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To cite this article: Enda Susanty *et al* 2019 *J. Phys.: Conf. Ser.* **1282** 012028

IOP Conf. Series: Journal of Physics: Conf. Series **1282** (2019) 012028 doi:10.1088/1742-6596/1282/1/012028

Calculation of burnup fuel cell uranium metallic with carbon dioxide cooled

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Abstract. This article presents the calculation of uranium metallic fuel cell (U-10%Zr) to be used in carbon dioxide cooled Gas-cooled Fast Reactor (GFR). This fuel cell is a cylindrical cell with a 1.4 cm diameter. The fuel cell is tested without enrichment and enrichment of 1-10%. Calculation of fuel cells based on neutron transport equation using SRAC. The calculation parameter observed is the value of infinite multiplication factor (K-inf), conversion ratio, and Pu-239 atomic density. The results of the fuel cell calculation with 6% enrichment showed the K-inf and conversion ratio is relatively stable during burn-up history. While the largest atomic density of Pu-239 occurs in fuel cells without enrichment because U-238 atoms undergoing transmutation to Pu-239.

1. Introduction

A large amount of energy is needed to support human life. Fossil energy is a major part of this energy. However, fossil energy has a problem, namely the limited amount of resources and global warming caused by carbon dioxide [1].

Nuclear energy is one of the alternative solutions in producing energy that is currently being developed. Nuclear energy produces a large amount of energy with the advantage of being non-polluting and economical [2]. The Nuclear Power Plant (NPP) is considered to be one of the alternative sources of electrical energy in Indonesia [3-5].

The working principle of nuclear power plants, in general, is no different from conventional fossil power plants that are utilizing heat generated by fuel as a steam generator. The difference between nuclear power plants and other power plants is from the fuel used. Nuclear power plants utilize nuclear fission that occurs inside the core of a nuclear reactor to produce thermal energy [4].

One of the IV generation nuclear reactors being developed is Gas-Cooled Fast Reactor (GFR). GFR is a reactor that is good in terms of durability because it has a closed fuel cycle that can operate at 850^{o}C so that it supports hydrogen production. The initial step in designing the core of a nuclear reactor is to do the preparation of fuel cells. Fuel cells consist of fuel, cladding, and coolant [6-10].

The preparation of fuel cells aims to produce evenly neutron flux from the results of chain fission reactions. The preparation is done through the calculation of fuel cells resulting in a burn-up parameter that determines the feasibility of fuel cells which is the starting point for the feasibility of a reactor core [8].

A nuclear reactor is a system where you can control and maintain a nuclear reaction. In nuclear reactors, the energy produced is in the form of heat. Therefore, the reactor requires cooling materials that are capable of absorbing heat well, low neutron absorption, stable in radiation and high temperatures. This cooler is used to channel the heat produced when a chain fission reaction occurs [4,5].

This paper describes the parameters of metallic uranium-based (UZr) fuel cell burn-up in a carbon-dioxide-cooled fast reactor. Used carbon dioxide because it has the advantage of being able to work at low pressures, and carbon dioxide has a high heat capacity and heat coefficient compared to helium.

An analysis was carried out on the value of neutrons infinite multiplication factors, conversion ratio and Pu-239 atomic density in fuel cells. After knowing the neutron characteristics, it's hoped that this paper will provide the benefit of being the basis for designing the GFR reactor core [6,8].

2. Literature Review

The fuel cells used as a nuclear energy source is fissile fuel. Fissile material is an element or atom that directly divides when it captures neutrons. The material that is widely used as nuclear fuel is uranium. Natural uranium found in the earth's crust consists of three isotopes, namely U-238 (99.275%), U-235 (0.720%) and U-234 (0.005%). Just only U-235 is a fissile material [12].

Fertile U-238 atoms can be used as fuel by converting them to fissile by neutron capture reactions. U-238 absorbs neutrons will change to U-239 which naturally decays to Np-239 and will produce Pu-239 which is a fissile product. If fissile material like U-235 capture neutron, it will occur a fission reaction. A fission reaction is a radioactive decay process in which the nucleus of an atom splits into smaller parts (lighter nuclei) and followed of release a very large amount of energy [13].

Metallic alloys in the form of the element Zr (Zirconium) is added to the fuel to increase the corrosion resistance, increase the solidus temperature and increases the dimensional stability. UZr has a high density that allows fuel to operate at a much higher burn-up [7,11]. The fission reaction that occurs will produce a neutron flux. In order to produce the desired electrical power, the distribution of neutron flux must be evenly distributed on the reactor core. Therefore, the preparation of the fuel cell must be precise by calculating the fuel cell. The reactor is always maintained so that the neutron flux is always safe in a critical state. The effective multiplication factor (K-eff) can be shown in equation (1) below:

$$k \equiv \text{Multiplication factor} \equiv \frac{\text{Number of neutrons in one generation}}{\text{Number of neutrons in preceding generation}} \quad (1)$$

Criticality of the reactor is the ability to control neutron populations as long as the reactor operates. So that the number of neutrons produced with those produced during the fission process is balanced. [4,5] The reactor is declared supercritical if K-eff > 1, which means the number of neutrons increases as a function of time, sub-critical K-eff < 1, which means the number of neutrons decreases as a function of time and is critical if K-eff = 1 means the number of neutrons does not decrease or increase but constant (fixed) [4,13].

During the burn-up process, the composition of the fuel will always change because fissile isotopes will be produced and some will be consumed (reduced). The burn-up equation that states this is:

$$\frac{dN_A}{dt} = -\lambda_A N_A - \left[\sum_g \sigma_{Ag}^A \ \emptyset_g\right] N_A + \lambda_B N_B + \left[\sum_g \sigma_{Ag}^C \ \emptyset_g\right] N_C \quad (2)$$

Tribes $\lambda_A N_A$ are parts that are lost due to radioactive decay, whereas $\left[\sum_g \sigma_{Ag}^A \emptyset_g\right] N_A$ is the part that is lost because of the neutron capture, $\lambda_B N_B$ is an additional nuclide, nuclide A is obtained from decay B to be A and $\left[\sum_g \sigma_{Ag}^C \emptyset_g\right] N_C$ there is a change in C to A through neutron catches.

Feasibility of fuel performance can be seen from several parameters, namely the effective multiplication factor (K-eff), infinite multiplication factor (K-inf), burn-up level and conversion ratio value which shows the level of comparison of production and consumption of fissile fuel.

3. Research Methods

The calculation of fuel cell is done using the PIJ module on the SRAC (System Reactor Analysis Code) developed by JAERI (Japan Atomic Energy Research Institute) with the JENDL 3.2 library [14]. Fuel cell design parameters can be seen in Table 1.

Table 1. Fuel cell design parameter specifications

Parameter	Value/Description
Fuel Material	U-10%Zr
Cladding Material	Stainless Steel
Coolant Material	Carbon Dioxide
Type of Geometry Cells	Cylinder
Pin Pitch	1,4 cm
Volume Fraksi (Fuel: Cladding: Coolant)	65% : 10% : 25%
Smear Density	85% TD

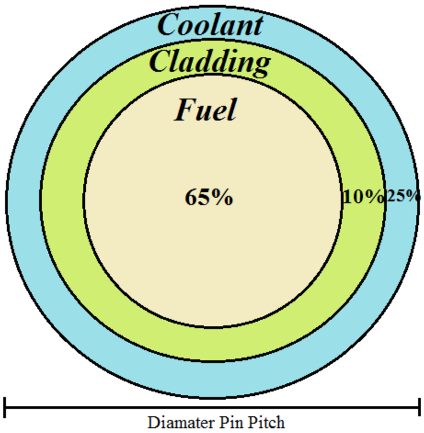

Figure 1. The geometry of fuel cells

4. Results and Discussion

The result of the fuel cell calculation is done for 100 years burn-up history. The composition fuel cells are no enrichment and with enrichment U-235 which varies from 2%, 4%, 6%, 8%, and 10%.

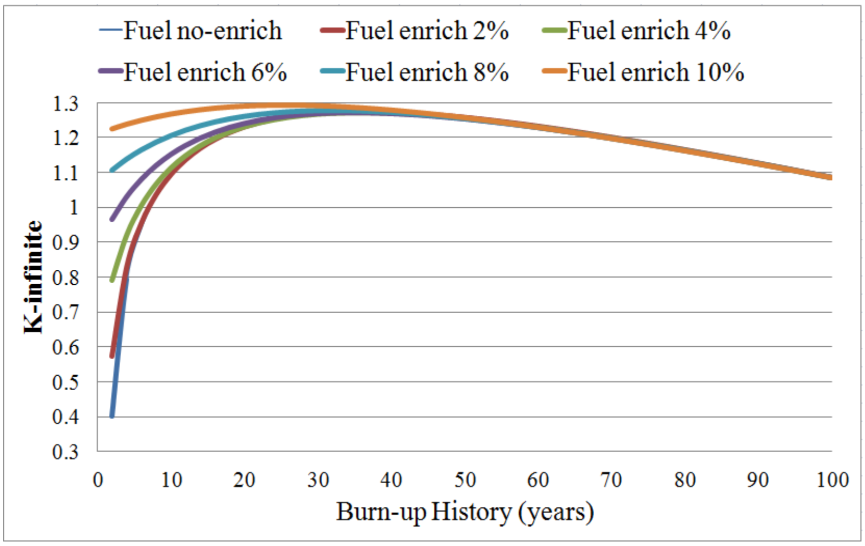

Figure 2. Infinite multiplication factor (K-inf) during burn-up history

Figure 2 shows the infinite multiplication factor (K-inf). K-inf is a measure of the increase or decrease in neutron flux calculated without leakage. The varies enrichment the U-235 related to the value K-inf. For the higher percentage of U-235 enrichment, the value K-inf will be larger in the first time of burn-up history. The peak value of k-inf for all of the fuels was reached at the 30th year, after that, all of them decreased significantly.

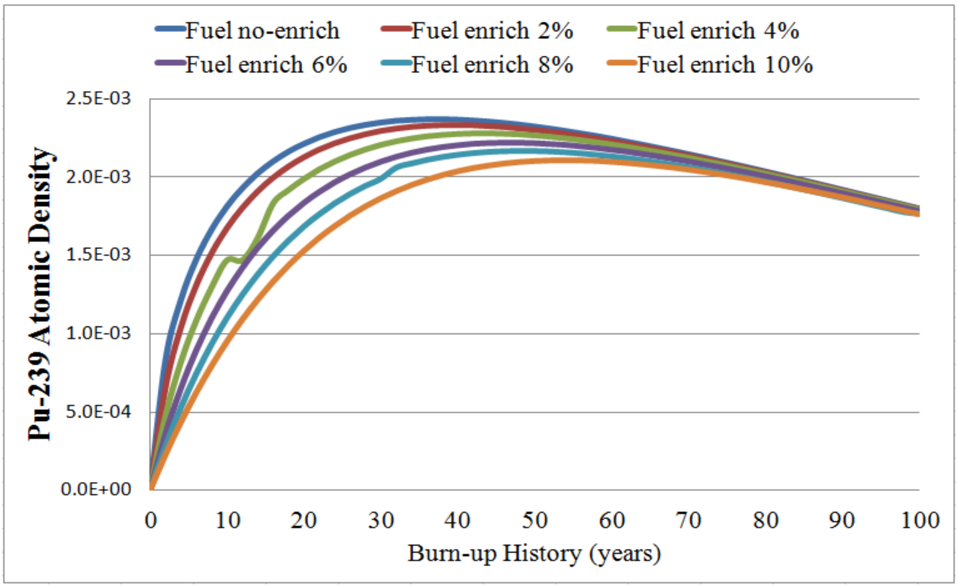

Figure 3. Pu-239 atomic density during burn-up history

Figure 3 shows changes in the concentration of Pu-239 during burn-up with variations in fuel cells that are no enrichment and with enrichment. During burn-up, atomic density U-235 and U-238 atoms decrease because they change to other atoms according to their respective fission reaction chains. The concentration of Pu-239 higher in fuel cells no enrichment because the Pu-239 atom comes from the U-238 atom which experiences a fission reaction by absorbing neutrons. The higher enrichment U-235, the smaller the U-238 atom so that it causes a smaller atomic density of Pu-239.

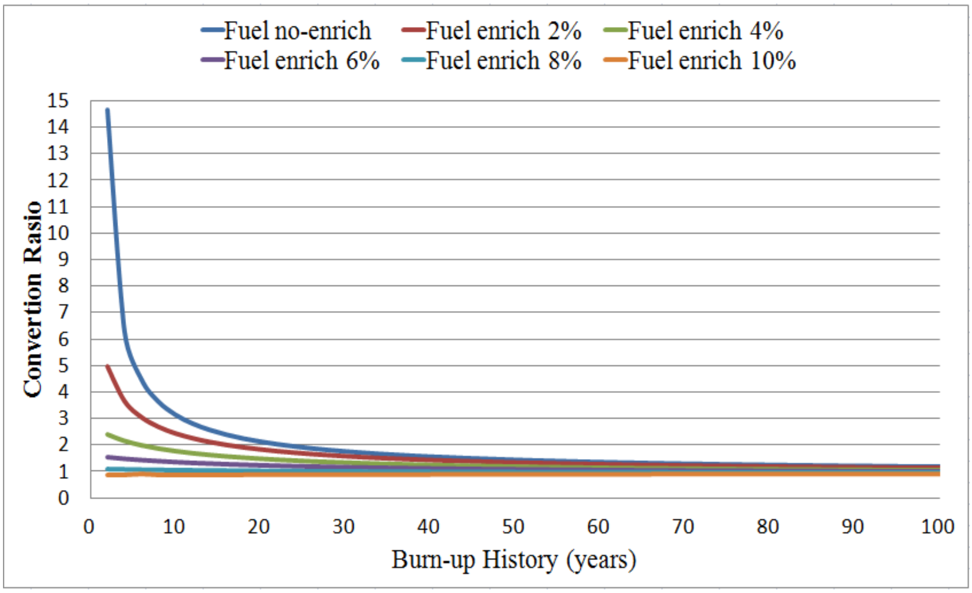

Figure 4. Change in Conversion Ratio during burn-up history

Figure 4 shows that the change in conversion ratio during burn-up. The conversion ratio is defined as the ratio of production rate to depletion rate of fissile nuclides. If the conversion ratio = 1 then the amount of fissile fuel produced is equal to the amount of fissile fuel consumed. If the conversion ratio > 1 then the amount of fissile fuel produced is more than the amount of fissile fuel consumed. In the first 10 years, the conversion ratio for all fuel cell variations declined sharply but after more than ten years the conversion ratio decreased slowly during burn-up history. If you look at one of the characteristics of a generation IV reactor as a breeding reactor, where this reactor must have a conversion ratio greater than or equal to 1. The calculating result shows that fuel with enrichment ≤ 6% has a conversion value ratio > 1 while the enrichment above 6% has a conversion value ratio <1. It means that uranium fuel cells with enrichment ≤ 6% was suitable used to fuel cycle for breeder reactors.

5. Conclusion

Calculation of metallic uranium fuel cells with enrichment 6% U-235 at fuel volume fraction of 65%, cladding of 10% and coolant 25% from the 4th burn-up year produced a critical value of K-inf = 1.03 which continues to increase until the value of K-inf = 1.27 in the 30th year, then decreases again to the value of K-inf = 1.08 in the year 100.

Acknowledgements

This research was financially supported by the Ministry of Research, Technology, and Higher Education, the Republic of Indonesia under the PDUPT (2017-2018) through Sriwijaya University.

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