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Research Article

Analyzing (Th-^{233}U-^{235}U)O_2 fuel performance in various assembly configurations: A comparative neutronic study

Fadi El Banni[1], Bogbe L. H. Gogon[2], Ouadie Kabach[3], El Mahjoub Chakir[3]

1 *IREN, Nangui Abrogoua University, Abidjan, Cote d'Ivoire*
2 *LASMES, SSMT, Félix Houphouet Boigny University, Abidjan, Cote d'Ivoire*
3 *LPMS, Faculty of Sciences, Ibn Tofail University, Kenitra, Morocco*

Corresponding author: Ouadie Kabach (ouadie.kabach10@gmail.com)

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Abstract

This article investigates alternate fuel options for Pressurized Water Reactors (PWRs), focusing on thorium use to address safety, efficiency, and waste issues associated with standard UO_2 fuel. Challenges in thorium utilization, such as the lack of a fissile isotope, are handled using approaches such as homogeneous mixtures and heterogeneous arrangements, promoting the exploration of (Th-^{233}U-^{235}U)O_2 fuel in various assembly configurations. According to recent research, the annular dual-cooled assembly design has promising results in terms of fuel efficiency and safety while lowering the requirement for higher fissile enrichment levels. Studies additionally demonstrate that annular dual-cooled duplex fuel configurations can produce higher discharge burnup and lower power peaking factors than traditional UO_2 fuel. The purpose of this work is to analyze and compare the performance of (Th-^{233}U-^{235}U)O_2 fuel in various configurations against conventional UO_2 fuel, focusing on key characteristics such as reactivity change, criticality, discharge burnups, and reactivity feedback coefficients.

Keywords

Thorium-based nuclear fuel, PWR, Dual-cooled annular assembly, Neutronic evaluation, Criticality, Safety coefficients

Introduction

PWRs predominantly utilize low-enriched uranium dioxide (LEU) fuels; however, numerous alternative fuel options have been proposed over decades to address various concerns. These include enhancing safety parameters, developing fuels resilient to severe operating and irradiation conditions, minimizing nuclear waste, and increasing burnup efficiency. Among these alternatives, there's a growing interest in exploring the potential of thorium utilization. However, utilizing thorium poses challenges. For instance, pure ThO_2 lacks a fissile isotope, rendering it incapable of initiating a fission chain reaction in thermal reactors. Consequently, the initial utilization of thorium necessitates the incorporation of fissile material from the uranium cycle. In response to this challenge, numerous studies have investigated methods to incorporate thorium, either through different fuel mixtures or innovative fuel assembly arrangements (Galahom 2018; Galahom et al. 2021).

In general, the approaches for introducing fissile material to produce a thorium-based fuel are well recognized, including homogeneous mixtures and heterogeneous

arrangements (Du Toit et al. 2024). The fuel used in the homogeneous configuration is a mix of thorium and fissile material, such as uranium oxide (UO_2). Still, this technique demonstrated that the poor neutronic property would increase fuel cost over the cycle, and such a method requires higher ^{235}U enrichment to match the normal cycle length in PWR. The aforementioned heterogeneous (i.e. seed & blanket) structure incorporates distinct classes of fissile material and thorium at any fuel level: pellet, pin, or assembly (Uguru et al. 2021). This spatial separation of fissile material and thorium has the potential to improve the effectiveness of new fissile breeding (i.e. ^{233}U) and easier its extraction during reprocessing. It is also recommended since this strategy can be used on a small or large basis. Uranium and thorium are separated within a fuel assembly or a fuel rod using a micro-heterogeneous configuration, such as a duplex fuel rod. In contrast, the macro-heterogeneous fuel concept aims for spatial separation at the fuel assembly level (Galahom 2020).

Recent research indicates that using (Th-^{233}U-^{235}U)O_2 fuel in an annular dual-cooled 13 × 13 assembly design can be a feasible alternative to traditional solid UO_2 in homogeneous mixes (Benrhnia et al. 2022; Bouassa et al. 2023; Lkouz et al. 2023a, 2023b; Kabach et al. 2024). This fuel composition, which combines ^{232}Th, ^{233}U, and recovered ^{235}U, seeks to capitalize on the advantages of both ^{233}U and ^{235}U. This technique reduces the need for higher fissile enrichment and compensates for the reduced delayed neutron percentage and negative moderator temperature coefficients, which are principally influenced by ^{233}U (El Banni et al. 2022). Furthermore, the dual-cooled assembly design enhances the cooling of the fuel rods, mitigating the risk of overheating and cladding damage. This contributes to accident prevention and enhances reactor safety. The enhanced cooling of the fuel rods in the dual-cooled design can extend fuel lifetimes by reducing susceptibility to damage from overheating, thus allowing for higher power density and increased reactor efficiency (Hejzlar and Kazimi 2007; Yang et al. 2009; Shin et al. 2012; Deng et al. 2016). Research outcomes indicate that incorporating ^{233}U and ^{235}U with ^{232}Th enhances fuel burnup and augments the net delayed neutron fraction, while also introducing negative feedback into the overall moderator temperature coefficient. Moreover, (Th-^{233}U-^{235}U)O_2 fuel yields lower levels of ^{239}Pu and minor actinide concentrations.

Previous studies have also looked into the advantages of annular dual-cooled fuel and duplex fuel configurations in a Westinghouse SMR assembly, which is a scaled-down version of the AP1000 reactor. These investigations intended to improve the present performance criteria for such reactor types. To account for the dual-cooled feature, the study investigates both duplex configurations: ThO_2 in the inner region and UO_2 in the outer region, as well as UO_2 in the inner region and ThO_2 in the outer region (El Kheiri et al. 2023; Kheiri et al. 2023). This study found that annular duplex fuels can produce higher discharge burnup than annular UO_2 fuel, despite having approximately comparable reactivity values at the beginning of the cycle. Annular duplex fuels also have lower power peaking factors, which are beneficial for reactor operation.

Given these considerations, one of the main objectives of this paper is to analyze and comprehend the performance of (Th-^{233}U-^{235}U)O_2 fuel in three different configurations: simple annular dual-cooled, dual-cooled macro-heterogeneous, and dual-cooled micro-heterogeneous. This examination will be carried out by comparing them to conventional LEU-UO_2 fuel. Key characteristics to assess include reactivity change, criticality time, discharge burnups, power distribution, delayed neutron fractions, and reactivity feedback coefficients.

Materials and methods

Dimensions of reference and suggested assemblies

In this study, we examine two distinct fuel types, both possessing the same cumulative enrichment level of 4.95 wt. %, across five proposed fuel assembly (FA) configurations. The first fuel option is UO_2, while the second comprises a composite of ^{232}Th, ^{233}U, and ^{235}U, forming (Th-^{233}U-^{235}U)O_2 fuel. Table 1 outlines the characteristics of these investigated fuels.

Table 1. The main properties of the investigated fuel types (Kabach et al. 2024).

Fuel	Density	Fertile	Fissile	Enrichment
UO_2	10.53 (g/cm^3)	^{238}U	^{235}U	4.95 wt.% (^{235}U)
(Th-^{233}U-^{235}U)O_2	9.54 (g/cm^3)	^{232}Th	^{233}U & ^{235}U	2.475 wt.% (^{233}U)
				2.475 wt.% (^{235}U)

A single fuel assembly based on the Westinghouse-designed AP1000 was taken as the reference design (FA-1). This assembly contains a 17 × 17 grid, accommodating 264 fuel rod positions, 24 guide thimbles for control rods, and one central guide thimble for core instrumentation. The fuel rods are with UO_2 fuel with 95.5% theoretical density, each with a radius of 0.409575 cm. The fuel pellets are separated from the cladding by a 0.008 cm gap. The cladding is made of 0.06 cm thick Zirlo™, and the fuel rods are pitched at 1.26 cm (Schulz 2006).

Given the objectives outlined in the introduction, the primary focus of this paper is to analyze and understand the performance of (Th-^{233}U-^{235}U)O_2 fuel across various configurations. Prior research suggests that utilizing (Th-^{233}U-^{235}U)O_2 fuel in dual-cooled assembly designs enhances fuel rod cooling, thereby mitigating the risk of overheating and cladding damage (Benrhnia et al. 2022; Bouassa et al. 2023). Building upon the previous research, a 13 × 13 annular dual-cooled assembly was designed and loaded with (Th-^{233}U-^{235}U)O_2 fuel in this study. Three methods were employed: in FA-2 and FA-3, both fuels were distributed homogeneously. FA-4 uses a macro-heterogeneous concept, with the assembly divided into a Blanket region fueled by (Th-^{233}U-^{235}U)O_2 and a Seed region fueled by UO_2. However, FA-5 uses a micro-heterogeneous concept,

with the inner zone fueled by UO_2 and the outer by (Th-^{233}U-^{235}U)O_2. The geometric configurations of the assemblies are illustrated in Fig. 1. Supplementary structural material data can be found in Tables 2, 3, while Table 4 offers a comprehensive overview of the cases analyzed in the neutronic study. Additionally, Table 5 shows detailed mass compositions of the different fuel assemblies at BOC.

Used codes

The methodology employed in this study builds upon the approaches used in prior research (Benrhnia et al. 2022; Bouassa et al. 2023; Lkouz et al. 2023a, 2023b; Kabach et al. 2024). Both the DRAGON Version 5 and MCNP codes were utilized throughout this investigation. DRAGON code, an open-source package developed by École Polytechnique

Table 2. Geometrical specification of rod and guide in study assemblies (Benrhnia et al. 2022; El Kheiri et al. 2023).

Parameter	Solid	Dual-cooled	Dual-cooled duplex
Rod lattice pitch (cm)	1.260	1.648	1.648
Inner clad inner radius (cm)	-	0.431650	0.431650
Inner clad outer radius (cm)	-	0.488650	0.488650
Inner helium gap outer radius (cm)	-	0.494850	0.494850
Fuel 1 outer radius (cm)	0.409575	0.705150	0.598697
Fuel 2 outer radius (cm)	-	-	0.705150
Outer helium gap outer radius (cm)	0.417750	0.711350	0.711350
Outer clad outer radius (cm)	0.474750	0.768350	0.768350
Total fuel area per rod (cm2)	0.527007	0.792812	0.792812
Guide tube			
Guide tube inner radius (cm)	0.56	0.71	0.71
Guide tube outer radius (cm)	0.60	0.77	0.77

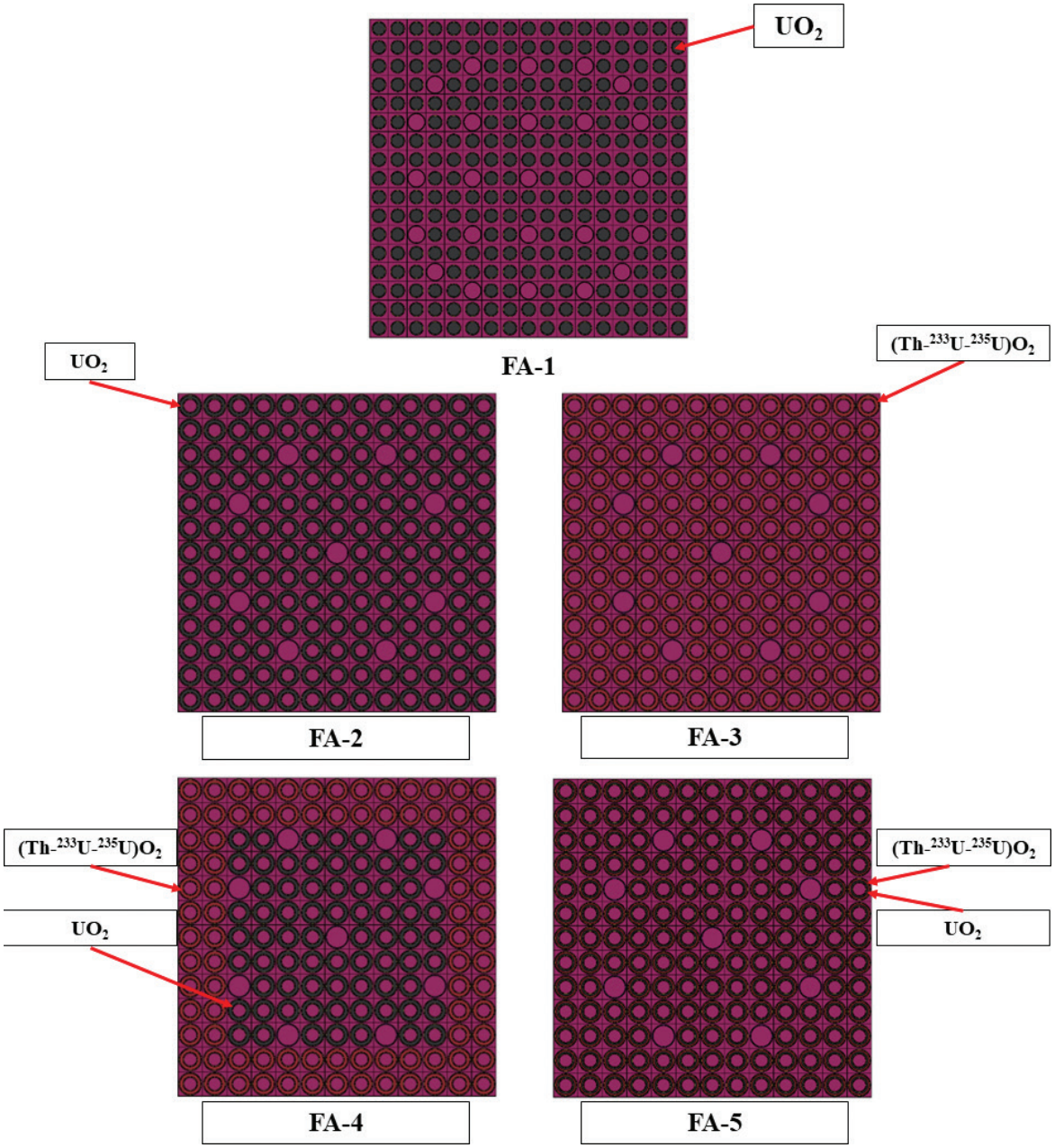

Figure 1. Horizontal cross-section of the suggested assembly models.

Table 3. Other data for reference assemblies.

Zone	Parameter	Value
Cladding	Material	Zirlo™
	Density	6.50 g/cm^3
Gap	Gap material	Helium
	Gap density at 600 K	0.012 g/cm^3
Moderator	Material	Light water
	Density at 573 K	0.721 g/cm^3
Guide tube cladding	Material	Stainless Steel type 304
	Density	8.03 g/cm^3

Table 4. Considered cases for the neutronic study.

Case Name	Configuration and compositions
FA-1	Solid (17 × 17) with 264 UO_2 fuel rods per assembly.
FA-2	Dual-cooled (13 × 13) with 160 UO_2 fuel rods per assembly.
FA-3	Dual-cooled (13 × 13) with 160 (Th- ^{233}U, ^{235}U)O_2 fuel rods per assembly.
FA-4	Dual-cooled (13 × 13) with 72 UO_2 and 88 (Th- ^{233}U, ^{235}U)O_2 fuel rods per assembly in macro-heterogeneous configuration.
FA-5	Dual-cooled (13 × 13) with 160 fuel rods per assembly in micro-heterogeneous configuration (UO_2 & (Th- ^{233}U, ^{235}U)O_2).

Table 5. Mass compositions of the different fuel assemblies at BOC.

Case Name	Nuclide	Mass (kg)
FA-1	^{235}U	27.25
	^{238}U	523.30
FA-2	^{235}U	24.85
	^{238}U	477.20
FA-3	^{232}Th	431.40
	^{233}U	11.23
	^{235}U	11.23
FA-4	^{232}Th	237.30
	^{233}U	6.18
	^{235}U	17.36
	^{238}U	214.70
FA-5	^{232}Th	237.30
	^{233}U	6.18
	^{235}U	17.36
	^{238}U	214.70

de Montréal, is employed to solve the neutron transport equation at the assembly level, employing deterministic methods like the collision probability method, discrete ordinate method, or method of characteristic (Hébert 2008; Marleau et al. 2021). DRAGON utilizes various modules for different functions associated with solving transport or diffusion equations, including LIB, GEO, EXCELT, MCCGT, SHI, ASM, FLU, EVO, and EDI. For this study, the cross-section library ENDFB-VIII ref.0 (SHEM-361) accessible via the download page was utilized.

Additionally, MCNP, a Monte Carlo code, was employed in this study to verify and validate the DRAGON depletion results, particularly in terms of the eigenvalue (k_{inf}). MCNP, utilizing the Monte Carlo probabilistic method, is designed to simulate the transport of large particles such as neutrons, photons, and electrons in complex geometric models (Werner 2017). The simulations in this research were conducted using a custom-developed library based on ENDFB-VIII.0 nuclear data (Brown et al. 2018; Kabach et al. 2019, 2021). The MCNP burnup runs comprised 80,000 neutrons per cycle with 150 inactive and 150 active cycles for each time step, resulting in a maximum uncertainty of 19 pcm.

Calculation methodology

As per available literature, large PWRs typically exhibit neutron leakage levels ranging from 3% to 4%, with the initial reactivity showing an approximately linear variation. In the present study, the constructed models include reflective boundary conditions on outer surfaces, hence neglecting neutron leakage in the eigenvalue calculation. To address neutron leakage, a 3% reactivity correction is applied. As a result, the single-batch discharge burnup is adjusted to yield a k_{inf} value of ≈1.03. The Linear Reactivity Model (LRM) is then used to convert the single-batch discharge burnup into that of a three-batch core (Burns et al. 2020; Akter et al. 2021; Hossain et al. 2022). The relationship between the discharge burnup for an N_{RB}-batch core ($B_{Discharge}$), the single-batch discharge burnup (B_{SC}), and N_{RB}, the number of fuel batches in the fuel assembly shuffling scheme is given by:

$$B_{Discharge} = N_{RB}B_{NC} = \frac{2N_{RB}}{(N_{RB}+1)}B_{SC} \quad (1)$$

The corresponding cycle lengths were calculated using the formula:

$$T_{cycle} = \frac{2}{(N_{RB}+1)}\frac{B_{SC}}{\rho} \quad (2)$$

where T_{cycle} represents the cycle length in Effective Full Power Days (EFPDs), and ρ is the specific power density of 40 kW/kg. In our cases, the discharge burnup and cycle length for each fuel model were modeled as calculated from the LRM under the assumption of a 3-batch core with 3% neutron leakage (Kabach et al. 2024).

The dynamics of a nuclear reactor during normal operation can be approximated by considering the number of delayed neutron emissions resulting from the decay of fission products. Safety parameters for reactors predominantly fueled by UO_2 include a higher ratio of effective delayed neutrons (β_{eff}), which is crucial for licensing operations (Washington 2016). Since β_{eff} is influenced by the type of fuel used—whether fissile or fertile—it's essential to assess how altering or changing the fuel type affects this parameter throughout burnup. In this study, MCNP, with options invoked on the KOPTS card, is utilized to compute β_{eff} (Werner 2017). This approach allows for the analysis of the impact of fuel variations on reactor dynamics and safety characteristics as burnup progresses.

Fuel and moderator temperature coefficients are two critical reactivity coefficients. The fuel temperature coefficient (FTC) is defined as the fractional change in k_{inf} per unit change in fuel temperature, while the moderator temperature coefficient (MTC) is defined similarly for

moderator temperature changes. In this study, FTC and MTC were evaluated at the beginning of cycle (BOC) and end of cycle (EOC) steps. The evaluated models are compared for the following temperature perturbation scenarios:

- Increasing the fuel temperature by 100 K over the baseline of 900 K.
- Increasing the moderator temperature by 20 K above the 573 K baseline.

The reactivity coefficients are then determined using Eq. (3), where α represents the evaluated coefficient, X_1 and X_2 are the changes in the considered parameter, and k_1 and k_1 are the multiplication factors arising from the states X_1 and X_2. These factors are multiplied by 10^5 to yield results in pcm (Bouassa et al. 2023).

$$\alpha = \frac{1}{X_2 - X_1}\left(\frac{k_2 - k_1}{k_2 k_1}\right) \cdot 10^5 \quad (3)$$

Results and discussion

Fuel burn-up

The neutronic calculations for the analyzed fuel type assemblies were conducted at a boron concentration of 0 ppm, with all control rods withdrawn. Fig. 2 depicts the k_{inf} as a function of fuel burnup for the investigated assemblies, comparing results from the DRAGON and MCNP code. The general pattern across the analyzed scenarios is consistent: k_{inf} falls dramatically during the early burnup stage due to the production of ^{135}Xe and ^{149}Sm, which have high absorption cross-sections. k_{inf} then decreases linearly with burnup, albeit at various rates, as a result of the burning of fissile material. According to previous studies, (Th-^{233}U-^{235}U)O_2-fueled models produce less ^{135}Xe- and ^{149}Sm than regular UO_2 fuels (Benrhnia et al. 2022; Lkouz et al. 2023b; Kabach et al. 2024).

As shown in Fig. 2 and outlined in Table 6, FA-3 scenario fuel has a higher reactivity at BOC and sustains criticality longer than FA-1, despite the fuel types having the same enrichment and the total fuel volume being lowered when transitioning from solid 17 × 17 to a dual-cooled 13 × 13 assembly. This discrepancy can be attributed to the advantageous characteristics of ^{233}U as a fissile material in a thermal reactor compared to ^{235}U. Specifically, ^{233}U demonstrates the lowest capture-to-fission ratio, resulting in higher neutron production per neutron absorbed in the fuel (η). The η value for ^{233}U is approximately 2.497 ± 0.004, whereas for ^{235}U it is about 2.435 ± 0.002 (Abdelghafar Galahom et al. 2024).

Transitioning from an all (Th-^{233}U-^{235}U)O_2-fueled assembly to the macro-heterogeneous (FA-4) and micro-heterogeneous (FA-5) configurations reveals distinct patterns in terms of initial reactivities and criticality, even though the total volumes occupied by UO_2 and (Th-^{233}U-^{235}U)O_2 are the same for FA-4 and FA-5 (see Tables 2, 4). FA-4 sustains criticality 0.24% longer than FA-1 and 9.77% longer than FA-2, despite having lower reactivity at BOC than FA-1. However, the criticality is reduced by 5.41% compared to FA-1, primarily due to the decreased concentration of ^{233}U. FA-5 sustains criticality for a short-

Table 6. Comparison of k_{inf} at BOC and EOC, and the single-batch discharge burnup (B_{SC}).

Case Name	k_{inf} (BOC)		k_{inf} (EOC)		B_{SC} (MWd/kg)	
	DRAGON	MCNP	DRAGON	MCNP	DRAGON	MCNP
FA-1	1.41672	1.41245	0.89381	0.90146	38.38	39.39
FA-2	1.37707	1.37474	0.89087	0.89803	34.73	35.82
FA-3	1.41836	1.41753	0.91448	0.92239	40.67	42.00
FA-4	1.39207	1.39239	0.90579	0.91291	38.48	39.81
FA-5	1.31661	1.31482	0.91461	0.92149	33.93	35.14

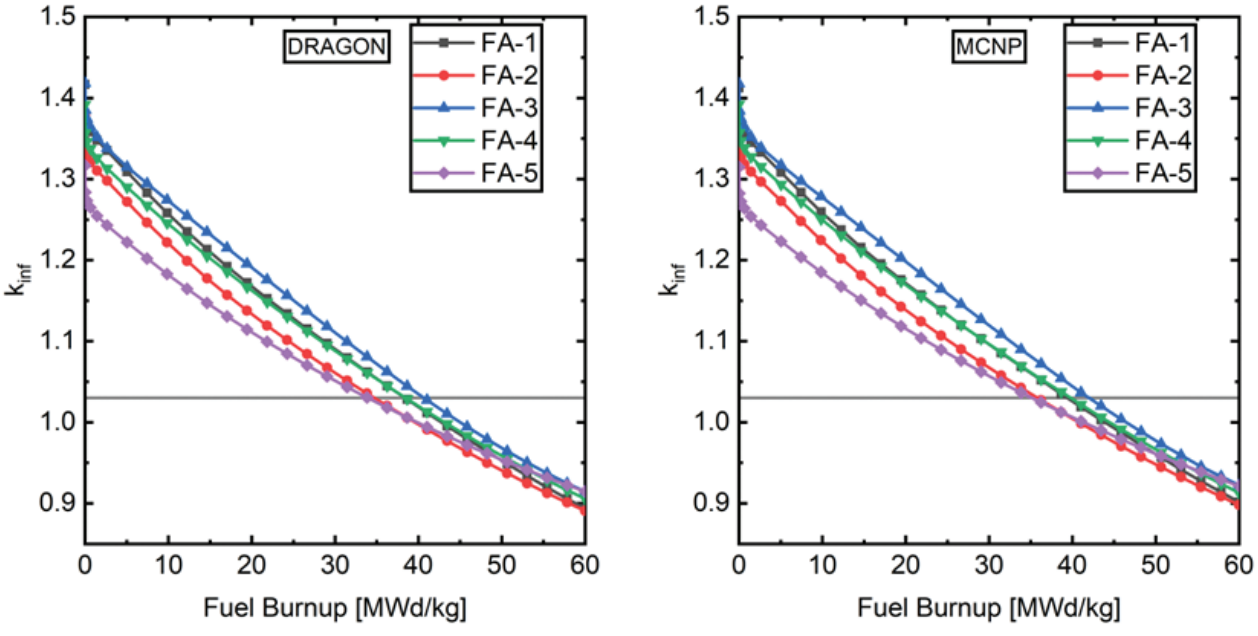

Figure 2. Infinite multiplication factor trend for the analyzed assembly models.

er duration than all other cases, which can be attributed to the duplex design's enhanced self-shielding effect.

As previously mentioned, the LRM was utilized to accurately represent key cycle parameters, including cycle burnup, discharge burnup, and cycle length. The results are visually depicted in bar chart representations in Figs 3–5. Fuel cycle parameters for FA-2 exceeded those for UO_2 assemblies (i.e. FA-1 and FA-2), as anticipated. Similarly, the FA-4 case exhibited enhanced fuel cycle parameters compared to UO_2 assemblies. Given that cycle burnup, discharge burnup, and cycle length are linked to the single batch criticality period, it follows that a decrease in the criticality in the FA-5 case leads to reductions in the values of these parameters as well.

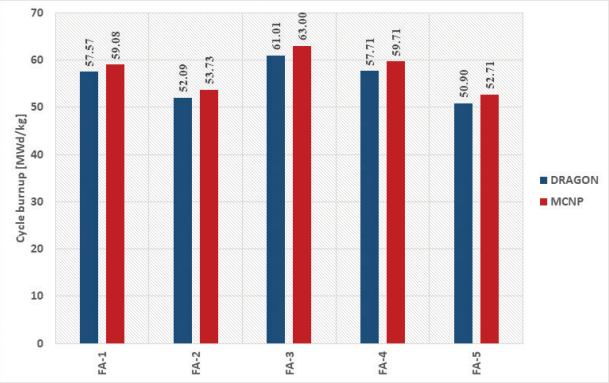

Figure 3. Comparison of cycle burnup.

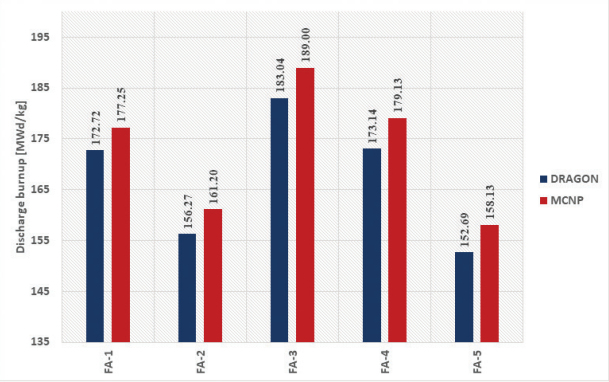

Figure 4. Comparison of discharge burnup.

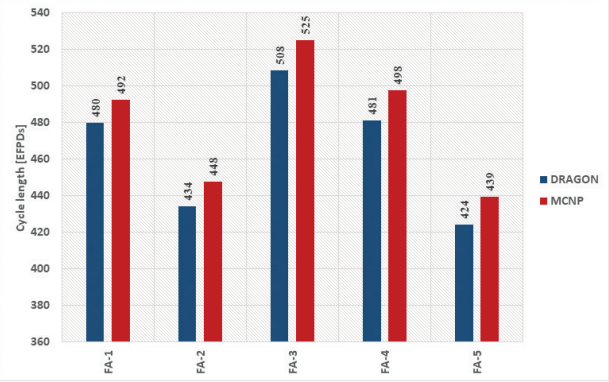

Figure 5. Comparison of cycle length.

Fissile inventory ratio

The fissile inventory ratio (FIR) is a crucial metric for reactor operation and is an effective means to evaluate a reactor's breeding capability. This ratio represents the proportion of the fuel's fissile content to its initial fissile content, serving as an indicator of the conversion ratio. The fissile nuclei considered in this study to measure the conversion ratio include ^{233}U, ^{235}U, ^{239}Pu, and ^{241}Pu (Galahom 2019). Additionally, ^{233}Pa and ^{233}Np were considered fissile materials to calculate the FIR because they will eventually decay into ^{233}U and ^{239}Pu, respectively, after the assembly has been discharged (Alam et al. 2019).

Fig. 6 illustrates the variation of the FIR with burnup. It is observed that the ratio in UO_2-based assemblies is smaller compared to (Th-^{233}U-^{235}U)O_2-based assemblies. Interestingly, the FIR for FA-5 is higher, with the difference increasing with burnup. This is attributed to the use of a micro-heterogeneous "duplex" configuration, which allows for a higher conversion rate of fertile material into fissile material. Furthermore, the double cooling system enables a greater moderator volume, enhancing neutron thermalization. This improvement in neutron utilization results in a better conversion of fertile material to fissile material, as shown in Fig. 7.

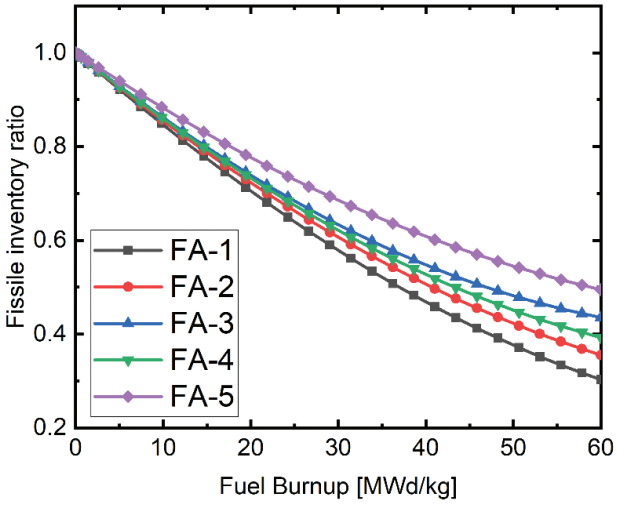

Figure 6. Variation of FIR with burnup for the different investigated assembly types.

Neutron spectra profiles

The neutron spectrum is influenced by the balance between neutron moderation and absorption. Fig. 8 illustrates the neutron flux per unit lethargy, the spectra are plotted using the DRAGLIB_SHEM-361-group energy structure. The micro-heterogeneous configuration (FA-5) results in a harder neutron spectrum due to the increased self-shielding effect inherent in its duplex design. Since a harder spectrum facilitates the efficient conversion of fertile material to fissile material, the FIR values are higher for FA-5, as depicted in Fig. 6.

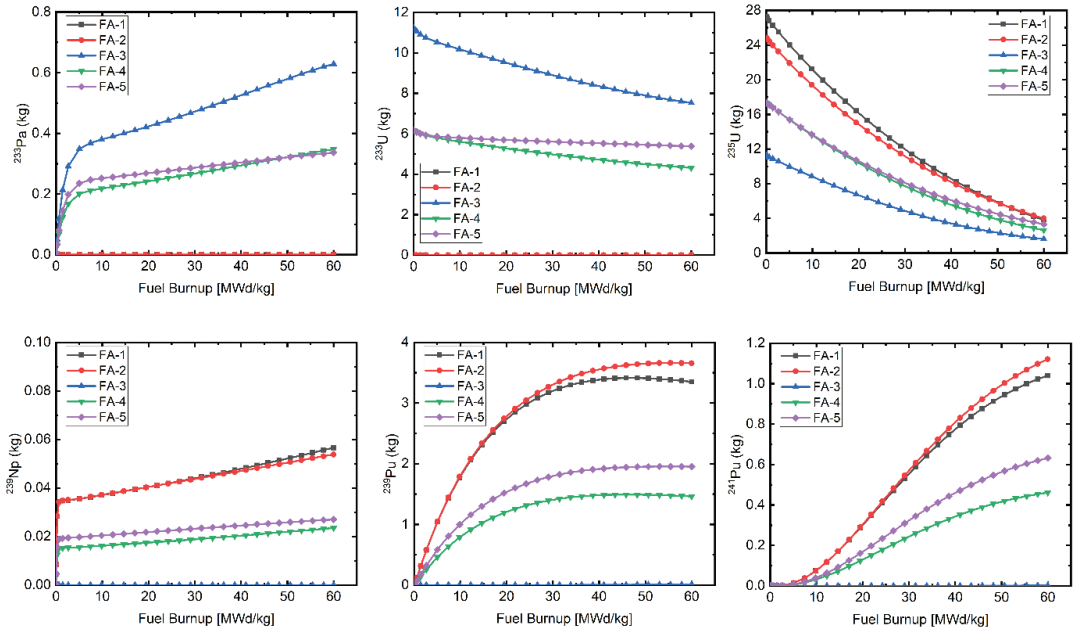

Figure 7. Variation of mass (in kg) of ^{233}Pa, ^{233}U, ^{235}U, ^{239}Np, ^{239}Pu, and ^{241}Pu with burnup.

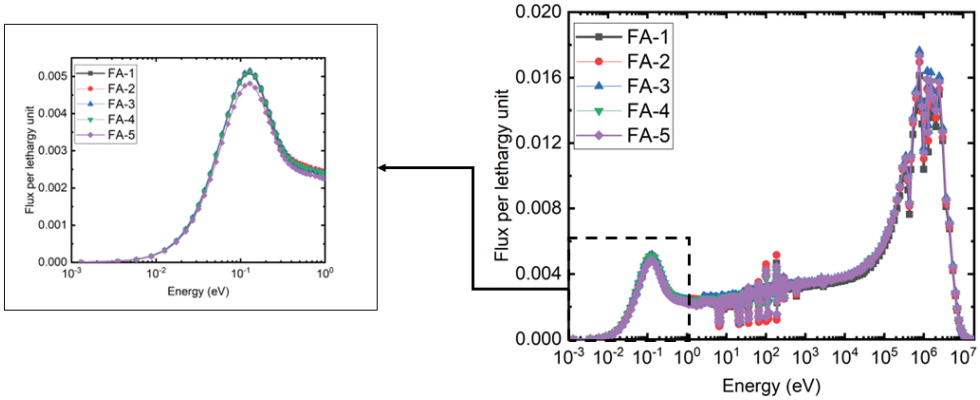

Figure 8. Normalized neutron spectra.

Radial power distribution

Fig. 9 depicts the relative pin power of the evaluated assembly cases at both BOC and EOC. The numerical values show the pin power ratio to the assembly's average pin power, with red numbers representing hot channels. The maximum pin powers at BOC and EOC for FA-3 are lower when compared to FA-1 and FA-2. However, using the macro-heterogeneous assembly (FA-4) presents a considerable issue because the maximum pin power increases due to EOC. This is linked to increased plutonium production in the UO_2 region, which creates a power imbalance between the blanket and seed regions. In contrast, the micro-heterogeneous assembly (FA-5) maintains constant pin power values throughout the burnup cycle. This shows that the annular duplex fuel design provides safety benefits, as the fuel temperature remains lower than in other cases.

Delayed neutrons

Changes in β_{eff} values with fuel burnup are depicted in Fig. 10. It is expected that (Th-^{233}U-^{235}U)O_2-based assemblies would exhibit lower β_{eff} values at BOC and during burnup compared to all UO_2 assemblies, as ^{233}U has a lower delayed neutron fraction than ^{235}U (Benrhnia et al. 2022). However, despite this expectation, the addition of UO_2 improves the β_{eff} values, as evident in the FA-4 and FA-5 cases compared to FA-3. The decrease in β_{eff} values with burnup observed in FA-4 and FA-5 cases is also observed in UO_2 assemblies, attributed to the production of ^{239}Pu, which has a lower delayed neutron fraction. Nevertheless, it is worth noting that the favorable fuel temperature coefficients discussed below may compensate for the lower delayed neutron fraction.

Reactivity feedback coefficients

Fig. 11 illustrates the FTC values for the investigated assembly models at BOC and EOC. The illustration shows highly negative FTC values, with a greater influence shown in (Th-^{233}U-^{235}U)O_2 assemblies than in UO_2 assemblies, notably at BOC. Additionally, the FTC values for FA-5 followed by FA-4 are more negative than in the other cases. The aforementioned phenomenon can be explained by the increased influence of the Doppler effect, which is amplified when

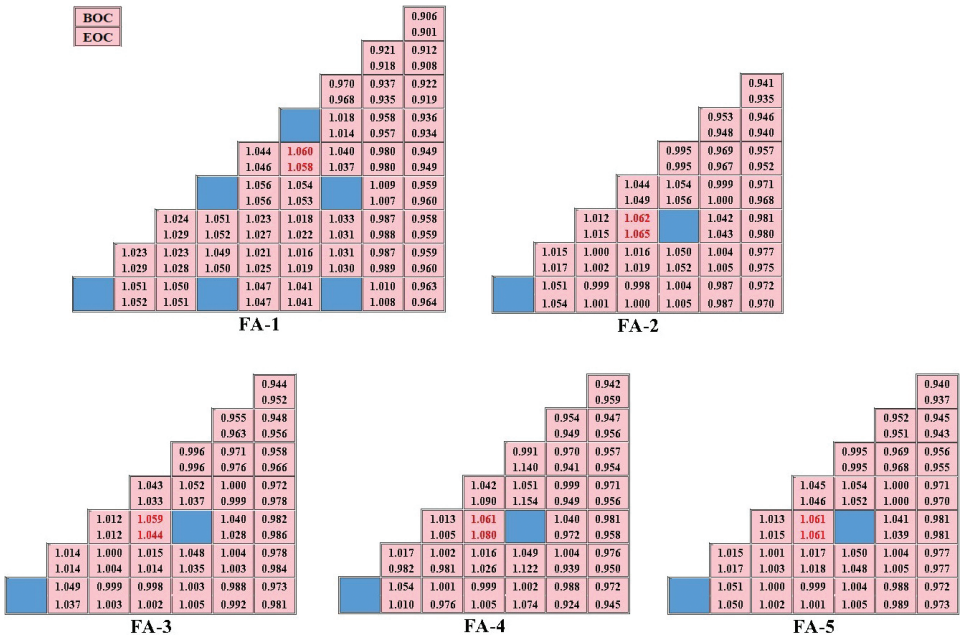

Figure 9. Radial power factors distribution for the studied cases at BOC and EOC.

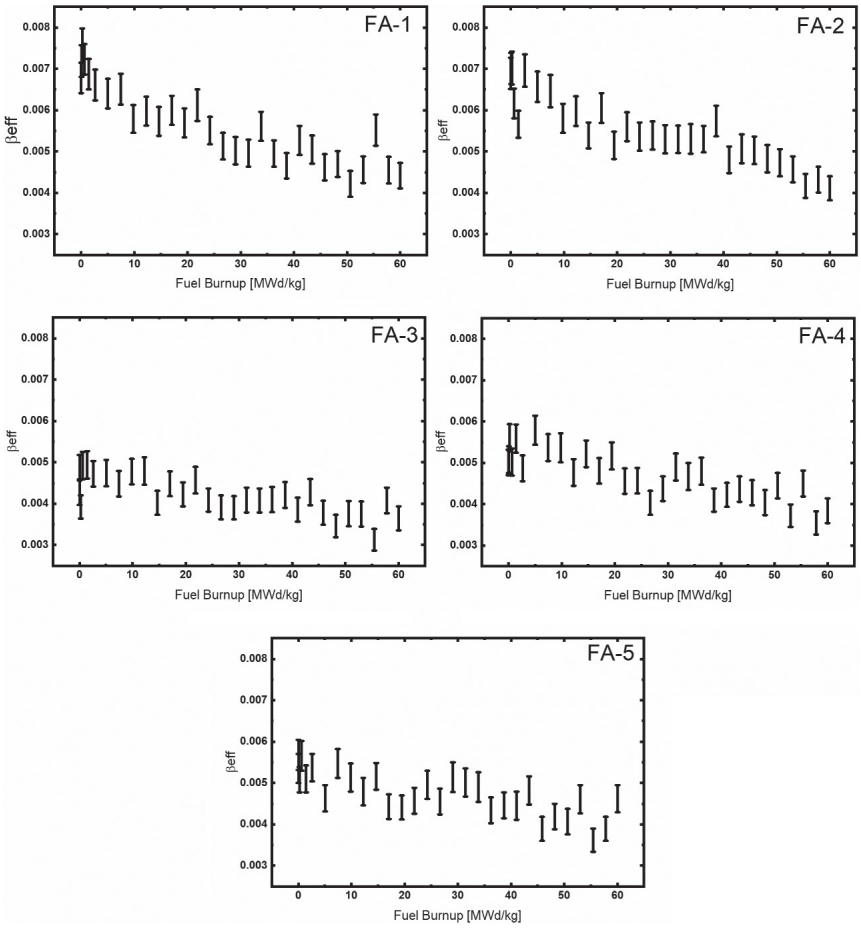

Figure 10. Change in β_{eff} with fuel burnup.

^{232}Th is coupled with ^{238}U. Furthermore, FA-5 has higher negative FTC values than FA-4 because of the increased self-shielding effect inherent in the duplex design.

Previous research studies have shown that using ^{233}U as fissile material in LWRs is a problem because of its potential to produce lower negative or even positive MTC values. This behavior is essentially related to the epithermal fission resonance cross-section of ^{233}U at higher neutron temperatures. However, as shown in Fig. 12, the combination of (Th-^{233}U-^{235}U)O_2 and UO_2 in the assembly improves MTC values in FA-4 and FA-5, with a stronger influence in FA-5 due to the greater self-shielding effect

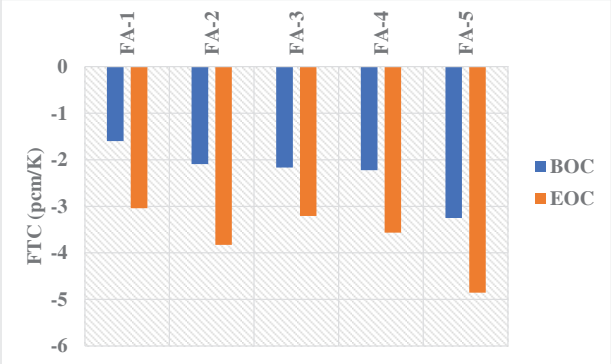

Figure 11. FTC at BOC and EOC for the analyzed assembly cases.

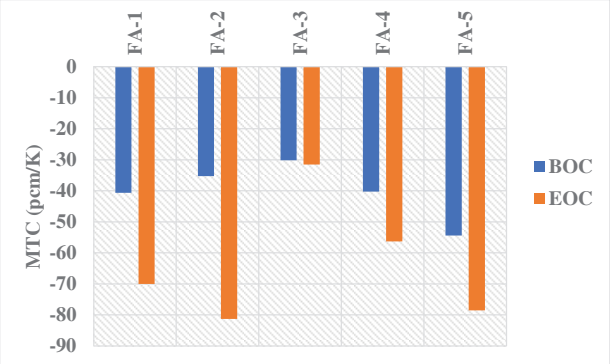

Figure 12. MTC at BOC and EOC for the analyzed assembly cases.

in the duplex design. As a result, based on the obtained FTC and MTC values, FA-5 provides increased stability and control in terms of reactivity feedback coefficients.

Conclusion

The increasing need for thorium-based fuel in PWRs highlights the significance of exploring novel approaches for enhancing nuclear energy performance. One such approach includes combining ^{233}U and ^{235}U with ^{232}Th to create (Th-^{233}U-^{235}U)O_2 fuel. This study aimed to compare the neutronic safety properties of (Th-^{233}U-^{235}U)O_2 fuel in various configurations within a dual-cooled annular 13 × 13 assembly. The study examined criticality, cycle length parameters, power distributions, delayed neutron fractions, and reactivity feedback coefficients for various assembly configurations. The study focused on the simple annular dual-cooled (FA-3), dual-cooled macro-heterogeneous (FA-4), and dual-cooled micro-heterogeneous (FA-4) configurations of (Th-^{233}U-^{235}U)O_2 with UO_2.

The findings show that the micro-heterogeneous configuration exhibits lower criticality and cycle length compared to all-UO_2 assemblies. In contrast, the macro-heterogeneous design has higher criticality and cycle parameters than all-UO_2 assemblies. However, radial pin power data showed minor increases in the macro-heterogeneous assembly, resulting in power imbalances between the seed and blanket regions. On the other hand, the micro-heterogeneous assembly resulted in lower and more consistent pin power than UO_2. However, to fully comprehend the impact of the fuel and assembly change, a thermal-hydraulic analysis is required. The delayed neutron fractions were also estimated, and it was found that using macro-heterogeneous or micro-heterogeneous configurations improved the net delayed neutron fraction.

In terms of moderator and fuel temperature coefficients, simulations indicated that both macro-heterogeneous and micro-heterogeneous assemblies exhibit more negative coefficients compared to the all-(Th-^{233}U-^{235}U)O_2 assembly. Moreover, these designs demonstrated performance equivalent to traditional UO_2 fuel. Therefore, the results of this paper can serve as essential data for safety analyses of PWRs with different fuel options.

Future work

Future work will focus on detailed full-core analyses to further evaluate the safety and performance of thorium-based fuel assemblies in PWRs. Investigating future safety parameters such as reactivity control will be essential. Additionally, studies could address the thermal-hydraulic behavior and burnup characteristics of these assemblies to provide a more comprehensive understanding of their viability.

Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

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