Study of Neutronic Performance of Sodium-Cooled Fast Reactor Design for Various Output Power Using Radial Fuel Shuffling Strategy

Mohammad Ali Shafii1,*, Elsa Yolanda Putri1, Feriska Handayani1, Dian Fitriyani1, Zaki Su’ud3

1Department of Physics, Andalas University, Padang, Indonesia
2Nuclear and Biophysics Laboratory, Bandung Institute of Technology Bandung Indonesia

*masafii@sci.unand.ac.id

Abstract. Study of neutronic performance analysis of Sodium-Cooled Fast Reactor (SFR) design based on the variations of output power using radial fuel shuffling strategy has been performed. SFR is a type of the generation IV reactor that is widely researched for application on a commercial scale. The reactor design uses natural uranium as a fuel and Sodium as a coolant. The research has been carried out by using the SRAC code and JENDL-3.2 as a library with two dimensional R-Z shuffling strategy of cylinder core for variations of power output 300, 350, 400, 450 and 500 MWTh. The neutronic parameters such as a multiplication factor (k-eff and k-inf) and burn-up analysis are observed. The SFR core is separated into 10 regions having the same volume fraction in the radial direction. At beginning of burn-up process, the reactor core is only filled with natural uranium fuel called fresh fuel and prepared for the one cycle of 10 years of burn-up time. The burn-up result in the first region is shifted in the second region, the burn-up result in the second region is shifted into the third region, and so on until burn-up result into the tenth region. The burn-up result in the tenth region is removed from the reactor core, then the first region can be filled with fresh natural uranium fuel and so on up to 10 times fuel cycles of reactor operation. The neutron calculation results indicate that the multiplication factors (k-eff and k-inf) in a critical condition are occurred on the 300 MWTh of output power. Overall, the output power of 300 MWTh has requirements and a greater chance of being operated for SFR designed.

1. Introduction

The most important segment in the design of the advanced nuclear reactor is to analyze the behavior of the neutrons when they interact with the material that occurs in the reactor core. In general, the neutron analysis needs to be done to determine the distribution of neutron flux in the reactor core represented by the neutron transport equation [1,2]. The solution of the neutron transport equation determines the distribution of neutrons in the reactor to predict neutron flux as a function of space and energy for any given geometry and material distribution [3]. However, neutron analysis is not only about flux distribution, but also about the fuel burn up process. In fast reactors, the burn up process is a very important point to be studied because it is related to the effectiveness of nuclear fuel and the
nuclear reactor life time. A fast reactor is a candidate for the future reactor called Generation IV reactor that is currently a worldwide concern. SFR is a type of the generation IV reactor that is widely researched for application on a commercial scale that are recommended to be operated. SFR is the type of reactor that best meets the requirements of Generation IV, given the experience accumulated over the years [4]. SFR is ready for operation and has active research projects for advanced SFR design and construction [5]. Even, the commercial type of Japan Sodium-cooled Fast Reactor (JSFR) is planned to be operated in 2050 [6]. The advantage of fast reactors such as SFR is that the cycle of fuel can use natural Uranium. The advantages of SFR coolant are in the form of solid at room temperature, melting at a temperature of 98.1˚C and starting to evaporate at 892˚C, having a wide operating temperature compared to ordinary water and having a large specific heat, so it is very good used as heat transfer media [7].

Shuffling strategies can be applied to reactors that operate using natural uranium fuel. In general, radial fuel shuffling strategy is a conventional technique that is still frequently used today, in which the fuel assembly is removed from one region and put into another region [8]. Research on the burn up process with shuffling strategies has been carried out by [9] for traveling wave reactor where natural uranium is fed in by a radial fuel shuffling on SFR. Futhermore, original Modified CANDLE reactor that intensively investigated in Japan is developed using axial [10,11], radial [12] and axial-radial combined shuffling strategy [13].

In this study, neutron analysis on SFR design is based on output power variations of 300, 350, 400, 450 and 500 MWTh. The output power variation is important to determine the optimal power from the designed SFR. The aims of this study is to analyze the neutron characteristics such as neutron multiplication factors and burn up analysis based on output power variations using shuffling strategy in the radial direction. The study is done by using the SRAC (Standard thermal Reactor Analysis Code system) code from the JAEA (Japan Atomic Energy Agency) [14].

2. SFR Core Design
The reactor design uses natural uranium as a fuel and Sodium as a coolant. The neutronic parameters such as a multiplication factor (k-eff and k-inf) and burn up analysis are observed. The general specifications and parameters of the SFR design used in this study can be shown in Table 1.

| Parameters          | Specification               |
|---------------------|-----------------------------|
| Cell Geometry       | Cylinder                    |
| Core Geometry       | 2-D Cylinder                |
| Number of regions   | 10 regions                  |
| Refueling period    | 10 years                    |
| Fuel                | Natural Uranium             |
| Cladding            | SS316                       |
| Coolant             | Sodium                      |
| Fuel fraction       | 51%                         |
| Cladding fraction   | 35%                         |
| Coolant fraction    | 14%                         |
| Pin pitch           | 1.4 cm                      |
| Active height core  | 250 cm                      |
| Active core diameter| 200 cm                      |
| Reflector wide      | 50 cm                       |
In the radial direction of shuffling strategy, the reactor core is separated into 10 regions having the same volume fraction. At beginning of burn-up process, the reactor core is only filled with natural uranium fuel called fresh fuel and prepared for the one cycle of the 10 years of burn-up time. The burn-up result in the first region is shuffled into the second region, the burn-up result in the second region is shuffled into the third region, and so on until burn-up result in the tenth region. The burn-up result in the tenth region is removed from the reactor core, then the first region can be filled with fresh natural uranium fuel \(^1\) and so on up to 10 times fuel cycles or 100 years of reactor operation. The mechanism of shuffling strategy in the radial direction is shown in Figure 1.

![Figure 1. The shuffling strategy in the radial direction.](image)

3. Calculation Methods
The SRAC code system covers production of the effective microscopic and macroscopic cross-sections, cell and core calculations including burn-up and fuel management \(^1^0\). The used library data in the study is the JENDL-3.2 with two dimensional \(R-Z\) shuffling strategy of cylinder core for variations of 300, 350, 400, 450 and 500 MWt power output.

Facilities used in the SRAC code for neutron calculations are the PIJ module to calculate burn-up process of the fuel cell and the CITATION module for multi-group diffusion calculation. The mechanism of the neutronic calculation is as follows; the power density is determined first in each region. Furthermore, the calculation of the burn-up process of cells \((k\text{-inf})\), burn up level, integral conversion ratio, and fuel nuclide density for each fuel is carried out by the PIJ module. The results of cell calculations of the PIJ module are homogenized and collapsed into the 8 energy groups of macroscopic cross section data to be implemented in two-dimensional \(R-Z\) geometry in the multi-group diffusion calculation. The discretization is carried out by dividing the reactor core with a radius of 100 cm into several meshes, with one mesh length of 5 cm in the module CITATION. From CITATION data, the new power level values will be obtained later used in the PIJ module. The average power density in each region of SFR core is recalculated by SRAC code for the calculation of the fuel cell burn-up \(^1^0\). The process is repeated to the first step until a homogeneous power level is obtained. This iteration is repeated for another power density level until the convergence is fulfilled.
4. Results and Discussions

The neutronic performance calculation of the SFR design were investigated for five types of output power i.e. 300, 350, 400, 450 and 500 MWTh. Figure 2 shows the effective multiplication factor ($k_{\text{eff}}$) of reactor with different output power level. Output power of 300 MWTh provides the highest multiplication factor compared to the others. The $k_{\text{eff}}$ value per burn up period remain in subcritical conditions and tends to be critical in the next burn-up period. Burn-up period is carried out for 10 years because the calculation is done once 2 years in 5 periods. The output power of 500 MWTh requires more fission reactions to produce greater energy, consequently more neutron production is needed. This event arises because in fast reactors the fission reaction rate only occur in the region that have a fairly large of fuel distribution \[1\]. Figure 2 also indicates that the greater output power of the reactor produces smaller $k_{\text{eff}}$. It is shown that the reactor can be operated for 5 periods without refueling or fuel shuffling.

![Figure 2. Relation of $k_{\text{eff}}$ with burn-up period for various power output](image2)

The infinite multiplication factor ($k_{\text{inf}}$) is a comparison of number of neutrons in one generation with the previous generation without neutron leakage. Figure 3 is the result of the data obtained from cell calculations, it can be seen that in the initial 20 years the period of burn up the value of the $k_{\text{inf}}$ has not reached the critical condition. In the homogenization of cell calculations, the distribution of neutron flux decreases in the cladding and coolant for fast energy region \[2,3\].

![Figure 3. Relation of $k_{\text{inf}}$ with burn-up period for various power output](image3)
After more than 20 years of burn-up period, the $k$-inf has gradually increased to reach the supercritical condition, namely at an output power of 300 MWTh, while at 350 MWTh, 400 MWTh, 450 MWTh and 500 MWTh has decreased gradually and has a value of $k$-inf in subcritical condition. It is occurred because of at the beginning of the burn up period, the fissile material inside the reactor core is still small, along with the increase in the level of burn up the fertile material $^{238}$U has produced fissile material $^{239}$Pu which results in an increased neutron population to reach the critical condition. In general, $k$-inf for all output power reaches the value of 1 at almost the same time.

![Figure 4. Level of burn up during burn-up period for various power output](image)

Burn-up analysis is needed for matters relating to operating conditions such as initial fuel composition, refueling period, fuel distribution pattern, power changes during reactor operation and control reactivity. Figure 4 show the pattern of burn-up level of the burn-up period as a variation of output power. The burn-up process increases during the burn-up period. The value of the level of burn-up is proportional to the increasing of burn-up period, its mean that the fuel to be used effectively. For 500 MWTh output power, the level burn up value is greater than the others. The greater the output power produced by the reactor, the greater the amount of fuel burned per day. This condition occurs because to produce more power, more neutrons are needed.

![Figure 5. Integral conversion ratio during burn-up period for various power output](image)

The integral conversion ratio (Inte.C.R.) is the ratio of the amount of change in fertile material ($^{238}$U) to fissile material ($^{239}$Pu). There is no significant difference for the output power variation.
Figure 5 shows the integral conversion ratio as a function of burn-up period for output power of the reactor. The profile shows that the pattern of Inte.C.R is inversely proportional with the infinite multiplication factor as seen in the Figure 2. Output power of 500 MWTh after several years of the burn-up period has Inte.C.R. greater than the power of 300MWTh, 350 MWTh, 400 MWTh, and 450 MWTh. It means that the greater output power requires a change in fertile material into more fissile material in the operation of the reactor.

The small output power of 300 MWTh requires a change in the fertile material to a relatively smaller fissile material. At the end of the burn up period, the output power of 500 MWTh has Int. C.R is smaller than the output power of 300 MWTh, due to the output power of 500 MWTh the fertile materials available to convert to fissile material are diminishing.

5. Conclusions.
Study of neutronic performance analysis of SFR design based on the variations of output power using radial fuel suffling strategy has been successfully applied based on the indicators neutronic parameters such as a multiplication factor (k-eff and k-inf) and burn-up analysis. The neutron performance calculation results indicate that the multiplication factors (k-eff and k-inf) are in a critical condition occurring for 300 MWTh of output power. The other parameter such as burn-up level and Inte. C.R. at 300 MWTh has a better value from the beginning to the end of the burn up period. Overall, the output power of 300 MWTh has requirements and a greater chance of being operated for SFR designed.

Acknowledgments
The author thanks to Directorate of Research and Community Service, Deputy for Strengthening Research and Development, Ministry of Research and Technology/National Research and Innovation Agency, with Decree No: 25/E1/Kpt/2020 and Contract No: 163/SP2H/AMD/LT/DRPM/2020 for supporting fund in the Fundamental Research scheme.

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