Comparative Study of UO$_2$ and (Th,U-233)O$_2$ Performance in Small Long-Life PWR Fuel Cell

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ABSTRACT

A study was performed comparing the performance of UO$_2$ and (Th,U-233)O$_2$ 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-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 ($k_{\text{inf}}$) 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-233)O$_2$ had better performance than UO$_2$, because only 5% U-233 was needed in (Th,U-233)O$_2$ while UO$_2$ needed 9% U-235 to gain criticality in 10 years of operation.

1. INTRODUCTION

Nuclear energy is one form of new and renewable energy that is able to meet the needs of increasing energy demands nowadays, especially electrical energy. One way of harnessing nuclear energy in the form of heat energy is by using nuclear power reactors that utilize controlled fission reactions in the fuel containing fissile radionuclides. The most common type of nuclear reactor used to produce electrical energy is Pressurized Water Reactor (PWR). In 2021, 69.3% (303 out of 437) of total nuclear power reactors in the world were PWRs (IAEA, 2022). A small long-life PWR is a PWR that could be used to generate electricity in rural areas without a refueling process for quite a long time, up to 30 years, with an output power of 300 MW$e$ or below (IAEA, 2010). The most typical fuel type utilized in PWRs is uranium dioxide (UO$_2$) with U-235 enrichment from 3% to 5% that should be replaced every 1 to 2 years in the refueling process (IAEA, 2020b). So far, many studies about neutronic analysis have been conducted to investigate the performance of nuclear fuel used in a PWR. In 2014, Aziz and Massoud conducted a neutronic calculation of uranium dioxide (UO$_2$) fuelled pin cell of PWR and showed that U-235 enrichment of 6.2% gave an initial effective multiplication factor ($k_{\text{eff}}$) of 1.52 and could operate the reactor in critical condition for two years (Aziz & Massoud, 2014).
In another study, Hoang et al. (2021) increased U-235 enrichment in PWR fuel pin cells to 15% to achieve higher burn-up and increase the reactor’s operation period.

Thorium-232 (Th-232) is a fertile isotope of thorium that has a similar role to U-238 in nuclear fuel. Th-232 could be converted into Uranium-233 (U-233), which is a fissile radionuclide more efficiently than the conversion of U-238 to Plutonium-239 (Pu-239) via neutron capture. Another advantage of Th-232 is that its abundance in nature is almost three times Uranium’s in the earth’s crust. The most superior advantage of Th-232 is the produced U-233 is better than U-235 and Pu-239 because U-233 has the lowest capture-to-fission ratio in the thermal spectrum, that is, the ratio of neutron capture cross-section to fission cross-section in the thermal spectrum (NEA, 2015). So far, thorium has been used in many types of reactors in the world for experimental and commercial purposes, 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).

Several reactors in the world have utilized thorium-based fuel in their operation. Shippingport and Indian Point are two American PWRs that utilize the oxide form of Th + U-233 or (Th,U-233)O2 as their fuel (Maiorino & Carluccio, 2004; Humphrey & Khandaker, 2018). Besides that, many studies have been conducted to investigate the performance of thorium-based fuel used in PWR. It has been shown that Th-232 and U-233 oxide or (Th,U-233)O2 fuel has excellent performance to a long-life PWR, those are longer operation period without refueling, higher internal conversion ratio, and higher fuel burn-up (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).

In this study, the performance of UO2 fuel (with U-238 and U-235 oxide) and (Th,U-233)O2 fuel (with Th-232 and U-233 oxide) are compared using a similar design for the fuel pin cell. By varying the enrichment of fissile nuclide (U-235 and U-233) and the fuel volume fraction, the infinite multiplication factor (k-inf) and conversion ratio (CR) are investigated and compared.

2. METHOD

The neutronic calculation in this study is conducted by using SRAC2006 code developed by Japan Atomic Energy Agency (JAEA) and JENDL-4.0 as the nuclide data library. The main purpose is to compare the performance of two types of fuel, those are UO2 and (Th,U-233)O2 in the fuel cell. To achieve the goal, the calculation is restricted to fuel cell calculation only by comparing the infinite multiplication factor (k-inf) and Conversion Ratio (CR) of those fuels used. The fuel burn-up calculation conducted in a fuel cell is carried out by PIJ module that uses Collision Probability Method (CPM) for square pin cell geometry, as shown in Figure 1. Fuel burn-up calculation by PIJ module is done by varying U-235 enrichment in UO2 and U-233 enrichment in (Th,U-233)O2 from 1% to 20% and by varying the fuel volume fraction in the fuel cell from 40% to 60%. Fuel pin cell design parameters and the volume fractions for fuel cell components are shown in Tables 1 and 2.

![Figure 1 Fuel pin cell geometry](image)

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Table 1 Fuel pin cell design parameter

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value/Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel Material</td>
<td>Uranium dioxide (UO$_2$)</td>
</tr>
<tr>
<td></td>
<td>Thorium-Uranium dioxide (Th,U-233)O$_2$</td>
</tr>
<tr>
<td>Cladding Material</td>
<td>Zircaloy-4</td>
</tr>
<tr>
<td>Coolant Material</td>
<td>Water (H$_2$O)</td>
</tr>
<tr>
<td>Moderator</td>
<td>Water (H$_2$O)</td>
</tr>
<tr>
<td>Fuel pin cell geometry</td>
<td>Square</td>
</tr>
<tr>
<td>Pin pitch</td>
<td>1.26 cm</td>
</tr>
<tr>
<td>Helium gap width</td>
<td>0.0084 cm</td>
</tr>
<tr>
<td>Cladding thickness</td>
<td>0.057 cm</td>
</tr>
<tr>
<td>U-235/U-233 enrichment</td>
<td>1 – 20%</td>
</tr>
<tr>
<td>Fuel volume fraction</td>
<td>40 – 60%</td>
</tr>
<tr>
<td>Power level</td>
<td>109.65 Watt/cm</td>
</tr>
<tr>
<td>Refueling period</td>
<td>10 years</td>
</tr>
</tbody>
</table>

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

<table>
<thead>
<tr>
<th>Fuel Volume Fraction (%)</th>
<th>Helium Gap Volume Fraction (%)</th>
<th>Cladding Volume Fraction (%)</th>
<th>Coolant &amp; Moderator Volume Fraction (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>FF 40%</td>
<td>40.00%</td>
<td>1.51%</td>
<td>10.97%</td>
</tr>
<tr>
<td>FF 45%</td>
<td>45.00%</td>
<td>1.60%</td>
<td>11.59%</td>
</tr>
<tr>
<td>FF 50%</td>
<td>50.00%</td>
<td>1.69%</td>
<td>12.17%</td>
</tr>
<tr>
<td>FF 55%</td>
<td>55.00%</td>
<td>1.77%</td>
<td>12.73%</td>
</tr>
<tr>
<td>FF 60%</td>
<td>60.00%</td>
<td>1.84%</td>
<td>13.25%</td>
</tr>
</tbody>
</table>

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).

\[
k \equiv \text{multiplication factor} = \frac{\text{Number of neutrons in one generation}}{\text{Number of neutrons in the preceding generation}} \quad (1)
\]

or

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

\[
CR = \frac{\text{Average rate of fissile atom production}}{\text{Average rate of fissile atom consumption}} \quad (3)
\]

3. RESULTS AND DISCUSSION

The calculation for the fuel cell is done for ten years of operation because the refueling period for most small long life reactors design developed is below ten years (IAEA, 2020a). The fuel burn-up calculation is conducted to 1% to 20% of U-235 enrichment in UO$_2$ for 40% to 60% fuel volume fraction. The result obtained for infinite multiplication factor ($k$-inf) shows that as U-235 enrichment increases, the $k$-inf value also increases. The result also showed that the minimum enrichment needed for UO$_2$ to gain criticality in 10 years is 9%. This minimum enrichment value is similar to all fuel volume fractions, as shown in Figures 2a and 2b. Another behavior shown in Figures 2a and 2b is the decrease of $k$-inf over time as the fissile nuclide is depleted in the operation period. Figure 3 shows the exception observed at 1% and 2% of U-235 enrichment in 60% fuel volume fraction where the $k$-inf value is increased from the second year onward for 1% enrichment and the fifth year onward for 2% enrichment. As time passes, U-238 in fuel is transmuted into Pu-239. This leads to Pu-239 builds up that happened in every enrichment. But the effect of increased $k$-inf is only observed at the lowest
enrichment in the lowest moderator content (highest fuel volume fraction, 60%) because the increase of Pu-239 balance the decrease of U-235, that is, the nuclide number of Pu-239 and U-235 are in the same order of magnitude, as shown in Figure 4.

![Figure 2](image_url)

**Figure 2** The UO₂ 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 UO₂ fuel to be critical in 10 years period is 9%.

Figure 4 also demonstrates how fissile nuclide density (number of nuclides per cm³ of fuel), U-235, and Pu-239 changed in 10 years. The build-up of Pu-239 is mainly affected by U-238 present in the fuel. That's why the change of U-235 enrichment doesn't affect Pu-239 builds that constantly happened in the order of magnitude 10²⁰ nuclide per cm³. In 1% and 2% U-235 enrichment, it is shown that after several years there is more Pu-239 than U-235, even more than the initial load of U-235 in the fuel. This explains why k-inf for 1% and 2% enrichment of U-235 are increasing in Figure 3.
Figure 3 The $k_{\text{inf}}$ value over the years for 1% and 2% of U-235 enrichment in 60% and 40% fuel volume fraction.

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 5 shows the conversion ratio (CR) of UO₂ fuel with various U-235 enrichment in 60% fuel volume fraction. The CR value for low enrichment has a higher value than the CR value for higher enrichment. In 10 years, the 1%, 2%, and 3% enrichment could be observed with CR values higher than 1. According to equation 3, it means that there are more fissile nuclides produced than consumed in that period. It confirmed the results obtained in Figure 4a and 4b. This condition results in fuel breeding, where the discharged fuel will have more fissile nuclides content than the fresh fuel in initial loading.

The calculation results obtained for 45%, 50%, and 55% fuel volume fraction showed similar behavior as obtained for 40% and 60% fuel volume fraction shown in Figures 2a and 2b where $k_{inf}$
value increased as U-235 enrichment increased, and where the \( k_{\text{inf}} \) value decreased over time. The results show that the minimum U-235 enrichment needed to get critical condition in 10 years for all fuel volume fractions is 9%. Figure 6 demonstrates how various fuel volume fractions affect \( k_{\text{inf}} \) and CR values in 9% U-235 enrichment. From Figure 6, it can be seen that 60% fuel volume fraction exhibits the best performance because it has the lowest value in \( k_{\text{inf}} \) and the highest value in CR.

**Figure 6** (a) The \( k_{\text{inf}} \) 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 6](image_url)

**Figure 7** The (Th,U-233)O\(_2\) \( k_{\text{inf}} \) value for 1% to 6% of U-233 enrichment in 40% fuel volume fraction in 10 years operation.

The next calculation is done to (Th,U-233)O\(_2\) fuel with 1% - 20% U-233 enrichment in 40%, 45%, 50%, 55%, and 60% fuel volume fraction. The results are shown in Figures 7 and 8. Only 1% to 6% U-233 enrichment in 40% and 60% fuel volume fraction as a result for every fraction gave similar behavior. Figures 7 and 8 show that the \( k_{\text{inf}} \) value increases as U-233 enrichment increases, as in the case of UO\(_2\) with U-235 enrichment. Figures 7 and 8 show that the \( k_{\text{inf}} \) value decreased over time, except for 1% enrichment, where the \( k_{\text{inf}} \) is increased. Similar to what happened in UO\(_2\), fissile nuclide build-up is also in (Th,U-233)O\(_2\), but the difference is that only one fissile nuclide has a major role in (Th,U-233)O\(_2\), that is U-233. Th-232 is transmuted into U-233, while U-233 is consumed as the
main fissile nuclide. The change of U-233 nuclide density in the fuel as a combination of its consumption and production for 1% - 6% U-233 enrichment value is shown in Figure 9.

![Figure 8 The (Th,U-233)O₂ k-inf value for 1% to 6% of U-233 enrichment in 60% fuel volume fraction in 10 years operation](image)

**Figure 8** The (Th,U-233)O₂ 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 cm³) change in 60% fuel volume fraction for (Th,U-233)O₂](image)

**Figure 9** U-233 nuclide density (number per cm³) change in 60% fuel volume fraction for (Th,U-233)O₂

Figure 9 exhibits the change of U-233 nuclide density (number of nuclides per cm³ of fuel) in (Th,U-233)O₂, where higher U-233 enrichment results in higher U-233 density. The initial load value in fresh fuel is shown at the zeroth year. Unlike U-235 in UO₂ which is depleted over time, U-233 is depleted and also is built up as Th-232 captured neutrons while the reactor operated. The nuclide density for 1% and 2% enrichment is increased over time because the production rate of U-233 is higher than its consumption rate. The increase in enrichment value means that the initial load for U-233 is also increased, so there will be more U-233 to be consumed than it is produced by Th-232 transmutation. This also could be seen by (Th,U-233)O₂ CR value shown in Figure 10.

The CR value of (Th,U-233)O₂ with 60% fuel volume fraction in Figure 10 showed similar behavior as the CR value of UO₂ in Figure 5. As U-233 enrichment is increased, the CR value is decreased. As the reactor operated for ten years, 1% and 2% enrichment gave higher than 1 CR value, while higher enrichment gave a lower CR value.
Figure 10 The (Th,U-233)O₂ CR value for 1% to 6% of U-233 enrichment in 60% fuel volume fraction in 10 years operation.

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)O₂ in 40%, 45%, 50%, 55%, and 60% fuel volume fraction.

The minimum value of U-233 enrichment needed for (Th,U-233)O₂ fuel to reach critical condition in 10 years are 6% in 40% fuel volume fraction and 5% in 45%, 50%, 55%, and 60% fuel volume fraction. Figure 11 shows how fuel volume fraction variation affects k-inf and CR values. The enrichment is set to be 5%, for it is the least enrichment value needed in most fuel volume fractions tested. The k-inf and CR have similar behavior with UO₂, where higher fuel volume fraction gave lower k-inf. Similar results were also shown by Napirah & Su’ud (2020), where 60% is the best volume fraction because it gave lower k-inf and reactivity swing (Napirah & Su’ud, 2020). Higher fuel volume fraction results in higher CR at the beginning of life (BOL). While k-inf is decreased over time, the CR is increased over time. The best performance is given by 60% fuel volume fraction because it produces the lowest k-inf and highest CR value.
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Figure 12 The k-inf value for 5% and 9% enrichment of U-235 in UO₂ and U-233 in (Th,U-233)O₂ in 60% fuel volume fraction

Figure 13 The CR value for 5% and 9% enrichment of U-235 in UO₂ and U-233 in (Th,U-233)O₂ in 60% fuel volume fraction

The best performance for both fuels is obtained with 60% fuel volume fraction while the enrichments are 9% U-235 in UO₂ and 5% U-233 in (Th,U-233)O₂. Figures 12 and 13 show both fuels’ k-inf and CR values in 5% and 9% enrichment. In Figure 12, it is shown that in similar enrichment, (Th,U-233)O₂ has a higher k-inf value than UO₂. Similar results were shown by Galahom et al. (2022) after comparing (Th,U-233)O₂ and UO₂ k-inf in PWR fuel cell and proved that this happened because the average number of neutrons produced in U-233 fission is larger than in U-235 fission (Galahom et
al., 2022). The advantage of this behavior is the U-233 enrichment needed in (Th,U-233)O₂ is lower than the required U-235 enrichment in UO₂, as shown in Figure 12. As for CR comparison, Figure 13 shows that 5% U-233 has a higher CR value than 9% U-235.

There are three important points obtained from the results: (1) (Th,U-233)O₂ is better than UO₂ because the enrichment of fissile nuclide needed in (Th,U-233)O₂ is lower than the required enrichment in UO₂ to give critical condition in 10 years, (2) UO₂ has lower k-inf value than (Th,U-233)O₂ in the minimum enrichment needed for both fuels to gain criticality in 10 years, (3) (Th,U-233)O₂ has higher CR value than UO₂ in the minimum enrichment needed for both fuels to gain criticality in 10 years. By the results obtained and the analysis conducted, it could be concluded that (Th,U-233)O₂ has better performance than UO₂.

4. CONCLUSION

The performance of UO₂ and (Th,U-233)O₂ fuel has been evaluated for a small long-life PWR fuel cell in 10 years of operation. The performance is evaluated based on the fuel's infinite multiplication factor (k-inf) and conversion ratio (CR). Calculation results show that 60% fuel volume fraction gave critical condition with the lowest k-inf and highest CR for both fuel types. While in terms of fissile nuclide enrichment needed, (Th,U-233)O₂ has better performance than UO₂, because only 5% U-233 is needed in (Th,U-233)O₂ while UO₂ needs 9% U-235 to gain criticality in 10 years of operation.

REFERENCE


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