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Editors • Ratna Dewi Syarifah, Sutisna and Wenny Maulina

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Preface: The 3rd International Conference on Physical Instrumentation and Advanced Materials (ICPIAM) 2021

Physics Department – University of Jember, Indonesia, 27 October 2021

The 3rd International Conference on Physical Instrumentation and Advanced Materials (ICPIAM) 2021 is a serial international seminar held as a form of collaboration bi-annual event between the Faculty of Mathematics and Natural Sciences, University of Jember and the Faculty of Science and Technology (FST) Universitas Airlangga (UNAIR).

This conference (ICPIAM 2021) was held as a means for academics, scientists and researchers to discuss, share and exchange information, experiences, methods and research findings as well as the latest innovations at the international level. Through this conference, links will also be formed between researchers and academics to establish cooperation and collaboration both in the fields of education and research internationally.

This conference was attended by 60 presenters who have submitted paper from their studies and researches. They come from Japan, Malaysia, and mostly from Indonesia such as (BPPT, LAPAN, UNAIR, UNEJ, ITB, UNAND, UNCEN, and many more). All the presenters will convey their speech in the 6 different parallel rooms as given in the book of abstract. All presented and reviewed paper will be considered to be published at the AIP Conference Proceedings (Scopus indexed proceedings). We deeply thank the authors for their enthusiastic and high-grade contribution.

The 3rd ICPIAM 2021 would not be possible running without the dedicated efforts of many people especially all organizing committee members who have made planning and organizing the programs. We are grateful to IsDB and Jember University for the funding support, also Physics Student Association (HIMAFI) and volunteers who contributed to the various processes that make up the conference and it would not be possible for me to name them all in this short message.

Bowo Eko Cahyono, Ph.D.
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Preliminary study of 300 MWth pressurized water reactor with carbide fuel with addition neptunium 237 using SRAC-COREBN code

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Arindi Kumala Sari, Ratna Dewi Syarifah and Artoto Arkundato

Preliminary Study of 300 MWth Pressurized Water Reactor with Carbide Fuel with Addition Neptunium 237 Using SRAC-COREBN Code

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Abstract. The research on the design of a pressurized water reactor with a power of 300 MWth and using uranium carbide (UC) fuel has been carried out. Neutronic analysis was carried out using the System Reactor Atomic Code (SRAC) program, which was operated on a set of computers with Linux Operating System (OS), SRAC codes, namely PIJ and COREBN. PIJ is used to calculate the fuel cell level, and COREBN is used to calculate the reactor core level. The reactor is designed to produce critical conditions. The parameters analyzed in this study were fuel enrichment (U-235), reactor core size, multiplication factor, reactor core configuration, and criticality. This study calculates the neutronic calculation of UC fuel with the addition of Neptunium 237 (Np-237), which aims to reduce the value of the effective multiplication factor (k-eff). Neutronic analysis of UC fuel with critical conditions at F1 = 4.5%, F2 = 5%, F3 = 5.5%, fuel fraction = 55% with the addition of neptunium 237 0.5%.

INTRODUCTION

The need for energy in Indonesia is increasing along with the increase in the world's population. The increasing energy demand is very influential in the rapid development of technology so that the required fuel is also getting more significant. In contrast, commercial energy resources limit the use of coal, oil, and natural gas [1,2]. Renewable energy sources such as hydro, wind, solar, and wave power have not been able to help Indonesia's electrical energy needs in terms of technology and capacity. At the same time, it is estimated that the demand for electrical energy in 2025 will increase to more than three times the current demand for electrical energy. Indonesia, which has a large population with a large land area, requires environmentally friendly energy sources and high-intensity, such as Nuclear Power Plants (PLTN) [3].

The development of nuclear reactors has been achieved from Generation-I up to IV. In this research, PWR nuclear reactor is used, which is generation II. The second-generation nuclear reactors do not spring that has been developed towards standard designs and are equipped with appropriate safety systems [4]. A nuclear reactor is a place where a controlled fission chain reaction occurs to produce energy. A fission chain reaction occurs when a fissile or fissile element nucleus (Uranium-235) reacts with thermal neutrons. Then, the reaction results will become other elements and produce 2-3 neutrons and heat energy [5, 6]. Pressurized Water Reactor (PWR) is the most widely used power reactor globally, which is about 63 percent. Pressurized Water Reactor (PWR) is a thermal nuclear reactor that uses light water as a moderator and coolant. The moderator functions to slow the rate of neutrons in the reactor core, while the coolant absorbs heat from the fission reaction in the reactor core [7].

Previous research on the design of a pressurized water reactor (PWR) has been carried out [8-14] with Uranium Carbide fuel [14] using the COREBN method [15]. Based on previous studies, this study designs a small PWR-type reactor with Uranium Carbide (UC) fuel estimated to have a long life without refueling.

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DESIGN CONCEPT AND CALCULATION METHODS

Neutronic calculation in this study uses SRAC2006 to start the calculation of neutronic analysis of various types of reactors. SRAC (Standard Reactor Analysis Code) was developed by JAERI Japan Atomic Energy Research Institute). The first calculation is the burn-up of cells with SRAC-PIJ to produce a MACRO file with a PDS format. The output has a value (k-inf). Then create a history file (HIST), HIST functions to convert macroscopic cross-sectional data files in PDS format to PS to be read by COREBN. COREBN only runs if it is in PS format[14]. The parameters analyzed in this study were fuel enrichment, burn-up, core configuration (homogeneous and heterogeneous), criticality, and addition of neptunium-237.

Table 1 shows the design specifications of the reactor used for U-235 fuel enrichment from 3% to 15% with an interval of 1%. The optimation results were obtained by adding the neptunium 237 (Np-237) to reduce the multiplication factor value. This study uses zirconium cladding because the properties of zirconium are corrosion resistance. So, it can absorb neutrons. The cell is divided into fuel, cladding, and coolant regions. The division of the hexagonal cell type region is shown in Figure 1.

TABLE 1. Reactor design specifications

Design Parameters	Specifications
Power	300 MWth
Burn-up	Ten years
Fuel	Uranium Carbide(UC)
Cladding	Zirconium (ZIRCO)
Coolant	Air (H_2O)
Fuel pin cell shape	Sel Silinder
Pin pitch	1.45 cm
Active core height	100 cm
Active core diameter	80 cm
Fuel fraction (%)	35%-60%
Cladding fraction (%)	10%
Coolant fraction (%)	25%-50%
Pin pitch (cm)	1.45 cm

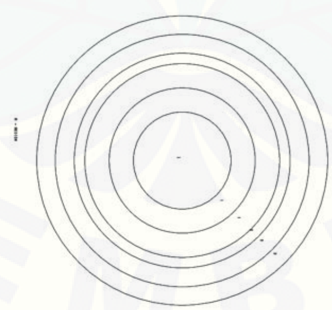

FIGURE 1. Regional division of cylindrical cell types [16].

The three-dimensional design of the fuel core (X-Y-Z) at the HIST and COREBN inputs is shown in Figure 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.

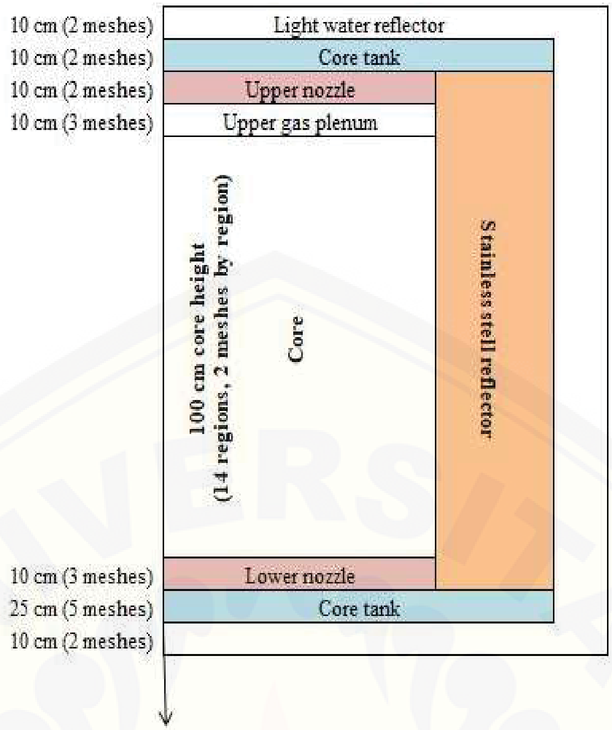

FIGURE 2. Burn-up calculation model types [17].

RESULTS AND DISCUSSIONS

This study uses a PWR reactor in a module with UC fuel. The neutronic calculation begins with calculating the homogeneous core configuration, the two heterogeneous core configurations, the third addition of Neptunium 237 (Np-237), the four variations in the volume of fuel, and the last is fuel optimization. After that, the results were obtained with the optimization value of Uranium Carbide fuel in Pressurized Water Reactor (PWR).

The first calculation carried out in this study is to calculate the fuel cell without considering the rate of neutron leakage and analyze the value of the multiplication factor in the analysis of the homogeneous core configuration based on the percentage variation of U-235. Figure 3 is a graph of the k-inf value based on the variation in the percentage of U-235 from 3% to 15% using a homogeneous core configuration. Figure 4 is a graph of the k-eff value based on the variation in the percentage of U-235 from 3% to 15%. Figure 4 shows that the most sloping rate is U-235 5%, with the highest k-eff value of 1.108089.

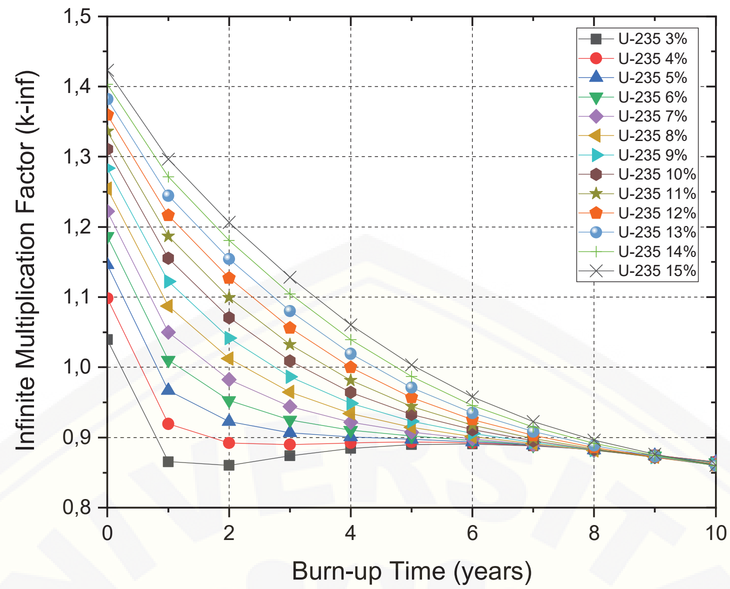

FIGURE 3. k-inf value in a homogeneous core configuration

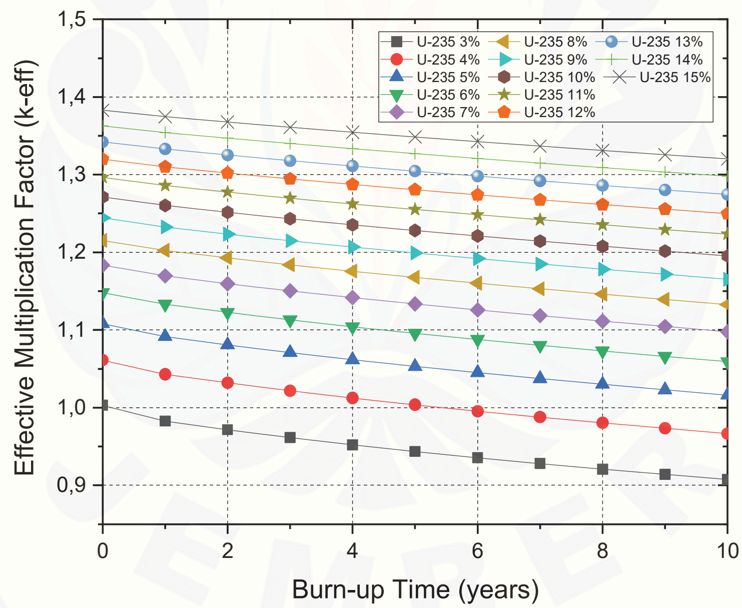

FIGURE 4. k-eff value in a homogeneous core configuration

After obtaining the most sloping value in the homogeneous terrace configuration, the next calculation is the heterogeneous core configuration with three types of fuel percentages. Figure 5 shows a graph of the k-eff value using a core configuration where there are variations in the three types of fuel percentages shown in Table 2. In Figure 5 shows the most sloping value for the variation in the percentage of fuel F1 = 4%, F2 = 5%, F3 =6% with an excess reactivity value of 8.86%.

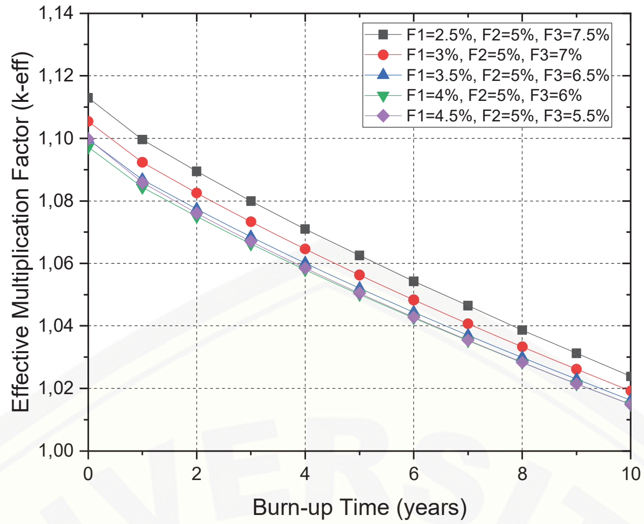

FIGURE 5. k-eff value in a heterogeneous core configuration.

TABLE 2. Variations in fuel percentage

Name	Percentage U-235			Average
	Fuel 1	Fuel 2	Fuel 3	
Case 1	2.5%	5%	7.5%	5%
Case 2	3%	5%	7%	5%
Case 3	3.5%	5%	6.5%	5%
Case 4	4%	5%	6%	5%
Case 5	4.5%	5%	5.5%	5%

Figure 6 compares power density values using homogeneous and heterogeneous core configurations. It means that the value of the average power density in the radial direction is too high in the center of the reactor in the calculation of the homogeneous core configuration. After calculating using a heterogeneous core configuration, the high power density in the center reactor will decrease to a more gentle one in the center of the reactors.

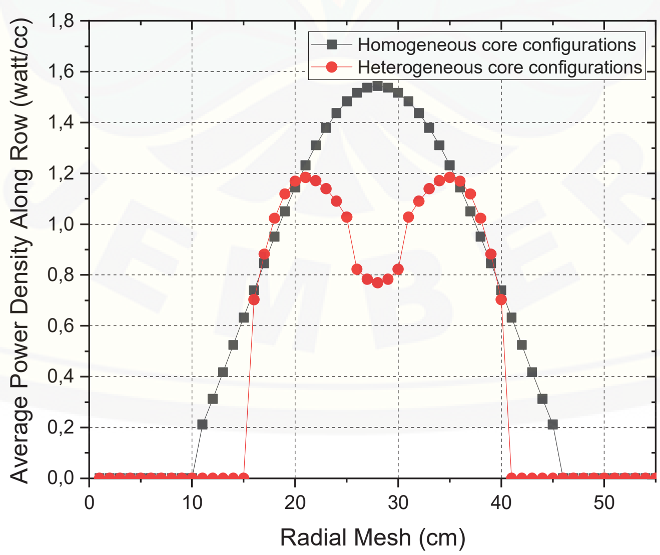

FIGURE 6. The power distribution to the radial direction of the mesh in the homogeneous and heterogeneous core configurations

Figure 7 shows a graph of the k-eff value with fuel variations from 35% to 60% with variations in the percentage of fuel F1 = 4%, F2 = 5%, F3 = 6%. Figure 7 shows that as the volume of the fuel fraction increases, the k-eff value decreases. The value of k-eff will decrease as the volume of the fuel fraction increases. So, it means that the value is close to the critical value in the reactor, namely at 55% fuel volume.

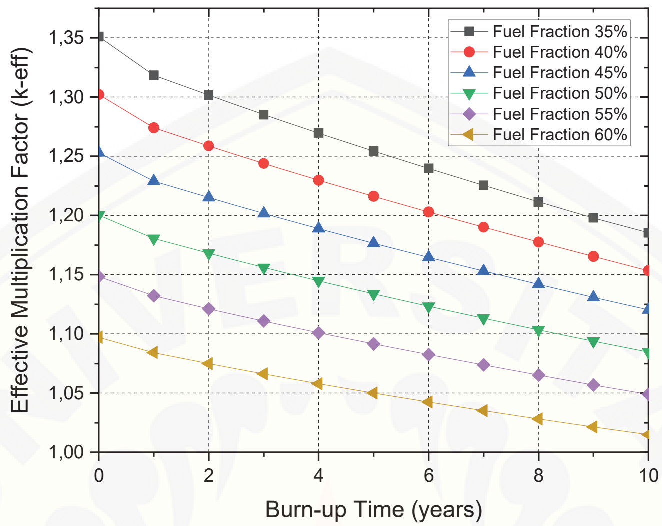

FIGURE 7. The k-eff value with variations fuel fractions (35%-60%).

In this study, the addition of neptunium 237 was also carried out. It can reduce the value of k-eff. Figure 8 shows the k-eff value in acquiring neptunium 237 with variated from 0% to 0.6% with an interval of 0.1%. Figure 8 shows that the value of k-eff decreased 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.

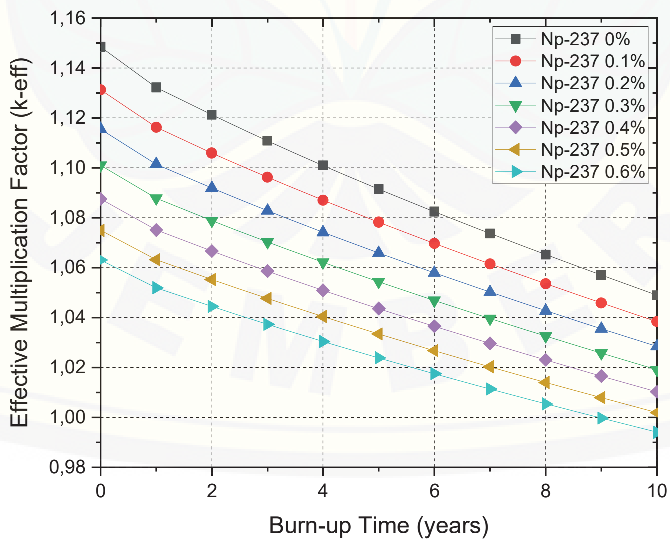

FIGURE 8. The k-eff value with the addition of Np (0%-0.5%).

The addition of Neptunium 237 can be used as an additive in this study because the variation in the percentage of neptunium 237 is 0.5% in the interpretation of plutonium F1 = 4%, F2 = 5%, F3 = 6% with a fuel volume fraction of 55% the maximum value of k-eff worth 1.074966 with an excess reactivity value of 6.97% and a maximum power density value of 50.80 Watt/cc. Then the optimization of the results of this study is shown in Figure 9.

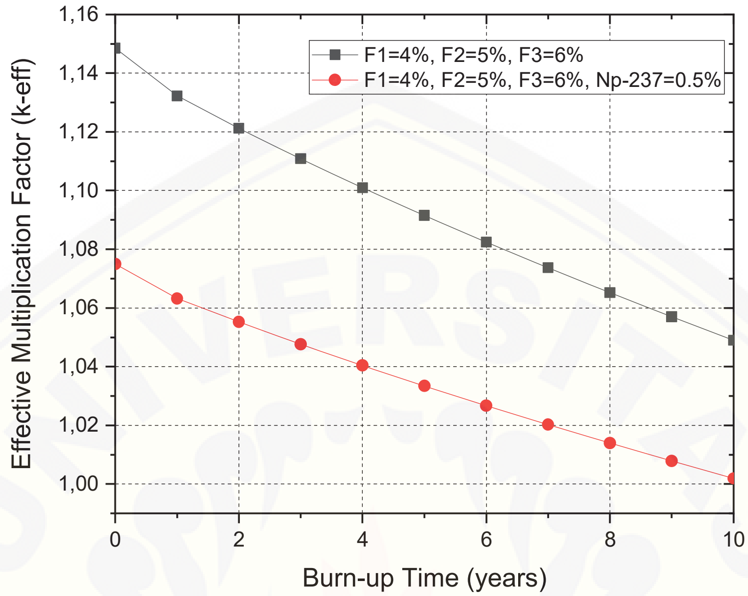

FIGURE 9. Optimization results design k-eff value.

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. This optimal condition is obtained when the variation of fuel used is F1:4%, F2:5%, F3:6%, the active core height is 100 cm, diameter is 80 cm, and the power is 300 MWth. 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 6.97%.

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