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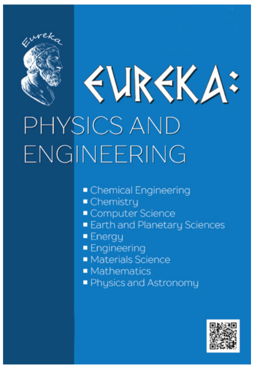

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CONTENT

NEUTRONICS ANALYSIS OF UN-PUN FUEL FOR 300 MW PRESSURIZED WATER REACTOR USING SRAC-COREBN CODE

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Abstract

Nuclear Power Plants (NPP) is one of alternative energy that can be used to replace fossil energy. NPP is a clean energy that doesn't emit CO_2 as a residue from the power plant. Pressurized Water Reactor (PWR) is the most widely used commercial reactor to date. This reactor usually uses UO_2 as fuel with enrich U-235 and produce plutonium as a waste nuclear. In this study, the fuel uses natural uranium and the addition of plutonium from the spent fuel of PWR, calling it Uranium-Plutonium Nitride (UN-PuN) fuel. Plutonium from spent fuel PWR (U-fueled) consists of Pu-238, Pu-239, Pu-240, Pu-241 and Pu-242, which Pu-239 and Pu-241 are fissile material. The aim of using plutonium is to reduce the amount of nuclear waste and prevent the use of plutonium as a nuclear weapon. The addition of neptunium and protactinium is also carried out in the fuel. Neptunium is one of the minor actinide materials which have high toxicity, so add Np-237 it is very useful to reduce the amount of neptunium in the world. And also, the addition of Np-237 and Pa-231 aims to decrease the k_{eff} value both in the Beginning of Life (BOL) and End of Life (EOL) in the reactor. This research calculates the neutronic calculations for PWR use SRAC2006 with COREBN Code and Japanese Evaluated Nuclear Data Library 4.0 (JENDL-4.0) for the nuclear data library. The COREBN code is a calculation method based on interpolation of macroscopic cross-section and finite-difference diffusion theory. The optimum design reached the maximum k-eff value 1.017077 and the maximum power density was 58.47 W/cc. The results obtained are the excess reactivity value of 1.68 % smaller than previous study which produces an excess reactivity value of 11.94 %. So this design has the potential to be applied in the PWR design.

Keywords: PWR, UN-PuN, SRAC-COREBN, neutronics calculation, minor actinide.

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1. Introduction

The need for energy in Indonesia has increased by 6 % per year [1]. The increasing need for electrical energy sources is in line with the number and level of the population and science and technology from year to year. It requires discovering alternative energy sources, one of which is nuclear energy as a Nuclear Power Plant [2]. For Indonesia, especially Java island, large or medium

Nuclear Power Plants (NPP) are recommended to be built in the future because the Indonesian population is concentrated on this island [3].

NPP is a power plant that uses one or several nuclear reactor units to produce steam for turning an electric turbine. The development of nuclear reactors starts from generation I to generation IV. Generation I reactors are early prototypes that aim to prove that nuclear energy can be put to good use. Generation II reactors are designed to be economical and reliable such as BWR, PWR, and CANDU. Generation III reactors is a development from the previous generation (Generations I and II) in terms of the length of reactor operation. Generation III+ reactors are an evolutionary development of generation III reactors, while generation IV reactors are designed to supply electrical power and supply thermal energy for industry [4].

The reactor is distinguished by two types of material that undergo fission or fission, called fissionable material, namely fissile material and fertile material. Fissile material, which is when a neutron shoots it with a certain amount of energy, will undergo fission. In contrast, fertile material, which is a material that captures neutrons by radioactive decay, will turn into fissile material [5]. The fission of uranium produces neutrons. The resulting neutrons can shoot back the uranium nucleus so that the subsequent fission reaction occurs. The mechanism takes a swift time to form an uncontrollable chain reaction that releases a large amount of energy. The release of energy can be used as a power plant if the fission chain reaction is controlled [6].

Pressurized Water Reactor (PWR) is a nuclear power reactor using light water that functions as a coolant and moderator. The water pressure reactor was initially developed by the Westinghouse Bettis Atomic Power Laboratory and the United States Government Research Center in Argonne. Previously, the PWR was developed as submarine propulsion. Nautilus is the name of a nuclear-powered submarine that operated from 1954 to 1980. Making submarine reactors led the Westinghouse Company to build a power plant reactor with a power of 100 MWe [7].

Previous research on the design of a PWR has been carried out. The research used the thorium cycle as fuel with SRAC Code Calculation. The calculation is divided into two steps, i. e., PIJ calculation, to calculate cell calculation, and second, CITATION calculation, to calculate core calculation [8–11]. Moreover, there was research using thorium fuel and SRAC-COREBN calculation [12]. The other research also has been calculated the neutronics analysis using uranium nitride (UN) and uranium-plutonium nitride (UN-PuN) fuel for PWR [13]. The results of this study indicate that the use of UN-PuN fuel can increase the k-eff value and reduce nuclear waste because it uses plutonium as a fuel mixture [13].

Based on previous studies, this research designs a small PWR with Uranium Plutonium Nitride (UN-PuN) fuel using SRAC-COREBN calculation for long-life fuel without refueling. This research uses natural uranium fuel, and plutonium left over from the Light Water Reactor (LWR) waste to reduce spent nuclear fuel in the world. In this research, also add Protactinium 231 and Neptunium 237.

Previous research on the addition of minor actinides in the fuel in the Gas-Cooled Fast Reactor (GFR) has been carried out [14–16]. The study found that the addition of minor actinides could reduce the k-eff value. Therefore, in this study, Np-237 will also be added as a minor actinide with a half-life of 2.1×10^6 years.

Neutronic calculation in this study uses SRAC2006 which is a code system for analysis of neutronic calculations in several types of thermal reactors under the Linux Operating System. The data library uses the JENDL4.0 nuclear data library. The first calculation is fuel pin cell calculation (PIJ method) and reactor core calculation (COREBN method). The core combustion calculation is optimized in X-Y-Z geometry by COREBN to reduce moderator with lattice model based on cell geometry [17]. The parameters analyzed in this study were fuel enrichment, burn-up, core configuration (homogeneous and heterogeneous), criticality, and power density distribution. The criticality of the reactor is indicated by the value of the influential multiplication factor (k-eff). The reactor core in this study was designed using a cylindrical core model.

2. Materials and methods

In a nuclear reactor, determining the distribution of neutrons is a major problem, because determining the number of neutron distributions will affect the rate of nuclear reactions in the reactor.

The distribution of neutrons in a reactor can be analyzed using the neutron transport theory. Neutron transport describes the movement (flow) of neutrons in the reactor core, the occurrence of neutron scattering with the atomic nucleus, the absorption of neutrons by the atomic nucleus and the leakage of neutrons from the reactor. Most of the reactor studies treat the movement of neutrons as a diffusion process. As a result, neutrons diffuse from areas of high neutron density to areas of low neutron density. The description of neutron transport as a diffusion process has limited validity, but in the case of reactor operation the diffusion equation is quite accurate [18]. SRAC code uses a multigroup diffusion equation approach to calculate the distribution of neutrons in the reactor.

Based on the neutron balance equation, the diffusion equation mathematically can be written as follows:

$$\frac{1}{\upsilon}\frac{\partial \Phi_g}{\partial t} = \nabla.D_g\nabla\Phi_g - \sum\nolimits_{ag}\Phi_g + S_g - \sum\nolimits_{sg}\Phi_g + \sum_{g'=1}^{G}\Sigma_{sg'g}\Phi_{g'} \tag{1}$$

with D – the diffusion constant; S_g – neutron source; Φ – neutron flux depends on space and energy; $\nabla.D_g\nabla\Phi_g$ – leakage; $\sum\nolimits_{ag}\Phi_g$ – absorption rate; $\sum\nolimits_{sg}\Phi_g$ – number of incoming neutrons due to scattering; $\sum_{g'=1}^{G}\Sigma_{sg'g}\Phi_{g'}$ – the number of neutrons decreases due to scattering.

The reactor design specifications used in this research are shown in **Table 1**. The specifications include specifications for the reactor core and the fuel used. This study uses cylinder-shaped fuel pin geometry. The use of cylindrical cell geometry in this study, because the cylindrical cell type does not provide space between other fuel cells when arranged in the reactor core. The cell is divided into three regions consisting of fuel, cladding, and coolant (moderator). The division of the cylindrical cell type region is shown in **Fig. 1**.

The three-dimensional design of the fuel core (*X-Y-Z*) at the HIST and COREBN inputs is shown in **Fig. 2**. The fuel element consists of an upper nozzle, an upper gas plenum, fuel passes through 14 nodes, and a lower nozzle to simulate the fuel assembly.

This research uses Uranium Plutonium Nitride (UN-PuN) as fuel. The uranium used is natural, namely U234, U235, and U238, with content of 0.005 % total U234, 0.72 % total U235, and 99.27 % U238. In contrast, the plutonium used is from the rest of the light water reactor waste fuel (Light Water Reactor). Plutonium is taken from waste fuel PWR, which does a burn-up of 33 MW/ton, aiming to reduce the amount of plutonium in the world. The percentage of plutonium isotopes from the amount of plutonium in PWR waste is shown in **Table 2**.

Table 1
Reactor design specifications

Design Parameters	Specifications
Power	300 MWth
Burn-up	Ten years
Fuel	UN-PuN
Cladding	Zirconium
Coolant	Water (H_2O)
Fuel pin cell type	cylindrical
Pin pitch	1.45 cm
Height active core	100 cm
Diameter active core	80 cm
Fuel fraction	40–65 %
Cladding fraction	10 %
Moderator fraction	25–50 %
Pin pitch	1.45 cm

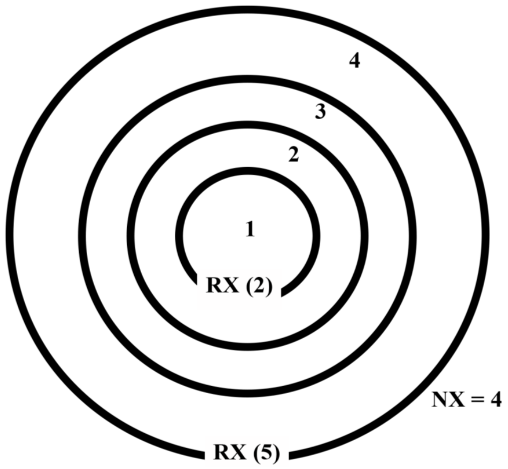

Fig. 1. Regional division of cylindrical cell types [17]

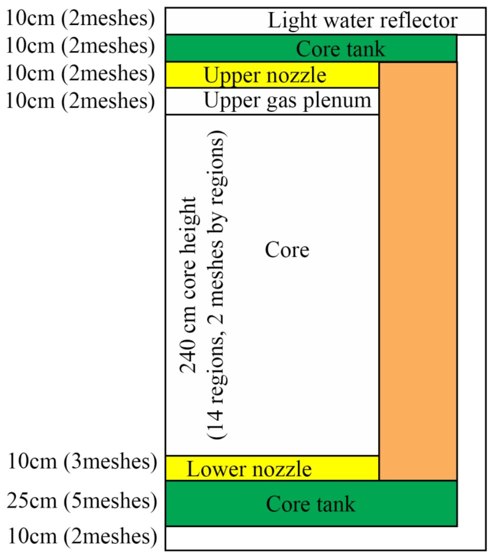

Fig. 2. Burn-up calculation model types

Table 2
Plutonium percentage from PWR (U-fueled) Burn-up 33 MWd/ton and the half-life [19]

Isotope	PWR (U-fueled) Burn-up 33 MWd/ton	Half-life (years)
Pu-238	1.8 %	87.74
Pu-239	58.7 %	24100
Pu-240	24.2 %	6560
Pu-241	11.4 %	14.4
Pu-242	3.9 %	376000

3. Results and discussion

This study uses a PWR reactor in the form of a module with UN-PuN fuel. The uranium used is natural, while the plutonium was from spent nuclear fuel from LWR. The neutronic calculation begins with the calculation of the homogeneous core configuration, secondly calculated heterogeneous core configurations, thirdly calculated the addition of Protactinium-231 (Pa-231) and Neptunium-237 (Np-237), fourthly calculated the variation of the fuel volume fraction, fifthly calculated the power variation and sixth is the fuel optimization.

The homogeneous core configuration use variation of percentage plutonium from 3 % up to 15 %. **Fig. 3** shows the *k*-inf value, which results from fuel cell calculations without considering the level of neutron leakage. **Fig. 3** shows that the more significant plutonium percentage made the *k*-inf value also more fantastic. After the cell calculation used PIJ calculation, the following was reactor core calculation (COREBN calculation). **Fig. 4** shows the *k*-eff value of homogeneous core configuration. Based on the *k*-eff value obtained, the variation of plutonium 6 % indicates the most critical condition with the flattest trend line. The maximum excess reactivity value was obtained at 7.36 %.

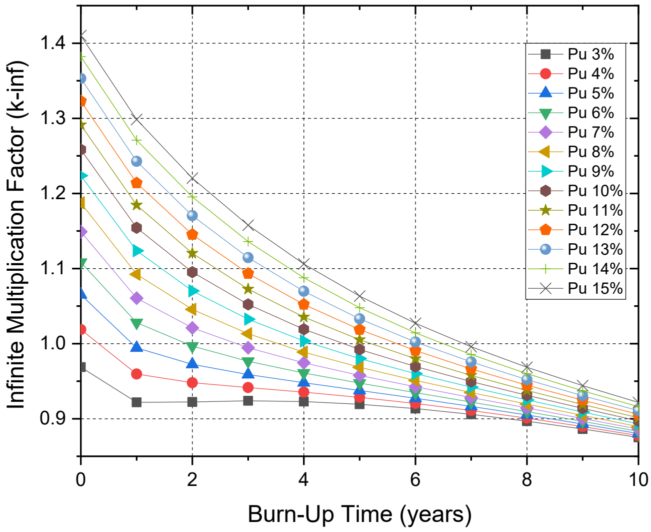

Fig. 3. The *k*-inf value of the homogeneous core configuration

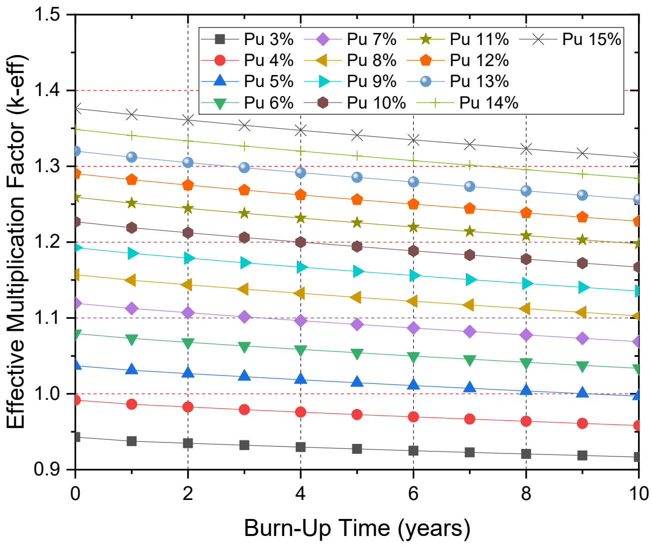

Fig. 4. The *k*-eff value of the homogeneous core configuration

Fig. 5 shows the *k*-eff value in heterogeneous core configurations. The most critical condition of the reactor was obtained when the percentage variation of plutonium was $F1 = 5$ %, $F2 = 6$ %, $F3 = 7$ % with the maximum *k*-eff is 1.067487, and the maximum excess reactivity value is 6.32 %.

The comparison of the power density distribution in the radial direction for homogeneous and heterogeneous core configurations can be seen in **Fig. 6**. There is a peaking power density in a homogeneous core configuration. The peaking power density is decreased by using a heterogeneous core configuration. The comparison of homogeneous and heterogeneous power density values is shown in **Table 3**.

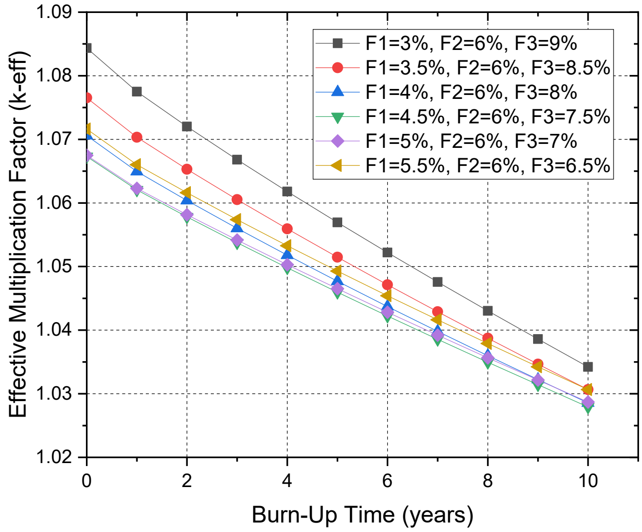

Fig. 5. The k-eff value of heterogeneous core

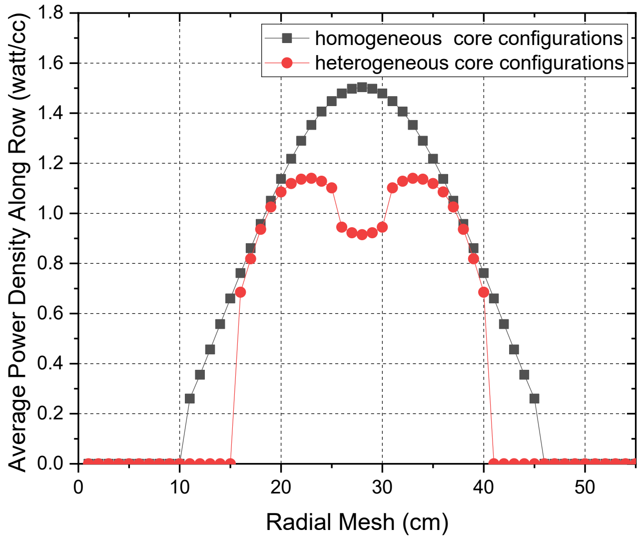

Fig. 6. The power distribution comparison of homogeneous and heterogeneous core

Table 3
Comparison of homogeneous and heterogeneous power density values

No.	Core configuration	Percentage of Plutonium	Power density (Watt/cc)	
			Average	Maximum
1	Homogeneous core	$F1=6$ %, $F2=6$ %, $F3=6$ %	49.71	74.73
2	Heterogeneous core	$F1=5$ %, $F2=6$ %, $F3=7$ %	49.92	56.90

To decrease the excess reactivity value, the authors added the Pa-231 and Np-237. The Pa-231 was also called burnable poison because it could absorb the neutron, reduce the k-eff value, and decrease the excess reactivity.

Fig. 7 shows the k-eff value of addition Pa-231 with variations of Pa-231 from 0 % up to 3 % with an interval of 0.5 %. The results experienced a decrease in the overall k-eff value from the beginning to the end of the year. At the percentage of Protactinium 231 (Pa-231), 0.5 % has experienced a subcritical state, which means the *k*-eff value is less than one. In Protactinium 231 (Pa-231), a maximum power density value is produced, that the greater the variation in the percentage of protactinium, the greater the maximum power density.

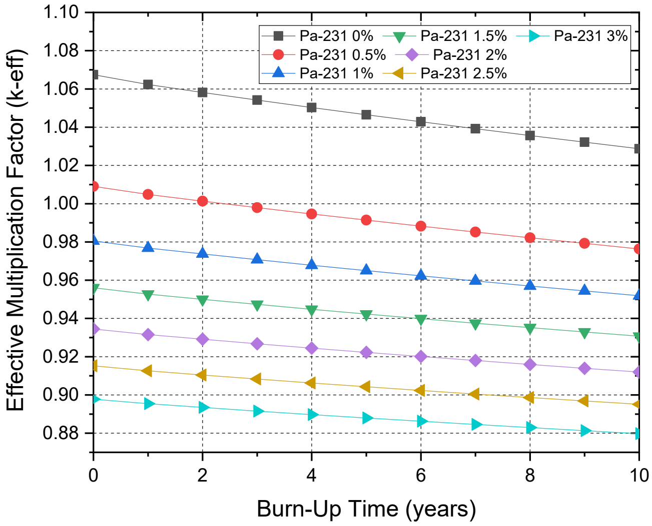

Fig. 7. The *k*-eff value of addition Pa-231

In this study, the addition of neptunium 237 was also carried out, reducing the value of *k*-eff. **Fig. 8** shows the relationship between multiplication factors and burn-up time with the addition of Neptunium 237 (Np-237).

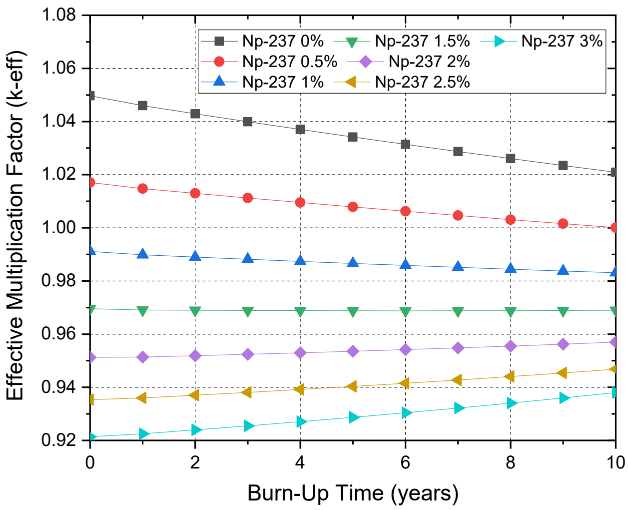

Fig. 8. The *k*-eff value with the addition of Np-237 from 0 % up to 3 %

Fig. 8 shows the *k*-eff value with the addition of neptunium 237. The variation of neptunium 237 is from 0 % to 3 % with an interval of 0.5 %. The **Fig. 8** shows that the *k*-eff value decreases at the beginning of the burn-up year due to Np-237 absorbing neutrons. The greater the variation in the percentage of neptunium, the greater the maximum power density value, but not significant. The chain burn-up from Neptunium 237 (Np-237) up to Plutonium 239 (Pu-239) shows in **Fig. 9.**

The addition of Neptunium 237 can be used in this study because it can reduce the *k*-eff value. The optimum *k*-eff when we variated the Np-237 from 0 % up to 3 % is the variation in the percentage of neptunium 237 is 0.5 % in the variation of plutonium $F1 = 5$ %, $F2 = 6$ %, $F3 = 7$ % with a fuel volume fraction of 65 % the maximum value of *k*-eff is 1, 017077 with an excess reactivity value of 1.68 % and a maximum power density value of 58.47 Watt/cc. From the calculation of the addition of neptunium 237, it is the most critical or stable result compared to other calculations such as the calculation of the homogeneous core configuration, the heterogeneous core configuration, and the calculation of the addition protactinium 231. Then, the next step of the calculation is the variation of fuel volume fraction. The fuel volume fraction is varied from 40 % to 65 %, as shown in **Fig. 10**.

Fig. 10 shows that the greater the variation in the volume fraction of the fuel, the smaller the k-eff value. The k-eff value produced by adding Neptunium 237 (Np-237) can reduce the k-eff value to close to the value. The results obtained are the excess reactivity value of 1.68 % smaller than previous study [13], which produces an excess reactivity value of 11.94 %.

In this study, variations of reactor power were also carried out using nine variations from 100 MWth to 500 MWth with 50 MWth intervals. The greater the power supplied, the greater the maximum power density value. The relationship of k-eff to burn-up time is shown in **Fig. 11**, which shows that the greater the power supplied, the smaller the k-eff value.

$$ {}^{237}_{93}Np \xrightarrow{(n,\gamma)} {}^{238}_{93}Np \xrightarrow{\beta} {}^{238}_{94}Pu \xrightarrow{(n,\gamma)} {}^{239}_{94}Pu $$

Fig. 9. The chain burn-up Np-237 become Pu-239

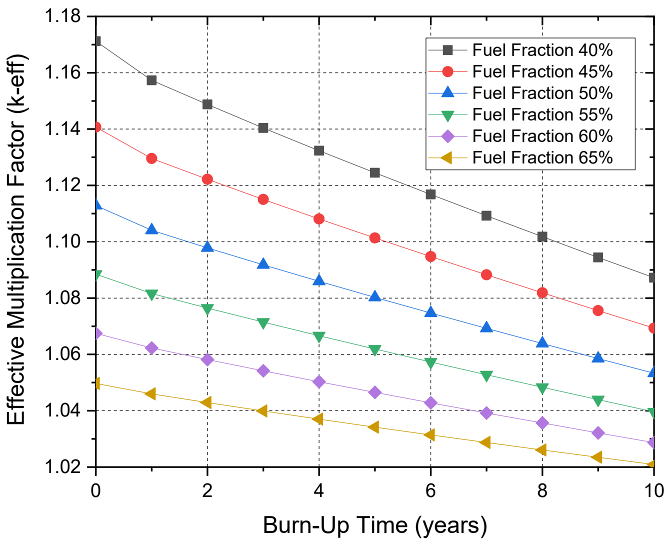

Fig. 10. The k-eff value with variations fuel fractions (60–65 %)

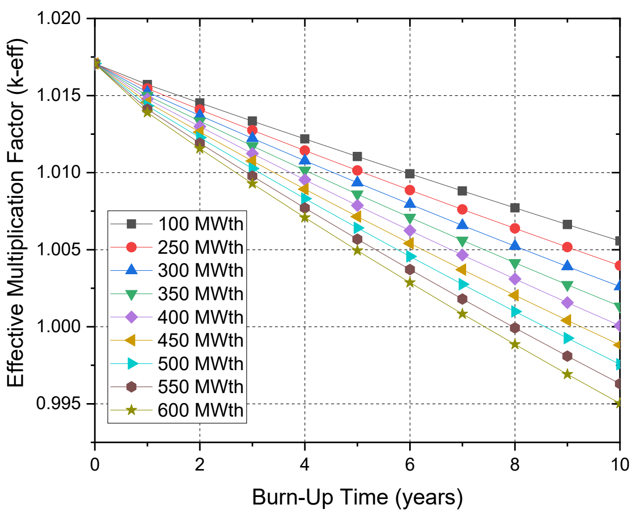

Fig. 11. The k-eff value with variations power from 100 up to 500 MWth

The analysis of nuclear reactors is divided into 3, namely, neutronic analysis, thermal-hydraulics, and safety. The neutronic analysis is related to fuel control. Burn-up analysis was carried out to determine the characteristics of isotopic changes in the reactor. Combustion analysis also provides atomic density parameters that indicate changes in reactor fuel density [20, 21]. The burn-up analysis is a calculation that focuses on fuel, namely the size of the fuel, the combustion process, processing, and the amount of energy produced per unit weight of fuel expressed in Mega Watt-days (MWd) of each tonne of fuel [21]. **Fig. 12** shows the burn-up level value for power variations from 100 MW up to 500 MW. It can be seen in the Figure that the greater the power supplied, the greater the burn-up level value.

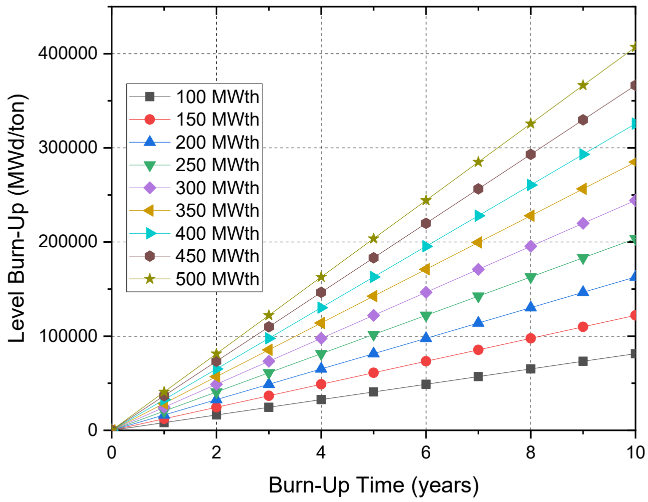

Fig. 12. The level burn-up value with variations power from 100 up to 500 MWth

The last step in this research is to optimize the value of k-eff, whose excess reactivity value is close to the value of one. The optimization was done by calculating homogeneous and heterogeneous core configurations and adding Protactinium 231 (Pa-231) and Neptunium 237 (Np-237) in the fuel. Heterogeneous core configuration with the percentage of plutonium $F1 = 5$ %, $F2 = 6$ %, $F3 = 7$ % resulted in a k-eff value of 1.0674875 with an excess reactivity value of 6.32 %. Heterogeneous core configuration at the percentage of plutonium $F1 = 5$ %, $F2 = 6$ %, $F3 = 7$ % with the addition of a percentage of Protactinium 231 (Pa-231) 0.5 % to obtain a k-eff value of 1.0091004 and an excess reactivity value of 0.90 %, but is in a subcritical state. While the heterogeneous core configuration at the percentage of plutonium $F1 = 5$ %, $F2 = 6$ %, $F3 = 7$ % with the addition of a percentage of Neptunium 237 (Np-237) 0.5 % to get a k-eff value of 1.0717777 and an excess reactivity value of 1.68 %. **Fig. 13** shows the results of the optimization calculations that have been carried out.

Fig. 13 in the black graph is a calculation of the heterogeneous core configuration, which shows that the k-eff value is the largest than the others, with the excess reactivity value far from the value of one so that it must reduce the k-eff value with the addition of Protactinium 231 (Pa-231) and the addition of Neptunium 237 (Np-237). In the calculation of the addition of Protactinium 231 (Pa-231), which is shown in the red graph where the k-eff value is smaller than the graph but experiences a subcritical state which means the k-eff value is less than one, so it is not used as an additive in this study. While the calculation with the addition of Neptunium 237 (Np-237) shown in the blue graph is the most stable result with the excess reactivity value approaching the value of one. The k-eff value generated by the addition of Neptunium 237 (Np-237) can reduce the k-eff value to close to a value of one, for that the fuel volume fraction is increased to 65 % so that the k-eff value and excess reactivity decrease until the results obtained are excess reactivity is 1.68 % smaller than previous research which produces an excess reactivity value of 11.94 % [13]. All calculations in this study, with an optimal design and can operate for 10 years, namely the addition of Neptunium 237 (Np-237) at a percentage of 0.5 % at the percentage of plutonium $F1 = 5$ %, $F2 = 6$ %, $F3 = 7$ % with fuel volume fraction 65 %. The following **Fig. 14** shows the optimization results using Uranium Plutonium Nitride (UN-PuN) in a Pressurized Water Reactor (PWR), which can operate for ten years without refueling.

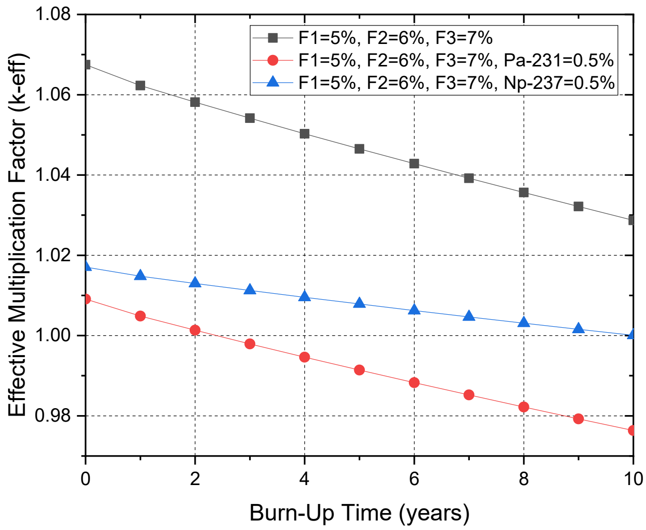

Fig. 13. The comparison of k-eff value by adding Pa-231 and Np-237

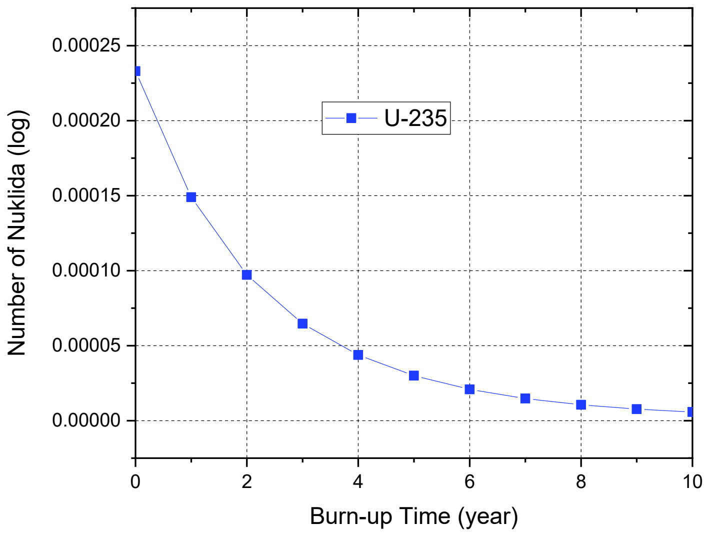

Fig. 14. Graph of the relationship between the number of U-235 nuclides and the burn-up time

Fig. 14, 15 show the number of essential nuclides, namely U-235, U-238, and Pu-239. **Fig. 14** shows that the atomic density of U-235 (fissile material) decreases due to a fission reaction in the reactor. Meanwhile, the number of U-238 nuclides decreased to form Pu-239. Pu-239 was formed due to the capture of fast neutrons in the reactor by U-238. The atomic density of Pu-239 increases due to the presence of U-238 (fertile material) passing through Np-239 to form Pu-239 (fissile material), as shown in **Fig. 16**.

This research is limited to the neutronic calculation in the reactor, it is necessary to develop further on the thermal-hydraulic aspects of the designed reactor design and the safety analysis calculations for ULOF (unprotected loss of flow) and UTOP (unprotected rod run-out transient overpower) analysis, so that studies are carried out not only on the neutronic aspect, but also on the hydraulic thermal analysis to ensure that the resulting reactor design is quite reliable.

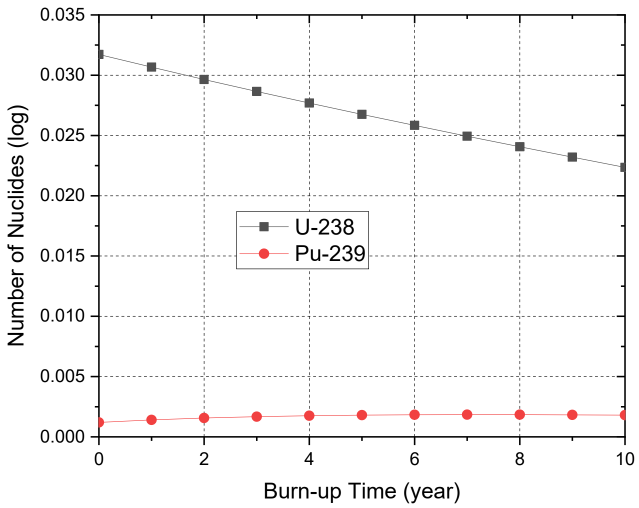

Fig. 15. Graph of the relationship between the number of U-238 and Pu-239 nuclides to burn-up time

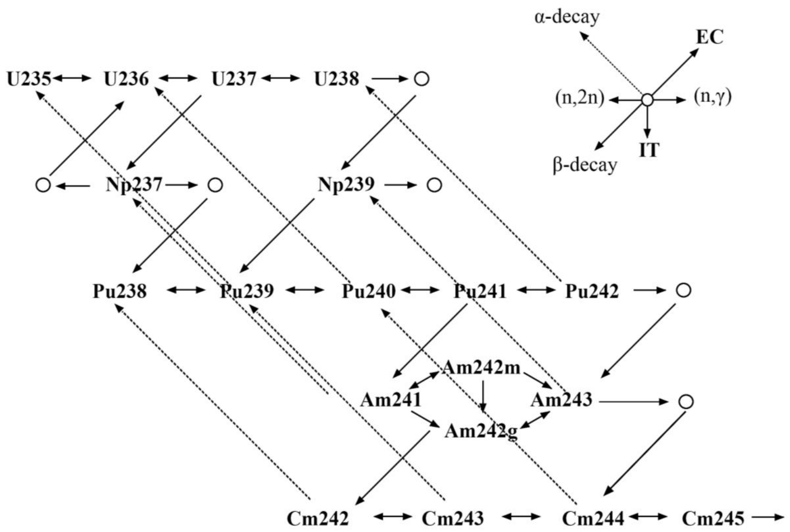

Fig. 16. Nuclide burn-up chain of Uranium [17]

4. Conclusions

The neutronic calculation of the PWR type module reactor using the COREBN 3D method shows that the PWR reactor used can operate for up to 10 years without refueling. The results with optimal conditions were obtained when the variation of the fuel used was $F1$:5 %, $F2$:6 %, $F3$:7 %, the active core height was 100 cm, the diameter was 80 cm, power was 300 MWth, fuel fraction was 65 % with the addition of neptunium-237. The effect of the addition of Np-237 can reduce the value of k-eff because the nature of neptunium 237 absorbs neutrons, resulting in an excess reactivity value approaching the value of 1.68 %.

Conflict of interest

The authors declare that there is no conflict of interest in relation to this paper, as well as the published research results, including the financial aspects of conducting the research, obtaining, and using its results, as well as any non-financial personal relationships.

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