Conceptual Analysis of \((\text{Th},\text{U-233})\text{O}_2\) Fueled Small Long-Life PWR with Np-237 and Pa-231 as Burnable Poison

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Abstract. Conceptual analysis of \((\text{Th},\text{U-233})\text{O}_2\) fueled small long-life PWR with Np-237 and Pa-231 as burnable poison has been performed. The fuel used is \((\text{Th},\text{U-233})\text{O}_2\) with Th-232 and U-233 as fertile-fissile nuclides pair with 8\% U-233 and addition of Pa-231 and Np-237 as burnable poison (BP). This study aims to investigate optimum design of fuel and core configuration and the performance of reactor. The calculation is done by SRAC2006 code developed by JAEA and is based on JENDL-4.0 library. Cell calculation conducted by PIJ module shows that optimum design is reached at 60\% of fuel volume fraction and 6\% of burnable poison. Core optimization calculation by CITATION module for two dimensional (R – Z) core geometry shows that reactor could reach 60 years of lifetime at 150 MWt power level which correspond to 25.36 power density. Calculation for heterogenous reactor shows that reactor with BP combination of 8\% of Pa-231 at center zone and 6\% of Pa-231 at outer zone has better performance with 60 years of operation period than reactor with combination of 8\% of Np-237 at center zone and 6\% of Pa-231 at outer zone that reaches 46 years lifetime.

1. Introduction

Small reactors are reactors that have power rating of 300 MW(e) or less, while small reactors without on site refuelling or small long-life reactors are reactors that have a capability to operate in a long period of time without reloading or fuel shuffling in their cores, so that there is no fresh or spent fuel being stored at the site during reactor operation. This type of reactor are being developed for several reactor lines, such as water cooled reactors, sodium cooled fast reactors, lead bismuth cooled reactors, etc [1]. Pressurized water reactor (PWR) is the most used type of reactor in producing electricity in the world. Conventional PWR used enriched uranium as fuel with 1.5 – 2.5\% enrichment and 1.5 years for one fuel batch loading [2]. The idea of developing small reactors, especially small PWR is at early stages, so it needs further research and development (R&D) steps.

Thorium as PWR fuel is chosen because of its superiority such as: (1) thorium abundance is three times greater than uranium abundance; (2) Thorium-232 (Th-232) and Uranium-233 (U-233) are the best fertile – fissile materials for thermal reactors; (3) thorium’s conversion ratio (CR) is higher than uranium in thermal spectrum, so that the reactor could be operated for a longer time; and (4) it’s thermal absorption product, U-233, has higher thermal fission factor [3] [4] [5].

In reactors operation, there is some kind of material that is added at the fuel as doping that is known as burnable neutron absorber or burnable poison. The main purposes of burnable poisons addition are: (1) to lower the excess reactivity at the beginning of fuel irradiation; (2) to increase the
achievable release of reactivity (due to higher fuel burn-up) at the end of fuel irradiation; and (3) to flatten the heat release field in the core [6]. The common material used for burnable poisons are gadolinium, erbium, and boron, but their daughter nuclides don’t play any role in affecting the neutronics characteristics of the core.

Protactinium-231 (Pa-231) and Neptunium-237 (Np-237) are the promising nuclides that could be used as burnable poisons. The use of these nuclides could reduce initial reactivity excess, increase fuel lifetime, and reach high level fuel burn-up [7]. Pa-231 could also play a role as fertile material, for its ability to produce U-233 [8]. Based on the reasons mentioned above, this study was conducted to investigate the neutronic behaviour of small long life PWR with Thorium cycle that use Pa-231 and Np-237 as burnable poisons.

2. Theoretical Background

Theoretical concept of neutronics calculation for reactors design are neutron diffusion equation, multiplication factor, and reactor power distribution [2].

2.1. Neutron Diffusion Equation

Diffusion equation of neutron describes how neutron acts inside the core. In SRAC2006 code system, the group-dependent diffusion equation in the $I$-th region is defined mathematically as

\[
-D_{gI} \nabla^2 \Phi_g \left( \vec{r} \right) + \Sigma_{n, gI} \Phi_g \left( \vec{r} \right) = S_g \left( \vec{r} \right)
\]

where $D_{gI}$ is averaged diffusion coefficient, $\Phi_g \left( \vec{r} \right)$ is flux of group $g$, $\Sigma_{n, gI}$ is macroscopic total cross-section of group $g$, and $S_g \left( \vec{r} \right)$ is neutron source term.

2.2. Multiplication Factor

Multiplication factor is a parameter indicates neutron population inside the core and mathematically defined as

\[
k = \frac{\text{neutron numbers in one generation}}{\text{neutron numbers in preceeding generation}}
\]

or

\[
k = \frac{P(t)}{L(t)}
\]

where $P(t)$ is neutron production rate and $L(t)$ is neutron loss rate. If $k > 1$, reactor is supercritical. If $k = 1$, reactor is critical, and if $k < 1$, reactor is subcritical. Ideally, reactor should be on critical condition.

2.3. Reactor Power Distribution

Reactor power distribution shows how power is distributed spatially inside the reactor and represented by

\[
q^s \left( r \right) = \sum_i w_j^{si} N_i \left( r \right) \int_0^\infty dE \sigma_f^{si} \left( E \right) \varphi \left( r, E \right)
\]

where $w_j^{si}$ shows energy amount released per fission reaction and $N_i \left( r \right) \int_0^\infty dE \sigma_f^{si} \left( E \right) \varphi \left( r, E \right)$ is fission reaction rate density for isotope $i$.

3. Reactor Design Parameter

Reactor design parameter used in this research is shown on the table below.
Table 1. Reactor design parameter

| Parameter                      | Value                              |
|--------------------------------|------------------------------------|
| Thermal power output           | 150 – 300 MW(t)                    |
| Active core diameter           | 196 cm                             |
| Active core height             | 196 cm                             |
| Fuel                           | (Th,U-233)O₂                        |
| U-233 percentage               | 8%                                 |
| Burnable poison (BP)           | Pa-231 and Np-237                  |
| BP percentage                  | 2 – 10 %                           |
| Pin pitch                      | 1.4 cm                             |
| Fuel volume fraction           | 40 – 60 %                          |
| Cladding volume fraction       | 10 %                               |
| Moderator volume fraction      | 30 – 50 %                          |

Table 1 shows reactor design parameter used in this study. Parameters such as active core diameter and height, U-233 percentage, pin pitch, and cladding volume fraction is fixed at certain value. The other such as thermal power output, BP percentage, and fuel volume fractions are varied in certain range. As fuel volume fraction is varied, moderator volume fraction is changed.

In this study, the fuel used is (Th,U-233)O₂. Th-232 and U-233 is the fertile – fissile nuclide pair in this type of fuel. Pa-231 and Np-237 is used as burnable poison to decrease excess reactivity at BOC (beginning of cycle). U-233 value is fixed at 8% while BP percentage is varied in survey parameter step to examine how BP content affect reactor reactivity.

4. Calculation Method

The reactor’s neutronics calculation is done by SRAC2006 code developed by Japan Atomic Energy Agency (JAEA) and the nuclide data library used is JENDL-4.0. Fuel cell calculation is carried out by PIJ that use collision probability method. Cell burn-up calculation is also carried out in this module. The output obtained is infinite multiplication factor ($k_{inf}$). Core calculation is done by CITATION module that solve multi-dimensional diffusion calculation. The core geometry is two dimensional (R – Z) cylinder. There were two zones of fuel in the core, those were 1st fuel zone (F1) and 2nd fuel zone (F2). The BP concentrations were similar for both zones in homogen core calculation, but different in heterogen core calculation. Figure 1 below shows fuel configuration in reactor core.
5. Result and Discussion

First calculation step is parametric survey. In this step, the effect of parameters variation is investigated in ten years of operation. First calculation in parametric survey is conducted by varying fuel volume fraction from 40% to 60% with 5% interval at power output 250 MWt. For each variation, cladding volume fraction is fixed at 10%, resulting in the change of moderator volume fraction. The output investigated is $k_{inf}$ (multiplication factor without taking core geometry into account) for each year for fuel with and without BP.

Figure 1. Core design for 2 dimensional (R – Z) core.
Figure 2. Effect of fuel volume fraction (FF) variation on $k_{inf}$ change in (a) reactor without BP, (b) reactor with 2% Pa-231, and (c) reactor with 2% Np-237.

Figure 2 shows how variation in fuel volume fraction (FF) value affect reactor $k_{inf}$. From the figure above, it could be seen that the bigger fractions of FF yield lower value of $k_{inf}$. If cladding has constant volume fraction, then the change in FF also change moderator volume fraction that leads to the change of moderator to fuel ratio (MFR) value. High MFR value leads to high fuel burn-up level and rapid fissile nuclide depletion. This could lead to rapid change of $k_{inf}$ value (high reactivity swing). If fuel has bigger volume fraction and smaller moderator fraction, then the $k_{inf}$ value is low enough so that the $k_{inf}$ change is not so rapid. This could be seen from figure 2. FF 60% has lower $k_{inf}$ and its change is not as rapid as $k_{inf}$ of core with higher FF. From figure 2 it also could be seen that addition of BP in fuel could reduce $k_{inf}$ value.
The next calculation step is variations of BP concentration values from 2% to 10% with 2% interval. The fuel volume fraction used is 60% and the results is shown in the figures below.

**Figure 3.** Effect of burnable poison (BP) concentrations variation on $k_{\text{inf}}$ change in (a) reactor with Pa-231 as BP and (b) reactor with Np-237 as BP.

Figure 3 shows how change in BP concentration affects $k_{\text{inf}}$ value. From that figure, it could be seen that higher BP concentration gives lower $k_{\text{inf}}$ value. In the case of Np-237 as BP, $k_{\text{inf}}$ value is decreasing without having any increase. In other case, the using of Pa-231 as BP not only able to reduce $k_{\text{inf}}$ value at BOC but also could increase its value as the operation going on at 6%, 8%, and 10% Pa-231. Np-237 and Pa-231 could transmute into fissile nuclide as shown in the figure below.

**Figure 4.** Transmutation of Np-237 into Pu-239.

**Figure 5.** Transmutations of Pa-231 into U-233.

Pu-239 and U-233 are fissile nuclides. So the addition of Np-237 and Pa-231 could increase fissile nuclide content in the fuel. But from figure 3, it could be seen that addition of Np-237 doesn’t increase
$k_{\text{inf}}$ value as Pa-231 does. This is because Pu-239 has higher capture to fission ratio ($\alpha$) than U-233. Capture to fission ratio is the ratio of radiative capture cross section to fission cross section. In thermal energy range, $\alpha$ value of Pu-239 and U-233 is 0.4 and 0.09 respectively. Better fissile nuclide has lower $\alpha$ value. Beside that, the addition of Np-237 is also able to produce more plutonium isotopes. Odd isotopes of plutonium is good fissile nuclide and have low $\alpha$ value, but its even isotopes tend to act like poison that capture more electrons and have small fission cross sections (high $\alpha$ value). In this case, this explains why addition of Pa-231 from 6% to 10% could decrease $k_{\text{inf}}$ at BOC but also could increase its value as operation goes on and addition of Np-237 from 2% to 10% keep reduce $k_{\text{inf}}$ value without any increase.

The optimal BP concentration obtained for both Pa-231 and Np-237 is 6% because at this value, reactor could operate for ten years in critical condition. The next calculation step is conducted with varied power density with CITATION module to get $k_{\text{eff}}$ value (multiplication factor by taking core geometry into account). At the previous calculation steps, the variations is conducted at power 250 MWt. In this case, the power density is adjusted to obtain output thermal power of 150 MWt, 200 MWt, 250 MWt, and 300 MWt with fixed core dimension. The purpose is to observe how power density affect $k_{\text{eff}}$ value and how long the reactor could operate before it reaches subcritical condition. In this step, core configuration is shown at figure 1 with both F1 and F2 fuel zone used similar fuel with similar BP concentration, that was 6% (homogen fuel configuration). The calculation was conducted on two types of reactors, with only Pa-231 and only Np-237 as the BP for each type. The result is shown below.

![Figure 6](image-url)

Figure 6. Value of $k_{\text{eff}}$ on (a) reactor with 6% Pa-231 and (b) reactor with 6% Np-237.
From figure 6 it could be seen that lower power density give longer operation period. This is because lower power density leads to lower fuel burn-up level that reduce fissile nuclide depletion rate. From figure 6 it’s also known that reactor with Pa-231 as BP has longer lifetime (before reaching subcritical condition) for about 60 years than reactor with Np-237 that has about 12 years of lifetime. For the next calculation step, the power used is 150 MWt with power density 25.36 Watt/cc because it has longest operation period with lowest reactivity swing. Radial and axial power distribution for homogen fuel configuration are shown below.

![Radial Power Distribution](image1)

(a)

![Axial Power Distribution](image2)

(b)

**Figure 7.** Power distribution of homogen fuel configuration in (a) radial direction and (b) axial direction at beginning of cycle (BOC).

Figure 7 show radial and axial power distribution of reactor with homogen fuel configuration. Each configuration has different BP, those are Pa-231 and Np-237. From figures above, it could be seen that core with Np-237 as BP has slightly lower relative power that core with Pa-231. Maximum radial relative power for core with Pa-231 and Np-237 are 1.89 and 1.81 respectively, while maximum axial relative power for core with Pa-231 and Np-237 are 1.42 and 1.39 respectively. The radial and axial power distribution for homogen core configuration is not flat. To flatten power distribution profile, heterogen fuel configuration is used. There are two model that is investigated, model A and model B. For model A, F1 has 8% Pa-231 and F2 has 6% Pa-231 (Pa-Pa poisoned). While in model B, F1 has 8% Np-237 and F2 has 6% Pa-231 (Pa-Np poisoned). The keff obtained for each model is shown in the figure below.
Figure 8. $k_{\text{eff}}$ of heterogen reactor.

Figure 9. Power distribution of heterogen fuel configuration in (a) radial direction and (b) axial direction at beginning of cycle (BOC).

Figure 8 shows $k_{\text{eff}}$ values of homogen reactor for model A (6% Pa-8% Pa) and model B (6% Pa-8% Np). Model A has longer lifetime, about 60 years, while model B has 46 years of critical period. This is happened because model A has more Pa content than model B.
Figure 9 shows radial and axial power distribution for heterogen fuel configuration, model A and B. Maximum radial relative power for model A and B are 1.55 and 1.68 respectively, while maximum axial relative power for model A and B are 1.30 and 1.35 respectively. It’s clear that heterogen fuel configuration has lower maximum axial and radial relative power than homogen configuration because heterogen configuration has higher BP content in the center of the core that flatten the core power profile. From figure 9, it could also be seen that model A (Pa-Pa poisoned core) has flatter power profile than model B (Pa-Np poisoned core). This difference is caused by the different reactivity of fuel with Np-237 and fuel with Pa-231. From figure 3, it could be seen that 6% Pa-231 has higher $k_{inf}$ (and higher reactivity) than 8% Pa-231 and 8% Np-237. Putting fuel with higher reactivity in outer zone (F2) leads to power flattening. And then from figure 3, it’s also known that 8% Pa-231 has lower reactivity than 8% Np-237. So that Pa-Pa combination has lower neutron flux in the core center than Pa-Np combination, and this also leads to lower maximum axial and radial power distribution for Pa-Pa combination. From calculation results and the analysis conducted, the optimal combination of fuel –BP to be used in small long-life reactor is U-233 and Pa-231.

6. Conclusion
Conceptual analysis of (Th,U-233)O$_2$ fueled small long-life PWR with Np-237 and Pa-231 as burnable poison has been performed. Cell calculation results show that optimum design is reached at 60% of fuel volume fraction and 6% of BP concentration for both Pa-231 and Np-237. Homogen reactor calculation results show that reactor poisoned by 6% of Pa-231 has longer lifetime, especially the 150 MWt reactor that reaches 60 years of operation in critical condition. But from power distribution point of view, reactor poisoned with 6% of Np-237 has better performance with slightly lower maximum relative power in axial and radial direction. To flatten power distribution profile, heterogen fuel configuration is used. Based on calculation result and analysis conducted, the best configuration is model A with 8% Pa-231 in F1 zone and 6% Pa-231 in F2 zone. The maximum relative power in axial and radial direction at BOC are 1.30 and 1.55. This configuration also allow reactor to be operated in critical condition for 60 years.

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