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Comparative Studies of Nuclear Fuel Cell (U-Pu)N and (U-Pu)Zr Using OpenMC Software

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Abstract. This study analyzes pin-shaped fuel cells in two types of fuel mixtures, namely Uranium-Plutonium-Nitride and Uranium-Plutonium-Zirconium. Each fuel is varied by changing the percentage of Uranium-235 content as much as 1% to 10%. Using the Monte Carlo method in the OpenMC program code, calculations were performed to obtain the effective multiplication factor, rate of fission reaction, and distribution of neutron flux for two years of burn-up. The calculation results for the effective multiplication factor and the rate of fission reaction show that the greater the enrichment of uranium, the greater the value of these parameters. In the neutron flux distribution calculation fuel cells have the highest value in the middle area of the fuel, and its value gets smaller toward the edge of the cell. By doing this, fuel cell analysis so that later it can be used as a reference in the preparation of fuel cells in the reactor core that is safe and efficient.

INTRODUCTION

In modern society, the need for electricity is a basic need that must be fulfilled in everyday life, both in terms of work, study, and just entertainment, likewise, in Indonesia. Indonesian people, especially those who live in urban areas, are highly dependent on electrical energy to carry out their daily life [1]. According to the Coordinating Ministry for Human Development and Culture, the electricity supply in Indonesia in 2020 will reach 99.48%. But in fact, there are 433 villages that have not felt electricity.

To fulfill Indonesia's electricity supply, especially in rural areas, using a nuclear power plant can be the right choice. This power plant utilizes the fission reaction of fissile material, which is currently abundant on earth. The selection of nuclear power plants to fulfill electricity needs in addition to producing large amounts of energy also does not cause pollution and is economical, so it is appropriate to meet electricity needs in remote areas in Indonesia.

One of the nuclear power plants that can be applied in Indonesia is the Gas-Cooled Fast Reactor (GFR). GFR is included in the Generation IV reactor. The advantages of generation IV compared to previous generations include efficiency in using fuel, producing less waste, higher safety, and resistance to proliferation [2]. Like other fast reactors, GFR does not use a graphite moderator so that the core temperature can rise to 16,000ºC/min due to its mass and lower thermal conductivity than thermal reactors. This reactor has a high temperature, so the fuel required a mixture with ceramic carbide and nitrite such as SiC, ZrC, Tic, Tin, and ZrN to suppress extreme temperatures [3].

To build a nuclear reactor of any type, a simulation process is required to obtain optimum results and as a reference in nuclear safety issues. As must be controlled fission reactions that occur in the reactor. This control is done to limit the number of neutrons so that only one neutron is absorbed for the next nuclear fission. So that the balance of the number of neutrons in each generation is achieved, which is constant [4]. Therefore, this research focuses on the analysis of the effective multiplication factor (keff), the rate of fission reactions, and the distribution of neutron flux in a pin-shaped fuel cell. The calculation uses the OpenMC program code, which is a program code to simulate nuclear transport based on the Monte Carlo method.

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METHOD

Fuel cell calculations using OpenMC program code version 12.0. The program code was developed in 2011 by members of the Computational Reactor Physics Group (CRPG) at Massachusetts Institute of Technology (MIT), whose initial development focused on the critical calculations required for nuclear reactor simulations. As is the case for the calculation of the effective multiplication factor (keff), the rate of fission reactions, and the distribution of neutron flux can be easily simulated by OpenMC. Using the native HDF5 format generated from the ACE format as particle interaction data [5].

The simulation using two types of mixed fuel, where every kind of uranium enrichment is varied from 1% to 10% within a burn-up time of 2 years. Mixed fuels are used to obtain a fuel fraction with a high melting point, conduct heat well, resist corrosion, not easily crack, and be able to withstand loose fission products [6]. The fuel cell design parameters are attached in the following table:

TABLE 1. Fuel Cell Characteristics Data

Parameter	Description
Fuel cell geometry	Pin
Pitch geometry	2D-Hexagonal
Fuel type	Uranium-Plutonium-Nitride Uranium-Plutonium-Zirconium
Fuel radius	0.4256 cm
Cooler type	Helium
Gap thickness	0.0029 cm
Cladding material	Stainless Steel 316 (SS316)
Cladding thickness	0.05 cm
Enrichment	1 to 10%
Burn-up time	2 year

TABLE 2. Fuel, Clad, and Coolant Parameters (7–9)

Parameter	Fuel		Cladding	Cooler
	(U-Pu)N	(U-Pu)Zr	SS-316	Helium
Theoretical density (g/cm^3)	14.30	15.60	8.03	17.85×10^{-5}
Melting conductivity (ºC)	2780	1080	1370 – 1398	-
Thermal conductivity (W/m/ºK)	14.30	14.00	12.10	0.1513

RESULTS AND DISCUSSION

Fuel Cell Design

This research uses a pin-shaped fuel cell with a hexagonal pitch. The OpenMC program displays the fuel cell geometry in 2-D, as shown in Figure 1, and has a different color for each element. The pink color in the middle is the fuel, the gray is the cladding, and the pitch is purple.

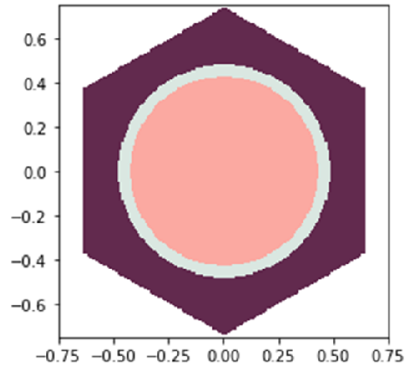


FIGURE 1. Fuel cell design

Effective Multiplication Factor (k_{eff})

The effective multiplication factor is the ratio of the number of neutrons produced in one generation to the neutrons lost through absorption or leakage in the next generation [10]. The value of the multiplication factor (k) can determine the state of the reactor. There are three conditions, namely a supercritical state (k>1) when the neutron population continues to increase, a critical state (k=1) where the neutron population remains, and a subcritical state (k<1), which means the neutron population has decreased [11]. This research was conducted to get the fuel cell in a critical state.

Both fuels show the same behavior; firstly, the greater the enrichment used, the greater the value produced; this is because the more uranium enrichment, the greater the fuel fraction, which has an impact on the fission products produced [12]. And secondly, during the two years of burn-up, over time, the value of the effective multiplication factor decreases because the density of the fissile and fertile fuels used decreases. The number of neutrons produced in each generation also decreases [13]. Figures 2 and 3 show that the effective multiplication factor for both fuels for two years is not critical, but Uranium-Plutonium-Nitride fuel without enrichment has the best keff value because the value is closer to 1.0.

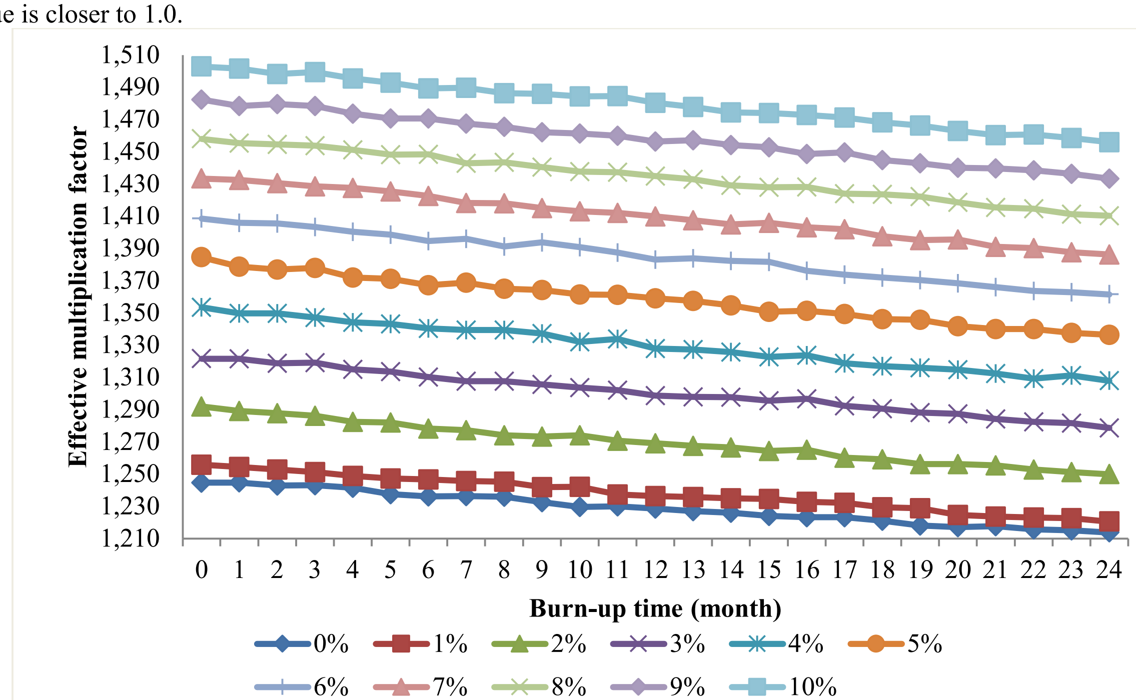


FIGURE 2. Effective multiplication factor for (U-Pu)N fuel

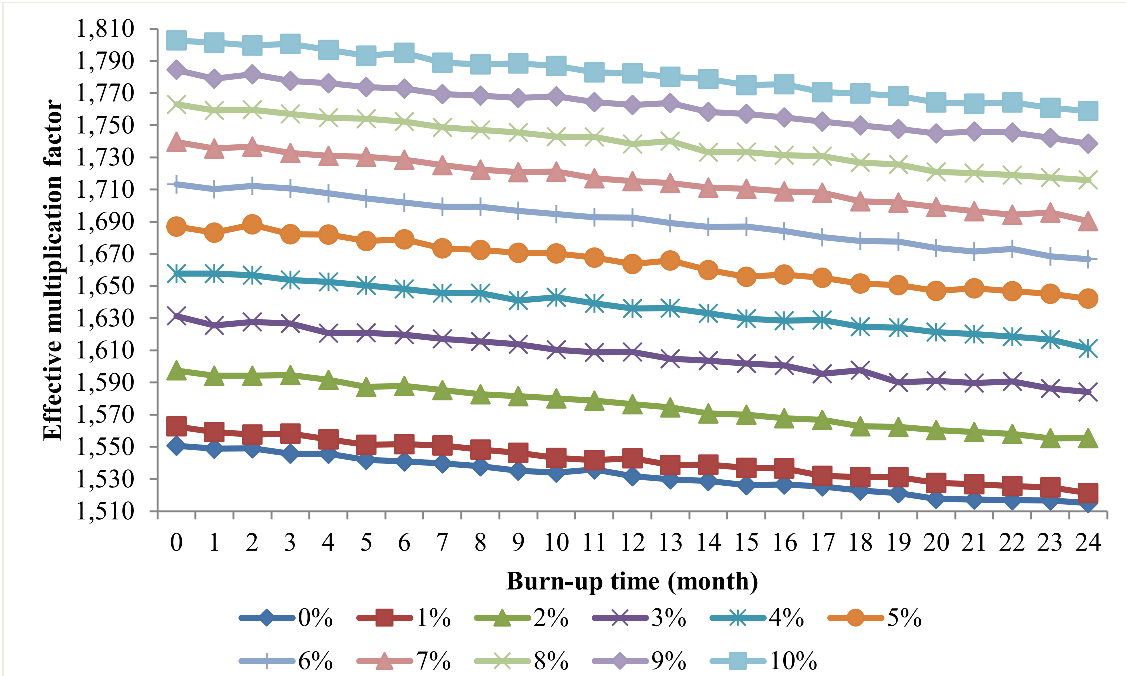
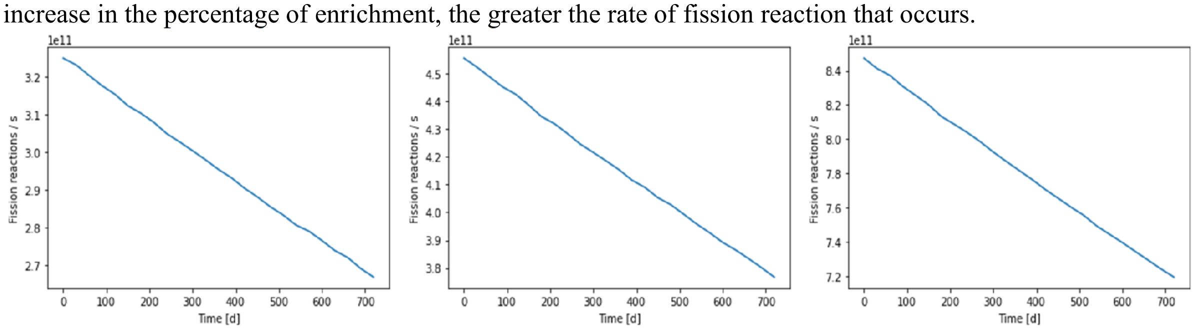


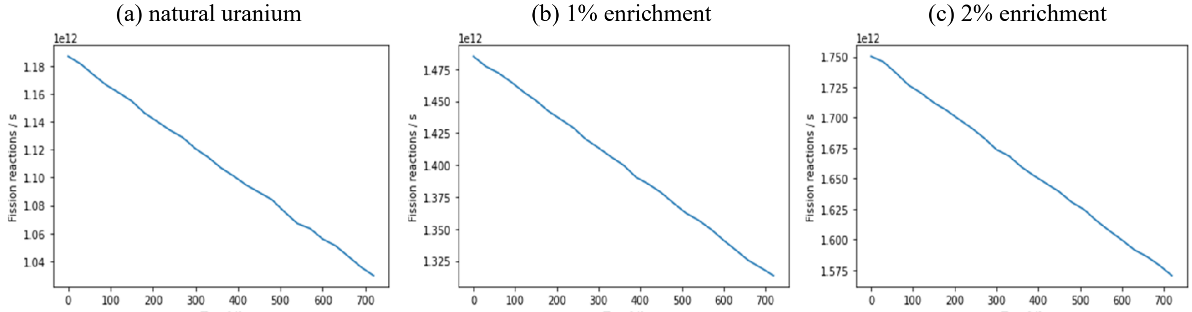
FIGURE 3. Effective multiplication factor for (U-Pu)Zr fuel

Rate of Fission Reaction

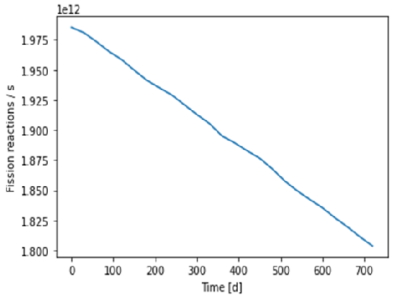
There is a decrease in the rate of fission reactions for Uranium-Plutonium-Nitride and Uranium-Plutonium-Zirconium fuels. This is due to the shrinkage of the fuel used so that the fission reaction that occurs is also reduced, and the number of neutrons produced is getting less and less. The fission reaction is also influenced by the amount of fissile fuel used. In this study, there are two types of fissile fuel used, namely Uranium-235 and Plutonium-239. Because the amount of Plutonium used is the same, but the amount of Uranium-235 used is different, this is what causes the rate of the fission reaction to be obtained is also different. It can be seen in Figures 4 and 5 that for every increase in the percentage of enrichment, the greater the rate of fission reaction that occurs.



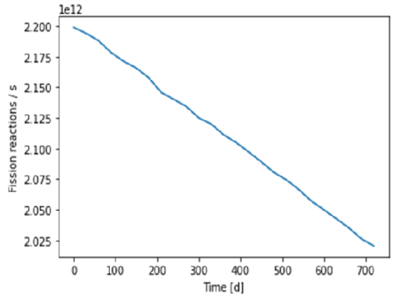
(a) natural uranium (b) 1% enrichment (c) 2% enrichment



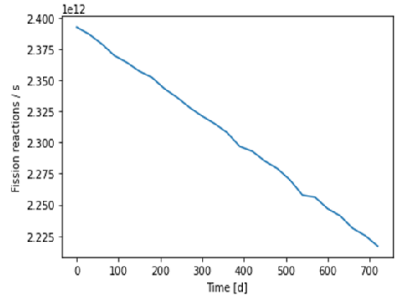
(d) 3% enrichment (e) 4% enrichment (f) 5% enrichment



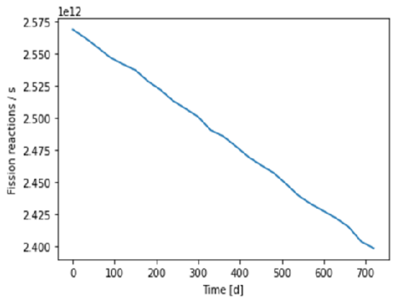
(g) 6% enrichment



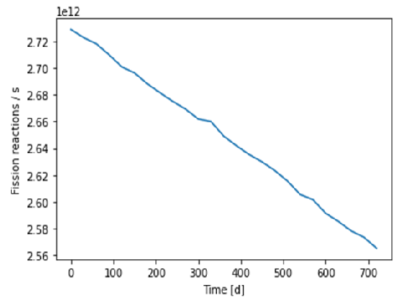
(h) 7% enrichment



(i) 8% enrichment

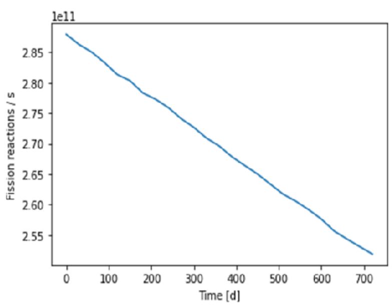


(j) 9% enrichment

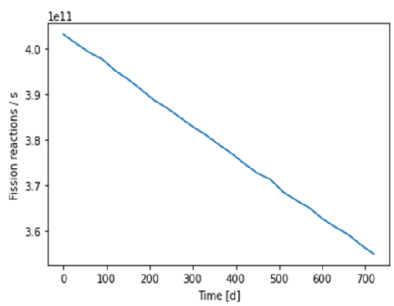


(k) 10% enrichment

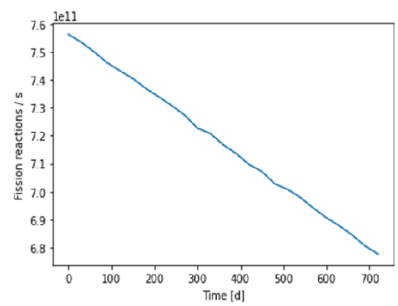
FIGURE 4 (a-k). The rate of fission reactions for (U-Pu)N fuel



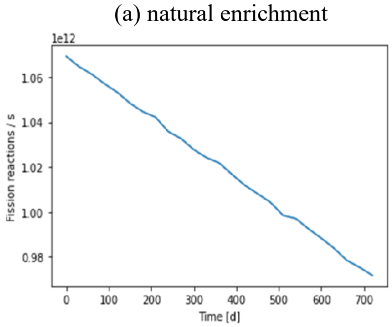
(a) natural enrichment



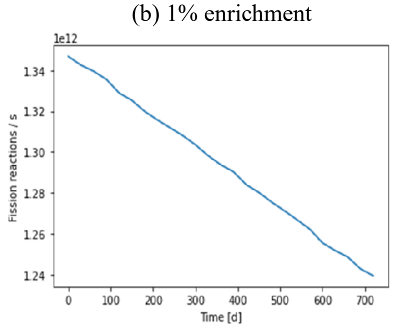
(b) 1% enrichment



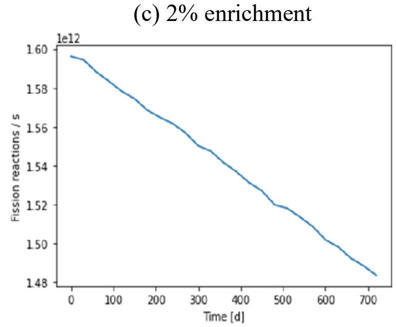
(c) 2% enrichment



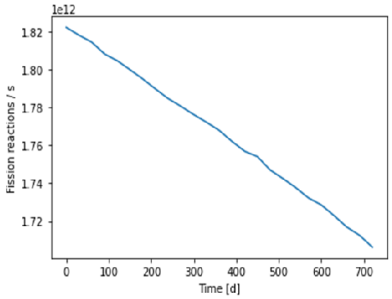
(d) 3% enrichment



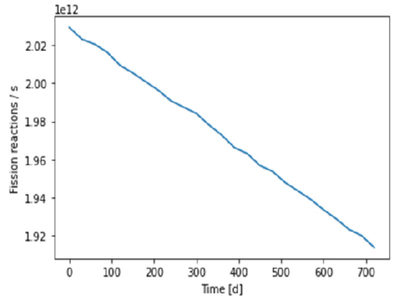
(e) 4% enrichment



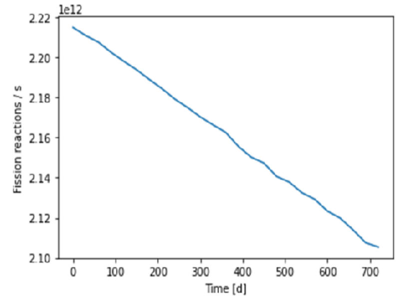
(f) 5% enrichment



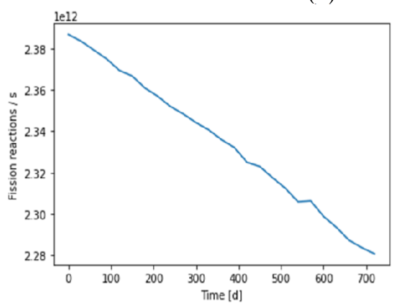
(g) 6% enrichment



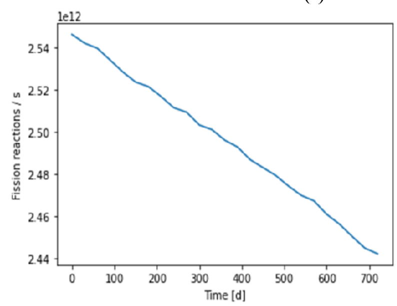
(h) 7% enrichment



(i) 8% enrichment



(j) 9% enrichment



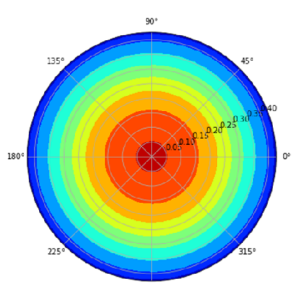
(k) 10% enrichment

FIGURE 5 (a-k). The rate of fission reactions for (U-Pu)Zr fuel

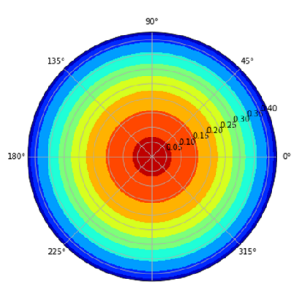
Distribution of Neutron Flux

To calculate the neutron flux at a location, the Monte Carlo method uses track length estimation. This method takes data on the length of the neutron path during the simulation. Where a random number generator randomly generates the probability of a neutron interacting with another medium. Furthermore, the number of neutrons passing through a location is calculated, the number of neutron counts is the neutron flux at the observed point [10]. The number of neutrons per unit area per unit time is called the distribution of the neutron flux. The distribution of the neutron flux depends on the probability of a fission reaction occurring in a material [14]. In this study, the flux distribution was calculated at the initial state.

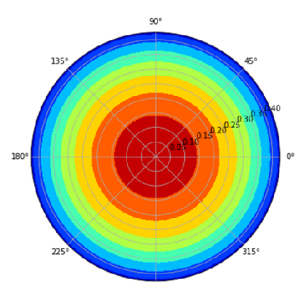
The distribution of the largest neutron flux occurs in the center area, and the farther outward, the fewer the neutrons. The cause of the reduced distribution of neutron flux at the edge of the cell pin can be caused by being scattered, absorbed by materials outside the fuel, or due to leakage of neutrons out of the system. This study aims to obtain an even distribution of flux on a pin cell from many experiments carried out. The cell pins that have the best flux distribution are images with a nearly similar color spectrum. Although all experiments obtained a good neutron flux distribution, the best flux distribution for Uranium-Plutonium-Nitride fuel is shown in Figure 6d with a Uranium enrichment of 3%, and for Uranium-Plutonium-Zirconium fuel shown in Figures 7d and 7g.



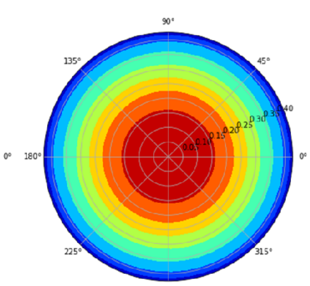
(a) natural uranium



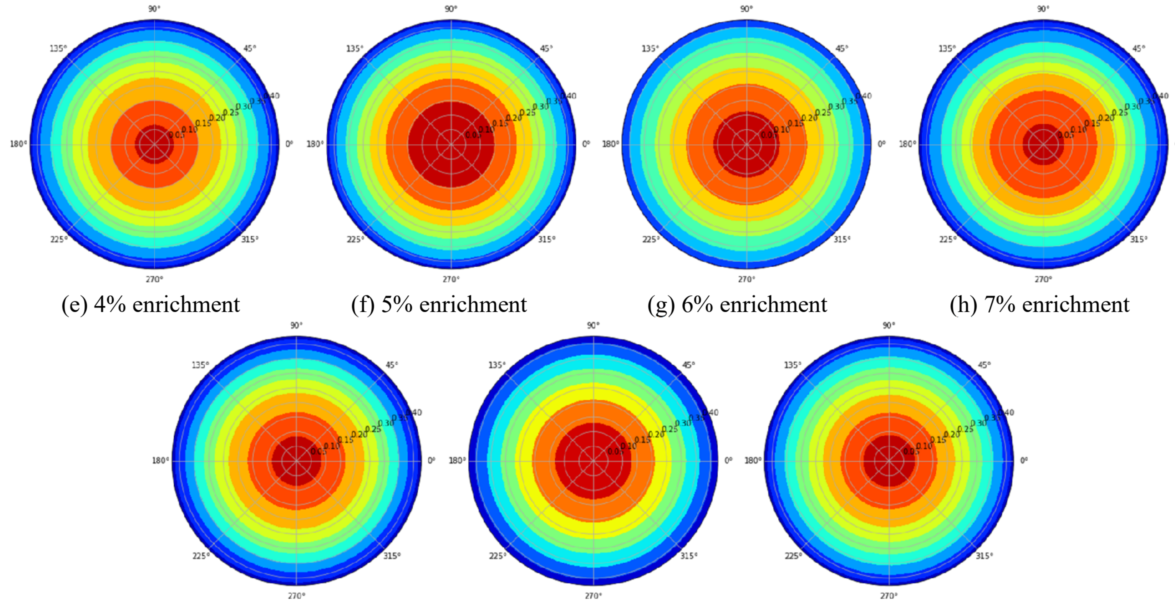
(b) 1% enrichment



(c) 2% enrichment



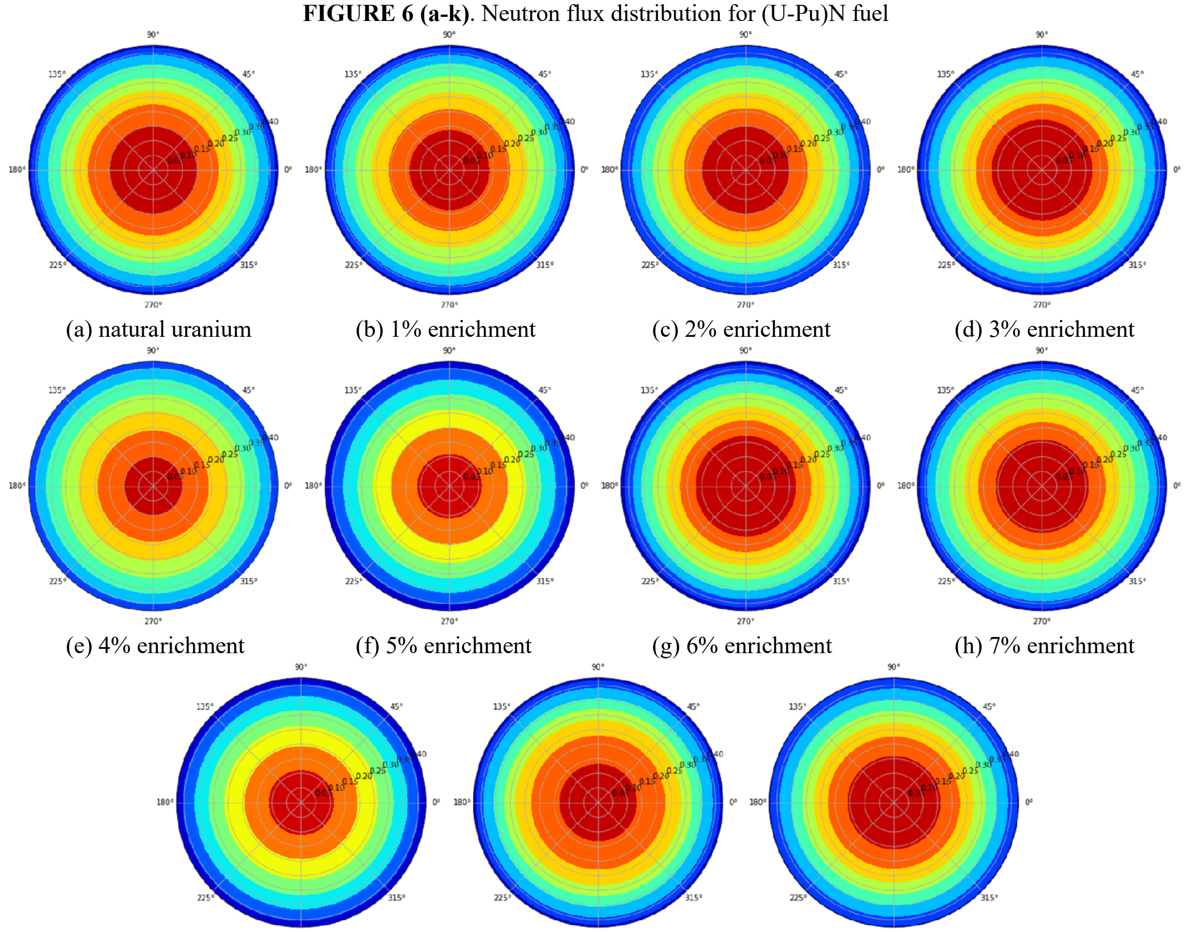
(d) 3% enrichment



(e) 4% enrichment (f) 5% enrichment (g) 6% enrichment (h) 7% enrichment

(i) 8% enrichment (j) 9% enrichment (k) 10% enrichment

FIGURE 6 (a-k). Neutron flux distribution for (U-Pu)N fuel



(a) natural uranium (b) 1% enrichment (c) 2% enrichment (d) 3% enrichment

(e) 4% enrichment (f) 5% enrichment (g) 6% enrichment (h) 7% enrichment

(i) 8% enrichment (j) 9% enrichment (k) 10% enrichment

FIGURE 7 (a-k). Neutron flux distribution for (U-Pu)Zr fuel

CONCLUSION

Based on research and analysis that has been carried out with three parameters: effective multiplication factor, rate of fission reaction, and distribution of neutron flux, it can be concluded that using natural uranium for a fuel mixture of Uranium-Plutonium-Nitride has better results to be applied to gas-cooled fast reactors than using Uranium-Plutonium-Zirconium as fuel. This is because the fuel has a keff value close to the critical state, the rate of fission reaction and flux distribution is the smallest.

ACKNOWLEDGMENT

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