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ANALYSIS OF CYCLE LENGTH FOR HIGH-FISSILE-DENSITY FUEL IN HTR-MMR

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ABSTRACT

Micro Modular Reactors (MMRs) are gaining significant attention in the nuclear power industry due to their compact design and mobility. However, the challenge of optimizing neutron economy in such reactors remains a critical issue. To address this, advanced fuel designs with higher fissile density are being explored to extend the operational cycle length of MMRs. This paper presents a detailed neutronics simulation of an MMR using High-Fissile-Density Fuel. A High Temperature Reactor (HTR)-based MMR was selected as the reference model. Several case studies involving different High-Fissile-Density Fuel types were conducted, analyzing key neutronic parameters such as cycle length, achievable burnup, power distribution, and nuclide inventory. The simulations also examined the fuel temperature reactivity coefficient to assess the safety and efficiency of the proposed fuels. Thermal power generation and thermal-hydraulic behavior were not considered in this study and are left for future work. Results indicate a significant improvement in cycle length and burnup with the implementation of High-Fissile-Density Fuel, along with a more favorable power distribution. These findings suggest that advanced fuel designs could play a crucial role in enhancing the performance and sustainability of MMRs, contributing to the advancement of small nuclear reactor technologies.

Keywords: Neutronics, MCNPX, High density fuel, MMR reactor

INTRODUCTION

Micro Modular Reactors

Nuclear power is a key component of the low-carbon energy solution, having grown significantly in the 1970s and 1980s. Countries like China and South Korea have effectively expanded their nuclear capacity, with China doubling its capacity to 54 GWe by 2020. Innovations in nuclear technology focus on faster construction through factory production, simpler operation, and inherently safer designs. Small Modular Reactors (SMRs), which produce up to 300 MWe, are developed for their cost-effectiveness, safety, and versatility. Micro-sized Modular Reactors (MMRs), producing less than 30 MWe, offer advanced features like unmanned operations and emergency power supply. These advancements aim to enhance nuclear power's efficiency, safety, and economic viability (Rabir, Ismail & Yahya, 2021).

The High-Temperature Reactor (HTR) is the most studied MMR type due to its potential and suitability. It is considered a leading innovative reactor design, capable of deployment within a decade. Studies over the past 30 years have confirmed the inherent safety features of micro modular HTRs. They can co-generate industrial heat along with electricity. Despite being a mature design, further research is needed to improve its fuel burnup performance, as smaller reactors suffer from larger neutron leakage, affecting their burnup performance and operational cycle length. To extend the operational cycle, small reactors often operate at reduced power density, which decreases their fuel economics compared to commercial reactors (Rabir, Ismail & Yahya, 2021).

Advanced fuel to prolong cycle length

Utilizing high enriched uranium (HEU) fuel poses significant proliferation risks. Since the 1970s, numerous HEU fuels have been converted to low enriched uranium (LEU). Any enrichment level exceeding 20% uranium-235 is classified as HEU (Van Den Berghe & Lemoine, 2014). While most commercial large reactors currently use fuel with less than 5% U-235, new High Assay Low Enriched Uranium (HALEU) fuels are being developed to enhance reactor performance. The higher concentration of fissile U-235 in HALEU allows for smaller fuel assemblies and reactors, reducing the frequency of refueling. This makes HALEU an attractive option for many SMRs and MMRs. When paired with high-fissile-density fuels like uranium silicide (U_3Si_2) or uranium mononitride (UN), HALEU maximizes fuel performance due to their higher uranium densities—11.3 g/cm³ for U_3Si_2 and 13.55 g/cm³ for UN, compared to UO_2's 9.75 g/cm³.

Both UN and U_3Si_2 are promising candidates for advanced systems, including liquid-metal-cooled fast reactors. For UN, high linear heat generation rates—42 kW/m in lead-cooled and 54 kW/m in sodium-cooled reactors—are achievable due to its exceptional thermal properties. U_3Si_2 also supports high heat generation rates and demonstrates excellent mechanical integrity under high temperatures and irradiation conditions. The superior thermal conductivity of both materials ensures lower fuel temperatures, reduced thermal stresses, and enhanced safety margins, making them strong contenders for improving the efficiency and safety of light water and advanced reactor designs (Yang et al., 2021; Khoshahval, 2024) .

This research aims to simulate and compare the neutronics performance of three HTR-MMR core models loaded with different fuel types: UO_2, U_3Si_2, and UN. The study will analyze key parameters, including reactivity, neutron flux spectrum, and burnup, to determine the advantages of each fuel type.

METHODS

Select High Temperature Reactor (HTR) MMR as references reactor

The reference reactor core parameters are based on a micro-sized HTR prismatic block reactor, commonly known as the U-Battery, initially developed in 2008 and recognized for its commercialization potential among micro-HTR designs. This reactor technology uses graphite blocks as the moderator, with fuel blocks featuring channels for helium coolant and fuel compacts containing TRISO (TRi-structural ISOtropic) fuel particles embedded within a graphite matrix. TRISO (TRi-structural ISOtropic) fuel particles are advanced nuclear fuel designed for high-temperature reactors (HTRs). Each particle consists of a fissile kernel surrounded by three protective layers: an inner porous carbon buffer, pyrolytic carbon layers (PyC), and a silicon carbide (SiC) layer. These layers ensure excellent fission product retention, with containment efficiencies exceeding 99.9%, and enable operation at temperatures up to

1600°C. The particle size is typically ~0.92 mm, and their design enhances safety and fuel integrity under extreme conditions (Kabach et al. 2021).

The fuel quantity depends on the concentration of TRISO particles, characterized by the "packing fraction" parameter. The TRISO packing fraction refers to the volume fraction of TRISO fuel particles within a fuel compact or matrix, typically ranging from 30% to 40% in high-temperature reactor designs to balance fuel performance, heat transfer, and structural integrity. The annular core configuration consists of six fuel columns surrounded by reflector blocks, with an additional central reflector block, as illustrated in Figure 1 (Ding & Kloosterman, 2011). For this study, each column comprises two stacked fuel blocks, resulting in a total of 12 fuel blocks, reduced from the original larger configuration to minimize simulation time since core design optimization is not the study's focus. The U-Battery's fuel blocks are modeled after the gas turbine-modular helium reactor (GT-MHR) design, with detailed parameters provided in Tables 1 and 2 (Ding & Kloosterman, 2013).

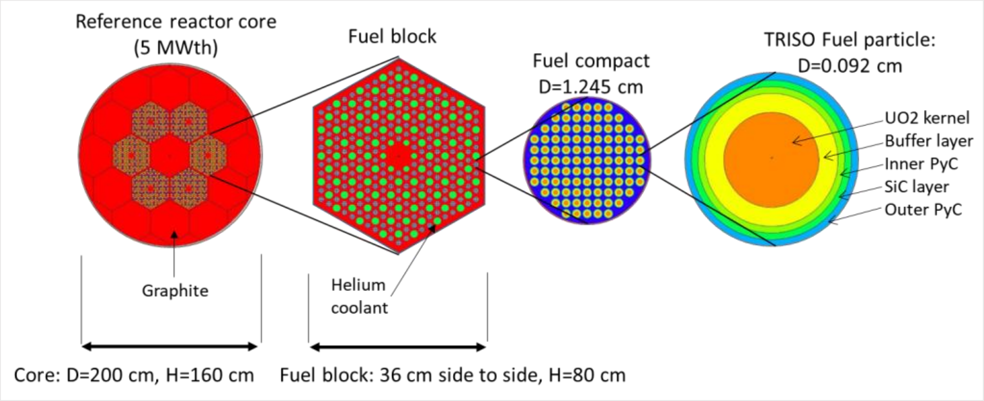

Figure 1 Reference reactor core configuration of the U-Battery

Table 1 MMR reference reactor core parameters

Parameter	Details
Reactor type	Block-type HTR
Thermal power	5 - 10 MWth
Cycle length	5 – 10 years
Core configuration	Annular core, 6 columns
Fuel type	UO_2, TRISO particle
Fuel enrichment	< 20 wt% ^{235}U/U
Coolant	Helium
Inlet/outlet temperature	250/750 ˚C
Energy utilization	Electricity, process heat

Table 2 Fuel block parameters

Parameter	
Width across flats (cm)	36
Height of block (cm)	80
Fuel kernel radius (µm)	250
Buffer layer thickness (µm)	100
Buffer layer density (g/cm3)	1
Inner PyC layer thickness (µm)	35
Inner PyC layer density (g/cm3)	1.9
SiC layer thickness (µm)	35
SiC layer density (g/cm3)	3.2
Outer PyC layer thickness (µm)	40
Outer PyC layer density (g/cm3)	1.97
Fuel compact diameter (cm)	1.245
Fuel compact height (cm)	4.93
TRISO packing fraction (%)	30

Simulation tool and model

All calculations were conducted using the MCNPX 2.7 code with the ENDF/B-VII.0 nuclear data library for steady-state reaction simulations. Key parameters such as neutron flux, reaction rates, and eigenvalues were calculated, while isotope depletion and updated isotopic compositions of the fuel were determined using the CINDER90 burnup code integrated with MCNPX. Due to the core's geometric symmetry, only 1/6 of the core was simulated, representing a single column of two stacked fuel blocks. This column was divided into segments, with each segment serving as a burnup zone where material changes occur independently, corresponding to spatial flux variations during the burnup period. Reflective boundary conditions were applied, with no burnable absorbers and a standardized fuel block temperature of 600°C. The MCNPX model simulated a high-temperature reactor core with detailed geometry, including the compact fuel's explicit TRISO arrangement. Burnup calculations using MCNPX KCODE achieved standard deviations ranging from 0.00040 to 0.00050 (40–50 pcm) with 50000 neutrons per cycle, 150 cycles, and 50 skipped cycles.

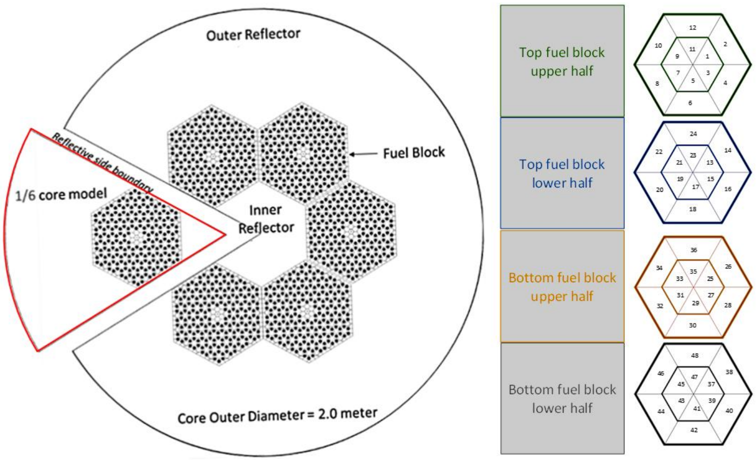

Figure 2: Top view of the referenced core illustrating the 1/6 model used in the MCNPX simulation (left) and the burnup zoning scheme applied (right), divided into four axial segments and six radial segments, numbered from 1 to 48.

Neutronics parameters

In this work, we analyzed several key neutronics parameters to highlight the differences between the fuel materials evaluated. The effective multiplication factor (k_{eff}) measures how the neutron population changes from one generation to the next in a nuclear reactor. The k_{eff} determines how long a reactor can sustain a chain reaction, directly influencing cycle length. Cycle length and exit burnup, measured in effective full power years (EFPY) and gigawatt-days per ton of heavy metal (GWd/tHM), are critical for determining the operational lifespan of the reactor core and the energy extracted per unit of fuel. EFPY is a unit of time that represents how long a nuclear reactor or fuel has operated at 100% full power continuously. It accounts for actual operating conditions by normalizing part-power or intermittent operation to an equivalent period at full power. The neutron flux energy spectrum expressed in neutrons flux unit with energy in MeV, describes how neutrons are distributed across various energy levels within the reactor. Fissile inventory, quantified in kilograms or grams, indicates the total mass of fissile material in the core, which is essential for assessing reactivity and fuel consumption. The peaking factor represents the peak local power within the reactor core, reflecting the maximum power density at specific locations compared to the average power across the core. Finally, the fuel temperature coefficient (FTC), expressed in pcm/°C, represents how reactivity changes with temperature variations in the reactor core. This analysis is limited to neutronics, as the study is in a preliminary stage; aspects related to heat management and thermal-hydraulic behavior are not included and are left for future investigation.

RESULTS AND DISCUSSION

Cycle length and burnup

Figure 3 illustrates the variation of k_{eff} over time for UO_2, U_3Si_2, and UN-loaded cores. As power is generated from fission, the buildup of xenon leads to a rapid decrease in k_{eff}. Subsequently, fuel depletion occurs, resulting in a slower, more gradual decline in k_{eff}. Among the fuel types, UO_2 exhibits the highest k_{eff} at the beginning of cycle (BOC), despite having a lower fissile mass than the other configurations. This behavior can be attributed to the reduced resonance capture associated with the lower U-238 mass in UO_2, which allows for a greater availability of thermal neutrons to induce fission.

In contrast, fuels with higher fissile densities, such as U_3Si_2 and UN, experience increased competition between resonance capture and neutron thermalization. This competition results in a lower k_{eff} at BOC as fissile density increases, despite the higher U-235 content. Nevertheless, the k_{eff} trends for U_3Si_2 and UN cores exhibit a gentler slope, indicating slower depletion compared to the UO_2 core. This slower depletion rate contributes to an extended cycle length, as evidenced by the UN-loaded core achieving the longest EFPY of 3.45. Meanwhile, the U_3Si_2-loaded core begins with the lowest k_{eff} at BOC. Although its depletion rate is less pronounced, it achieves a similar cycle length to the UO_2-loaded core, approximately 2.75 EFPY.

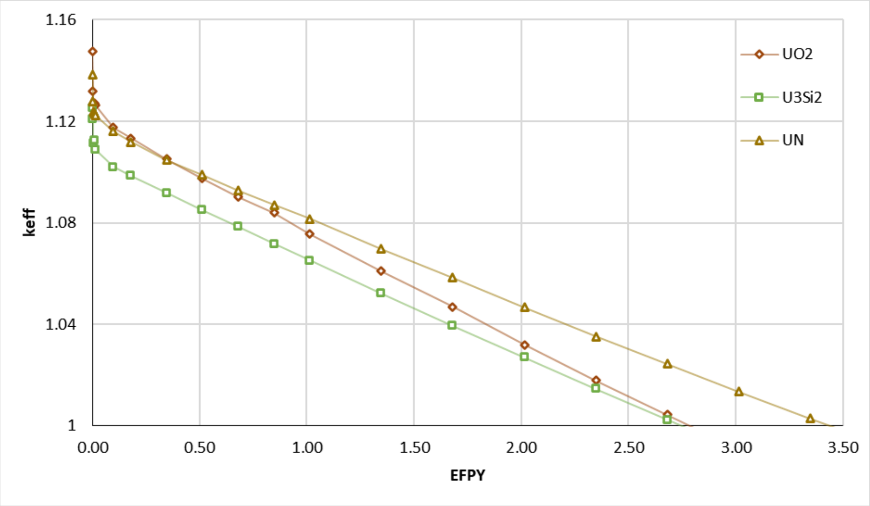

Figure 3: Evolution of k_{eff} over time for UO_2, U_3Si_2, and UN-loaded cores

Figure 4 depicts the average core burnup over time, measured in effective full-power years (EFPY), for the three cases. Since all cores operate at the same thermal power (simulated at 5 MWth), cores with higher fuel density exhibit a lower burnup gradient due to their larger initial fuel mass. Notably, the UN-loaded core achieves an exit burnup of approximately 60 GWd/MTU, which is remarkably high and within the range of advanced fuel burnup performance. In comparison, the UO_2-loaded core achieves an exit burnup of around 47 GWd/MTU, similar to the U_3Si_2-loaded core. Despite its lower burnup gradient compared to UO_2, the U_3Si_2-loaded core demonstrates a steeper gradient than the UN-loaded core.

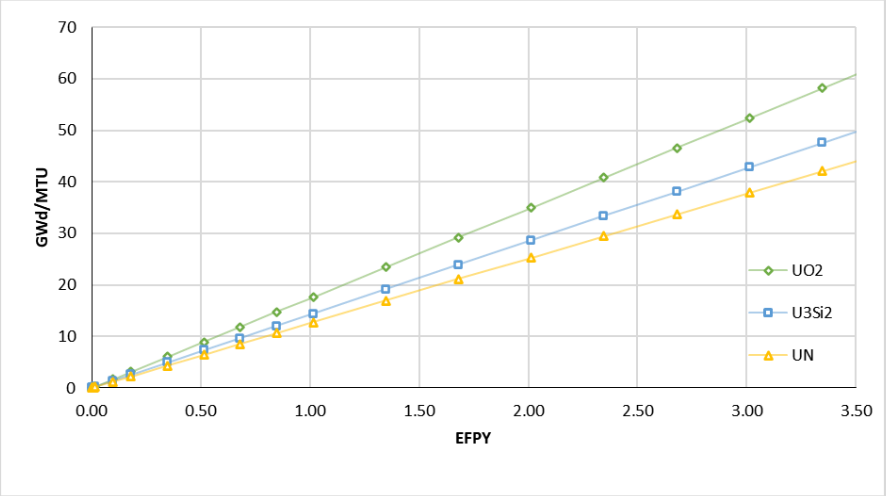

Figure 4: Average core burnup as a function of time (EFPY)

Neutron spectrum

The spectrum profile aligns closely with the typical neutron spectra reported in the literature for HTR prismatic fuel blocks. Unlike water-moderated reactors, the transition from fast to thermal energy occurs relatively more rapidly, whereas HTRs with graphite moderation require longer scattering times. Consequently, a significant fraction of neutrons remains within the resonance energy region in the spectrum (Chiang et al., 2014; Kim, Cho & Venneri 2007). Higher fissile density fuels with greater U-238 mass, corresponding to increase resonance capture, result in fewer neutrons reaching thermal energy levels. This trend is illustrated in

Figure 5, where higher fissile density correlates with a reduction in neutron flux at energies below approximately 1E-4 MeV. To demonstrate the effect of increased fissile density, Figure 5(b) illustrates that, while UO_2 exhibits a higher peak in the thermal neutron group at the EOC compared to the BOC, indicating an increase in thermal neutron flux, the other two materials show slightly lower peaks at EOC relative to BOC. In UO_2, the primary contribution to neutron absorption is from U-235. As U-235 depletes, the thermal neutron absorption reaction is significantly reduced. This trend is also expected to occur in U_3Si_2 and UN; however, in these materials, the contribution of resonance capture remains significant compared to the absorption in U-235.

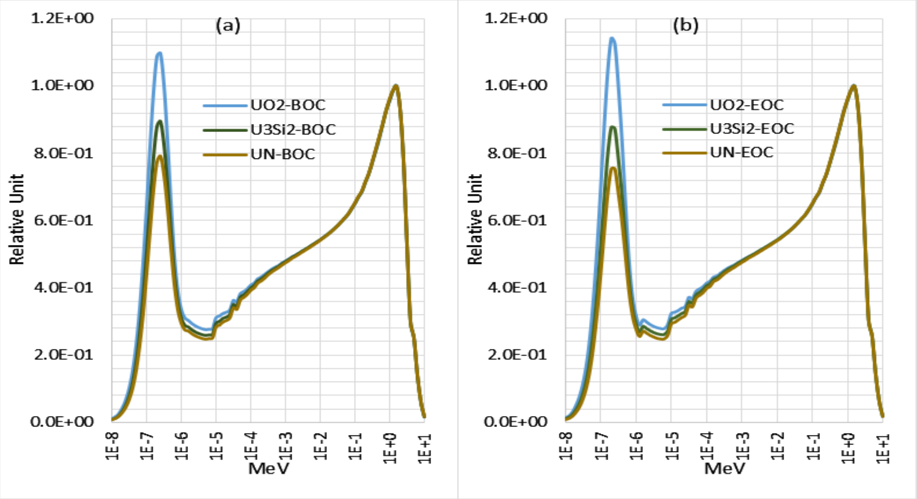

Figure 5: Neutron energy spectrum at beginning of cycle (BOC) and end of cycle (EOC).

Power distribution

Figure 6 illustrates the trends in the highest radial power peaking factor and its variation over time, expressed in effective full power years (EFPY). The results do not represent the actual spatial power distribution or its variation as a function of area or volume. Instead, they provide a single data point reflecting the highest power peak relative to the average and its evolution over time, as well as with burnup and depletion. It should be noted that none of the core models are optimized and all share the same basic configuration and fuel block loading pattern. As a result, the power distribution is expected to show minimal variation. However, fuels with higher fissile density exhibit slightly elevated maximum radial power peaking trends across the entire cycle length, which can lead to higher local fuel temperatures due to increased heat generation in those regions.

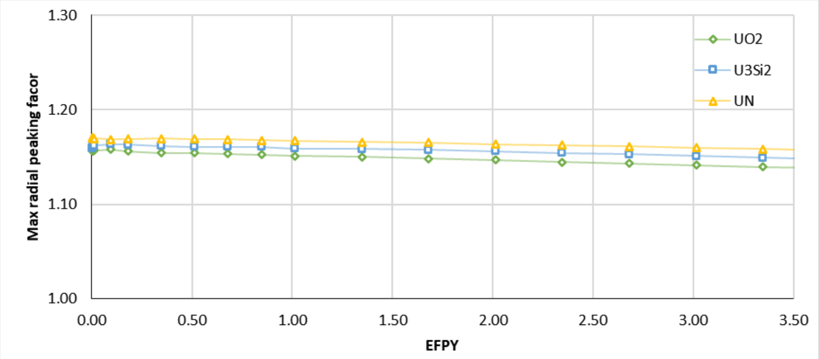

Figure 6: Comparison of the highest power peaking factor and its variation over time (EFPY).

Major fissile inventory

The concentrations of major fissile actinides as a function of cycle length are presented in Figure 7. The exponential depletion trends of U-235 for all core models are shown in Figure 7(a). It can be observed that fuels with relatively lower initial U-235 mass exhibit a progressively steeper depletion trend. As expected, Figures 7(b) and 7(c) indicate that Pu-239 build-up is highest in U_3Si_2 and UN loaded cores due to their higher U-238 content. With an increased production of fissile isotope Pu-239, the depletion of U-235 is relatively slower in these fuels, as the plutonium contributes significantly to the total fission process (Uguru et al., 2020).

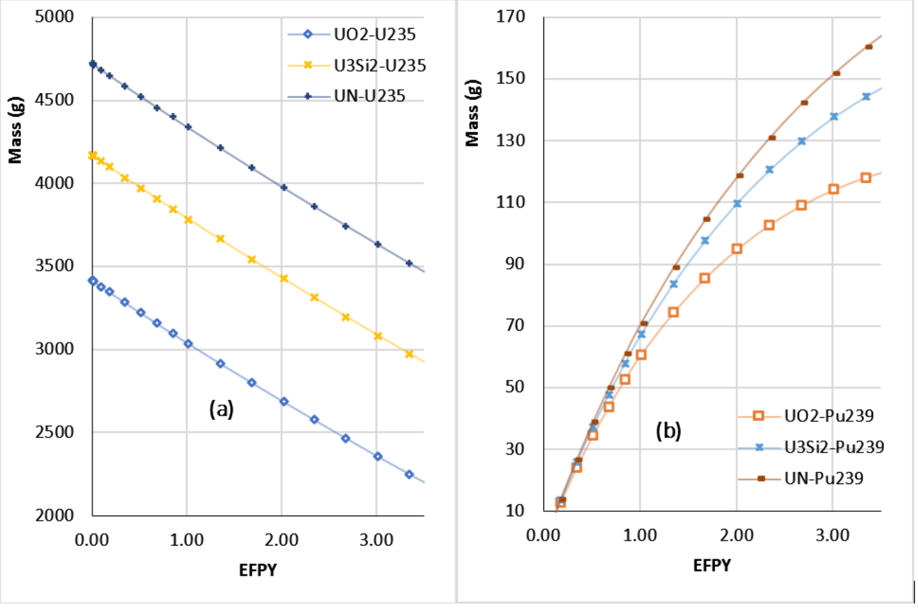

Figure 7: Reactors major fissile inventory comparison

Fuel temperature reactivity coefficient

The FTC calculated in this work are derived from differences in k_{eff} results obtained through simulations conducted at two temperatures: 600°C and 900°C. In this section of the calculation, the k_{eff} results obtained from MCNPX simulations were converted to reactivity (ρ) expressed in pcm (1E5 Δk/k). Figure 8 illustrates the simulated FTC values plotted against cycle length for three core models, all of which exhibit consistently negative FTCs throughout irradiation. As burnup progresses, the Doppler effect becomes more pronounced (FTC values become increasingly negative) due to contributions from U-238 and the accumulation of other non-fissile actinides (Choi, 2011). The FTC values obtained in this study align well with results reported for reference reactors in previous studies (Ding et al. 2011; Rabir, Ismail & Yahya, 2021). In comparison to UO_2, cores loaded with U_3Si_2 and UN demonstrate relatively stronger Doppler effects. This is evident from the more negative FTC values scattered throughout the cycle length for these fuels. The enhanced Doppler effect is attributed to the higher U-238 mass in U_3Si_2 and UN fuels, leading to increased resonance capture as the temperature rises.

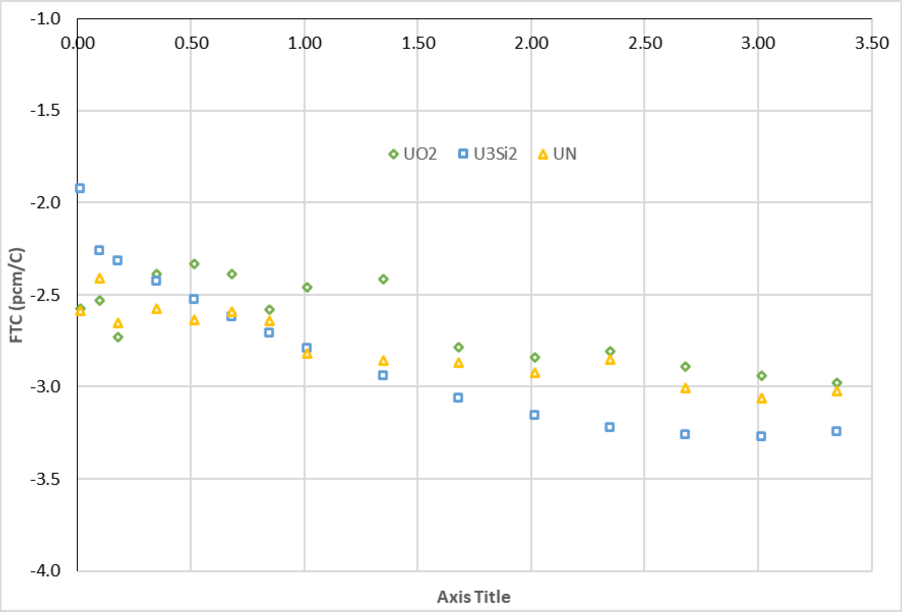

Figure 8: FTC trends comparison

Application-Oriented Fuel Selection

Based on the findings, UN (Uranium Nitride) emerges as the most suitable fuel candidate for nuclear power applications, particularly in Micro Modular Reactors (MMRs), due to its superior cycle length, high burnup capability, and strong negative fuel temperature coefficient, which enhances safety. Its high fissile density and thermal conductivity make it ideal for power reactors where long operating cycles and thermal performance are critical. Conversely, U_3Si_2 (Uranium Silicide), with its favorable neutronic behavior and moderate burnup, may be more appropriate for research reactors, where shorter cycles, higher neutron flux, and flexibility in reactivity management are often prioritized.

CONCLUSIONS

MMRs are gaining prominence in nuclear power applications due to their compact design and mobility. However, optimizing neutron economy remains a significant challenge. This study evaluates the performance of High-Fissile-Density Fuels (U_3Si_2 and UN) in HTR-based MMRs through detailed neutronics simulations. Key parameters assessed include reactivity, burnup, neutron flux spectrum, and safety metrics such as power distribution and the fuel FTC.

The results indicate that UN-loaded cores achieve the longest cycle length (3.45 EFPY) and highest exit burnup (60 GWd/MTU). While the U_3Si_2 core begins with a lower k_{eff} at the BOC compared to UO_2, it achieves a similar cycle length (2.75 EFPY) due to its slower depletion rate. Neutron spectrum analysis reveals that fuels with higher fissile density and greater U-238 content exhibit increased resonance capture, resulting in reduced thermal neutron flux. This is particularly evident in U_3Si_2 and UN fuels, which display lower thermal peaks at EOC compared to BOC. Higher-density fuels also exhibit slightly elevated maximum radial power peaking factors.

All core models demonstrate consistently negative FTCs throughout irradiation, with more pronounced negative values for U_3Si_2 and UN cores, attributed to their higher U-238 content. This indicates stronger Doppler effects, enhancing reactor stability at elevated temperatures. While High-Fissile-Density Fuels show potential for improving cycle length, burnup, and safety characteristics, further research is required to optimize material composition and core design before these findings can be generalized.

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