Neutronic Performances of 100 MWe MSR with Weapon Grade Plutonium Fuel

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Abstract. An advanced nuclear reactor Generation IV, called Molten Salt Reactor (MSR), has been developed with Thorium utilization for a sustainable energy system. In this paper, the study is focused in the neutronic calculation of 250 MWt/100 MWe MSR with Thorium-Plutonium (Th-Pu) fuel salt for 5 years time operation. Fuel salt is composed of a eutectic FLiBe, Th, and Pu as a coolant, fertile, and fissile nuclide, respectively. Pu loaded is a weapon-grade which consist of $^{238}\text{Pu}$, $^{239}\text{Pu}$, $^{240}\text{Pu}$, $^{241}\text{Pu}$, $^{242}\text{Pu}$, and $^{241}\text{Am}$. The reactor is operated in the thermal energy range and moderated by the graphite. The reactor design is calculated in neutronic terms with program code CITATION in SRAC 2006 with JENDL 4.0 as a nuclear data library. The result shows some neutronic parameter changes with increasing Plutonium loaded and the reactor criticality is obtained for 5 years by minimal loaded of PuF$_4$ of 2.41%mol. The utilization of Plutonium is described as a capability of MSR in burning a high-level waste of nuclear and radioactive isotopes. This system can be dedicated to future cleaning energy production in a nuclear reactor.

1. Introduction
The nuclear reactor physics is an elaborate physics analysis of the neutron reaction system in the reactor. As a disciplinary study, this analysis has been mature developed for a principle behavior of the neutron, nuclear data, and a computational method. The neutron behavior in the reactor can be analyzed by describing the population, distribution, energy, and flux of the neutron. This analysis is also called a neutronic calculation that has an important role in maintained and controlled the neutron reaction in the reactor.

One of the theories that modeled the neutron distribution in the reactor is the diffusion neutron theory. This theory assumes that the neutron motion in the reactor as a diffusion process. In nuclear reactor analysis, there is some computational method in the programming tool that used the diffusion neutron theory to solve the neutronic calculation. In 1986, JAERI (Japan Energy Research Institute) has been developed a code system for neutronic calculation of many reactor types. That is called as SRAC (Standard Reactor Analysis Code)[1]. The latest and revised version of the code is SRAC 2006 that updated in 2007. The code system in this program adopted both of the transport and diffusion neutron calculation. The nuclear data is also considered in the calculation to obtain the results.
sufficient similar to real cases. Several nuclear data has been revised and validated based on the experimental and the calculation, such as JENDL, ENDF/B, and JEFF [2–4].

The SRAC code can be used to calculate the neutronic design of a steady-state MSR system. MSR is selected as one of the six of the advanced nuclear reactor system in Generation IV in 2002[5,6]. The reactor fuel is mixed with liquid salt and fissile nuclide that distributed as homogenous salt in the reactor. The neutronic performance for the utilization of some fissile nuclides has been reported in our previous study, such as the utilization of $^{233}$U in FUJI-U3 reactor[7], the utilization of some Pu isotopes[8-12] and U-enriched in FUJI reactor[13]. The problem of the spent fuel from the reactor is also motivating the researcher to investigate the utilization of Plutonium in the fuel of the reactor[14,15].

MSR has a deep burn process in the fuel salt that can be used to incinerate the waste nuclear[16]. This process is the capability of the MSR system in handling the spent fuel without a complicated fuel fabrication. Plutonium is a spent fuel that is produced from the nuclear reactor such as PWR[17]. The evaluation of handling Plutonium isotope is a complex issue regards to the high decay heat and spontaneous fission. As one of the grades Plutonium, weapons-grade Plutonium has a high concentration of $^{239}$Pu as fissile nuclide that can be considered in the utilization as a fuel. This scheme can be considered as a way to reduce nuclear weapon worldwide. In the present study, a new design of MSR is proposed and calculated with a power electric of 100 MWe and used a varied concentration of weapon-grade Plutonium as fuel. The neutronic performance, such as the effective multiplication factor, burnup value, reactivity, and neutron spectra, are analyzed in detail.

2. Methodology

The main parameters design of 100 MWe MSR are tabulated in table 1, in which the active core consists of fuel pin assemblies. The fuel duct and reflector are included in the neutronic calculation for a detailed result in the simulation. The fuel salt used is composed of FLiBe salt mixed with Th and weapon-grade Plutonium. Meanwhile, the moderator and reflector are made from graphite. All of the nuclides used are calculated in the nuclide density form as an input on the neutronic design simulation.

| Specification               | Value  |
|-----------------------------|--------|
| Power Output                | 250 MWth |
| Electric Power              | 100 MWe  |
| Core Dimension              |        |
| - Radius                    | 118 cm |
| - Height                    | 233 cm |
| Fuel Duct                   |        |
| - Thickness                 | 5 cm   |
| Reflector                   |        |
| - Thickness                 | 30 cm  |
| - Density                   | 1.84 gram/cm$^3$ |
| Fuel Volume Fraction        | 40%    |

The simplified geometrical configuration of the reactor is shown in figure 1. This design is developed in the simulation code to obtain the neutronic performances. In this model, the fuel salt is a homogeneous liquid in the active core that gives benefit in power distribution in the reactor. The reflector used is set in the outer side to prevent a neutron leakage from the active core. Meanwhile, the fuel duct is a fuel channel to connected fuel pin assemblies.
The operating temperature of the reactor is 900K, while the melting point of FLiBe salt used is 733 K[18]. Therefore, the fuel salt in the active core is confirmed in the molten state condition. The molten state of fuel salt provides flexibility on the adjustment of fuel salt concentration and does not require a fuel fabrication. The fuel salt concentration is varied to optimize the neutronic performances of the reactor. The concentration of the fuel salt and the isotopic composition of weapon-grade Plutonium are tabulated in table 2 and table 3, respectively.

**Table 2. Fuel salt composition.**

| Fuel salt   | LiF    | BeF₂  | ThF₄  | PuF₄  |
|-------------|--------|-------|-------|-------|
| Concentration 1 |        |       | 10.04% | 2.20% |
| Concentration 2 |        |       | 9.99%  | 2.25% |
| Concentration 3 | 71.76% | 16%   | 9.94%  | 2.30% |
| Concentration 4 |        |       | 9.89%  | 2.35% |
| Concentration 5 |        |       | 9.83%  | 2.41% |

**Table 3. Isotopic composition weapon-grade Plutonium.**

| Isotopes | ²³⁸Pu | ²³⁹Pu | ²⁴⁰Pu | ²⁴¹Pu | ²⁴²Pu | ²⁴¹Am |
|----------|-------|-------|-------|-------|-------|-------|
| Concentration | 0.01% | 93.82% | 5.80% | 0.13% | 0.02% | 0.22% |

The SRAC 2006 is a program code to calculate detailed neutronics design for many reactor types such as PWR, BWR, HTGR, CANDU, and etc. The application of SRAC code to calculate MSR design is a simplified study for describing the neutronic performances of the reactor. In our previous study, the neutronic calculation for the MSR system has been verified with the other simulation for MSR FUJI-U3, which has a similar initial keff of 1.024[19]. The basic assumptions are used for the calculation, such as two-dimensional RZ geometry and the flow rate of the fuel salt has the same velocity in each point the reactor core as if the fuel is not moving in the reactor. The energy group consists of 107 groups and compressed into 30 groups, for fast neutron and thermal neutron of 24 and 6 groups, respectively. The active core is divided into 32 radial and 16 axial regions.

The operation time of the reactor is about five years, which corresponds to the graphite moderator and reflector lifetime. The graphite used in the active core undergoes high irradiation during reactor operation. In the pin cell calculation, graphite and fuel salt number densities are calculated using the PIJ module and generated macroscopic data. The data is used for the full core calculation using the CITATION module and resulted in several neutronic parameters. The nuclear data library used in this
calculation is JENDL 4.0 as the latest version of the JENDL library. The JENDL 4.0 has been developed by the JAEA nuclear data center, which consists of 406 nuclides.

3. Results and Discussion
The neutronic design has been calculated in the diffusion neutron method and applied to the 100 MWe MSR. Figure 2 shows the neutronic parameters, the effective multiplication factor (keff) as a function of the operation time reactor. The keff is defined as a ratio of neutron population in a generation to the neutron population in the generation before. According to figure 2, the change of keff value is affected by the increasing of the PuF₄ concentration that loaded in the reactor, in which the keff increases with increasing PuF₄ concentration. The criticality of the reactor is obtained with minimal fuel of 2.41%mol of PuF₄ that must be loaded in the reactor, which it can be seen from the keff value is above unity during time operation. The trend of the keff is similar for all cases, which decrease with increasing time operation, which due to the decreasing fissile material that used to maintain the fission in the active core.

![Figure 2. The effective multiplication factor for different PuF₄ concentration.](image)

The burnup value for the optimum PuF₄ concentration in loaded fuel is shown in figure 3. The burnup values is defined as the fission energy release divided by the total mass of the initial fuel loaded. The maximum burnup is obtained of 6×10³ MWDay/Ton for five years time operation. The fact means that 2.41%mol of PuF₄ concentration in the fuel salt reactor can be operated with this burnup scheme.

![Figure 3. Burnup value for optimum fuel salt loaded for 100 MWe MSR with weapon-grade Plutonium fuel.](image)
The excess reactivity changes during time operation for the case with variation in PuF₄ concentration is shown in figure 4. The negative value of the reactivity can be seen with the PuF₄ concentration of 2.20%mol-2.35%mol and the positive reactivity is shown in the case of 2.41%mol of PuF₄ concentration. This fact means that the excess reactivity of the reactor can be controlled by adding or reducing fuel salt concentration in the reactor. For the optimum concentration of PuF₄ of 2.41%mol, the reactivity is under 2% which means that the reactor is in the equilibrium state.

Figure 4. The reactivity change for different PuF₄ concentration.

The neutron spectra as a function of log energy with the variation of PuF₄ concentration is shown in figure 5. In this paper, the relative flux per unit lethargy is chosen for neutron spectra parameters that the value is a multiplication of the neutron flux, the volume of the i-radial region, and the lethargy width of the g-group energy. As can be seen from the trend of the graph, the higher peak neutron spectra are in the fast energy range $10^{-5} \text{ eV} < E < 3.9279 \text{ eV}$ and gradually decrease in the thermal energy range $0.41399 \text{ eV} < E < 10 \text{ MeV}$. There was a significant correlation between the higher peak spectrum and the adding Plutonium in the fuel salt which due to the higher eta value of Plutonium isotope in the fast energy range. These results have also been reported in the previous work concerning Plutonium and minor actinides loaded in the reactor[10].

Figure 5. Relative flux per unit lethargy 100 MWe MSR with weapon-grade Plutonium fuel.
4. Conclusions
The neutronic performances of 100 MWe MSR with weapon-grade Plutonium fuel have been obtained and discussed. The reactor's criticality is achieved by minimal loading of weapon-grade Plutonium concentration of 2.41%mol. The maximum burnup is obtained about $6\times10^3$ MWDay/Ton in a 5-year time operation. From a safety point of view, the decreasing reactivity within five years reactor lifetime is merit in the inherent safety of the reactor. The neutron spectra are distributed in the fast energy spectrum due to the Plutonium fuel used in the active core reactor.

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