Investigation of Core Neutronics in a Multi-Purpose Research Reactor Utilizing VVR-KN Fuel

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Abstract: - Bangladesh aims to enhance its nuclear research capabilities through a proposed 10–20 MW multi-purpose research reactor (MPRR) utilizing VVR-KN low-enriched uranium (LEU) fuel, addressing the limitations of its existing 3 MW TRIGA Mark-II reactor, particularly fuel supply constraints. Despite the suitability of VVR-KN fuel for high neutron flux applications, comprehensive studies on its long-term neutronics behavior, including fuel burnup and neutron poison accumulation, remain limited, necessitating detailed analysis for reactor safety and efficiency. This research examines neutronics parameters for a 10 MW MPRR design using OpenMC with critical neutron flux analysis. The objectives include evaluating criticality, neutron flux, power distribution, and fuel depletion. Simulations reveal an effective multiplication factor (keff) of 1.10693 at the beginning of cycle (BOC), closely aligning with reference values, and an average neutron flux of 5.8×1014 neutrons/cm2·s, suitable for isotope production and neutron scattering. Power distribution analysis indicates an average power of 0.283 MW per fuel assembly, with a peaking factor of 1.197, reflecting balanced performance. Over a 90-day cycle, keff reduces to 1.003 due to U-235 loss and neutron absorber buildup (Xe-135, reaches equilibrium after 40-50 hours and, Sm-149, reaches equilibrium after 20 days), with U-238 converting to Pu-239 to sustain reactivity. Reactivity feedback analysis showed a negative fuel temperature coefficient (−2.82 pcm/K) and increasingly negative coolant temperature coefficients, confirming strong inherent safety. These findings validate the VVR-KN fuel's performance for MPRR applications, providing a robust foundation for safe and efficient reactor design in Bangladesh, with future research recommended for transient reactivity and multi-code validation.

Key-words: - VVR-KN fuel, MPRR, Neutronics analysis, OpenMC simulation, Low-enriched uranium

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1 Introduction

Research reactors are pivotal for advancing scientific research, education, and radioisotope production, supporting applications in materials science, medicine, and nuclear physics [1]. Unlike commercial power reactors, research reactors operate at comparatively low power levels, typically below 20 MW, and are primarily designed to provide intense neutron fluxes for experimental and applied research. Their applications include neutron scattering, neutron activation analysis, and the production of radioisotopes used in medical diagnostics and cancer treatment [1]. In Bangladesh, the 3 MW TRIGA Mark-II research reactor has been in operation since 1986 [2] and has supported a wide range of activities, including neutron activation analysis, neutron radiography, radioisotope production such as iodine-131 and technetium-99m, as well as the training of nuclear engineers and reactor operators [3], [4]. However, the TRIGA reactor faces sustainability challenges due to limited availability of its uranium-zirconium hydride fuel, despite resumed production by TRIGA International. To address these limitations and to strengthen its national nuclear research capabilities, Bangladesh has begun considering the development of a higher-power, multi-purpose research reactor in the range of 10–20 MW. Proposed designs favor the use of VVR-KN low-enriched uranium fuel, which is consistent with current international non-proliferation requirements [5], [7].

The VVR-KN fuel, developed in Russia, employs a hexagonal plate-type geometry with a uranium-235 enrichment of 19.75 percent. This design provides a high neutron flux while offering improved safety margins compared to earlier highly enriched uranium fuels, such as VVR-S [5], [6]. Operational experience and conversion studies, particularly for reactors such as Kazakhstan's VVR-K, which was successfully converted from HEU to LEU fuel, have demonstrated favorable neutron-physical performance and acceptable safety characteristics [6[[7]. More recent studies, such as Nguyen et al. (2020), have proposed a 10 MW MPRR design using VVR-KN fuel, validating its suitability through neutronics and thermal-hydraulic analyses. However, comprehensive analyses of long-term neutronics behavior, including fuel burnup, neutron poison accumulation (e.g., Xe-135, Sm-149), and actinide production (e.g., Pu-239), remain limited due to the fuel's recent development [5], [7]. The complex hexagonal geometry of VVR-KN fuel also poses challenges for accurate neutron transport modelling, necessitating high-fidelity simulations and multi-code validation [8].

Bangladesh's developing nuclear program, which is focused on the VVER-1200 reactors at RNPP, calls for indigenous expertise in isotope manufacturing, materials testing, and nuclear science. For teaching, safety research, and experimental validation of nuclear data, a domestic MPRR can offer a controlled setting. In keeping with the pattern of Russian-designed research reactors, the use of VVR-KN fuel in hexagonal geometry identical to the RNPP Russian-designed VVER-1200 hexagonal geometry [9], [10] guarantees compatibility with local nuclear infrastructure.

This study addresses these gaps by investigating the neutronics parameters of a conceptual design for a 10 MW MPRR using OpenMC with the nuclear data library ENDF/B-VIII.0. The research evaluates criticality, neutron flux, power distribution, temperature feedback coefficients and depletion characteristics to ensure reactor safety and efficiency, supporting Bangladesh's ambition to develop a robust nuclear research infrastructure compliant with international standards.

The novelty of this study lies in its comprehensive and quantitative evaluation of core neutronics for a VVR-KN fuelled Multi-Purpose Research Reactor (MPRR) specifically adapted to the Bangladeshi context. It establishes a neutronic correlation framework linking the MPRR with the RNPP VVER-1200 core, enabling cross-validation of key reactor parameters. The study also provides detailed burnup-dependent spectral data and reactivity coefficients for VVR-KN fuel assemblies, which are rarely reported in existing literature. Moreover, it demonstrates how research reactor operations can directly contribute to data verification, operator training, and the overall advancement of commercial VVER system applications in Bangladesh.

2 Materials and Methods

This section outlines the computational tools, reactor design specifications, and simulation methodologies used to investigate the neutronics parameters of a 10 MW MPRR with VVR-KN fuel that is a low-enriched uranium fuel. The study employs the OpenMC Monte Carlo simulation that applies on vast quantities of random data and statistical analysis [11], [12] to model neutron transport and depletion, ensuring high-fidelity analysis of criticality, neutron flux, power distribution, and fuel burnup characteristics.

2.1 Computational Tools

Neutronics simulations were performed using the OpenMC Monte Carlo code, developed by MIT's Computational Reactor Physics Group [13]. OpenMC uses a stochastic approach to track neutron interactions probabilistically, supporting continuous-energy transport and depletion calculations for both instantaneous reactor behavior and fuel burnup. It was chosen for its ability to accurately model complex geometries, such as the hexagonal VVR-KN fuel assemblies, without excessive simplifications.

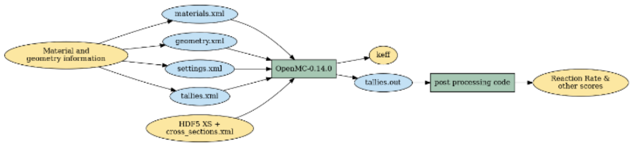

Fig. 2: 1: Working Procedure of OpenMC

The simulations employed the ENDF/B-VIII.0 nuclear data library, providing comprehensive neutron, photoatomic, and thermal scattering datasets across various temperatures, processed using NJOY 2016.68 for accurate cross-section treatment [14]. The JEFF-3.3 library supplemented photoatomic data to ensure robust evaluation of photon interactions. For burnup analysis, a PWR depletion chain in XML format, including decay data, fission product yields, and isotopic evolution information, was used from OpenMC's official repository, ensuring compatibility with the simulation framework.

By integrating OpenMC's high-fidelity Monte Carlo capabilities with advanced nuclear data libraries, the study performed detailed neutronic and burnup analyses of the VVR-KN fuel assemblies, providing reliable predictions of reactor behavior under operational conditions.

2.2 Reactor Core and Fuel Design

The MPRR has an open pool-type design with a thermal power of 10 MW, where light water (H2O) is used as coolant as well as moderator. Beryllium rods serve as a neutron reflector [4]. The compact core is comprised of 61 cells with 36 hexagonal fuel assemblies that contain 27 fuel assemblies labelled as FA-1 with eight fuel elements and nine control assemblies labelled as FA-2 with five fuel elements and a guide tube for control rods insertion in each FA-2. The VVR-KN fuel that has an enrichment of 19.75% U-235, consists of UO2 dispersed in an aluminum matrix with a uranium density of 2.8 g/cm³, clad in SAV-1 aluminum alloy [4], [6]. Each fuel element has a length of 600 ± 2 mm and a diameter of 0.7 mm with a total of six ribs on hexagonal and three ribs on cylindrical fuel elements and a clad thickness of 0.45mm [15]. The core dimensions are 180 cm in height and 113.5 cm in radius, with nine control rods, six B4C shim rods, two B4C safety rods, and one stainless steel regulating rod.

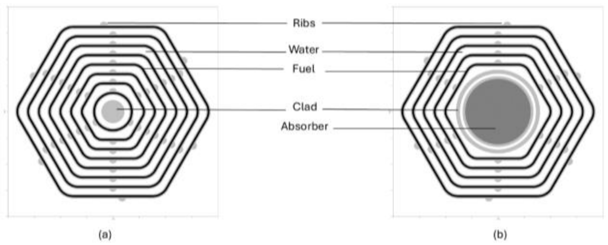

Fig. 2: 2: Design of FA-1 (a) and FA-2 (b)

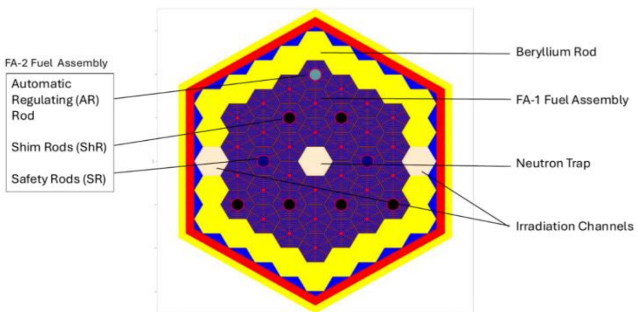

Fig. 2: 3: Configuration of the designed MPRR core

2.3 Neutronics Simulation Methodology

Neutronics simulations were conducted using OpenMC to evaluate key parameters: effective multiplication factor (k_{eff}), neutron flux, power distribution, and fuel burnup. The Monte Carlo method tracked millions of neutron histories to model scattering, absorption, and fission interactions, ensuring accurate representation of the VVR-KN fuel's hexagonal geometry. Simulations were performed in continuous-energy mode using ENDF/B-VIII.0 cross-section data. Criticality calculations determined k_{eff} at the beginning of cycle (BOC), while neutron flux and power distributions were computed using spatial tallies and heating tallies via OpenMC's DistribCell feature. Burnup analysis spanned a 90-day operational cycle, modeling U-235 depletion, neutron poison (Xe-135, Sm-149) buildup, and actinide (Pu-239) production. The simulation workflow included geometry definition, material specification, and iterative neutron transport calculations, validated against reference values from similar reactor designs [4].

2.4 Depletion Analysis

Depletion simulations tracked changes in core composition over 90 days, using OpenMC's depletion module with the PWR depletion chain. The analysis modeled the reduction of U-235 due to fission, accumulation of neutron poisons (Xe-135, Sm-149), and transmutation of U-238 to Pu-239 via neutron capture and beta decay. Key outputs included k_{eff} trends, U-235 and U-238 concentrations, and Xe-135 and Sm-149 buildup profiles, providing insights into reactivity control and fuel cycle planning. Simulations accounted for equilibrium states of neutron poisons, with Xe-135 stabilizing after 40–50 hours and Sm-149 after approximately 20 days.

3 Results and Discussions

This section presents the neutronics analysis results for a 10 MW multi-purpose research reactor (MPRR) using VVR-KN low-enriched uranium (LEU) fuel, simulated with the OpenMC Monte Carlo code and the ENDF/B-VIII.0 nuclear data library. Key parameters, including effective multiplication factor (k_{eff}), neutron flux, power distribution, and fuel burnup characteristics, are evaluated to assess reactor performance and safety. Results are compared with reference values from prior studies to validate the computational model and highlight the suitability of VVR-KN fuel for Bangladesh's nuclear research infrastructure.

3.1 Criticality Analysis

The effective multiplication factor (k_{eff}) at the beginning of cycle (BOC) was calculated to evaluate criticality of the reactor. Table 3.1 presents the k_{eff} results, showing a calculated value of 1.10693 using ENDF/B-VIII.0, compared to a reference value of 1.107 from ENDF/B-VII.0, with an absolute error of 6 pcm. This close agreement, well below the 50 pcm threshold typical for high-fidelity models [4], it is clear from here that there is no significant error in the simulation model preparation and is within expected limits to characterize this type of reactor design. The minor deviation likely stems from differences in nuclear data evaluations between ENDF/B-VIII.0 and ENDF/B-VII.0, justifies and supports another aspect of its applicability to MPRR type reactor designs.

Table 3. 1: Effective Multiplication Factor at BOC

Calculated Value	Reference Value	Error in pcm
1.10693 (ENDF/B-VIII.0)	1.107 (ENDF/B-VII.0)	6

3.2 Neutron Flux Distribution

The neutron flux distribution is of vital importance to ensure optimal utilization of MPRR in applications, such as isotope production and neutron scattering. representation of the radial distribution of the neutron flux in figure 3.1 shows that it attains a value of 4.8×10^{14} neutrons/cm²·s in the central part of the core and tapers towards the edge due to geometrical buckling. The average neutron flux, calculated as 5.8×10^{14} neutrons/cm²·s, deviates by 4% from the reference value of 5×10^{14} neutrons/cm²·s reported by Nguyen [13]. This high flux confirms the MPRR's suitability for research applications, with the small inconsistency attributable to differences in core modeling assumptions.

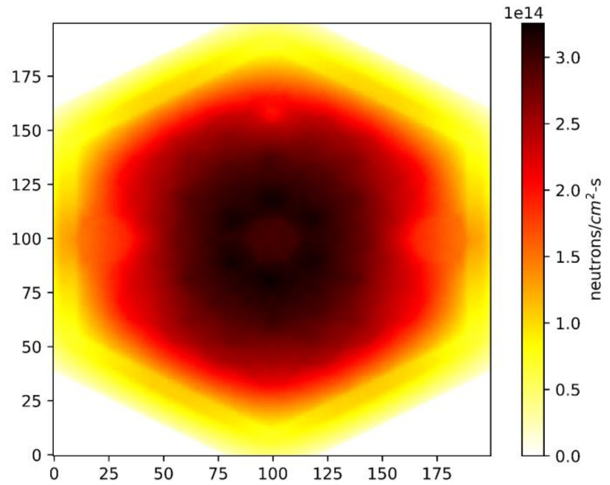

Fig. 3: 1: Radial Neutron Flux Profile (neutrons/cm²·s)

3.3 Power Distribution

Power distribution analysis provides insights into the thermal performance of the reactor. Using OpenMC's DistribCell feature, it is possible to find out how much power every fuel assembly produces, with results shown in Table 3.2. The average power of each assembly is 0.283 MW, closely matching the reference value of 0.278 MW with an error of 1.77% which indicates a balanced core design [4]. The highest power output, 0.316 MW, occurs in the assemblies adjacent to the central neutron trap, likely due to enhanced neutron moderation, resulting in a power peaking factor (PPF) of 1.119. The overall distribution is relatively uniform with gentle radial variation, reflecting a typical compact research reactor core.

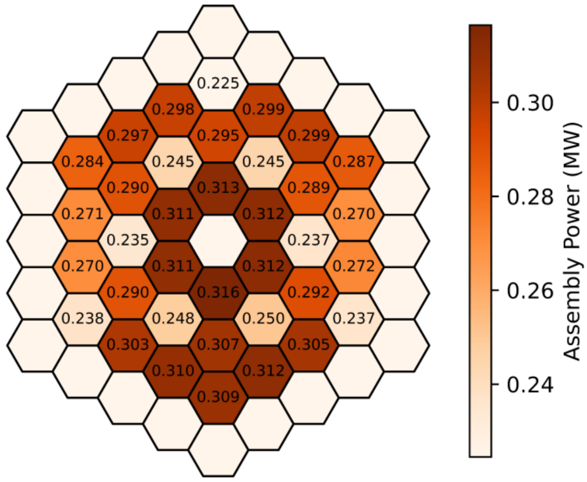

Fig. 3: 2: Power (MW) for each fuel assembly

Table 3. 2: Power fraction in each assembly type

Assembly Type	Power Fraction
FA1	78.87%
ShR	14.52%
SR	4.51%
AR	2.1%

3.4 Neutron Energy Spectrum

The neutron energy spectrum, critical for understanding the MPRR's thermal reactor characteristics, is presented in Figure 3.2. The flux peaks in the thermal energy region (<1 eV) at 0.1 eV, which corroborates that the neutron moderation by light water is very effective, while the contribution of fast neutrons (>100 keV) is minimal. Consistent with Nguyen et al. (2020), this forms a thermal spectrum ideal for neutron activation analysis and the production of radioisotopes. In this way, it increases the versatility of the MPRR for different research applications.

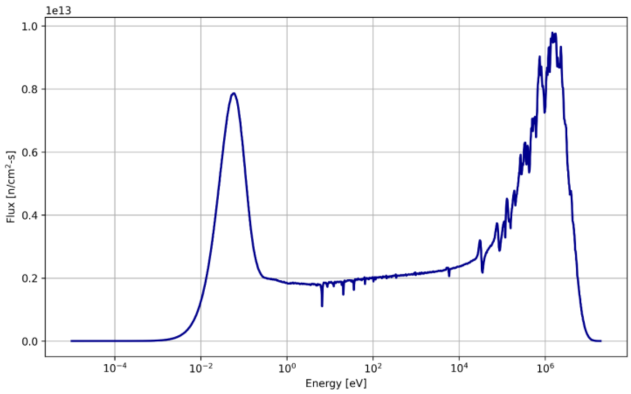

Fig. 3: 3: Energy-Dependent Neutron Flux Distribution Spectrum

3.5 Fuel Burnup and Depletion

Fuel burnup analysis over a 90-day cycle provides information about reactivity and core lifetime. Figure 3.3 illustrates the k_{eff} trend, declining from 1.10 at BOC to 1.003 at the end of cycle (EOC), with an excess reactivity of 1048 pcm (0.86% error from the reference 1039 pcm). This decline, driven by U-235 depletion and neutron poison accumulation, which is typical for thermal reactors [4]. Figure 3.4 illustrates the U-235 concentration decrease due to fission, while Figure 3.5 and 3.6 show Xe-135 and Sm-149 buildup until it reached its equilibrium after 40–50 hours and 20 days, respectively, due to their high neutron absorption cross-sections. Figure 3.8 and 3.9 depict the gradual decline of U-238 and the corresponding increase in Pu-239 via neutron capture and beta decay, contributing to long-term reactivity. These are behaviours that have previously been studied by Arinkin [6], verifying the VVR-KN fuel performances for extended MPRR operation.

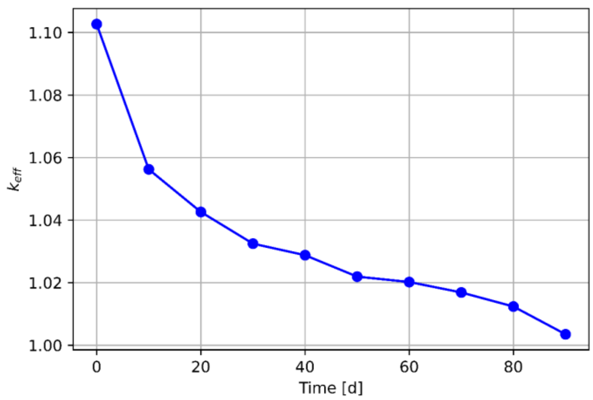

Fig. 3: 4: keff as a Function of Burnup Time

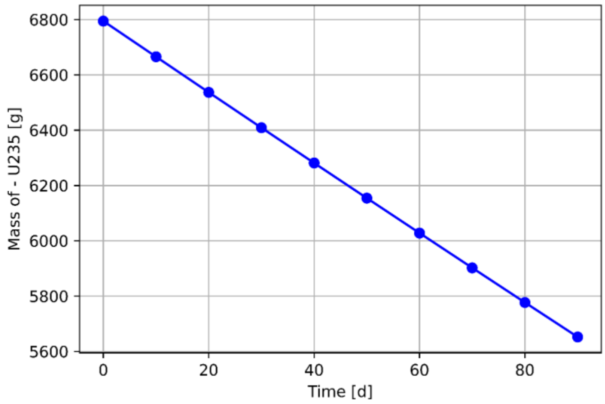

Fig. 3: 5: U-235 Concentration vs Burnup Time

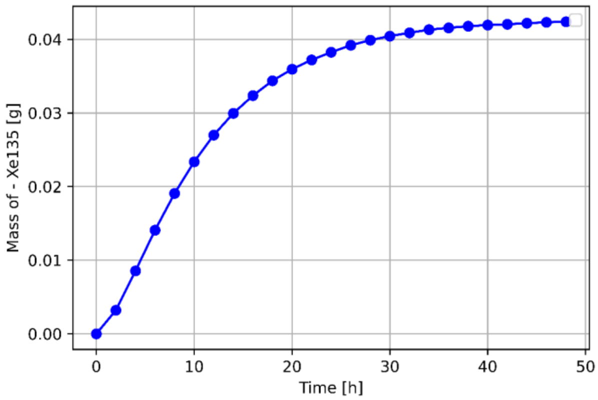

Fig. 3: 6: Xe-135 Concentration vs Burnup Time

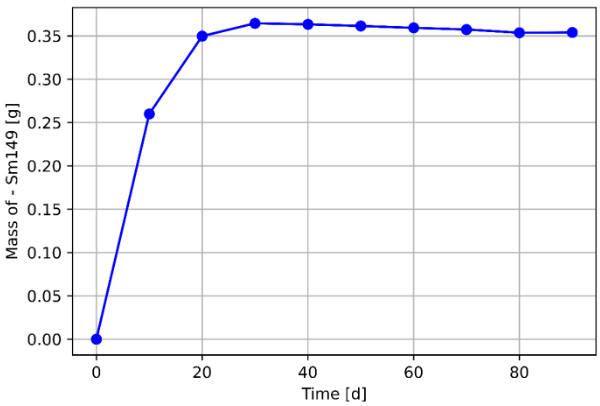

Fig. 3: 7: Sm-149 Concentration vs Burnup Time

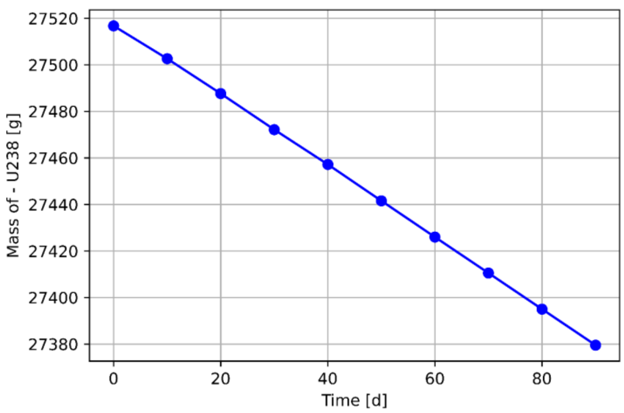

Fig. 3: 8: U-238 Concentration vs Burnup

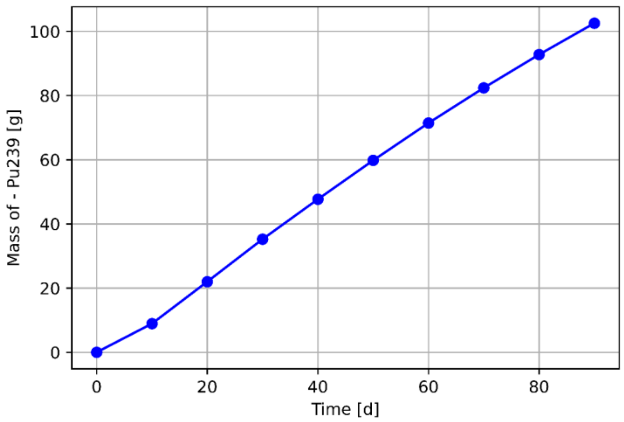

Fig. 3: 9: Pu-239 Concentration vs Burnup

3.6 Temperature Coefficients

As illustrated in Figure 3.10 and Figure 3.11, the reactor reactivity decreases as both the fuel and coolant temperatures increase, clearly demonstrating negative temperature reactivity feedback. Negative temperature coefficients are essential for reactor safety, as they provide an inherent and passive feedback mechanism that limits power excursions and promotes stable operation without reliance on active control systems.

For the evaluation of the fuel temperature coefficient, the variation of reactivity with fuel temperature shown in Figure 3.10 was fitted using a linear regression over the temperature range of 294 K to 380 K. The resulting fuel temperature coefficient was determined to be −2.8249 pcm/K, confirming the presence of a consistent negative reactivity feedback associated with fuel temperature increase.

In the case of the coolant temperature coefficient, Figure 3.11 reveals that the reactivity variation can be clearly divided into two distinct linear regions. Accordingly, the coolant temperature coefficient was calculated separately for the temperature intervals of 294-330 K and 330-380 K, within which linear behavior is maintained. The obtained coolant temperature coefficients are approximately −16.5455 pcm/K for the lower temperature range and −38.7512 pcm/K for the higher temperature range, indicating stronger negative feedback at elevated coolant temperatures.

Overall, the negative values of both the fuel and coolant temperature coefficients confirm that the reactor exhibits favorable inherent safety characteristics, with temperature-induced reactivity feedback that effectively stabilizes reactor power under varying operating conditions. These values satisfy the design targets and the safety requirements recommended in IAEA guidelines [16], [17].

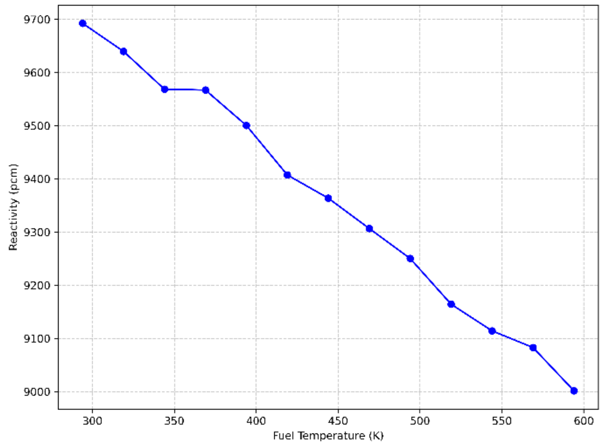

Fig. 3: 10:Fuel Temperature Coefficient (FTC)

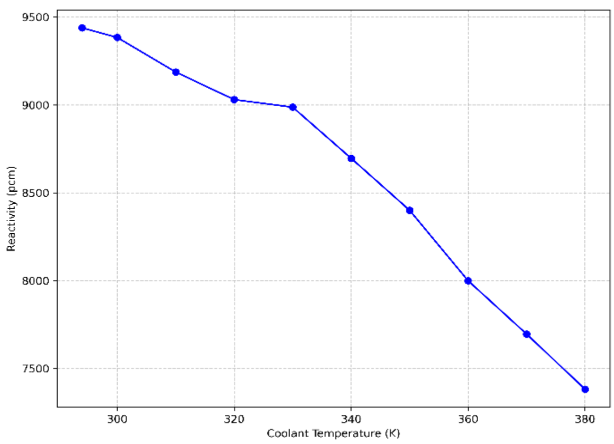

Fig. 3: 11: Coolant Temperature Coefficient (CTC)

4 Conclusion

Neutronics analysis of the 10 MW MPRR with VVR-KN LEU fuel demonstrates that it can meet the more demanding requirements of advanced research applications, with an effective multiplication factor of 1.10693 at BOC, an average neutron flux in the reactor core of 5.8×10^{14} neutrons/cm²·s, and a power profile of 0.283 MW per fuel assembly on average with a power peaking factor of 1.197. The neutron energy spectrum peaks at 0.1 eV, allowing for efficient use in thermal applications such as isotope production and neutron scattering. During the 90-day cycle, the k_{eff} decreases to 1.003 because of the depletion of U-235. Xe-135 and Sm-149 reach their equilibrium after 40-50 hours and 20 days, respectively; the increase of Pu-239 due to transmutation of U-238 improves the long-term reactivity, reaching an excess reactivity of 1048 pcm at the end of the cycle.

The Multi-Purpose Research Reactor (MPRR) can serve as an effective surrogate platform for RNPP-related experimental validation, particularly in areas such as control rod calibration and nuclear cross-section testing. In addition to its research function, the MPRR provides valuable training opportunities for RNPP operators by enabling hands-on experience in reactivity management, neutron flux measurement, and isotope handling under controlled conditions. Furthermore, the reactor facilitates benchmarking of VVER-type fuel behavior in smaller-scale cores, thereby enhancing confidence in the accuracy and reliability of VVER-1200 neutronic modelling and analysis within Bangladesh's nuclear program.

Future work includes integrating thermal-hydraulic coupling to analyze temperature and fuel–moderator effects, developing a digital twin for operator training, and validating flux distributions through activation foil experiments. Comparative studies between VVR-KN and LEU fuels will assess performance and feasibility, while collaboration with RNPP and BAEC will focus on nuclear data refinement and code benchmarking.

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Contribution of Individual Authors to the Creation of a Scientific Article (Ghostwriting Policy)

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Conflict of Interest

The authors have no conflicts of interest to declare that are relevant to the content of this article.