Nuclear Characteristics Analysis of MTR Fuel Assembly Using Monte Carlo Method Based KENO-VI Code

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1. Introduction

The research reactors (RRs) are widely used in various fields, including nuclear fuel and material test, RadioIsotope (RI) production, and Neutron Activation Analysis (NAA). Due to their high neutron flux and flexible operational characteristics, RRs play a crucial role in advancing nuclear science and technology. A key aspect of research reactor physics is the accurate modeling and analysis of fuel assemblies, as their design and performance directly impact reactor safety and efficiency.

Monte Carlo method based neutron transport codes (e.g., MCNP [1] and SERPENT [2]) are extensively utilized for criticality safety and reactor physics studies due to their ability to accurately simulate complex geometries and neutron interactions. Among these codes, the KENO-VI, a part of the SCALE package code system developed by Oak Ridge National Laboratory (ORNL), provides powerful capabilities for criticality analysis and neutron transport simulations [3].

This study focuses on the modeling and nuclear characteristics analysis of MTR Fuel Assemblies (FAs) using the KENO-VI. The primary objectives are to evaluate the change of neutron multiplication factors (k_{eff}) in accordance with the existence of Burnable Poisons (BPs) in FA. These results are compared with other results obtained from MCNP6 calculations and the differences between two calculation results are specifically analyzed to confirm their accuracy.

2. Methods and Materials

2.1 Main Characteristics of KENO-VI

Unlike its predecessor (KENO-V.a), KENO-VI offers advanced 3D arbitrary geometry modeling capabilities, allowing for precise representation of complex reactor structures, nuclear fuel, etc. One of KENO-VI's key features is its support for both multigroup and continuous energy cross-section calculations. In multi-group mode, users can utilize cross-section libraries such as the 238-group or 44-group, while in continuous energy mode, the code employs ENDF/B libraries for highly accurate neutron transport analysis. Beyond criticality (k_{eff}) calculations, KENO-VI can analyze neutron flux and reaction rates, perform reactivity calculations, and support nuclear fuel depletion analysis. It is also integrated with control sequences within the SCALE package, such as CSAS6 (Control System for Advanced Simulation) and T5-DEPL, enabling more comprehensive reactor system simulations.

To facilitate the visualization and validation of complex 3D geometries modeled in KENO-VI, the GeeWiz Graphical User Interface (GUI) is provided (See **Figure 1**). GeeWiz allows users to create, edit, and visually inspect their KENO-VI input models, ensuring that complex geometries are correctly implemented before running simulations. By providing an interactive and intuitive visualization environment, GeeWiz significantly enhances the usability of KENO-VI, reducing errors and improving workflow efficiency.

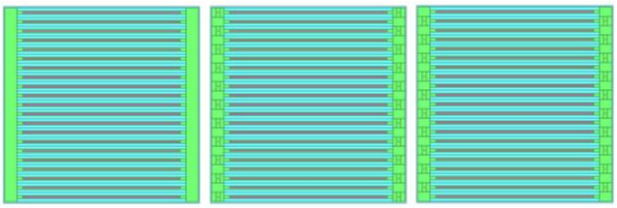
Figure 1. Geewiz Execution Screen

2.2 Unit Cell Model for MTR Fuel Assembly

The MTR FA generally consists of 21 fuel plates, which enhance cooling efficiency and ensure a uniform neutron flux distribution, making it well-suited for research reactors. Due to these characteristics, MTR fuel is widely used in applications such as the production of medical RIs (e.g., Mo-99), neutron scattering experiments, etc on new materials. The fuel meat consists of high-density nuclear fuel materials such as uranium-aluminum (U-Al) alloy, uranium-silicide (U_3Si_2), or uranium-molybdenum (U-Mo), with Low Enriched Uranium (LEU) of up to 19.75% enrichment [4]. Meanwhile, the FA including BPs (Cd wire) is designed to regulate neutron flux within the reactor core due to cadmium's strong neutron absorption properties.

The unit cell model was composed of fuel meat, cladding, side plates, BPs, and moderator region and

was designed with reference to the FAs used in the Argentina RA-6 and Japan MTR. The materials of fuel, cladding and side plate, BPs, and moderator are U_3Si_2-Al (4.8 gU/cm^3), aluminum, Cd (Diameter: 0.4mm and Length: 64cm), and pure water respectively. **Figure 2** shows standard MTR and other-type FAs including BPs, and the OPAL reactor uses the FAs containing 20 BPs.



(a) BP: 0 (b) BP: 22 (c) BP: 20

Figure 2. MTR FA Models Simulated by KENO-VI

3. Results and Discussions

Table 1 represents the criticality and standard divation of MTR FA Models evaluated by the KENO-VI and MCNP6 codes with the ENDF/B-VII.0 library. The reflective boundary conditions are applied to the four sides of X-Y section of the unit cell model, but not to the Z-axis section. Each evaluation was performed with 100,000 neutrons per cycle and an initial guess for k_{eff} of 1.0. The first 50 cycles were skipped before k_{eff} data accumulation, and a total of 450 cycles were run. As shown in the table, criticality values obtained from the two computational codes exhibit a difference ranging from a minimum of 1 pcm to a maximum of 110 pcm, with an average difference of approximately 60 pcm, indicating a high level of agreement. Additionally, the standard deviation of each criticality value remains less than 15 pcm, confirming that highly accurate calculations were performed. Furthermore, incorporating approximately 20 Cd wires into the fuel assembly is expected to reduce the initial criticality by ~100,000 pcm, significantly mitigating the reactivity swing during burnup. These results indicat that KENO-VI has a sufficient level of accuracy for application in nuclear fuel design and core design, comparable to MCNP6.

Table 1. Calculation Results Using KENO-VI and MCNP6

Unit Cell Model	KENO-VI	MCNP6	Difference (pcm)
Model A (BP:0)	1.53599 ±0.00013	1.53598 ±0.00012	1
Model B (BP:22)	1.43287 ±0.00013	1.43211 ±0.00012	76
Model C (BP:20)	1.44071 ±0.00013	1.43962 ±0.00012	109

4. Conclusions

In this study, the criticality analysis of MTR fuel assemblies was performed using the Monte Carlo-based KENO-VI and MCNP6 codes. The results showed that the calculated neutron multiplication factors (k_{eff}) from both codes exhibited a maximum difference of 110 pcm and an average difference of approximately 60 pcm, indicating a high level of agreement. Notably, incorporating approximately 20 Cd wires (Burnable Poisons) into the fuel assembly was found to reduce the initial criticality by approximately 100,000 pcm, effectively mitigating reactivity swing during burnup. These findings suggest that KENO-VI has a sufficient level of accuracy comparable to MCNP6, making it applicable for nuclear fuel and core design. In conclusion, KENO-VI can be effectively utilized for the criticality analysis and neutron transport simulation of MTR fuel assemblies, particularly in quantitatively evaluating the effects of BPs. Future research should focus on optimizing fuel design by analyzing reactivity variations and long-term burnup characteristics under various operating conditions.

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