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# The calculation of uranium metallic Fuel (U-10%wtZr) cell with helium coolant using SRAC 2K6

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**Abstract.** In this study, pin-shaped natural metallic uranium fuel cells with a diameter of 1.4 cm were designed with and without enrichment of 2-10% by using helium as a coolant. The calculations were conducted by using SRAC software and the results show that the greater the enrichment level, the more significant the value of the infinite-multiplication-factor ( $k_{inf}$ ), meanwhile the conversion ratio is smaller. Basically, uranium exists in two isotopic forms namely, U-235 and U-238 which are transformed into other element such as Pu-239 when they both go through fission and transmutation reactions in the reactor core. Therefore, to obtain the optimum reactor core design, the performance information of nuclear fuel cells from the variations of burn-up time needs to be taken into consideration.

## 1. Introduction

Alternative sources of energy are needed to ensure sustainability in the future, and one of the promising alternatives is nuclear energy [1], which are produced from nuclear reactions. These reactions occur in a reactor core through merging (fusion) or fission, when two atomic particles collide to create a different product from the initial one [2]. Consequently, heat energy is produced in the core, hence a material with low neutron absorption, stable radiation, high temperatures, which is capable of absorbing the heat is required.

Also, nuclear fuel is the main component in a reactor obtained from natural isotopes such as uranium and thorium, which splits during a reaction with neutrons [3]. Meanwhile, the most widely used fuel is uranium, having two isotopes in nature, namely U-238 (99.3% by weight) and U-235 (0.7% by weight). The U-238 is a fertile material which does not split or undergo fission reactions with neutrons. However, U-235 is fissile, which means it is able to split and react directly with neutrons to undergo nuclear fission reactions [4]. Consequently, fertile material from natural uranium is used more effectively in fast reactors, while the neutrons produced react with the U-238 to create a new fissile material (Pu-239) [5].

A neutronic analysis includes the population of neutrons, flux, distribution, and its behavior in the reactor core. Inside the reactor core, there is a chain reaction of fission, and each reaction produces energy of 200 MeV. This reaction occurs when a neutron hits a heavy nucleus [6], however, it hardly occurs spontaneously. Therefore, comprehensive analyses such as neutronic, thermal-hydraulic, and safety are needed when designing a nuclear reactor. For example, the neutronic calculations are required to find out information such as multiplication factor values, neutron flux distribution, and power distribution [7]. Also, the characteristics of the neutrons need to be known because they are particles capable of causing fission reactions with fissile isotopes [8][9].

In this study, the fuel cells made from natural metallic uranium was calculated and the content consists of 99.3% uranium-238 and 0.7% uranium-235. The content of uranium-235 is increased



through an enrichment process of 1-10% in order to utilize some nuclear reactors. Afterward, the uranium is combined with 10% zirconium (U-10%wtZr). Hence, this study compared survey parameters on fuel cell burn-up for 100, 120, 140, and 160 years. This comparison is carried out to determine the performance of nuclear fuel cells to ensure the continuous production of critical values at a predetermined burn-up time.

## 2. Methods

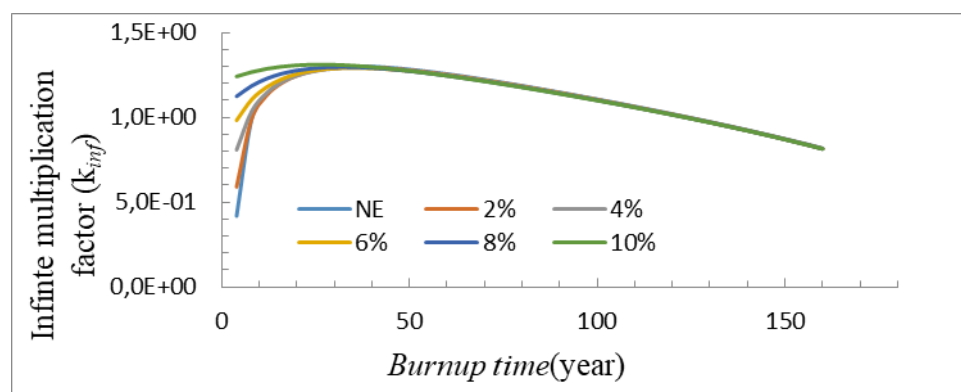
The program used is SRAC (Standard Reactor Analysis Code) 2K6 with JENDL 4.0 library [10][11]. The simulation was used to obtain the burn-up parameters of infinite multiplication factor ( $k_{inf}$ ), burn-up level, conversion ratio, and atomic density of U-235, U-238, and Pu-239. Nuclear fuel cells are pin-shaped with a cylindrical geometry, according to the design specifications in the GFR, as shown in Table 1. Also, this shape specification follows several existing studies.

**Table 1.** Fuel cell specifications

Parameter	Value/Description
Fuel (Fuel)	Uranium Metal (U-10% wtZr)
Cladding Material	Stainless Steel (SS316)
Coolant	Helium
Fuel cell geometry	Cylinder
Pitch pins	1.4 cm
Fuel volume fraction: cladding: coolant	55-65%: 10%: 25-35%
Burn-up time	100-160 years

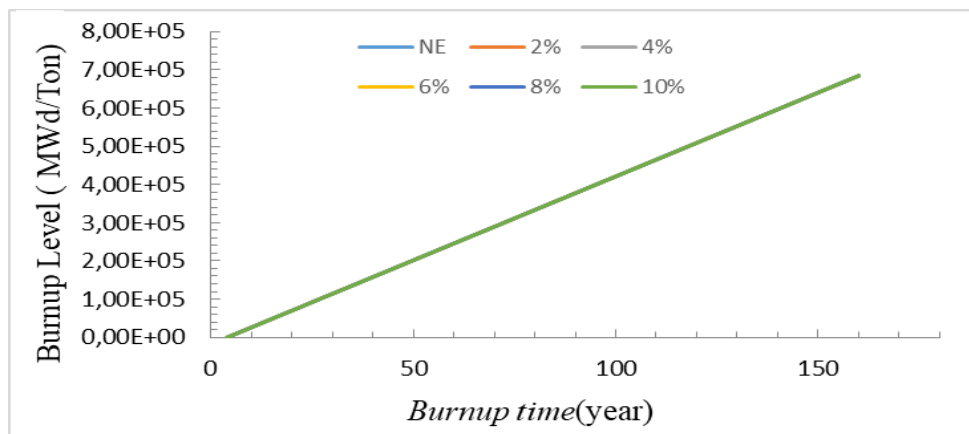
## 3. Results and Discussion

Figure 1 shows the change in the value of the infinite multiplication factor ( $k_{inf}$ ) with variations, without enrichment and enrichment at 2-10%. At 2-6% enrichment, the value of infinite multiplication factor ( $k_{inf}$ ) is  $<1$ , while 8-10% enrichment showed a value of  $k_{inf}$  as  $>1$ . The high level of enrichment causes the density of U-235 to increase and impacts the number of neutrons produced, thereby causing the value of the infinite multiplication factor ( $k_{inf}$ ) to be  $>1$ . This shows that the greater the enrichment of an element, the higher the value of the infinite multiplication factor ( $k_{inf}$ ) obtained.



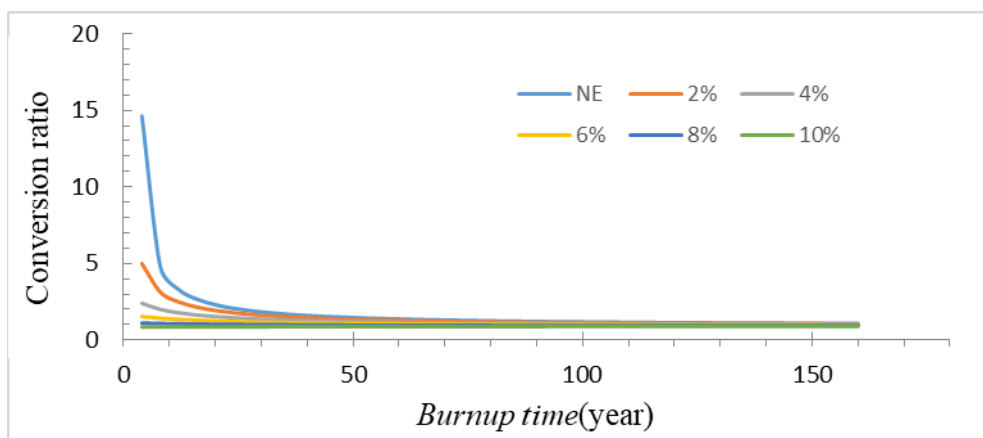
**Figure 1.** Infinite multiplication factor ( $k_{inf}$ ) for 100,120, 140, and 160 years of burn-up

The burn-up level is the energy released per unit mass of fuel as a result of combustion. According to Figure 2, the burn-up level increases continuously over time. For example, in the year 100, 120, 140, and 160 the burn-up level was at  $4.30E+05$  MWd/Ton,  $5.09E+05$  MWd/Ton,  $5.97E+05$  MWd/Ton and  $6.85E+05$  MWd/Ton respectively. These values show that as the burn-up period increases, the amount of fuel burned also increases.



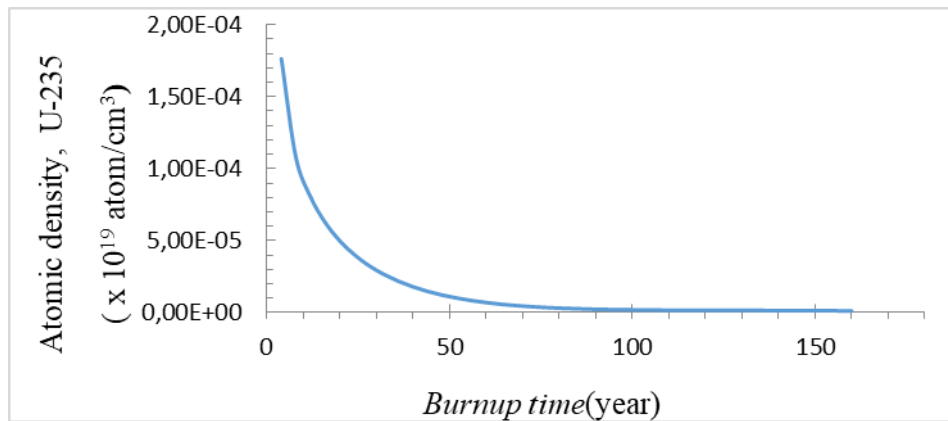
**Figure 2.** Burn-up levels for 100, 120, 140, and 160 years of burn-up time

The conversion ratio (CR) shows the ratio between the number of atoms of fissile fuel produced and the one consumed in the reactor. According to Figure 3, when the fissile atoms produced are more than the one consumed, the first ten years' conversion ratio for U-235 without enrichment experienced a very sharp decline, but afterward, it decreases slowly until the end of the burn-up time. Consequently, as the burn-up period increases, the fuel in the reactor runs out. Also, the enrichment level affects the value of conversion ratio obtained, hence the more the magnitude, the smaller the conversion ratio produced but the total density of U-235 is greater due to a large amount of enrichment.

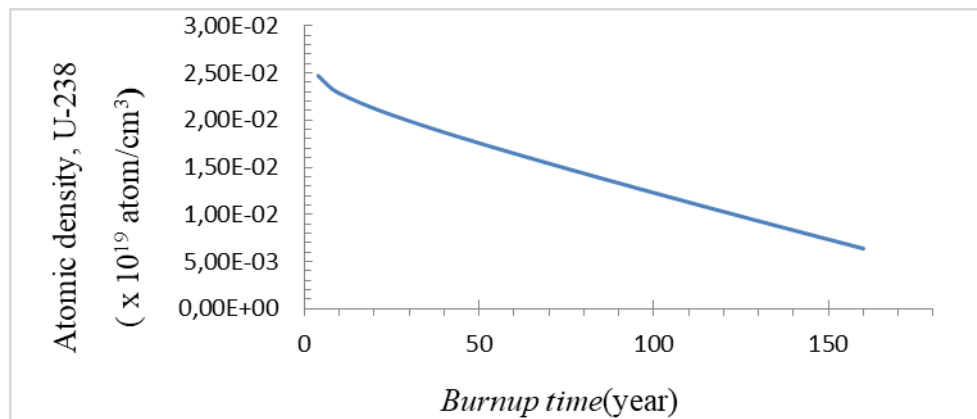


**Figure 3.** Conversion ratios for 100, 120, 140, and 160 years of burn-up

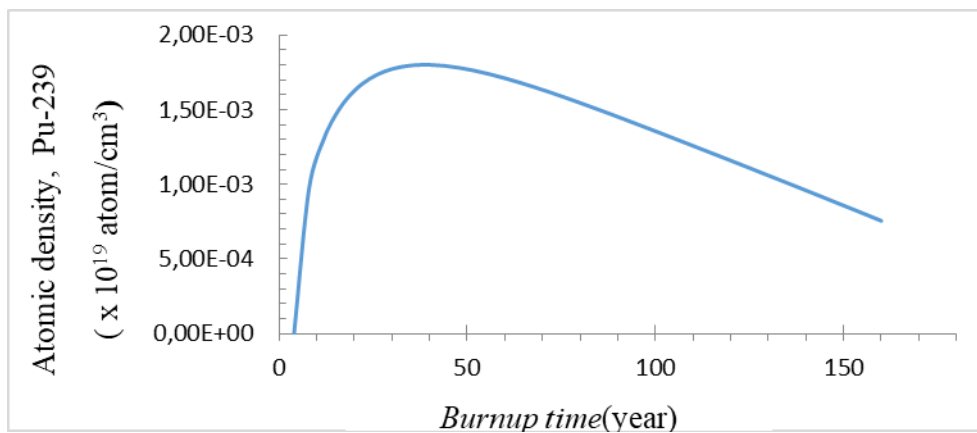
The atomic density distribution is used to obtain the population of the fission reaction in the fuel cell. Also, the fuel material in a nuclear reactor naturally undergoes a reaction while the device is operating. At a specific time, the fuel experiences a reduction in the amount due to the fission reaction and undergo nuclear transmutation. The primary materials used in this design are U-235 and U-238, which are transformed into other element such as Pu-239 during the reactions.



**Figure 4.** Distribution of atomic density of U-235 for 100,120, 140, and 160 years of burn-up



**Figure 5.** Distribution of atomic density of U-238 over 100, 120, 140, and 160 years of burn-up

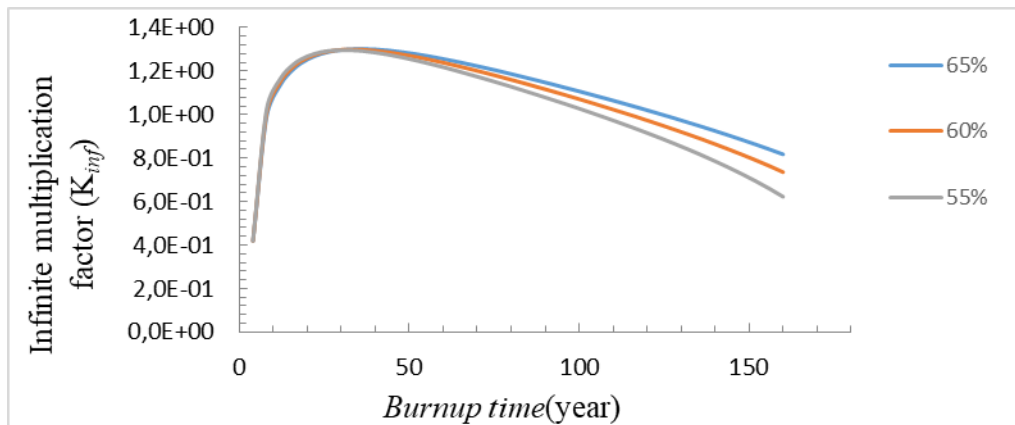


**Figure 6.** Distribution of the atomic density of Pu-239 over 100, 120, 140, and 160 years of burn-up

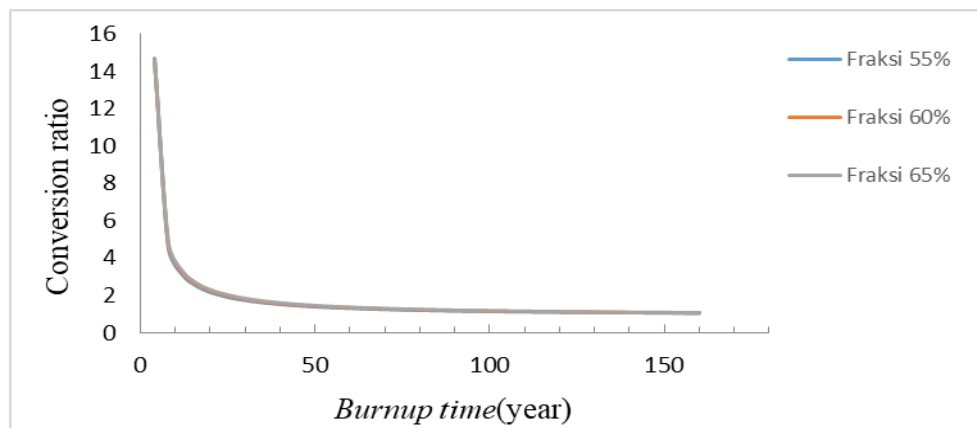
During the burn-up process, the atomic density of U-235 and U-238 changes to other elements due to the fission reaction in U-235 and the transmutation of U-238 which occurs in the fuel cell. Therefore, the atomic density of U-235 and U-238 decreases as the burn-up period increases, as shown in Figures 4 and 5. According to Figure 6, one of the resulting products, Pu-239 is absent because there is no fission reaction at the beginning, but when the burn-up process is running, it increases slowly and reaches its maximum value before decreasing significantly.

Basically, the fuel cell consists of three parts, which includes fuel, cladding, and coolant. The fuel functions in the ongoing fission reaction to produce the required heat, and the cladding determines the

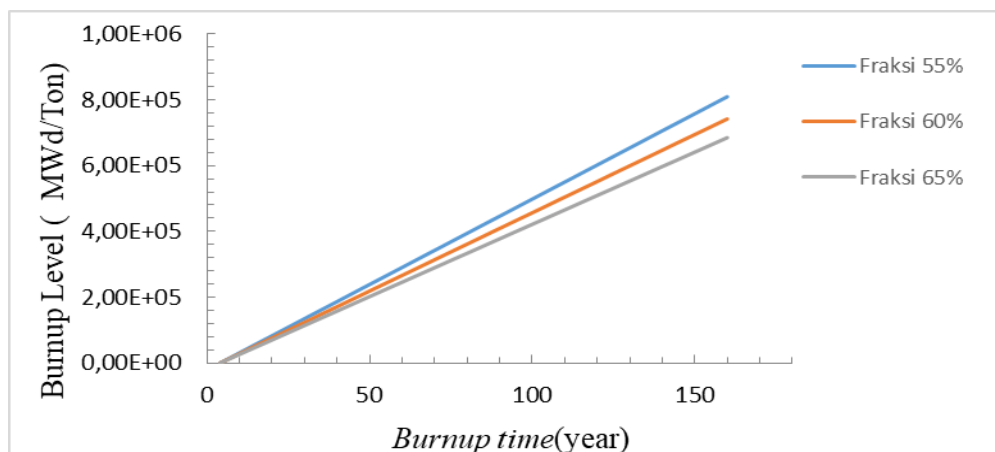
structural strength of the fuel cell to ensure there is no leakage, while the coolant anticipates the excess heat produced. The division of the volume fraction which is converted into 55%-65% fuel, 10% cladding, and 25%-35% coolant is carried out to ensure the comparison of the results obtained in each volume fraction division is changed.



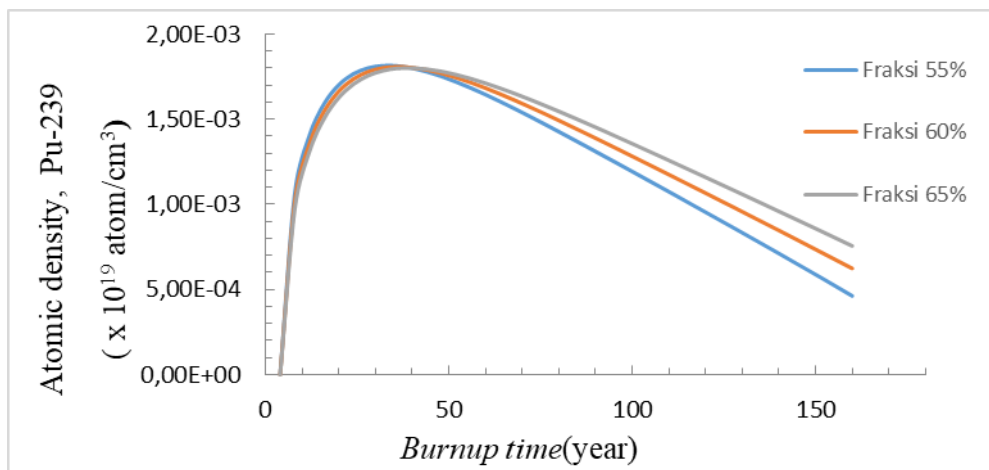
**Figure 7.** The infinite multiplication factor ( $k_{inf}$ ) was compared by dividing the volume fraction 55-65% fuel, 10% cladding, and 25-35% coolant.



**Figure 8.** Comparison of the conversion ratio value by dividing the volume fraction 55-65% fuel, 10% cladding, and 25-35% coolant.



**Figure 9.** Comparison of burn-up level values by volume fraction distribution of 55-65% fuel, 10% cladding and 25-35% coolant.



**Figure 10.** Comparison of the atomic density of Pu-239 with the volume fraction distribution of 55-65% fuel, 10% cladding, and 25-35% coolant.

According to Figure 7, the volume fraction of 65% fuel, 10% cladding, and 25% coolant obtains the highest infinite multiplication factor value in the thirty-sixth year of burn-up. However, for the volume fraction with 60% fuel, 10% cladding, and 30% coolant and the volume fraction of 55% fuel, 10% cladding, and 35% coolant, the multiplication factor value is smaller. This is because the large volume fraction of the fuel causes the fission reaction of the U-235 atom and the transmutation of the U-238 atom, as well as the capture of more neutrons.

Figure 8 shows no significant difference when the volume fraction is changed, hence the conversion ratio (CR) obtained is not affected. However, figure 9 shows that the value of the burn-up level obtained at the volume fraction of 55% fuel is greater than the one with 60% and 65%. This means that the smaller the volume fraction of fuel, the higher the burn-up level value obtained. Also, more fission reactions occur due to the small value of the fuel fraction used. According to Figure 10, the atomic density value of Pu-239 is more significant in the fuel volume fraction of 65% compared to the one of 55%-60%. Therefore, when the number of fuel volume fractions is large, the density of the number of Pu-239 atoms obtained is greater because more fission and neutron capture reactions occur.

#### 4. Conclusion

The calculation of uranium metal fuel cell (U-10%wt Zr) with 4% enrichment during the burn-up time shows a critical value of  $k_{inf} = 1,135$ . This increases continuously to 1,302 in the thirty-sixth year, before dropping towards the end of the burn-up time. Hence, a critical value is shown in each division of volume fraction comparison. However, the one with 65% fuel shows more stable results than the fuel volume fraction of 55% and 60%.

#### Acknowledgement

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