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Comparative Study of UO_2 and $(Th,U\text{-}233)O_2$ Performance in Small Long-Life PWR Fuel Cell

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ABSTRACT

A study was performed comparing the performance of UO2 and (Th,U-233)O2 fuel in small long-life PWR. The neutronic calculation carried out by PIJ module of SRAC2006 was done to a fuel cell in 10 years of operation. The calculation was conducted by varying the enrichment of U-235 in UO_2 and U-233 in $(Th,U\text{-}233)O_2$ for 1% - 20% and also by varying the fuel volume fraction for 40%, 45%, 50%, 55%, and 60%. The performance was observed by comparing the enrichment needed by each fuel type to gain criticality in 10 years, the infinite multiplication factor (k-inf) value, and the conversion ratio (CR) value. The calculation results showed that 60% fuel volume fraction gave critical conditions with the lowest infinite multiplication factor and highest conversion ratio for both fuel types. While in terms of fissile nuclide enrichment needed, $(Th,U\text{-}233)O_2$ had better performance than UO_2, because only 5% U-233 was needed in $(Th,U\text{-}233)O_2$ while UO_2 needed 9% U-235 to gain criticality in 10 years of operation.

1. INTRODUCTION

Nuclear power emerges as a dynamic and renewable energy source capable of fulfilling the escalating global demand for electricity. This energy is harnessed through nuclear power reactors that leverage the heat generated from controlled fission reactions in fuels imbued with fissile radionuclides. Predominantly, the Pressurized Water Reactor (PWR) stands out as the preferred choice for electricity generation, accounting for 69.3% (303 out of 437) of the world's nuclear reactors in 2021 (IAEA, 2022). Among these, the small long-life PWRs distinguish themselves as particularly suited for rural electrification, offering up to 30 years of service without the need for refueling and delivering power up to 300 MWe (IAEA, 2010). Uranium dioxide (UO_2) fuel, enriched with 3% to 5% U-235, is commonly deployed in PWRs, necessitating replenishment every one to two years (IAEA, 2020b). Scholarly exploration, particularly in neutronic analysis, has been pivotal in assessing PWR fuel performance. A notable study by Aziz and Massoud in 2014 demonstrated that elevating U-235 enrichment to 6.2% in UO_2-fueled pin cells could sustain a PWR in critical condition for a biennial cycle, with an initial effective multiplication factor (*k*-eff) of 1.52 (Aziz & Massoud, 2014). Further advancing this research, Hoang et al. (2021) probed the effects of increasing U-235 enrichment to 15% in fuel pin cells, targeting augmented burn-up and prolonged reactor operation.

Thorium-232 (Th-232), a fertile isotope abundant in the earth's crust, plays a critical role analogous to that of Uranium-238 (U-238) in nuclear reactors. It boasts a more efficient conversion process to Uranium-233 (U-233), a fissile radionuclide, than U-238's transformation into Plutonium-239 (Pu-239) through neutron absorption. Notably, Th-232 is nearly threefold more plentiful than uranium, presenting a significant advantage in terms of resource availability. The most compelling benefit of utilizing Th-232 lies in the resultant U-233's superior performance, evidenced by its lower capture-to-fission ratio in the thermal spectrum, implying a higher probability of fission compared to neutron capture (NEA, 2015). Reflecting its versatility and promise, thorium has been implemented in various reactor designs globally, encompassing experimental and commercial ventures such as Advanced Heavy Water Reactors (AHWR), Molten Salt Reactors (MSR), High-Temperature Reactors (HTR), Light Water Reactors (LWR), Liquid Metal Cooled Reactors (LMR), and Gas Cooled Reactors (GCR) (Ault et al., 2017; Vijayan et al., 2017; Humphrey & Khandaker, 2018).

Globally, several reactors have harnessed the potential of thorium-based fuel for their operations. Notably, Shippingport and Indian Point, two American Pressurized Water Reactors (PWRs), have adopted a fuel composition of thorium and Uranium-233 oxide (Th,U-233)O_2 (Maiorino & Carluccio, 2004; Humphrey & Khandaker, 2018). Extensive research has scrutinized the efficacy of this thorium-based fuel in PWRs. Findings indicate that the (Th,U-233)O_2 fuel mixture contributes to enhanced performance in long-life PWRs, delivering extended operational periods without the need for refueling, an elevated internal conversion ratio, and increased fuel burn-up, demonstrating its efficacy and sustainability (Subki et al., 2008; Subkhi et al., 2012; Subkhi et al., 2013; Subkhi et al., 2015; Hassan et al., 2020; Napirah & Su'ud, 2020; Lapanporo & Su'ud, 2022).

This research juxtaposes the performance of traditional UO_2 fuel, encompassing U-238 and U-235 oxides, against the innovative (Th,U-233)O_2 fuel, composed of Th-232 and U-233 oxides, within a comparable fuel pin cell design. By manipulating the enrichment levels of the fissile nuclides—U-235 and U-233—and altering the fuel volume fractions, the study meticulously examines and contrasts the infinite multiplication factor (*k*-inf) and the conversion ratio (CR).

2. METHOD

The neutron calculation in this study is performed using the SRAC2006 code developed by the Japan Atomic Energy Agency (JAEA) and the JAEA Nuclear Data Center (JENDL)-4.0 as the nuclide data library. The main purpose is to compare the performance of two types of fuel, namely UO_2 and (Th,U-233)O_2 in the fuel cell. To achieve this goal, the calculation is limited to the fuel cell calculation only by comparing the infinite multiplication factor (*k*-inf) and the conversion ratio (CR) of these fuels used.

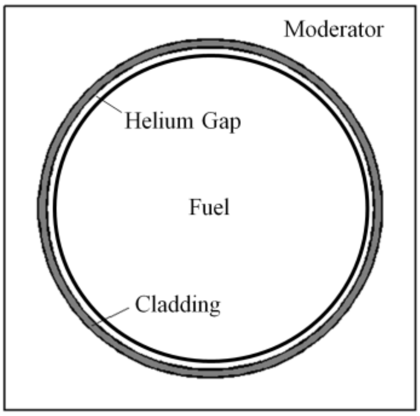


Figure 1 Fuel pin cell geometry.

Table 1 Fuel pin cell design parameter.

Parameter	**Value/Description**
Fuel Material	Uranium dioxide (UO_2) Thorium-Uranium dioxide $(Th,U\text{-}233)O_2$
Cladding Material	Zircaloy-4
Coolant Material	Water (H_2O)
Moderator	Water (H_2O)
Fuel pin cell geometry	Square
Pin pitch	1.26 cm
Helium gap width	0.0084 cm
Cladding thickness	0.057 cm
U-235/U-233 enrichment	1 – 20%
Fuel volume fraction	40 – 60%
Power level	109.65 Watt/cm
Refueling period	10 years

The fuel burn-up calculation in a fuel cell is performed by the PIJ module, which uses the Collision Probability Method (CPM) for square pin cell geometry, as shown in Figure 1. The fuel burnup calculation by the PIJ module is performed by varying the U-235 enrichment in UO_2 and the U-233 enrichment in $(Th,U\text{-}233)O_2$ from 1% to 20% and by varying the fuel volume fraction in the fuel cell from 40% to 60%. The design parameters for the fuel pin cell and the volume fractions for the fuel cell components are shown in Tables 1 and 2.

Table 2 Fuel cell component volume fractions. Helium gap width and cladding thickness are set to be constant, as in Table 1.

	Fuel Volume Fraction (%)	Helium Gap Volume Fraction (%)	Cladding Volume Fraction (%)	Coolant & Moderator Volume Fraction (%)
FF 40%	40.00%	1.51%	10.97%	47.52%
FF 45%	45.00%	1.60%	11.59%	41.81%
FF 50%	50.00%	1.69%	12.17%	36.14%
FF 55%	55.00%	1.77%	12.73%	30.51%
FF 60%	60.00%	1.84%	13.25%	24.90%

The output observed are infinite multiplication factor (*k*-inf) and conversion ratio (CR). The definition for *k*-inf and CR is shown in mathematical expressions as in Equations (1), (2), and (3) (Duderstadt & Hamilton, 1976).

$$k \equiv \text{multiplication factor} = \frac{\text{Number of neutrons in one generation}}{\text{Number of neutrons in the preceding generation}} \tag{1}$$

or

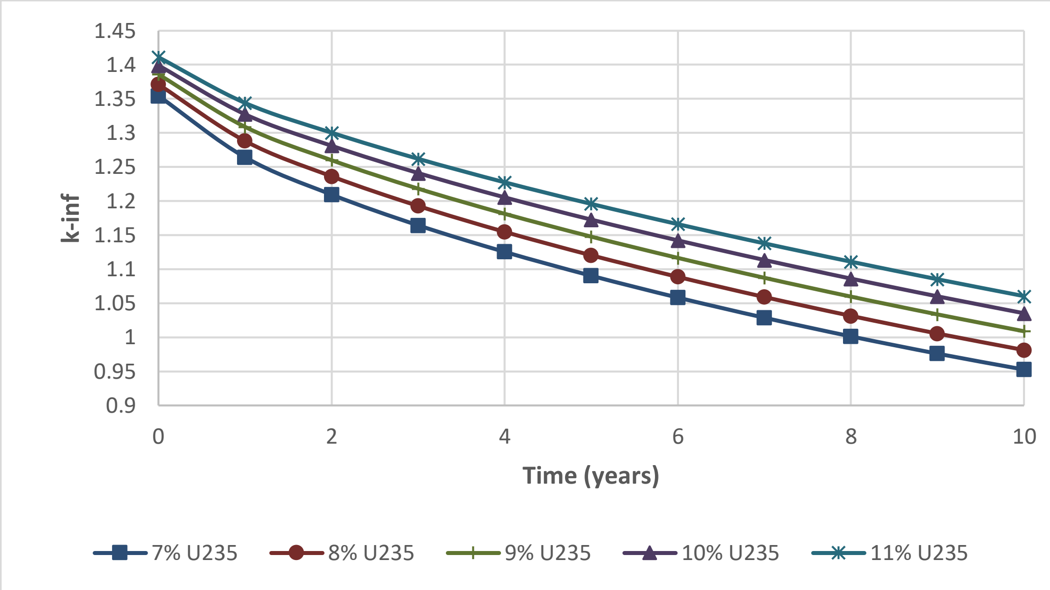
$$k \equiv \frac{\text{Rate of neutron production in a reactor}}{\text{Rate of neutron loss (absorption plus capture) in a reactor}} \equiv \frac{P(t)}{L(t)} \tag{2}$$

$$CR = \frac{\text{Average rate of fissile atom production}}{\text{Average rate of fissile atom consumption}} \tag{3}$$

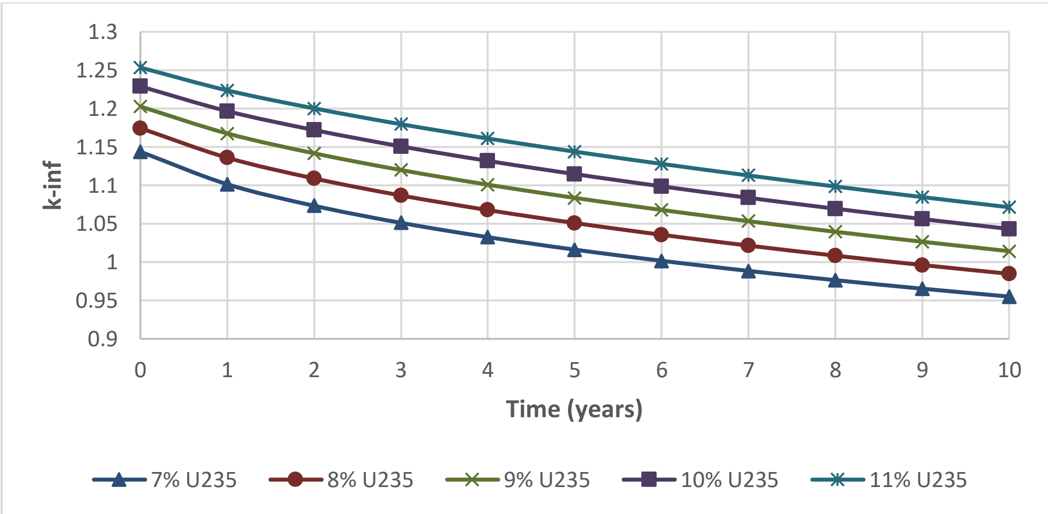
3. RESULTS AND DISCUSSION

The calculation for the fuel cell is made for ten years of operation, because the refueling time for most of the small long-lived reactor designs developed is less than ten years (IAEA, 2020a). The fuel burnup calculation is performed for 1% to 20% U-235 enrichment in UO_2 for 40% to 60% fuel volume fraction. It is shown that with increasing U-235concentration the value of *k*-inf also rises. In addition, it was found that a minimum enrichment of 9% is required to make UO_2 critical within a decade. This minimum enrichment value is similar for all fuel volume fractions as shown in Figures 2

(a) and (b). Another behavior shown in Figures 2 (a) and (b) is the decrease of *k*-inf over time as the fissile nuclide is depleted in the operation period.



(a)



(b)

Figure 2 The UO_2 k-inf value with 7% - 11% U-235 enrichment in (a) 40% fuel volume fraction and (b) 60% fuel volume fraction in 10 years operation. The minimum U-235 enrichment value needed in UO_2 fuel to be critical in 10 years period is 9%.

Figure 3 displays the anomaly detected for U-235 enrichment levels of 1% and 2% in a 60% fuel volume fraction. The *k*-inf value progressively increased starting from the second year for 1% enrichment and the fifth year for 2% enrichment. As a result, there is a buildup of Pu-239 during each enrichment cycle. This phenomenon can be attributed to the transmutation of U-238 into Pu-239 over time. However, the impact of an increased *k*-inf is solely observable at the lowest enrichment in the lowest moderator content (highest fuel volume fraction of 60%). This is because the increase in Pu-239 balances out the decrease in U-235, as the nuclide numbers of both Pu-239 and U-235 are roughly equal, as demonstrated in Figure 4.

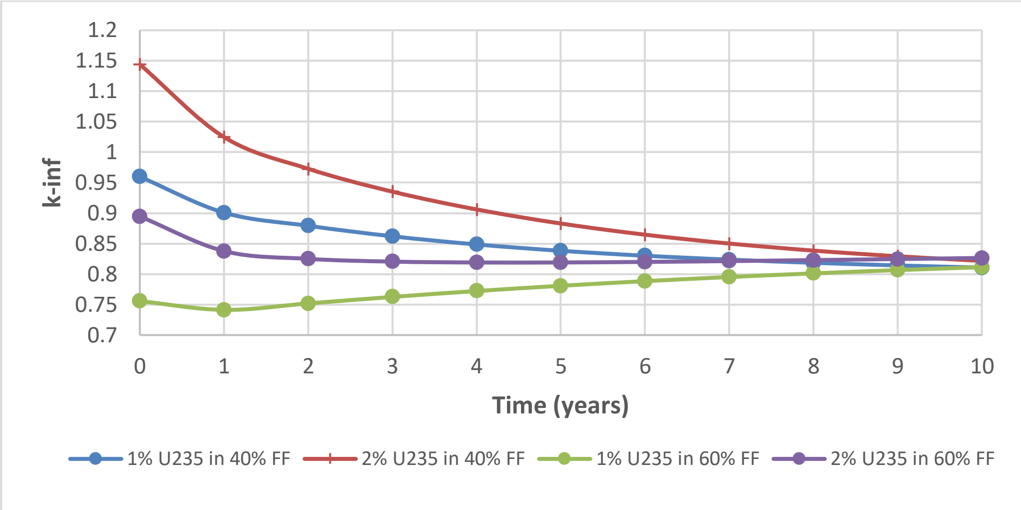
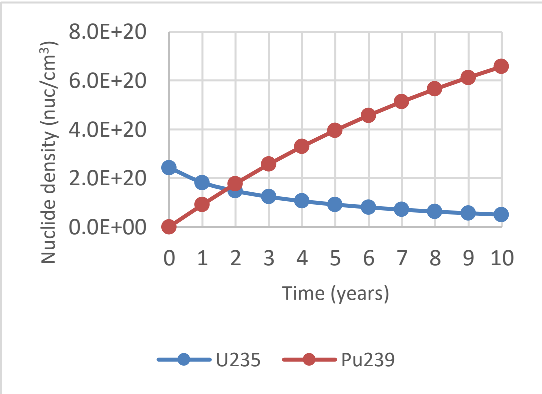
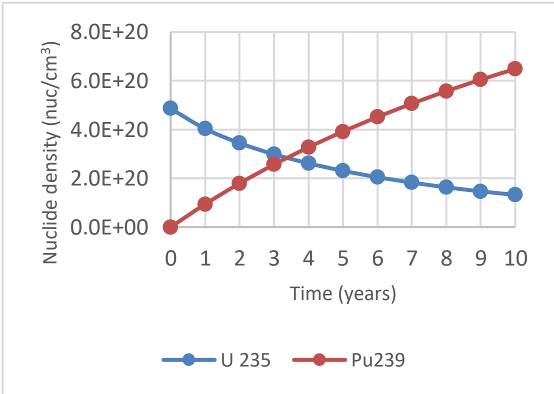


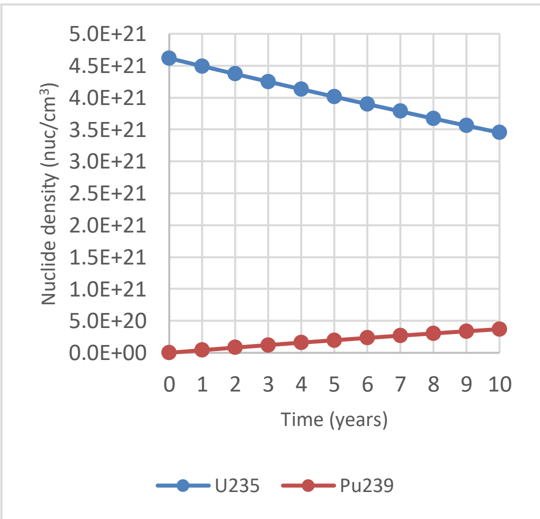
Figure 3 The k-inf value over the years for 1% and 2% of U-235 enrichment in 60% and 40% fuel volume fraction.



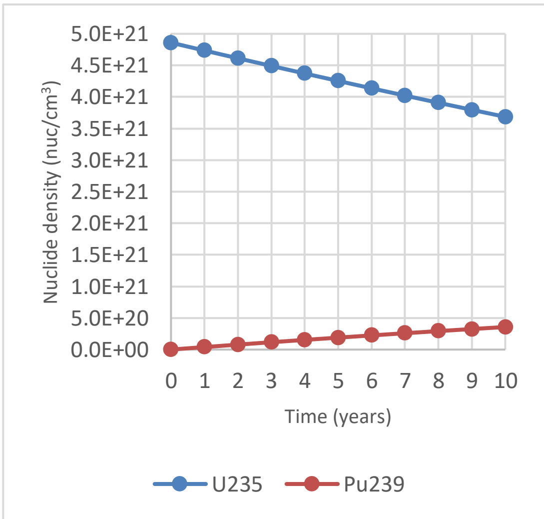
(a)



(b)



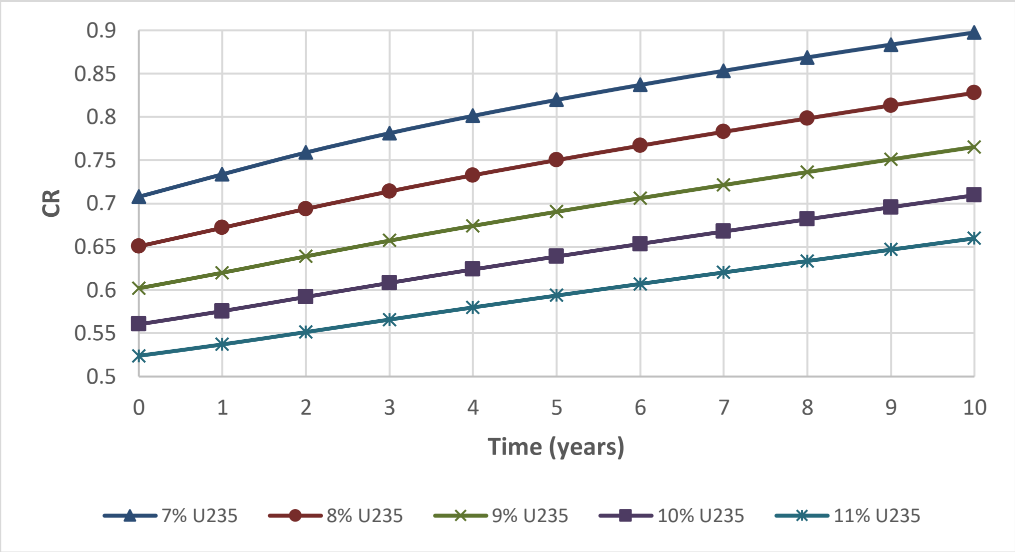
(c)



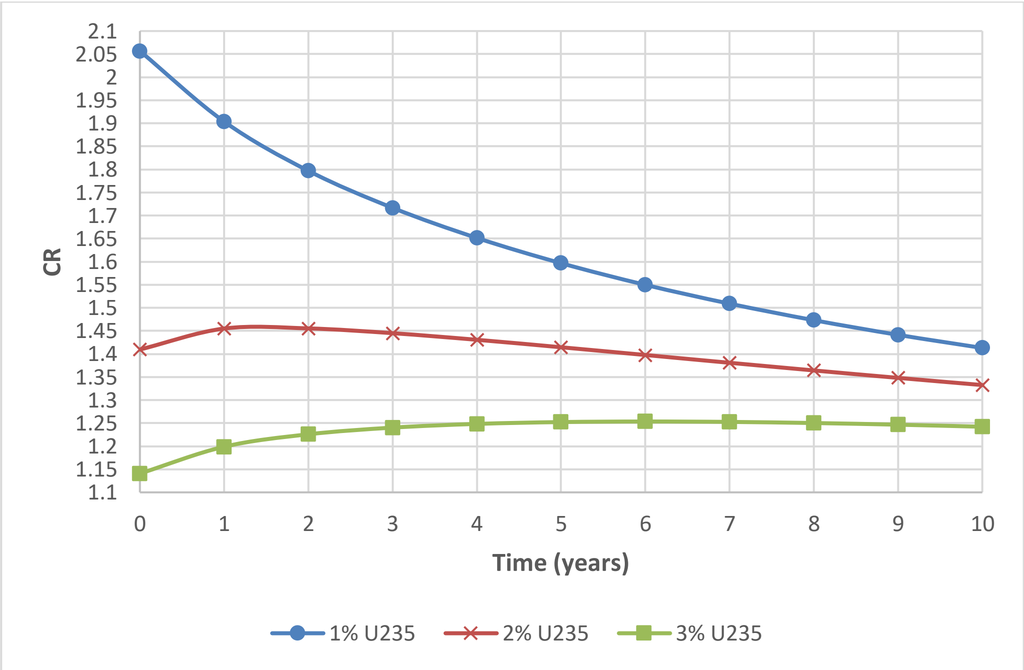
(d)

Figure 4 The fissile nuclide density (number per cm^3) change in 60% fuel volume fraction for UO_2 with (a) 1% U-235 enrichment, (b) 2% U-235 enrichment, (c) 19% U-235 enrichment, and (d) 20% U-235 enrichment.

Figure 4 illustrates the changes in fissile nuclide density (number of nuclides per cm^3 of fuel), U-235, and Pu-239 over a period of 10 years. The build-up of Pu-239 is primarily affected by the presence of U-238 in the fuel. Therefore, changes in U-235 enrichment do not impact the build-up of Pu-239, which consistently occurs at a rate of magnitude 1020 nuclides per cm^3. When fuel is enriched with 1% and 2% U-235, it is observed that after several years, there is more Pu-239 than U-235, even exceeding the initial load of U-235 in the fuel. This phenomenon explains the increase in *k*-inf for 1% and 2% U-235 enrichments, as shown in Figure 3.



(a)

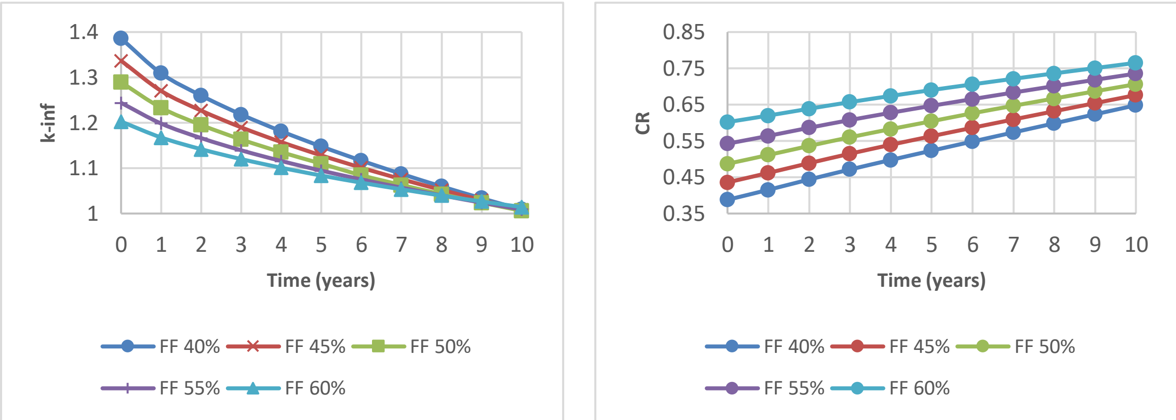


(b)

Figure 5 The UO_2 CR value for (a) 7% - 11% U-235 enrichment and (b) 1% to 3% U-235 enrichment in 60% fuel volume fraction in 10 years operation.

Figure 5 displays the conversion ratio (CR) of UO_2 fuel at different U-235 enrichments in a 60% fuel volume fraction. The CR value is higher for low enrichment than for high enrichment. Over 10 years, CR values above 1 were observed for enrichments of 1%, 2%, and 3%. According to Equation (3), this indicates that more fissile nuclides were produced than consumed during that period. These findings confirm the results presented in Figures 4 (a) – (b). This condition leads to fuel breeding, resulting in higher concentrations of fissile nuclides in the discharged fuel compared to the fresh fuel in its initial loading.

The results of calculations performed on fuel volume fractions of 45%, 50%, and 55% exhibited behaviors similar to those of fuel volume fractions of 40% and 60%, as illustrated in Figures 2 (a) – (b). These patterns demonstrate that the *k*-inf value increased with an increase in U-235 enrichment, but decreased over time. The results reveal that a minimum U-235 enrichment of 9% is necessary for critical conditions to be achieved over 10 years for all fuel volume fractions. The effects on the *k*-inf and CR values for uranium-235 enrichment at 9% are shown in Figure 6. Based on the data presented, the most favorable performance is achieved with a fuel volume fraction of 60%, which exhibits the lowest *k*-inf value and the highest CR value.



(a) (b)

Figure 6 (a) The *k*-inf value for 9% U-235 enrichment and (b) The CR value for 9% U-235 enrichment in UO_2 in 40%, 45%, 50%, 55%, and 60% fuel volume fraction.

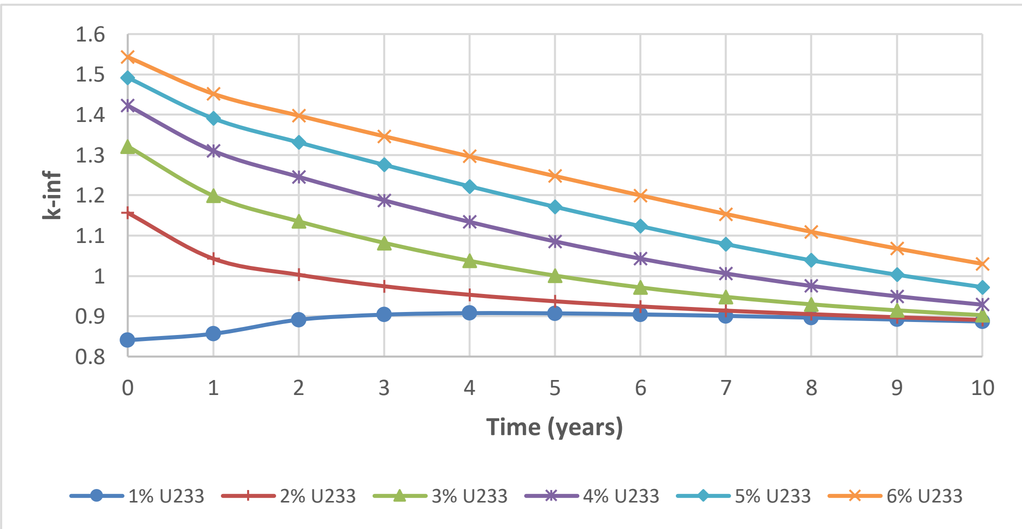


Figure 7 The $(Th,U\text{-}233)O_2$ *k*-inf value for 1% to 6% of U-233 enrichment in 40% fuel volume fraction in 10 years operation.

The (Th,U-233)O_2 fuel was calculated for U-233 enrichments ranging from 1% to 20% in fuel volume fractions of 40%, 45%, 50%, 55%, and 60%. The results are presented in Figures 7 and 8. U-233 enrichments between 1% and 6% in 40% and 60% fuel volume fractions exhibited similar behavior. As depicted by Figures 7 and 8, there was an increase in the *k*-inf value as U-233 enrichment escalated, similar to UO_2 with U-235 enrichment. Figures 7 and 8 illustrate a decrease in the *k*-inf value over time, except for at 1% enrichment where there is an increase. In (Th,U-233)O_2, a similar fissile nuclide build-up occurs as in UO_2. However, in (Th,U-233)O_2, only U-233 plays a major role as the fissile nuclide. Th-232 is transmuted into U-233, while U-233 serves as the primary fissile nuclide and is consumed. The change in U-233 nuclide density within the fuel, resulting from both consumption and production, is illustrated in Figure 9 for U-233 enrichment values ranging from 1% to 6%.

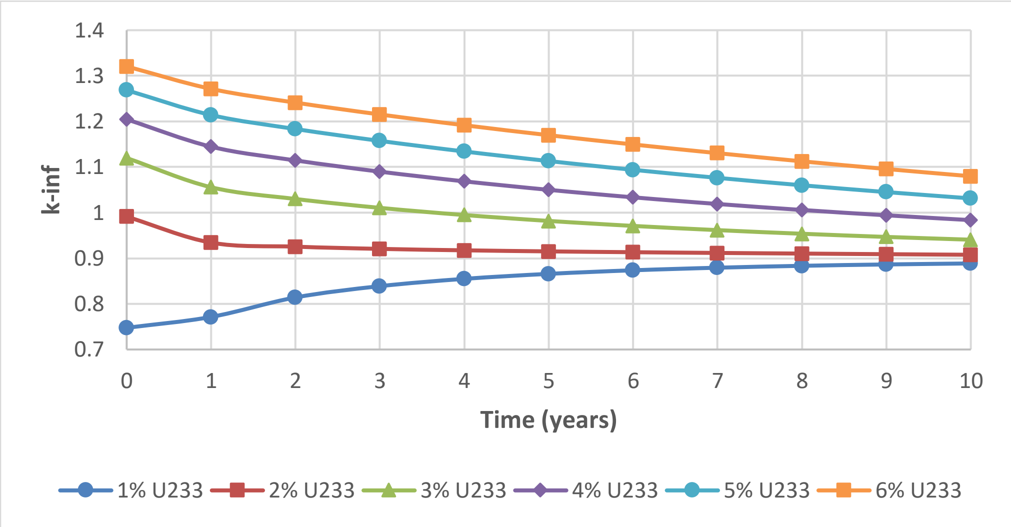


Figure 8 The (Th,U-233)O_2 *k*-inf value for 1% to 6% of U-233 enrichment in 60% fuel volume fraction in 10 years operation.

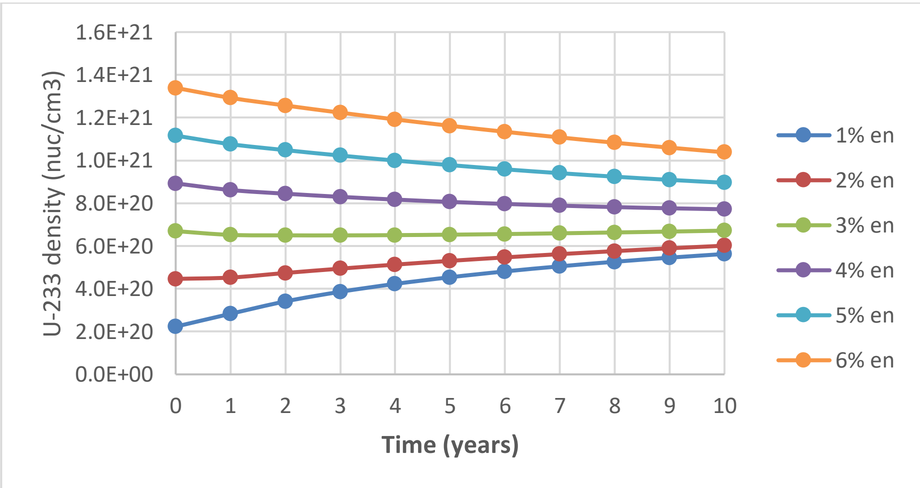


Figure 9 U-233 nuclide density (number per cm^3) change in 60% fuel volume fraction for (Th,U-233)O_2.

Figure 9 displays the alteration in U-233 nuclide density (number of nuclides per cubic centimeter of fuel) in (Th,U-233)O_2. Higher U-233 enrichment leads to increased U-233 density. The initial load value of fresh fuel is presented at year zero. Unlike U-235 present in UO2, which depletes over time, U-233 is both depleted and built up as Th-232 captures neutrons during reactor operation. The nuclide density increases over time for 1% and 2% enrichment due to the higher production rate of U-233 rather than its consumption rate. As the enrichment value increases, so does the initial load for U-233, resulting in more U-233 to be consumed than produced by Th-232 transmutation.

Figure 10 shows the CR value of (Th,U-233)O_2, which confirms the observation. The CR value of (Th,U-233)O_2 with a 60% fuel volume fraction in Figure 10 displayed similar behavior to the CR value of UO_2 in Figure 5. Increasing U-233 enrichment resulted in a decrease in the CR value. During the ten years of reactor operation, 1% and 2% enrichment gave higher CR values than 1, while higher enrichment produced a lower CR value.

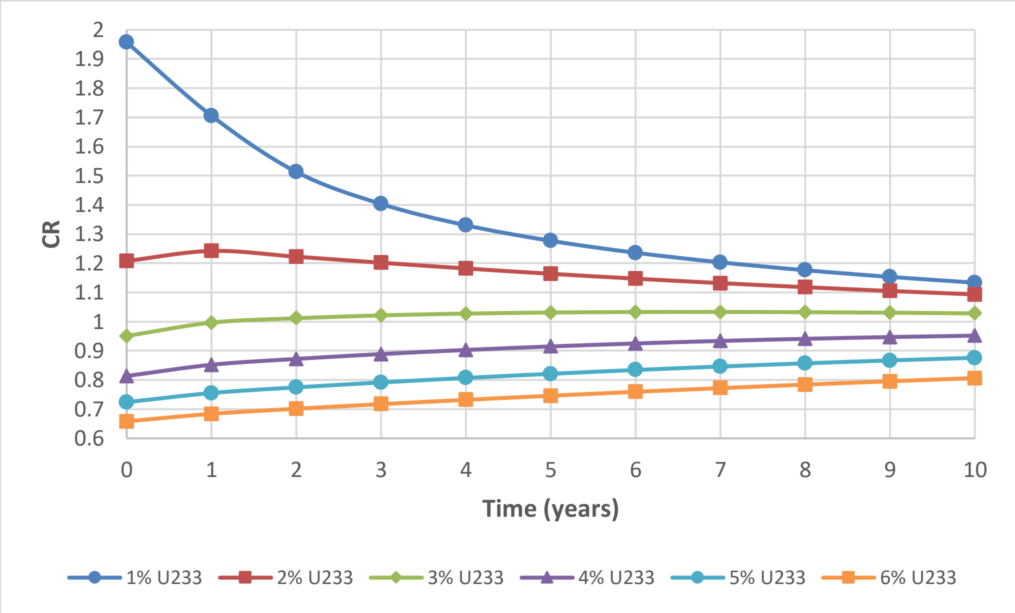
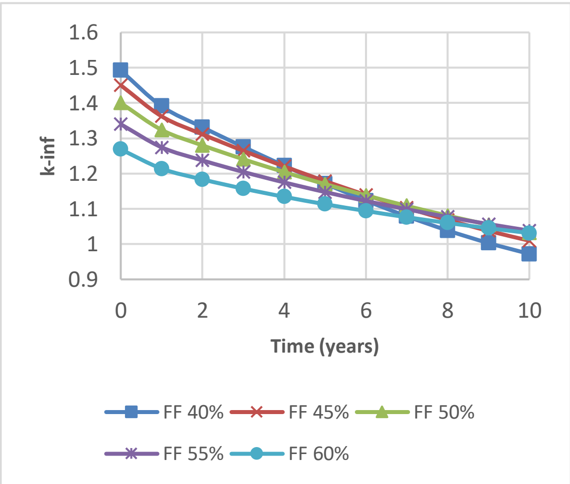
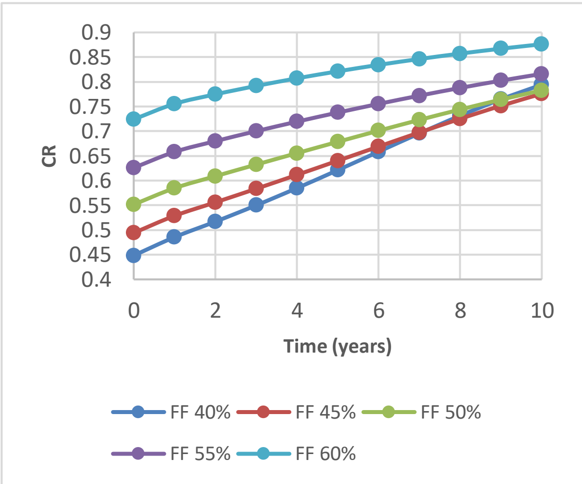


Figure 10 The (Th,U-233)O_2 CR value for 1% to 6% of U-233 enrichment in 60% fuel volume fraction in 10 years operation.



(a)



(b)

Figure 11 (a) The *k*-inf value for 5% U-233 enrichment and (b) The CR value for 5% U-233 enrichment in (Th,U-233)O_2 in 40%, 45%, 50%, 55%, and 60% fuel volume fraction.

Figure 11 illustrates the effect of fuel volume fraction on *k*-inf and CR values. The minimum U-233 enrichment required for (Th,U-233)O_2 fuel to reach criticality within 10 years is 6% in a 40% fuel volume fraction and 5% in 45%, 50%, 55%, and 60% fuel volume fractions. To test most fuel volume fractions, the enrichment is set to 5%, which is the lowest enrichment value required. The *k*-inf and CR values exhibit similar trends to UO_2, where higher fuel volume fractions result in lower *k*-inf. Similar findings were also reported by Napirah and Su'ud (2020), indicating that the optimum volume fraction is 60% due to its lower *k*-inf and reactivity swing compared to higher fuel volume fractions. Although *k*-inf decreases over time, CR increases, particularly at the beginning of life (BOL). Therefore, the best fuel performance is achieved with a 60% volume fraction, which produces the lowest k-inf and highest CR values.

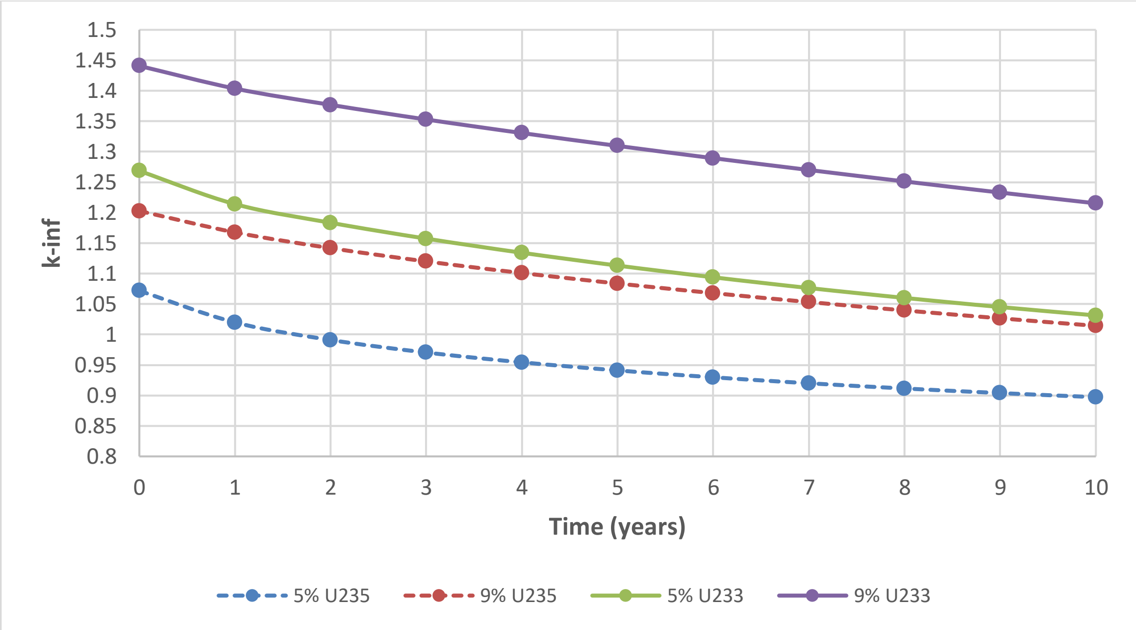


Figure 12 The *k*-inf value for 5% and 9% enrichment of U-235 in UO_2 and U-233 in $(Th,U\text{-}233)O_2$ in 60% fuel volume fraction.

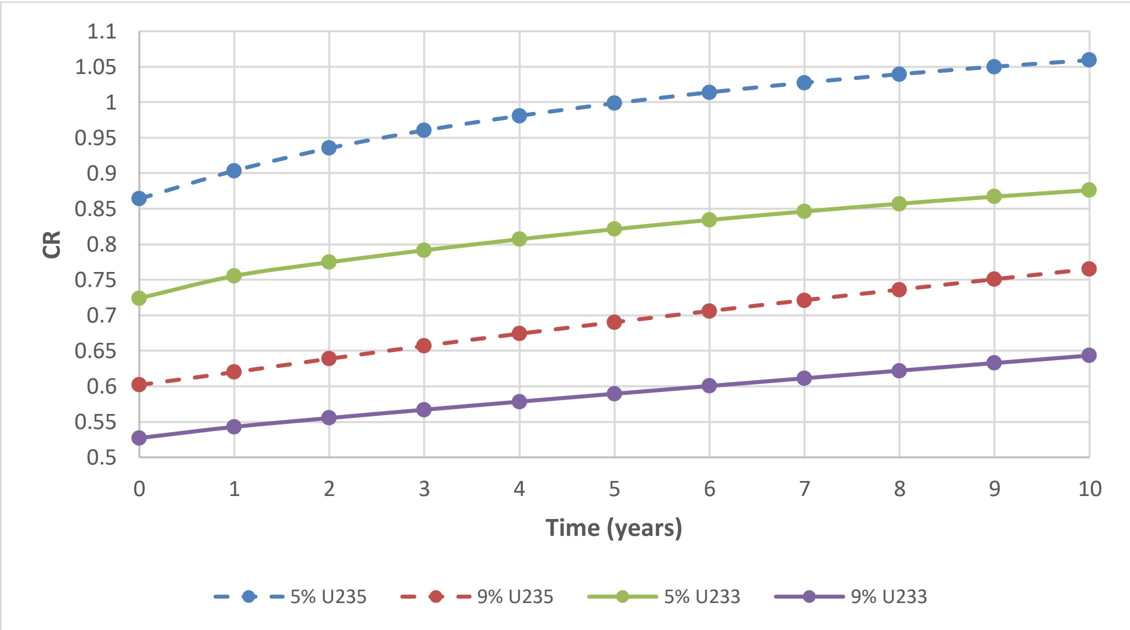


Figure 13 The CR value for 5% and 9% enrichment of U-235 in UO_2 and U-233 in $(Th,U\text{-}233)O_2$ in 60% fuel volume fraction.

The results of the calculation indicate that the most efficient performance for both fuels is acquired at a fuel volume fraction of 60%, with UO_2 enriched by 9% of U-235 and $(Th,U\text{-}233)O_2$

enriched by 5% of U-233. Both fuels' *k*-inf and CR values at 5% and 9% enrichment are displayed in Figures 12 and 13, respectively. Figure 12 demonstrates that in similar enrichment, (Th,U-233)O_2 has a higher *k*-inf value than UO_2. Similar findings were obtained by Galahom et al. (2022) when they compared *k*-inf in PWR fuel cell between (Th,U-233)O_2 and UO_2, which showed that the average number of neutrons produced in U-233 fission is greater than in U-235 fission (Galahom et al., 2022). As a result, the enrichment of U-233 that is necessary in (Th,U-233)O_2 is lower than that required for UO_2 enrichment, as illustrated in Figure 12. In terms of CR comparison, Figure 13 exhibits that 5% U-233 has a higher CR value than 9% U-235.

4. CONCLUSION

The conclusion encapsulates the core findings of a computational analysis comparing UO_2 and (Th,U-233)O_2 fuels within the operational framework of a small, long-lived PWR fuel cell over a decade. This investigation primarily focused on assessing two critical performance indicators: the infinite multiplication factor (*k*-inf) and the conversion ratio (CR). The outcomes revealed that a 60% fuel volume fraction is optimal for achieving criticality, concurrently yielding the lowest *k*-inf and highest CR for both types of fuel. However, a comparative advantage emerges for (Th,U-233)O_2 fuel, which requires only a 5% enrichment of U-233 to maintain criticality over ten years. In contrast, UO_2 fuel necessitates a higher enrichment, specifically 9% of U-235, to achieve the same operational standard. This distinction underscores the superior performance of (Th,U-233)O_2 fuel in terms of enrichment efficiency within the scope of the study.

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