NUCET NUCLEAR ENERGY AND TECHNOLOGY

Nuclear Energy and Technology 11(3): 195–208
DOI 10.3897/nucet.11.163189

Research Article

Preliminary performance evaluation of an LEU+ based dual-cooled duplex UO2-ThO2 fuel for advanced PWR assembly

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Academic editor: Hasan Özdoğan ♦ **Received** 25 June 2025 ♦ **Accepted** 10 September 2025 ♦ **Published** 25 September 2025

Citation: Rifai H, Kabach O, Chakir EM, Sadoune Z (2025) Preliminary performance evaluation of an LEU+ based dual-cooled duplex UO2-ThO2 fuel for advanced PWR assembly. Nuclear Energy and Technology 11(3): 195–208. https://doi.org/10.3897/nucet.11.163189

Abstract

This study presents a comprehensive neutronic analysis of dual-cooled annular duplex fuel assemblies incorporating UO_2-ThO_2 composition for application in advanced pressurized water reactors (PWRs). The investigation employs a 13 × 13 fuel assembly configuration to evaluate the operational enhancement potential of modern reactor systems. The research methodology focuses on the comparative assessment of burnup characteristics across varying uranium enrichment levels, benchmarking the neutronic performance of dual-cooled duplex fuel against conventional solid 17 × 17 and dual-cooled 13 × 13 UO_2 assemblies.

The results demonstrate that the proposed UO_2-ThO_2 dual-cooled duplex fuel configuration with 7 wt.% U-235 achieves discharge burnup equivalent to that of conventional solid UO_2 assemblies. Safety analysis encompasses the quantification of plutonium isotope production and minor actinide generation, revealing that dual-cooled duplex assemblies produce significantly reduced quantities of plutonium isotopes and lower concentrations of minor actinides, including neptunium (Np), americium (Am), and curium (Cm), relative to conventional all-UO_2 assemblies. Reactivity coefficient analysis confirms that both the fuel temperature coefficient (FTC) and moderator temperature coefficient (MTC) maintain consistently negative values throughout the operational cycle. These coefficients not only satisfy but exceed the established safety criteria for PWR operations, thereby demonstrating the enhanced safety margins and operational performance characteristics inherent to the dual-cooled duplex fuel assembly design.

Keywords

Dual-Cooled Duplex Fuel, Neutronic Performance, Safety Parameters, AP1000 Assembly

Introduction

The global pursuit of clean, reliable, and sustainable energy has intensified the focus on nuclear power as a critical component of the decarbonization strategy. However, the nuclear industry faces significant challenges in fuel efficiency, waste management, and economic competitiveness that demand innovative solutions. Traditional nuclear fuel assemblies, comprising solid UO_2 pellets encased in cladding, present notable limitations, including the formation of long-lived transuranic isotopes, particularly plutonium isotopes from U-238 decay, and heat accumulation

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issues in the central region of solid fuel that may necessitate premature fuel removal before achieving complete burnup (Kianpour et al. 2020). Furthermore, while current uranium resources can sustain power generation for over a century at present consumption rates, these resources remain finite and vulnerable to significant price fluctuations, highlighting the need for alternative fuel strategies.

Thorium emerges as an up-and-coming alternative to conventional uranium-based fuel systems, offering several compelling advantages for next-generation reactor designs. Thorium, which is three to four times more prevalent in nature than uranium, can be combined with fissile isotopes to optimize fuel utilization and extend burnup. Furthermore, upon irradiation, thorium produces U-233, a fissile material that can be recycled and reused in subsequent fuel cycles, thereby contributing to a more sustainable nuclear fuel strategy (Kabach et al. 2021b, 2024; Radi et al. 2024). The thorium fuel cycle produces significantly fewer radiotoxic isotopes compared to conventional uranium fuel cycles, enhancing both safety and waste management characteristics. Additionally, thorium-based fuels demonstrate superior proliferation resistance, addressing growing international concerns about nuclear security. However, in its natural form, thorium predominantly consists of Th-232, which is fertile rather than fissile. Consequently, it necessitates an external fissile material to initiate and sustain the chain reaction, with the process eventually being supported by the self-generated U-233. This fundamental characteristic has driven extensive research into optimal configurations for integrating fissile material with thorium dioxide (ThO_2) to achieve effective neutron economy and fuel sustainability (Anantharaman et al. 2008; Balakrishna 2012; Nuclear Energy Agency 2015; de Stefani et al. 2020; Shelley et al. 2022; Mammadzada and Kara 2024).

Among the various thorium utilization strategies, three principal configurations have emerged from the existing body of research: homogeneous mixtures, macro-heterogeneous geometries, and micro-heterogeneous geometries. Previous research indicates that the best utilization of ThO_2 in combination with UO_2 occurs in macro- and micro-heterogeneous geometries (Joo et al. 2004; Baldová 2014; Bourquin 2016; Rabir et al. 2020; Li et al. 2021; Kara and Mammadzada 2025). The micro-heterogeneous duplex pellet design proposed by MIT, particularly the radially separated configuration, has demonstrated excellent promise for enhancing fuel performance (Joo et al. 2004; Li et al. 2021). This configuration promotes a superior ratio of neutron capture in fertile material to neutron absorption in fissile material. This spatial arrangement enables enhanced neutron thermalization in the moderator-rich regions before interaction with the fertile thorium, improving the probability of Th-232 conversion to fissile U-233. However, achieving criticality and maintaining adequate excess reactivity throughout the fuel cycle requires still require increased enrichment levels according to multiple studies (Joo et al. 2004; du Toit and Naicker 2018; Li et al. 2021). The effectiveness of the duplex configuration in large PWRs depends critically on this enhanced fissile loading, which compensates for the initial neutron penalty associated with thorium introduction while enabling the long-term benefits of U-233 breeding and improved fuel utilization (Baldová 2014; Bourquin 2016; Rabir et al. 2020).

In this context, transitioning to Low Enriched Uranium Plus (LEU+) offers a promising solution for the implementation of thorium-based duplex fuel configurations in large commercial PWRs (Shaw and Clarity 2022; Khalefih et al. 2023; Sewell and Hayes 2023). LEU+, as defined by the IAEA, refers to uranium enriched between 5 wt.% and 10 wt.% in U-235, positioned between conventional Low Enriched Uranium (LEU, ≤ 5 wt.% U-235) and High Assay Low Enriched Uranium (HALEU, 10–20 wt.% U-235) as presented in Fig. 1. This intermediate enrichment category provides the additional fissile content necessary to overcome the initial reactivity deficit associated with thorium introduction (Burns et al. 2020; Carlson et al. 2020, 2022; Benrhnia et al. 2022; OECD/NEA 2024).

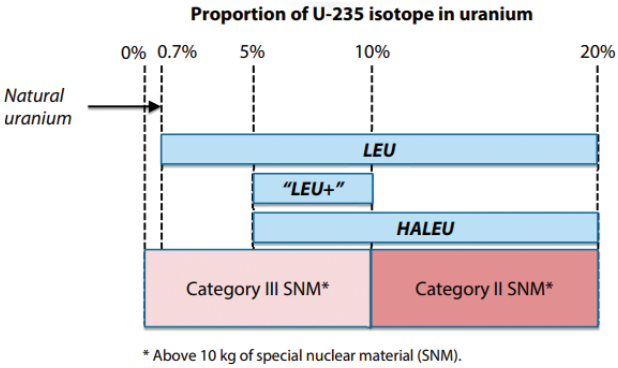

Figure 1. Definition of LEU, LEU+, and HALEU (OECD/NEA 2024).

Furthermore, historical data reveal a continuous progression in average burnup rates, from approximately 15 GWd/ton in the 1970s to over 45 GWd/ton currently, accompanied by a gradual increase in enrichment levels, with recent designs effectively reaching the 5 wt.% enrichment limits (4.95 wt.% nominally in practice) (Kropaczek and Turinsky 1991; Galperin et al. 1997; Joo et al. 2004). This trend illustrates a direct correlation between increased enrichment and improved fuel economy performance. The transition to LEU+ represents the logical continuation of this evolution, enabling even higher burnup rates while optimizing uranium resource utilization. Therefore, the integration of LEU+ with thorium-based duplex configurations presents a particularly attractive approach for large PWRs, as homogeneous thorium-uranium configurations typically require fissile material quantities to achieve standard cycle lengths (Todosow et al. 2005; Xu et al. 2007; Li et al. 2021). However, duplex configurations can achieve superior performance while maintaining enrichment levels within the 5–10 wt.% LEU+ range, making them both economically and regulatorily advantageous for large-scale commercial deployment.

In addition to improving the research direction of fuel materials, innovative fuel designs have also been proposed in recent years. In 2006, MIT proposed an internally and externally double-cooled annular fuel for high-power-density advanced PWRs, which allows a substantial increase in power density by maintaining or improving the safety margin (Fig. 2). This innovative geometry addresses the unique thermal and neutronic challenges that arise from combining enhanced fissile loading with spatially heterogeneous fuel compositions, representing a synergistic solution that maximizes the benefits of both technologies while mitigating their limitations (Kazimi and Hejzlar 2006; Hejzlar and Kazimi 2007; Kazimi 2007; Xu et al. 2007).

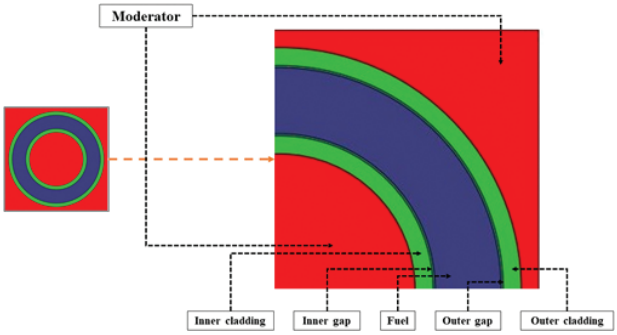

Figure 2. Schematic diagram of a model of a dual-cooled annular fuel unit cell (not to scale).

The implementation of LEU+ duplex fuel configurations creates distinctive thermal management challenges that conventional solid fuel geometries cannot adequately address. The radial separation of ThO_2 and UO_2 regions in duplex pellets generates non-uniform heat generation profiles, with the UO_2 regions typically exhibiting higher power densities due to their fissile content, while the ThO_2 regions initially contribute minimal thermal output but gradually increase in importance as U-233 builds up through neutron capture. This heterogeneous power distribution, combined with the higher overall enrichment levels characteristic of LEU+ fuel, creates localized thermal hot spots that can exceed the heat removal capabilities of conventional single-surface cooling systems. The dual-cooled annular design specifically addresses this challenge by providing enhanced heat removal capacity through both internal and external cooling channels, enabling more uniform temperature distributions across the radially heterogeneous fuel composition.

Furthermore, the geometric advantages of the annular configuration provide critical neutronic benefits that are particularly important for LEU+ duplex fuel systems. The central void region in annular fuel pellets allows for increased moderator volume within the fuel assembly, enhancing neutron thermalization efficiency. This improved moderation is especially beneficial for duplex fuel configurations, as thermal neutrons are essential for efficient Th-232 absorption and subsequent conversion to fissile U-233 (El Kheiri et al. 2023b, 2023a). The enhanced neutron thermalization provided by the annular geometry effectively increases the probability of productive neutron capture in the fertile thorium regions, thereby improving the overall conversion ratio and fuel utilization efficiency. Additionally, the dual cooling channels enable optimization of coolant flow distributions that can be tailored to the specific thermal characteristics of each duplex region, providing design flexibility to manage the evolving thermal profiles as U-233 accumulates and the fuel composition changes throughout the irradiation cycle (Alam et al. 2019a, 2019b).

The necessity of dual-cooled annular design for LEU+ duplex configurations also stems from power density considerations and safety margin requirements. The combination of increased enrichment levels and heterogeneous fuel composition tends to create more pronounced power peaking factors compared to conventional homogeneous fuel (Hejzlar and Kazimi 2007; Kianpour et al. 2020; Nejad and Ansarifar 2020). The dual cooling capability of annular fuel enables higher linear heat generation rates while maintaining acceptable fuel centerline temperatures, effectively providing the thermal margin necessary to accommodate the power peaks associated with LEU+ duplex fuel without compromising safety limits. The enhanced heat transfer coefficient achieved through internal surface roughening and the doubled heat transfer surface area collectively enable the system to operate at higher power densities while maintaining lower peak fuel temperatures than would be possible with conventional cooling arrangements. This thermal performance enhancement is crucial for realizing the full potential of LEU+ duplex fuel systems in large commercial PWRs, where economic competitiveness depends heavily on achieving high power outputs with extended fuel cycles.

While previous studies have individually examined dual-cooled annular fuel, thorium-based duplex configurations, and LEU+ applications, comprehensive neutronic analysis of the integrated LEU+ UO_2-ThO_2 dual-cooled annular duplex fuel system for large PWR applications remains limited. The proposed research addresses this critical knowledge gap by investigating the synergistic effects of combining these three advanced fuel technologies. This innovative fuel design promises to reduce waste generation through thorium utilization, enhance safety margins through improved thermal performance, and superior proliferation resistance while maintaining compatibility with existing large PWRs (Bouassa et al. 2023, 2024; El Kheiri et al. 2023a, 2023b; El Banni et al. 2024; Radi et al. 2024; Rifai et al. 2025). The primary objectives of this investigation are to: (1) conduct a comprehensive neutronic analysis of LEU+ enriched dual-cooled UO_2-ThO_2 fuel assemblies, (2) evaluate burnup characteristics and isotopic evolution throughout fuel cycles, (3) assess main safety parameters such as reactivity coefficients and power peaking factors, and (4) compare performance against conventional fuel assemblies.

Methodology and input parameters

Reference assembly model

The AP1000 reactor assembly, developed by the Westinghouse Electric Company in the United States, serves as a foundational benchmark for this study due to its advanced design and well-documented performance. This Generation III+ PWR possesses a nominal thermal power output of 3400 MWth and comprises 157 fuel assemblies (Westinghouse Electric Company LLC 2004). These assemblies feature an active fuel length of 4.27 m and a side length of 0.21417 m. Each assembly contains 264 fuel rod positions arranged in a 17 × 17 grid, along with 25 guide tubes designed to accommodate instrumentation and burnable absorber rods. The fuel rods themselves are composed of UO_2 fuel, with enrichments ranging from 2.35 wt.% to 4.95 wt.% at various locations within the core (Bouassa et al. 2023). The UO_2 fuel pellets have a diameter of 0.81915 cm, with a helium gap of 0.0082 cm thickness separating them from the cladding. ZIRLO™ cladding is employed, with a thickness of 0.057 cm. The fuel rods are arranged with a pitch of 1.26 cm. This reliable design has also significantly influenced subsequent advanced small modular reactor (SMR) designs, such as the AP300 reactor (IAEA 2024), rendering it an ideal reference point for this study. The key characteristic parameters are summarized in Table 1. Table 2 presents data for the other structural materials. These data were meticulously collected from open literature sources (Westinghouse Electric Company LLC 2004; Laranjo de Stefani et al. 2019).

Table 1. Geometrical Specification of the study assembly (Benrhnia et al. 2022)

Parameters	Value
Rod array	17 × 17
Number of fuel rods	264
Number of guide tubes	24
Assembly pitch (cm)	21.5
Rod lattice pitch (cm)	1.26
Fuel outer radius (cm)	0.409575
Helium gap outer radius (cm)	0.417750
Clad outer radius (cm)	0.474750
Guide tube inner radius (cm)	0.56
Guide tube outer radius (cm)	0.60

Table 2. Data for the examined fuel models and other material specifications (Benrhnia et al. 2022)

Zone	Parameter	Value
Fuel cladding	Cladding material	Zirlo™
	Cladding density	6.50 g/cm^3
Gap	Gap material	Helium
	Gap density	1.2049E-02 g/cm^3
Moderator	Moderator material	Light water
	Moderator density at 600 K	0.654 g/cm^3
	Moderator density at 575 K	0.718 g/cm^3
Guide tube cladding	Cladding material	Stainless Steel type 304
	Cladding density	8.03 g/cm^3

Investigated fuels and proposed assembly sizing

In this study, we examine three distinct assembly designs and various fuel pellet types, all of which operate under the same conditions. The first fuel type used is standard solid UO_2 fuel, which serves as a baseline for our parametric comparisons. The second fuel type is an annular UO_2 configuration, while the third is our innovative proposal – the dual-cooled annular duplex fuel, UO_2-ThO_2 (see Fig. 3), designed for the AP1000 reactor and featuring differing U-235 enrichments. As previously explored in prior research, the use of a dual-cooled fuel pellet design presents an attractive solution (Benrhnia et al. 2022; Bouassa et al. 2023; El Kheiri et al. 2023a). Such designs enhance fuel rod cooling, reducing the risk of overheating and cladding damage, thereby improving reactor safety and accident prevention. The dual-cooled assembly's design allows for greater power density, and the superior cooling characteristics of the fuel rods result in longer fuel lifetime, lowering the risk of overheating-related damage. In our current study, we propose using a 13 × 13 annular fuel assembly loaded with our proposed dual-cooled fuel (Hejzlar and Kazimi 2007; YANG et al. 2009; Shin et al. 2012; Deng et al. 2016). Fig. 4 depicts the geometric configuration of this assembly.

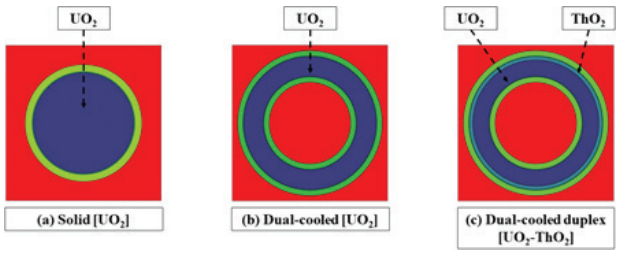

Figure 3. Cross-sectional views of **a.** a solid fuel pin; **b.** a dual-cooled fuel pin, and **c.** A dual-cooled duplex fuel pin.

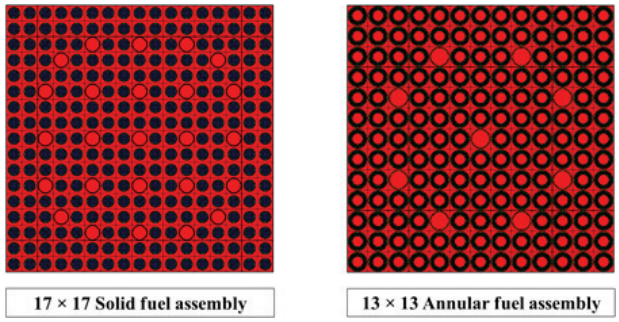

Figure 4. Geometric structure comparison of solid fuel and annular fuel assemblies.

Our motivation for exploring varying U-235 enrichments for the dual-cooled duplex UO_2-ThO_2 design stems from the unique requirements of a dual-cooled assembly. It requires a slightly higher fissile content to achieve adequate criticality at the beginning of the cycle (BOC), compensating for the presence of Th-232, which has significantly higher thermal absorption (approximately three times that of U-238) (Benrhnia et al. 2022). Furthermore, this strategy is used to achieve a discharge burnup equivalent to standard solid UO_2, while accounting for neutron leakage

from the assemblies. Table 3 summarizes the geometric specifications of the study's assemblies, while Table 4 provides detailed descriptions of the various models analyzed, including the suggested reference model. Table 5 shows the isotopic compositions at BOC for the investigated fuels.

Table 3. Geometrical Specification of the study assemblies (Bouassa et al. 2023; El Kheiri et al. 2023a)

Parameter	Solid	Annular	
Fuel	UO_2	UO_2	UO_2-ThO_2
Rod array	17 × 17	13 × 13	13 × 13
Number of fuel rods	264	160	160
Number of guide tubes	24	9	9
Assembly pitch (cm)	21.5	21.5	21.5
Rod lattice pitch (cm)	1.26	1.65	1.65
Inner clad inner radius (cm)	-	0.43165	0.43165
Inner clad outer radius (cm)	-	0.48865	0.48865
Inner helium gap outer radius (cm)	-	0.49485	0.49485
Fuel outer radius (cm)	0.409575	0.70515	0.70515
Outer helium gap outer radius (cm)	0.417750	0.71135	0.71135
Outer clad outer radius (cm)	0.474750	0.76835	0.76835
UO_2 area per fuel rod (cm^2)	0.527	0.793	0.634
ThO_2 area per fuel rod (cm^2)	-	-	0.159
Guide tube inner radius (cm)	0.56	0.71	0.71
Guide tube outer radius (cm)	0.60	0.77	0.77

Table 4. Properties of the analyzed fuel types

Case Name	Configuration
Reference	Solid (17 × 17) with UO_2 fuel [4.95 wt.% (U-235)]
Model 1	Dual-cooled (13 × 13) with UO_2 [4.95 wt.% (U-235)]
Model 2	Dual-cooled duplex (13 × 13) with UO_2-ThO_2 fuel [4.95 wt.% (U-235)]
Model 3	Dual-cooled duplex (13 × 13) with UO_2-ThO_2 fuel [6 wt.% (U-235)]
Model 4	Dual-cooled duplex (13 × 13) with UO_2-ThO_2 fuel [7 wt.% (U-235)]

Table 5. Isotopic compositions at BOC for the investigated fuels

Fuel composition	Nuclide	Atomic concentration (atoms/barn.cm)
ThO_2	Th-232	0.02198942
	O-16	0.04397885
UO_2 [4.95 wt.% (U-235)]	U-235	0.00117730
	U-238	0.02232158
	O-16	0.04699776
UO_2 [6 wt.% (U-235)]	U-235	0.00142701
	U-238	0.02207465
	O-16	0.04700331
UO_2 [7 wt.% (U-235)]	U-235	0.00166490
	U-238	0.02183940
	O-16	0.04700860

Code descriptions and simulation strategy

The simulation strategy employed utilized the latest version of the DRAGON code, in conjunction with nuclear data from the ENDF/B-VIII rel. 0 library. DRAGON is a specialized software tool designed to solve the Boltzmann neutron transport equation in two or three dimensions for a diverse array of reactor geometries (Hébert 2008; Marleau et al. 2021; Benrhnia et al. 2022; El Kheiri et al. 2023a). It offers various methodologies to solve the neutron transport equation by enforcing various numerical and estimation methods, including the characteristics method, the discrete ordinates method, and the collision probability method. These methodologies facilitate neutron flux solutions and additional computations, such as depletion calculations. The code also encompasses features such as microscopic cross-section interpolation, resonance self-shielding, multigroup neutron flux calculations, and the editing of condensed nuclear properties for reactor simulations, such as macroscopic cross-section, neutron fluxes... (Marleau et al. 2021; Khoshahval 2022; Paradis et al. 2022; El Kheiri et al. 2023a).

In this study, specific modules within DRAGON were selected to conduct precise analyses of assembly behaviors, including neutron flux, power density, and burnup. Geometric modeling was executed using the GEO module, employing a CARCEL-type geometry for the Cartesian mesh. Reflective boundary conditions were applied to the assembly's outer surfaces. Simulations for depletion characteristics were conducted at the temperatures listed in Table 2 to evaluate all potential reactivity coefficients pertinent to reactor safety, while maintaining a constant power density of 40 kW/kg in all scenarios. It is noteworthy that calculations using DRAGON assumed no neutron leakage. However, typical large PWRs, such as the AP1000, exhibit approximately 3% neutron leakage (Hernandez and Brown 2020; Rifai et al. 2025). Consequently, the single-batch discharge burnup in this study refers to the burnup period during which K_{inf} reaches 1.03. Finally, the simulation strategy was validated against results from well-documented 17 × 17 assemblies fueled with UO_2 enriched to 2.35 wt.% in a cold zero-power state, demonstrating favorable agreement with reference values, as presented in Table 6.

Results and discussion

Depletion characteristic

Fig. 5 and Table 7 illustrate the evolution of the K_{inf} as a function of fuel burnup for the five evaluated fuel configurations, with all models demonstrating the expected monotonic decrease in reactivity due to fissile depletion and fission product accumulation. The horizontal dashed line at K_{inf} = 1.03 represents the critical threshold. The reference solid UO_2 fuel exhibits the highest initial reactivity with a K_{inf} of 1.390875 at beginning of cycle (BOC), achieving a single-batch discharge burnup of 37.16 GWd/ton and an end of cycle (EOC) K_{inf} of 0.892616.

Table 6. K_{inf} value obtained at cold zero power state for the DRAGON simulation verification

Serpent (Laranjo de Stefani et al. 2019)	NODAL 3 (Pinem et al. 2018)	DRAGON (Khoshahval 2022)	DRAGON (This work)
1.331	1.330	1.317	1.302

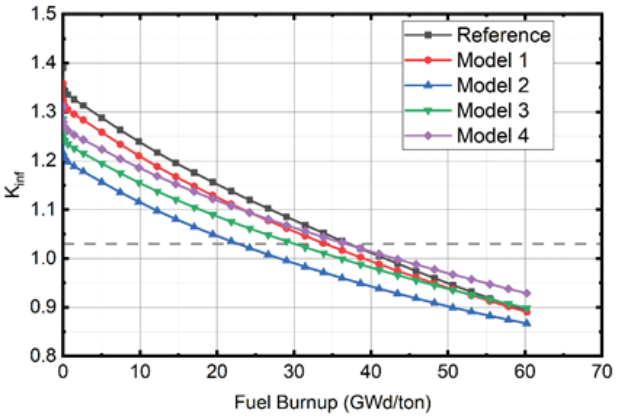

Figure 5. K_{inf} at different fuel burnups for the examined models. The horizontal dashed line represents K_{inf} = 1.03.

Table 7. K_{inf} at BOC, EOC, and single-batch discharge burnup

Case Name	K_{inf} (BOC)	K_{inf} (EOC)	Single-batch discharge burnup (GWd/ton)
Reference	1.390875	0.892616	37.16
Model 1	1.358797	0.890739	34.05
Model 2	1.249630	0.867146	22.73
Model 3	1.285318	0.898760	30.08
Model 4	1.311102	0.928976	37.69

Model 1, although it employs the same U-235 enrichment level (4.95 wt.%) and UO_2 fuel composition as the reference, shows a lower initial K_{inf} of 1.358797 and EOC K_{inf} of 0.890739. This reduction is attributed to the dual-cooled geometry (13×13) versus the solid configuration (17×17) of the reference, along with the presence of a moderator-filled annular gap, which enhances neutron moderation but also increases parasitic absorption. Additionally, the decreased fuel volume inherent to the annular design further limits reactivity, resulting in a discharge burnup of 34.05 GWd/ton.

Models 2, 3, and 4, which incorporate UO_2-ThO_2 fuel containing Th-232 as a fertile material, show significant reductions in initial K_{inf} due to thorium's high thermal neutron absorption cross-section (approximately three times that of U-238). Model 2, with the same enrichment as the reference (4.95 wt.% U-235), exhibits a dramatically reduced initial K_{inf} of 1.249630 and the poorest EOC performance with K_{inf} of 0.867146, achieving only 22.73 GWd/ton discharge burnup - the lowest among all configurations. Model 3, with 6 wt.% U-235 enrichment partially compensates for thorium absorption with a moderate initial K_{inf} of 1.285318 and EOC K_{inf} of 0.898760, achieving a discharge burnup of 30.08 GWd/ton. Model 4, using 7 wt.% U-235 demonstrates superior performance throughout the cycle despite starting with a lower K_{inf} of 1.311102 compared to the reference. Remarkably, Model 4 achieves the highest EOC value of 0.928976 and the highest single-batch discharge burnup of 37.69 GWd/ton, indicating that sufficient U-235 enrichment can effectively overcome thorium's parasitic absorption while benefiting from its breeding potential to U-233.

When extended to multi-batch core configurations using the Linear Reactivity Model (LRM), presented in equations 1 and 2, with a three-batch refueling scheme and 3% neutron leakage assumption, the performance characteristics translate into significant practical implications for reactor operations (Driscoll et al. 1991; Hedayat 2017; Burns et al. 2020; El Banni et al. 2024; Kabach et al. 2024). As presented in Fig. 6, the reference case establishes the baseline performance with a cycle burnup of 55.50 GWd/ton, discharge burnup of 166.50 GWd/ton, and cycle length of 462.50 EFPDs. Model 1 demonstrates reasonably competitive performance despite its geometric modifications, achieving a cycle burnup of 51.07 GWd/ton (92% of reference), discharge burnup of 153.21 GWd/ton (92% of reference), and cycle length of 425.58 EFPDs (92% of reference), indicating that the dual-cooled annular geometry maintains acceptable fuel utilization efficiency. Model 2 confirms the significant penalty associated with thorium incorporation at low enrichment levels, exhibiting substantially reduced performance with a cycle burnup of 34.09 GWd/ton (61% of reference), discharge burnup of 102.28 GWd/ton (61% of reference), and cycle length of 284.12 EFPDs (61% of reference), making this configuration economically unviable for practical applications. Model 3 shows marked improvement over Model 2 through increased enrichment, achieving a cycle burnup of 45.11 GWd/ton (81% of reference), discharge burnup of 135.34 GWd/ton (81% of reference), and cycle length of 375.94 EFPDs (81% of reference), demonstrating the effectiveness of enrichment optimization in thorium-based fuels. Most significantly, Model 4 not only matches but exceeds reference performance across all metrics, achieving a cycle burnup of 56.54 GWd/ton (102% of reference), discharge burnup of 169.62 GWd/ton (102% of reference), and cycle length of 471.17 EFPDs (102% of reference). This superior performance validates that the optimized

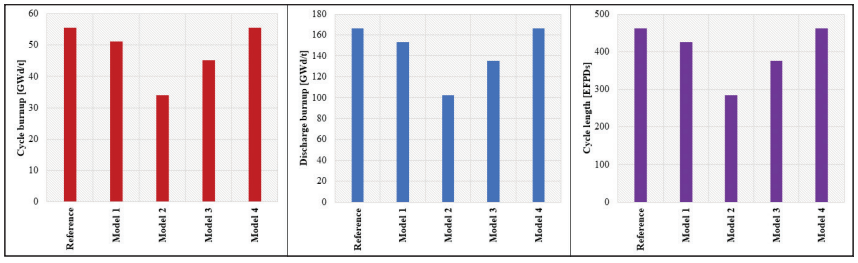

Figure 6. Comparison of cycle burnup, discharge burnup, and cycle length in EFPDs (effective full-power days) for a 3-batch refueling scheme.

UO_2-ThO_2 annular fuel design with 7 wt.% U-235 enrichment represents a viable option for enhanced fuel utilization and extended cycle operation.

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

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

where $B_{Discharge}$ is the discharge burnup for an n-batch core, B_{SC} is he single-batch discharge burnup, N_{RB} represents the number of fuel batches in the assembly shuffling scheme, T_{cycle} represents the cycle length in EFPDs (effective full power days), and ρ is the considered specific power density.

Evolution of the main actinides with burnup

The comprehensive actinide evolution data reveal fundamental differences in neutron physics operating across the investigated fuel configurations. This investigation explores how the transition from conventional solid uranium fuel to UO_2-ThO_2 designs creates entirely new neutron capture pathways and competitive mechanisms that reshape the actinide inventory throughout the fuel burnup. Fig. 7 presents the variation in the atomic density of key fissile and fertile isotopes during burnup for all the models examined in our study. It is well-established from extensive experience with UO_2 LWRs that certain minor actinides, such as Neptunium,

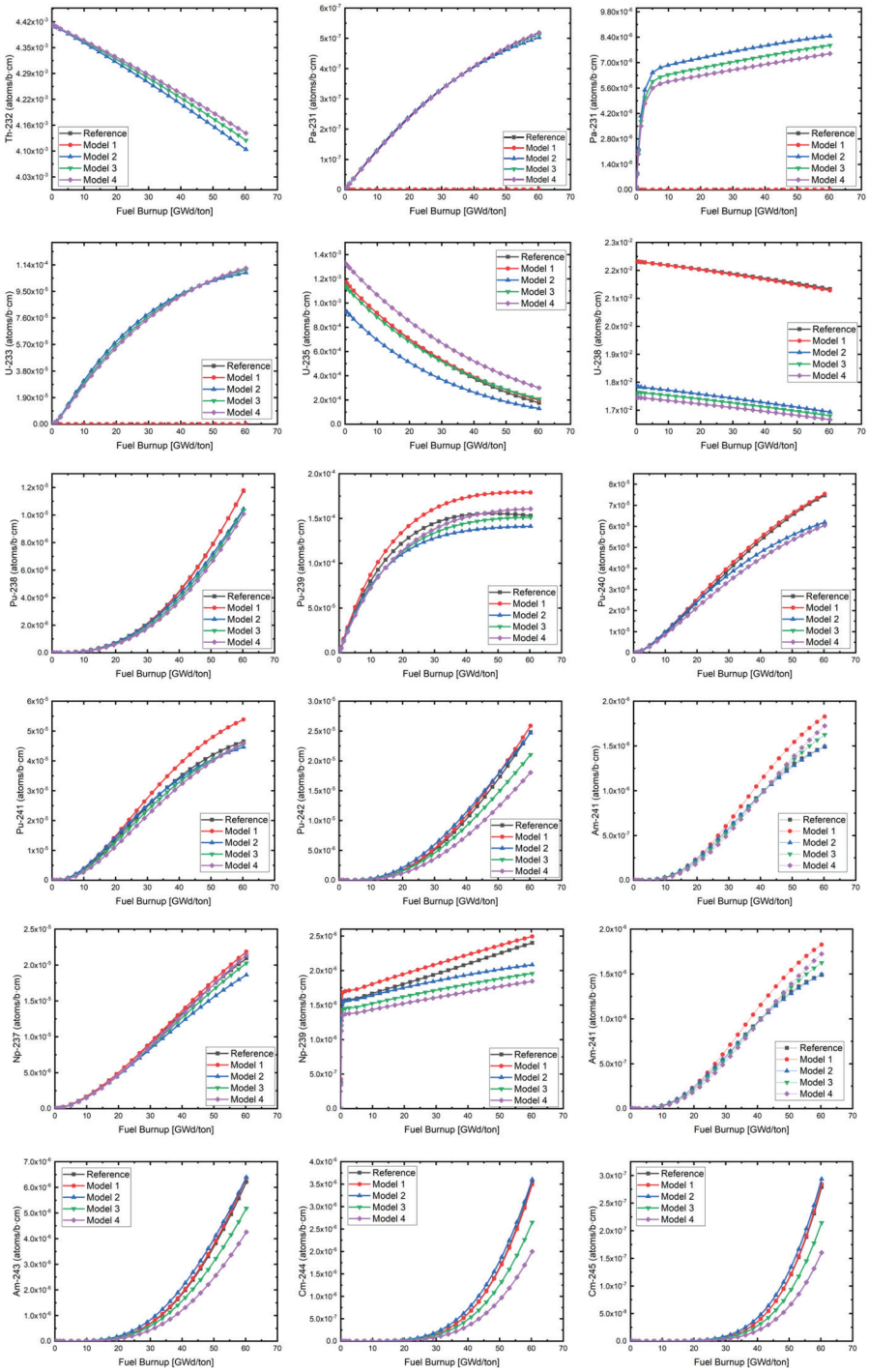

Figure 7. Variation in the atomic density of key fissile and fertile isotopes during burnup.

Americium, and Curium isotopes, assume significance as they are produced after the absorption of neutrons by U-238. However, it's essential to note that Protactinium isotopes play a crucial role as minor actinides in ThO_2-fueled reactors (Benrhnia et al. 2022; Bouassa et al. 2023, 2024; Lkouz et al. 2023; El Banni et al. 2024; Kabach et al. 2024; Radi et al. 2024; Rifai et al. 2025; Luhaib et al. 2026).

The investigation into uranium isotope behavior reveals complex interactions between initial fissile depletion and in-situ breeding processes. While U-235 follows predictable depletion patterns in all configurations, the rate of consumption varies significantly based on neutron spectrum modifications and capture competition. The reference solid fuel demonstrates the most aggressive U-235 depletion due to its hard neutron spectrum and lack of competing fertile materials. However, Model 1's dual-cooled geometry moderates this depletion despite identical initial enrichment, indicating that the annular configuration and enhanced moderation create more favorable neutron economics. The duplex configurations reveal a fundamentally different fissile material strategy. Despite starting with reduced U-235 inventories due to thorium dilution, these systems develop a dynamic fissile balance through U-233 breeding. The investigation shows that U-233 production accelerates rapidly during the first half of the cycle, reaching maximum generation rates around 20–30 GWd/ton before plateauing as neutron competition intensifies. Models 3 and 4, with higher initial enrichments, demonstrate superior breeding performance, suggesting that the initial fission rate directly influences the thermal neutron flux available for Th-232 conversion. The U-238 evolution investigation reveals striking differences between uranium-only and duplex systems. While the reference and Model 1 show steady consumption primarily through neutron capture leading to plutonium chains, the duplex models exhibit fundamentally altered behavior. The reduced U-238 inventory creates less competition for thermal neutrons, while thorium provides alternative absorption pathways that redirect neutron capture away from uranium isotopes.

The thorium cycle investigation demonstrates the complex interplay between fertile material utilization and neutron spectrum management. Th-232 concentrations remain essentially constant throughout all cycles, confirming its role as a neutron absorber rather than a consumable fuel. However, the investigation into conversion products reveals significant variations in utilization efficiency across different enrichment levels. Pa-231 formation serves as a sensitive indicator of thorium cycle activation. The investigation shows that protactinium buildup correlates directly with the thermal neutron flux intensity in the thorium blanket region. Higher enrichment cores generate more intensive neutron sources, leading to enhanced Pa-231 production and correspondingly higher thorium utilization rates.

The plutonium evolution investigation reveals one of the most significant advantages of duplex fuel configurations: systematic suppression of plutonium production while maintaining a favorable isotopic composition. The reference case follows classical Pu-239 buildup from U-238 capture, reaching peak concentrations. Duplex configurations demonstrate remarkable plutonium suppression through multiple mechanisms. The investigation shows that reduced U-238 inventory directly limits the source material for plutonium production. Simultaneously, thorium competes for thermal neutrons that would otherwise be captured by U-238. The combined effect reduces Pu-239 production by 15–25% compared to conventional. The investigation into higher-order plutonium isotopes reveals even more dramatic suppression effects. Pu-240 production shows proportional reduction with the overall plutonium inventory.

The minor actinide investigation reveals complex neutron capture competition effects that fundamentally alter transplutonium element formation. Np-237 production shows the expected correlation with U-238 content, but the investigation uncovers additional spectral effects that modify production rates beyond simple inventory changes. The duplex configurations consistently suppress Np-237 formation through dual mechanisms: reduced U-238 source material and neutron capture competition from thorium. However, the investigation reveals that spectral hardening in the thorium blanket region creates preferential (n,2n) reactions that further suppress neptunium production chains. This effect becomes more pronounced with increasing enrichment, as evidenced by the progressive Np-237 reduction from Model 2 through Model 4. Am-241 evolution presents a more complex investigation challenge due to its dual production pathways through both Pu-241 decay and direct neutron capture. The duplex configurations show modified americium behavior reflecting altered plutonium production and different neutron capture destruction rates. The investigation suggests that while initial Am-241 production decreases due to reduced plutonium inventory, the destruction rate through neutron capture also decreases due to spectral modifications, creating partially compensating effects. The curium isotope investigation reveals the most dramatic minor actinide suppression in duplex fuels. Both Cm-244 and Cm-245 show a substantial reduction compared to conventional designs. The investigation indicates that the multiple neutron capture chains required for curium production are particularly sensitive to neutron capture competition from thorium. The fertile ThO2 region effectively "starves" these higher-order formation pathways, reducing curium production by 40–60% depending on the isotope and enrichment level.

Most significant fission products

The types and concentrations of fission products depend on the type and concentration of the fissile nuclides used in the fuel. The significance of fission products arises from their vital effect either on the reactor operation or on the environment. Xenon-135 and samarium-149 are considered the most significant fission products because of their huge thermal neutron capture cross-section. The thermal capture cross-section values of Xe-135 and Sm-149 are (2,650,000 ± 110,000) and (40,140 ± 600) barn, respectively (Galahom 2018; Galahom and Ibrahim 2022). This investigation explores how the different fuel configurations fundamentally alter the production, accumulation, and neutron absorption characteristics of these critical fission products. Fig. 8 shows Xe-135 and Sm-149 evolution with burnup.

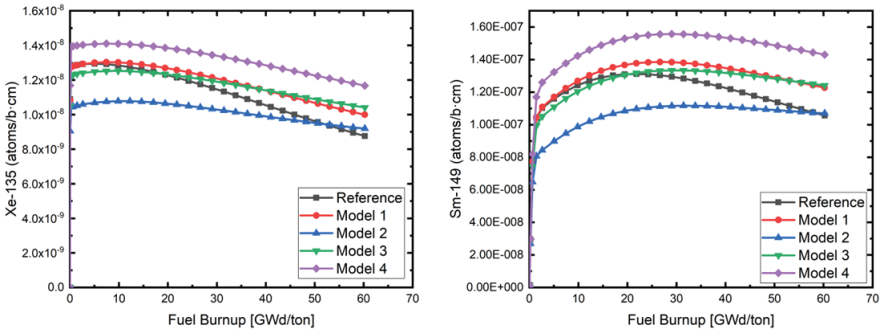

Figure 8. Xe-135 and Sm-149 evolution with burnup.

The Xe-135 evolution investigation reveals complex production and destruction mechanisms that vary significantly across fuel configurations. The investigation demonstrates that Xe-135 concentrations are governed by competing production pathways: direct fission yield (~6.1%) and decay chain production from I-135 (fission yield ~6.4%), balanced against destruction through neutron absorption and radioactive decay. The duplex configurations exhibit fundamentally different xenon behavior compared to conventional uranium fuels. Model 2 demonstrates the most favorable xenon characteristics, maintaining concentrations 15–25% lower than the reference case throughout the fuel cycle. At 30 GWd/ton burnup, Model 2 achieves Xe-135 concentrations of only 1.046×10^{-8} atoms/b·cm compared to the reference value of 1.287×10^{-8} atoms/b·cm. The investigation into high-enrichment duplex models reveals a complex relationship between initial enrichment and xenon behavior. Model 3 maintains Xe-135 concentrations slightly below the reference case early in life (1.231×10^{-8} atoms/b·cm vs 1.287×10^{-8} atoms/b·cm at 30 GWd/ton), while Model 4 shows the highest concentrations across all configurations (1.395×10^{-8} atoms/b·cm at 30 GWd/ton). This investigation uncovers that higher enrichments create competing effects: increased fission rates boost xenon production, but enhanced neutron flux also increases destruction rates through (n,γ) capture. The investigation reveals that beyond 6% enrichment, production effects dominate, leading to higher equilibrium xenon levels despite improved neutron economy.

The Sm-149 investigation reveals fundamentally different accumulation mechanisms compared to xenon poisoning. Unlike xenon's equilibrium-driven behavior, samarium accumulates steadily throughout the fuel cycle as a stable fission product with minimal destruction pathways. The duplex fuel investigation demonstrates systematic samarium suppression across thorium-bearing configurations, with Model 2 achieving the most significant reduction. At 30 GWd/ton burnup, Model 2 exhibits Sm-149 concentrations of 6.483×10^{-8} atoms/b·cm compared to the reference value of 7.730×10^{-8} atoms/b·cm, representing a 16% reduction. This investigation reveals multiple contributing mechanisms: reduced overall fission product yield due to U-233's different yield distribution, modified neutron spectrum affecting precursor decay chains, and altered fission rates due to the dual-cooled geometry. Conversely, Model 4 shows increased samarium accumulation (8.193×10^{-8} atoms/b·cm at 30 GWd/ton), indicating that the benefits of spectral modifications are eventually overcome by higher absolute fission rates at elevated enrichments.

Comparison of neutron spectra

The investigation of neutron spectra associated with various fuel configurations is a critical aspect of this study. The neutron spectrum represents a complex interaction of neutron thermalization and absorption phenomena (Alam et al. 2019a). Fig. 9 illustrates the neutron spectra at the BOC and EOC. It is noteworthy that the DRAGON code is capable of generating diverse neutron flux types; in this study, we derived our neutron spectra from integrated neutron fluxes within specific energy groups and their corresponding lethargy width. For this purpose, we employed a comprehensive data library comprising 172 energy groups, which justifies the use of the term "flux per unit lethargy" in this paper (El Kheiri et al. 2023a).

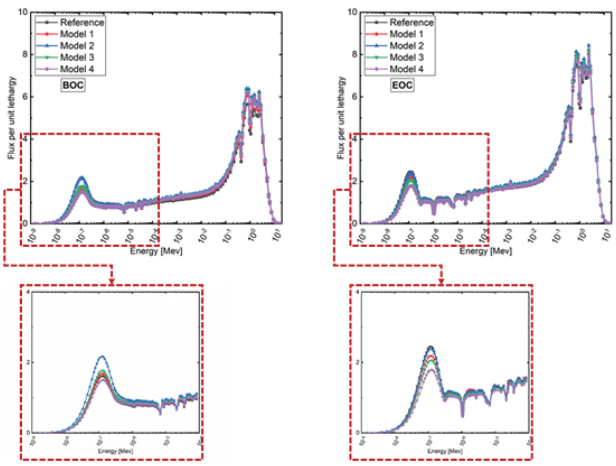

Figure 9. Neutron spectra for the analyzed fuel model at BOC and EOC.

The figures obtained from our analysis provide significant insights. They indicate a harder neutron spectrum throughout the entire burnup cycle, particularly in the case of dual-cooled duplex fuel with higher U-235 enrichment (Model 4 followed by Model 2). The staggered thermal peaks, which are notably lower for Model 4, exemplify this phenomenon. The reference case also exhibits a harder neutron spectrum, with this effect being most pronounced at BOC when compared to other cases. This outcome was anticipated, as the higher U-235 enrichment in Model 4

and the larger fuel volume in the reference configuration contribute to increased thermal neutron absorption, resulting in a hardened neutron spectrum. In contrast to BOC, the neutron spectra at EOC display a more pronounced peak in the thermal neutron region. This is primarily attributed to a reduction in fissile content within the fuel due to depletion. However, in comparison to the reference, the accumulation of fissile materials such as U-233 and Pu-239 results in a harder neutron spectrum for the dual-cooled and duplex fuel configurations. Both U-233 and Pu-239 exhibit the highest neutron production per neutron absorbed within the fuel, which contributes to the hardened neutron spectrum phenomenon.

Radial power peaking factor

Power Peaking Factors (PPFs) are essential neutronic parameters within an assembly that require careful consideration when altering fuel composition, enrichment levels, or assembly configurations. Precisely, radial PPFs are determined by dividing the average power by the power generated in each fuel rod within the assembly (Zhiwen Xu 2003; Kazimi and Hejzlar 2006; El Banni et al. 2024; Kabach et al. 2024; Radi et al. 2024). Fig. 10 illustrates the radial PPFs for both the reference and the analyzed fuel models. This analysis is concentrated on a single octant assembly, with the hot channels indicated by red numbers. The figure reveals that the proposed assembly cases exhibit slightly higher maximum PPF values compared to the reference cases. This observation can be attributed to two primary factors. Firstly, the reduction in the number of fuel pins in a dual-cooled assembly leads to an increased total amount of fuel within an individual pin, alongside the utilized enrichment level. Consequently, Model 4, with higher enrichment, exhibits the highest PPF value, whereas Model 2, with lower enrichment, shows the lowest PPF value. Secondly, alterations in the fuel mixture result in changes in neutron absorption rates. Enhanced absorption in the fuel can lead to elevated PPF values. Nonetheless, despite the higher PPF values in the proposed assembly cases, they remain within an acceptable range. According to references such as (Zhiwen Xu 2003; Kazimi and Hejzlar 2006), the acceptable PPF threshold for a Westinghouse PWR is typically around 1.65.

Main safety parameters

Fig. 11 presents the FTC and MTC values at BOC for the various fuel models under consideration. The FTC and MTC are critical safety indicators for nuclear reactors, necessitating their maintenance at negative values to ensure reactor safety and stability. In this study, we calculated the FTC and MTC by introducing temperature perturbations and employing the formula presented in equations 3 and 4 (Kabach 2021; Kabach et al. 2021a, 2024; Radi et al. 2024).

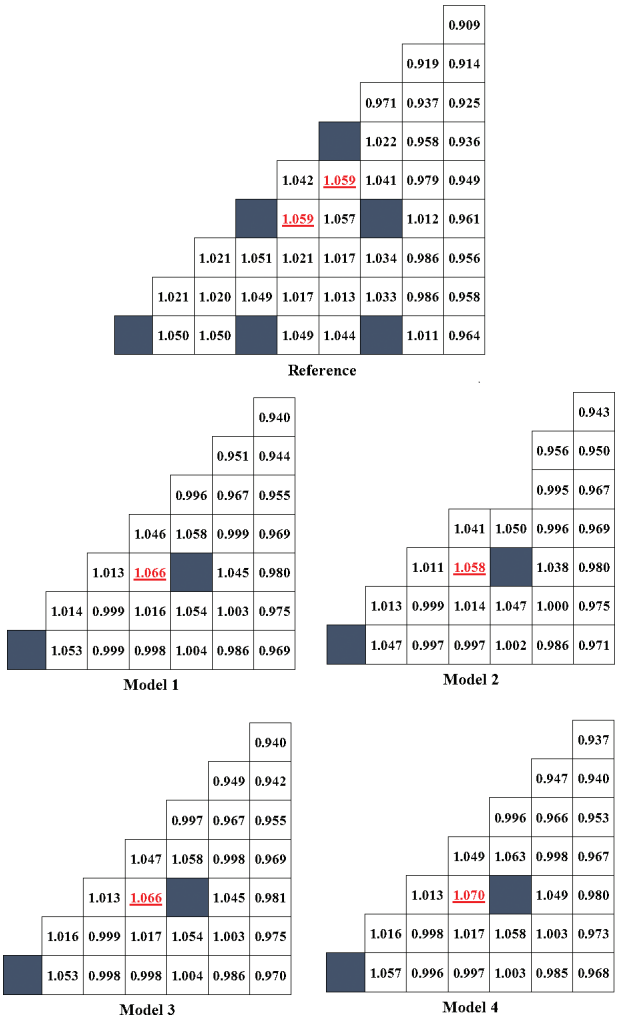

Figure 10. Radial pin power distribution at BOC for the analyzed models.

A detailed analysis of Fig. 11 indicates that all fuel models exhibit negative FTC and MTC values, underscoring their contribution to reactor safety. Notably, when compared to the solid UO_2 cases, the dual-cooled duplex fuel models demonstrate a more pronounced effect in this regard. This behavior can be attributed to the stronger influence of the Doppler effect, which is more significant for Th-232 than for U-238 in the case of the FTC. The slight decrease in FTC value when increasing the enrichment level of dual-cooled duplex fuel models is true due to the decrease in U-238 content, which is responsible for the negative values of the FTC.

Additionally, increased moderation absorption is crucial in the MTC. As the enrichment in the dual-cooled duplex configuration increases, the coefficients become slightly less negative. This effect occurs because, in this specific setup, the Doppler effect diminishes as the concentration of U-238 decreases. Therefore, the observed negative values of the FTC and MTC across the analyzed dual-cooled duplex models underscore their safety characteristics. The dual-cooled duplex models exhibit even more robust safety features,

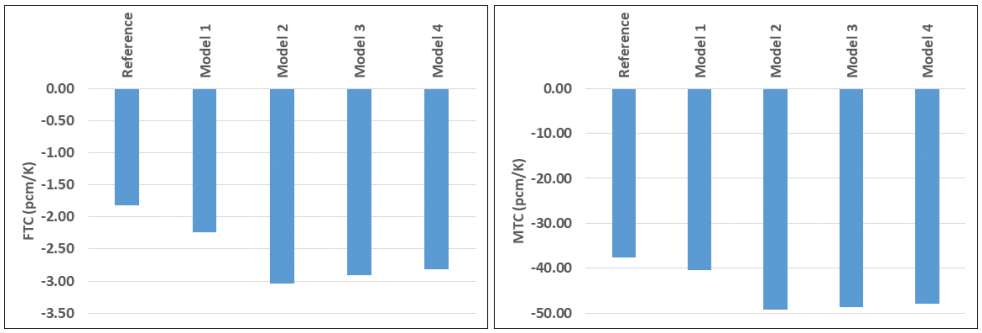

Figure 11. FTC and MTC at BOC for the analyzed models.

rendering them promising options for nuclear reactor fuel configurations.

$$\mathrm{FTC} = \frac{\mathrm{K}_{\infty}(\mathrm{T}) - \mathrm{K}_{\infty,0}}{\mathrm{K}_{\infty,0}\Delta\mathrm{T}} \quad (3)$$

where $K_{\infty}(T)$ is the multiplication factor at the perturbed fuel temperature, $K_{\infty,0}$ is the reference multiplication factor under standard conditions, and ΔT is the applied temperature variation. In this study, a perturbation of 100 K was applied to evaluate the FTC.

$$\mathrm{MTC} = \frac{\mathrm{K}_{\infty}(\mathrm{T},\rho) - \mathrm{K}_{\infty,0}}{\mathrm{K}_{\infty,0}\Delta\mathrm{T}} \quad (4)$$

where K_{∞} (T, ρ) is the multiplication factor under perturbed moderator temperature and density, with $K_{\infty,0}$ again representing the nominal multiplication factor. For the MTC calculation, a temperature perturbation of 27 K was applied to the moderator.

Conclusions and future work

The current research focuses on a preliminary study on assembly design and neutronic analysis of dual-cooled fuel, which is composed of UO_2-ThO_2, in AP1000-like reactors. This novel assembly, arranged in an optimal 13 × 13 configuration, was proposed as a potential solution for improving the current performance of these reactor types. The primary goal of this preliminary study was to investigate the burnup characteristics and compare the neutronic performance of dual-cooled duplex fuel with varying enrichment levels to solid 17 × 17 and dual-cooled 13 × 13 UO_2 assemblies. The most important finding of this study is that using UO_2-ThO_2 LEU+ fuel with an enrichment level of 7 wt.% (U-235) can achieve a criticality period comparable to solid UO_2. When compared to the solid reference UO_2 assembly, UO_2-ThO_2 results in smooth reactivity curves with a lower reactivity swing. From this perspective, this dual-cooled duplex fuel configuration appears to be a viable option for PWR reactors. Furthermore, in terms of safety, the study included the calculation of plutonium isotopes and other minor actinides. When compared to solid UO_2, the dual-cooled duplex configurations produced fewer plutonium isotopes and had lower concentrations of minor actinides such as Np, Am, and Cm. Furthermore, the power peaking factor results showed that the dual-cooled duplex designs could keep power peaking factors within PWR acceptable ranges, proving their viability. Finally, in the dual-cooled duplex models, the reactivity coefficients, both FTC and MTC, are predominantly negative and even more negative than in solid UO_2. Importantly, these coefficients consistently meet the stringent requirements of PWRs, highlighting the dual-cooled duplex fuel assembly's safety and performance advantages.

Future work will focus on extending this study toward a more comprehensive core-level analysis, including detailed 3D modeling, thermal-hydraulic coupling to evaluate heat removal capability, and fuel performance assessments under irradiation. Additionally, sensitivity studies on fuel cycle length and reactivity control strategies analysis with conventional UO_2 cores will be pursued to further validate the potential of dual-cooled duplex fuel in PWR applications.

Declaration of competing interests

The authors confirm that they have no known financial or personal conflicts of interest that could have influenced the integrity of the research presented in this paper. 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.

Credit authorship contribution statement

Hanan Rifa: Resources, Data curation, Investigation, Writing - original draft, - review & editing. Ouadie Kabach: Conceptualization, Data curation, Formal analysis, Investigation, Methodology, Supervision, Validation, Writing original draft- review & editing. El Mahjoub Chakir: Supervision, Formal analysis, review & editing Zouhair Sadoune: Supervision, Formal analysis, review & editing.

Data availability statement

This paper and the references therein contain all the data to reproduce and validate the presented results.

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