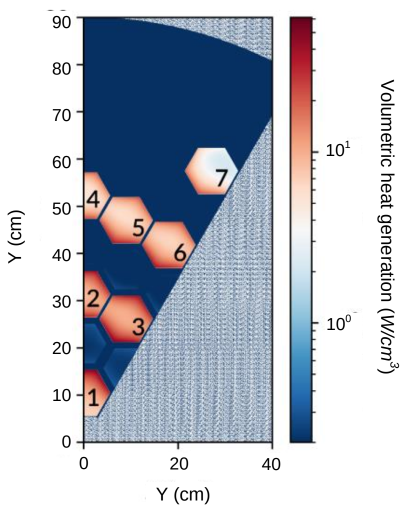

Fig. 5. Volumetric heat generation in central slice.

We observed that the heat generated is concentrated at the periphery of the fuel element facing directly towards the moderator, contrary to the standard fuel elements of thermal reactors where the concentration of power is in the middle. Additionally, we noted that even the moderators, especially those in the middle, generate heat at the ends facing the fuel. This is likely due to moderation, which reduces the neutron energy from a few MeV to a few eV or less. This explains why the designer placed the heat pipes between the moderator and the fuel rather than between the fuels.

In Fig. 6, we plotted the radial distribution of heat generation across all fuel elements. The axial heat generation distribution was calculated using a 10 cm mesh along each fuel element. Figure 7 shows a typical curvature with the maximum at the center. For both Figs. 6 and 7, the peak power appears very high. Therefore, we calculated the radial Power Peaking Factor (PPF) and axial PPF and presented them in Fig. 8.

The values of the radial PPF are very high. If the peak had not occurred at the periphery, it would have been unacceptable, as the cooling system removes heat directly from the peak location. However, a challenge for cooling remains because even the axial PPF is very high. The heat pipe, with its metal fluid (sodium), is highly conductive and may be capable of rapidly homogenizing the temperature along the fuel element.

From Fig. 5, it was observed that heat generation decreases as we move towards the center. This is due to the mean free path of thermal neutrons in the fuel, which is on the order of millimeters. With a fuel element apothem of 5 cm, only fast and epithermal neutrons can penetrate the middle of the fuel element. Therefore, we calculated the neutron spectrum for the central slice within the fuel, close to the center, and for an adjacent moderator element. The spectra are illustrated in Fig. 9.

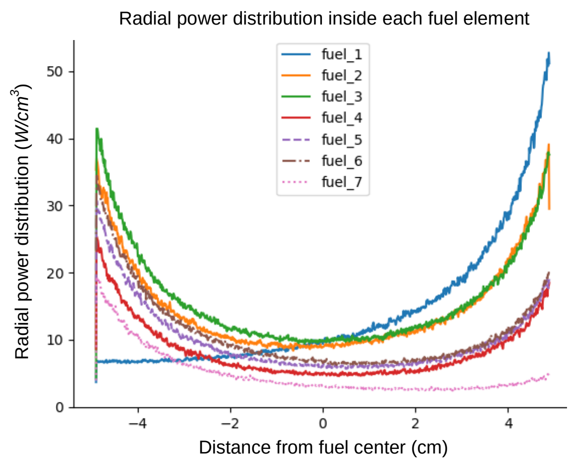

Fig. 6. Radial distribution of heat generation across all fuel elements.

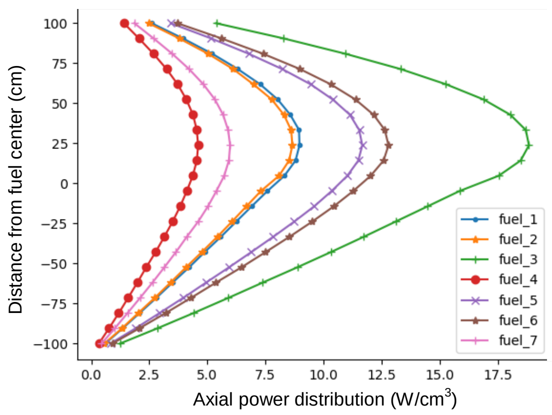

Fig. 7. Axial distribution of heat generation across all fuel elements.

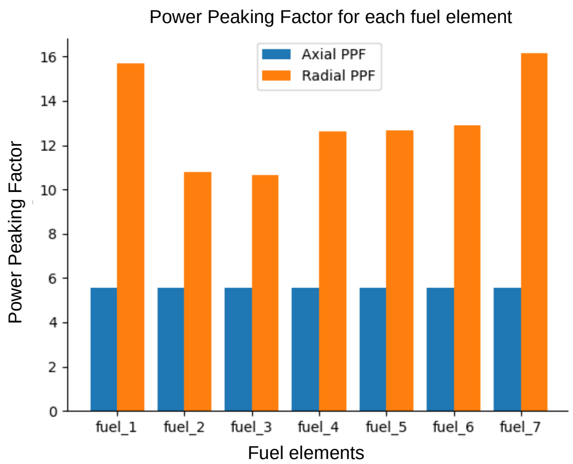

Fig. 8. Radial Power Peaking Factor (PPF) and axial PPF in different fuel elements.

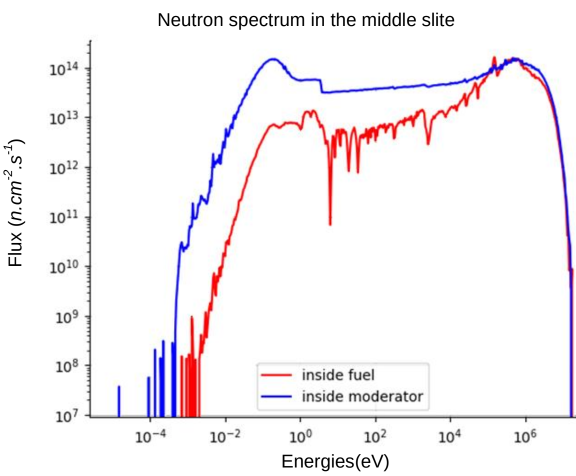

Fig. 9. Neutron spectrum inside central fuel element and adjacent moderator element.

It was noted that in the moderator, the spectrum is a mixture of neutron groups: thermal, epithermal, and fast. However, within the fuel element, the thermal component almost disappears. This leads us to believe that the heat generated at the center of the fuel is due to fission caused by fast and epithermal neutrons. Therefore, we conducted a flux distribution calculation for the three neutron groups:

$$E_{thermal} < 0.625 \text{ eV}$$
$$0.625 \text{ eV} < E_{epithermal} < 100 \text{ keV}$$
$$100 \text{ keV} < E_{fast} < 20 \text{ MeV}$$

The flux results are presentevd in Fig. 10.

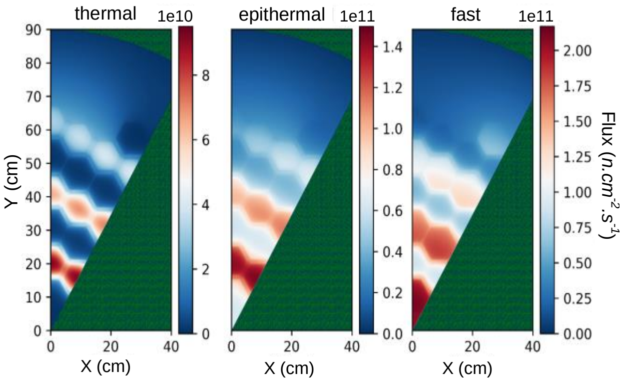

Fig. 10. Intensity of neutron flux by group flux in the core: (left) thermal neutrons; (middle) epithermal neutrons; (right) fast neutrons.

Furthermore, the fuel enrichment is 5 %, meaning the fuel contains a significant amount of the U-238 isotope, which is the target of fast fission. The OpenMC code provides powerful features for tallies, among which one can select specific interactions for study. Additionally, OpenMC allows for the selection of a specific isotope for statistical analysis. Therefore, we conducted another calculation for the fission rate distribution for the three uranium isotopes: U-235, U-238, and its daughter U-234. Results are presented in Fig. 11.

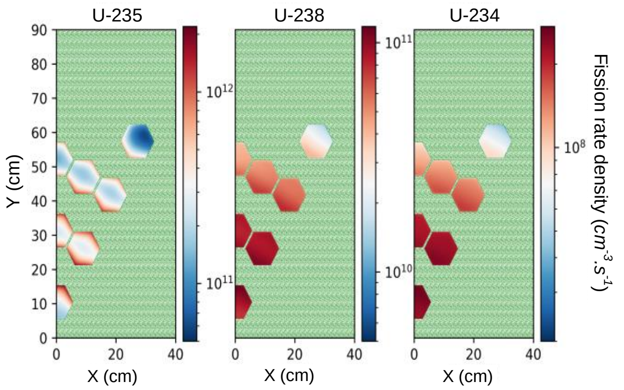

Fig. 11. Fission rate density: (left) fissions from U-235; (middle) fissions from U-238; (right) fissions from U-234.

It was observed that the values of intensity peaks for all three groups are close to each other. However, only fast neutrons exist deep inside the fuel element. The maximum thermal and epithermal fluxes are located at the moderator level. The fast flux increases as we move toward the core center and decreases as we move out of the fuel elements.

According to Fig. 11, it is clear that the fission rate of U-238 covers the entire fuel section with a very significant value, comparable to the contribution of the U-235 fission rate. This is an advantage that gives this reactor the property of a fast breeder, which uses fast neutrons to achieve high rates of conversion of fertile material into fissile material. Overall, the number of fissions of each isotope is summarized in Table 3.

Table 3. Fission rate fractions of uranium isotopes.

Isotopes	Fission rate fraction
U-235	85.74 %
U-238	14.22 %
U-234	0.04 %

This study is constrained by initial assumptions. A more precise investigation could be conducted using these results to perform a thermodynamic study for accurately determining the temperature distribution within the reactor core. Further studies could segment the fuel elements based on the intensity of each neutron group to calculate the burnup in each fuel region. Concurrently, the impact of neutron flux on other materials, such as the moderator, control device, sodium, and reflector, could be examined. However, this would necessitate extensive cross-sectional data for different temperatures.

CONCLUSION

This paper presents a comprehensive neutronic analysis of a LEU-fueled micro-reactor employing a sodium heat pipe cooling system, conducted using the OpenMC simulation code. This study focuses on evaluating the microreactor's performance under beginning-of-cycle conditions, with specific attention to the reactivity worth of the control device, the neutron spectrum, and power distribution.

Key findings reveal that the majority of the power is concentrated on the fuel surface interfacing with the moderator. This observation is crucial for the design and optimization of microreactors, particularly in the context of thermal-hydraulic and structural considerations. Furthermore, results indicate a mixed neutron spectrum comprising fast, epithermal, and thermal neutrons, with a significant fraction of fission occurring in the U-238 isotope.

Future research directions include conducting a more detailed thermodynamic analysis to precisely determine the temperature distribution within the reactor core. Additionally, segmenting the fuel elements based on neutron group intensity could provide further insights into reactor behavior and efficiency. Overall, this study contributes valuable insights into the design and operational aspects of microreactors, highlighting key areas for future research and optimization.

ACKNOWLEDGMENT

This paper is part of thesis work and was conducted at Materials and Energies Research Laboratory (MERL), Faculty of Science and Technology, University of Tamanrasset in partnership with Nuclear Research Center of Birine.

AUTHOR CONTRIBUTION

Aissa Bourenane: Conceptualization, Methodology, Software, Visualization, Writing Original draft preparation. Lylia Hamidatou: Supervision, Project administration. Mourad Dougdag: Supervision, Project administration, Writing- Review & Editing. Mohamed Laid Yahiaoui: Software, Validation, Data Curation.

REFERENCES

1. L. Kaustubh, L. Kasturi, Chem. Eng. Prog. **4** (2022) 44.
2. G. Black, D. Shropshire, K. Araújo *et al*, Nucl. Technol. **209** (2023) S1.
3. D. Pérez, D. Milian, L. Hernández *et al*, Atom Indones. **48** (2022) 75.
4. Nuryana, R. S. N. Mahmudah and A. Khakim, Atom Indones. **49** (2023) 77.
5. R. A. P. Dwijayanto and A. W. Harto, Atom Indones. 50 (2024) 9.
6. X. Yin, X. Wu, S. Zhou *et al.*, Ann. Nucl. Energy **180** (2023) 109506.
7. X. Chai, C. Guan, J. Deng, et al., Ann. Nucl. Energy **179** (2022) 109399.
8. J. Li, J. Cai, X. Li, Ann. Nucl. Energy **181** (2023) 109576.
9. P. K. Romano, N. E. Horelik, B. R. Herman *et al.*, Ann. Nucl. Energy **82** (2015) 90.
10. A. Islam, T.A. Rahim, A.S. Mollah, Atom Indones. **48** (2022) 193.
11. M. H. Kusuma, N. Putra, A. Rosidi *et al.*, Atom Indones. **45** (2019) 173.
12. S. Wada, R. Kimura, T. Yoshida *et al*., Trans. Am. Nucl. Soc. **115** (2016) 1145.
13. R. Kimura, T. Yoshida, S. Wada *et al.*, Trans. Am. Nucl. Soc. **115** (2016) 1141.
14. R. Kimura, K. Asano and T. Suzuki, *Atomic Energy Society of Japan 2019*, Annual Meeting (2019) 20.
15. Anonymous, Status Report - MoveluX (Toshiba Energy Systems & Solutions) Japan. https://aris.iaea.org/PDF/MoveluX_2020.pdf. Retrieved in April (2023).
16. Anonymous, JEFF-3.2 Evaluated Data Library - Neutron Data. https://www.oecd-nea.org/ dbforms/ data/eva/evatapes/jeff_32/Processed/ JEFF32-ACE-1000K.tar.gz, Retrieved in April (2023).
17. R. Kimura, S. Kanamura, Y. Takahashi *et al*., Nucl. Technol. **207** (2021) 1784