Natural Uranium Utilization in FUJI-U3 Molten Salt Reactor

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Abstract. Molten Salt Reactor (MSR) is one of the next-generation nuclear power systems known as Generation IV. In this paper, the FUJI-U3 MSR type is designed for neutronic analysis that natural uranium is used as a fuel. The standard reactor core model is adopted due to the flux flattening effect in the active core, as reported from previous research. FLiBe eutectic is used both as the coolant salt and moderator in the reactor core. The neutronic calculation is performed using the SRAC 2006 program: PIJ for pin cell calculation and Citation for core calculation. JENDL 4.0 is used as a nuclear data library that provides the cross-section data of the nuclides. The neutronic parameter results, such as the effective multiplication factor, conversion ratio, neutron spectrum, and power density, are obtained from the calculation. The reactor can achieve the criticality condition with a minimum loading Uranium Tetrafluoride (UF₄) of 3.6% mol for 2000 effective full power days (EFPD). The results calculation provide some parameter survey for the research and the future development of MSR.

1. Introduction
Nuclear energy is an alternative energy that can be applied in Indonesia, and many researchers have been interested in developing this energy because it has a high efficiency [1]. The availability of energy resources also encourages nuclear development to become a commercial scale. Nuclear fuel resources are abundant in Indonesia, as reported that the total amount of the Natural Uranium and Thorium of 70,000 tons and 133,000 tons, respectively [2]. One provider of nuclear energy sources is a nuclear power plant (NPP) that has been developed since 1950. In Indonesia, the development of nuclear energy has begun in 1954 and continues until now [3].

The nuclear power plant generation has been improved by studying the previous system and considering accidents that have occurred in the nuclear power plant. Generation IV is the latest generation of nuclear power plants with increasing security systems, safety systems, providing sustainable energy, nuclear waste minimizing, increasing economic value, proliferation resistance, and physical protection. There are six types of Generation IV NPP such as Gas-Cooled Fast Reactor, Lead-
Cooled Fast Reactor, Molten Salt Reactor, Sodium-Cooled Fast Reactor, Supercritical-Water-Cooled Reactor, and Very-High-Temperature Reactor [4].

The development of nuclear energy must have a very good level of accuracy, so it is necessary to do a simulation that can describe the nuclear power plant system. In this study, a neutronic calculation was carried out through the SRAC version 2006 simulation program for MSR systems[5]. MSR can be operated in thermal, epithermal, and fast neutron spectra which based on the moderator in the core. Some of the advantages of MSR include safety improvement, proliferation resistance, resource sustainability, burning waste and ability for hydrogen production [6][7][8]. This reactor uses liquid fuel type with the material in the form of molten salt. One type of MSR that is currently being developed is FUJI-U3 in Japan that this reactor uses three-division systems in the active core and graphite as a moderator [9]. The calculation uses the PIJ and CITATION module for pin cell and core calculation, respectively. The nuclides data used is available in JENDL 4.0 as nuclear library data.

2. Methodology

The specification of the FUJI-U3 reactor is described in Table 1. Compared with original FUJI-U3, the fuel salt researched in this study does not contain U-233 as initial fissile nuclide. The fissile nuclide for initial operation uses Natural Uranium enriched to replace U-233. In addition, Th-232 as fertile nuclides is loaded in the fuel, which can produce new fission products for subsequent fission reactions. The molten salt used is the type FLiBe (LiF-BeF₂) eutectic because this salt has high uranium solubility, low melting point, high heat capacity, and thermal conductivity [10][11].

Table 1. Specification of FUJI-U3 design [9][12]

| Parameters          | Specification     |
|---------------------|-------------------|
| Thermal Output      | 450 MWth          |
| Electric Output     | 200 MWe           |
| Thermal Efficiency  | 44.40%            |
| Reactor Vessel      |                   |
| - Diameter / Height (inner) | 5.40 m/5.34 m  |
| - Thickness         | 0.05 m            |
| Core                |                   |
| - Diameter / Height | 4.72 m/4.66 m    |
| - Fuel volume fraction(av.) | 36%           |
| Fuel path / Duct    |                   |
| - Width             | 0.04 m            |
| - Fuel volume fraction | 90vol%       |
| Reflector           |                   |
| - Thickness         | 0.30 m            |
| - Fuel volume fraction | 0.5 vol%      |

As shown in Figure 1, the active core reactor is divided into 3 regions based on the fuel volume fraction. The fuel volume fraction for core 1, 2, and 3 are 0.39, 0.27, and 0.45, respectively. This value will affect the total amount of both fuel and graphite moderator. This variation is expected to maintain the graphite moderator state during the reactor operation [9]. This reactor is also equipped with fuel duct, reflector, and vessel as a protector when fission occurs. The radius and height of each core can
be seen in Table 2. These values will be processed in the calculation and used as input to the program code.

![Figure 1. Configuration and design of active core FUJI-U3](image)

### Table 2. Radius and height of active core and reflector

|       | Radius (cm) | Height (cm) |
|-------|-------------|-------------|
| R1    | 116         | H1          | 123        |
| R2    | 80          | H2          | 70         |
| R3    | 40          | H3          | 40         |
| R4    | 30          | H4          | 30         |

The composition of the fuel used is 72 mol% LiF, 16 mol% BeF$_2$, and 12 mol% ThF$_4$+$\text{UF}_4$. ThF$_4$ and UF$_4$ concentrations were varied to obtain the reactor criticality values during 2000 Effective Full Power Days (EFPD). Natural uranium enriched in UF$_4$ is set of 19.75% mol of U-235. Thorium cycle is also expected to produce sustainable energy in reactors because it can produce fissile nuclides for subsequent fission reactions [13].

### Table 3. Fuel salt concentration

| LiF (mol%) | BeF$_2$ (mol%) | ThF$_4$ (mol%) | UF$_4$ (mol%) |
|------------|----------------|----------------|---------------|
| 71.76      | 16.00          | 9.50           | 2.50          |
|            |                | 9.25           | 2.75          |
|            |                | 9.00           | 3.00          |
|            |                | 8.75           | 3.25          |
|            |                | 8.40           | 3.60          |

3. Results and Discussion
In this section, the effective multiplication factor (keff) value becomes one of the focus analysis based on some concentration of UF$_4$ in the FUJI-U3 core. The keff value is a critical reactor benchmark that if the value is higher than one, the reactor can operate in critical condition. The composition of UF$_4$ loaded in the active core is varied from 2.5% up to 3.60% to obtain the critical condition of the reactor. Each UF$_4$ concentration has a similar trend of keff, as shown in Figure 2a. In general, all keff value decreases with increasing burnup value as well as operation time. For 2.5% of the UF$_4$ case, the keff value is under one from the beginning of operation (BOC) up to ending of operation (EOC) due to the less concentration of fissile nuclide in this case. The increasing of UF$_4$ concentration increases the keff value accordingly due to the increasing fissile material that can be burned in the active core. Based on the keff value, the reactor can operate on the critical condition for 2000 days by minimal loading UF$_4$ fuel of 3.60% or more with the final burnup is 2 GWDay/ton.

The contribution of UF$_4$ fuel to the breeding capability is shown in Figure 2b in terms of conversion ratio as a function of burnup for varied UF$_4$ fuel concentrations. The breeding capability of fuel is a survey parameter for sustainable energy in a nuclear reactor, especially in the MSR system. The obtained result for a conversion ratio of all UF$_4$ fuel concentrations gives high value for fuel breeding capability. The value is defined from the total produced fissile from the transmutation process from fertile into fissile divided by the total burned fissile in the reactor core. In this case, U-$^{238}$ can be converted to Pu-$^{239}$ and also Th-$^{232}$ can be transmuted to U-$^{233}$ in the reactor. Based on the result, the conversion ratio decreases with the increasing of UF$_4$ concentration and also the value from each concentration increase during operation time except for 2.50% of UF$_4$, which decreases with time and burnup. The high conversion ratio of low UF$_4$ concentration has estimated come from the high concentration of Th-$^{232}$ as the main fertile in the fuel salt. The high conversion ratio is also shown during reactor operation because some fissile is produced from fertile, which will be burned for maintaining reactor operation.

![Figure 2. (a) Effective multiplication factor vs burnup and (b) Conversion ratio vs burnup](image-url)

The neutron spectrum is also analyzed for obtaining the critical parameter in the neutronic point of view because the neutron population in energy terms can be evaluated. In this study, the reactor is operated in the thermal energy range because the moderator is used to moderating the fast neutron into the thermal neutron in the reactor core. The neutron spectrum of UF4 fuel as a function of energy is shown in Figure 3, where (a) for core 1 region, (b) for core 2 region, and (c) for core 3 region. In the thermal energy range, the spectrum neutron decrease with increasing UF4 concentration, meanwhile in the fast energy, there is no significant difference in the resulted neutron spectrum with increasing UF4 fuel. The decreasing neutron spectrum in the thermal energy is due to the increasing U-$^{238}$ in fuel salt, which has a high number of neutrons per neutron absorbed in the fast energy range. In addition, spectrum neutron behavior is different for core 1, core 2, and core 3 region, which can be compared.
from each spectrum peak in the thermal and fast energy range. The neutron spectrum peak for the core 2 region is relatively higher than in other regions due to the high moderator fraction in this region. The facts mean the neutron population in the core 2 region is distributed relatively in the thermal energy range while in the other regions, the spectrum becomes harder.

The power distribution of UF₄ fuel is conducted for the analyzing power distribution behavior at the beginning of operation (BOC) up to the ending of operation (EOC) for radial and axial direction. In the BOC, the reactor operates with fresh fuel while in the EOC, with the fuel composition after irradiation. In this case, the power distribution is analyzed for 3.60% of UF₄ fuel concentration because this concentration is the minimum UF₄ concentration loaded for reactor operation during 2000 days. Based on Figure 4, both in the radial and axial direction, power distribution is higher at the BOC than at the EOC because the decreasing burned fissile material in the reactor core, therefore, at the EOC the reactor will operate in subcritical condition. This behaviour can be identified as one of the advantages of the MSR system because the reactor has negative excess reactivity during the reactor operation.
4. Conclusions

The study of enriched Natural Uranium utilization in FUJI-U3 Molten Salt Reactor type on the neutronic point of view has been conducted. Some parameters survey is analyzed for the FUJI-U3 reactor, which gives a result as neutronic dependances. As the neutronic parameter, the k_{eff} value of varied UF4 concentrations is decreasing with decreasing burnup and operation time. The decreasing k_{eff} value is estimated from the decreasing burned fissile in the reactor core due to the maintenance of the fission reaction. Based on the k_{eff} analysis, the criticality of this reactor can be achieved by minimal loading fuel of 3.60\%mol of UF4, which with this concentration, the power distribution has a high value at the BOC than at the EOC in the radial and axial direction. In the conversion ratio analysis, UF4 fuel has a high breeding capability, and the value decreases with the increasing UF4 concentration. The increasing UF4 fuel is also affecting in the neutron spectrum, especially in the thermal energy range. For other parameter surveys, the difference of moderator fraction gives a varied peak of neutron spectrum, which the result shows the core 2 region has a high peak in the thermal energy range, and the other region has a hard neutron spectrum.

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