A study on the depletion behavior of molten salt reactor with OpenMC code

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1. Introduction

The Molten Salt Reactor (MSR) is a Generation IV reactor type that is currently under development. In this study, we examined how the reactor criticality varies through the reactor burnup calculation in order to assess how continuously MSR can be operated under specific core conditions. This type of reactor's nuclear fuel is fluid in the form of molten salt and circulates continuously between the in-core and the ex-core during operation. Since the OpenMC, a Monte Carlo code, simulates the reactor in a static state, an assumption was introduced to reflect this flow effect, and the change in criticality was compared after performing burnup calculations by applying it.

2. Methods

The Monte Carlo code, which does static calculations, uses a way of mixing the fuel composition inside the core and the fuel composition outside the core to model the flow of nuclear fuel. In other words, the fuel composition inside the core and the fuel composition outside the core are equal at the start of the calculation, and the burnup calculation is carried out until a specific point in time. The updated fuel composition in the core and the fuel composition outside the core before the burnup calculation are combined, taking into account the volume ratio of each region, to create the new fuel composition in the core. And the new fuel composition outside the core is set as the composition after the burnup calculation inside the core. If the results show similar behavior when the specific burnup time is changed, it is possible to infer that this assumption is valid for the flow of nuclear fuel.

The detailed mixing process is expressed as equations as follows. First, the number density of in-core is updated through the burn-up calculation of Equ. (1) and then, the updated number density of in-core is stored in a temporary variable, and the number density of in-core is mixed with the number density of ex-core in consideration of each volume ratio as Equ. (3). Finally, the number density of ex-core is updated with the number density stored in the temporary variable.

$$\mathbf{n}_{in}^{t+1} = \mathbf{B} \cdot \mathbf{n}_{in}^{t} \tag{1}$$

$$\mathbf{n}_{tmp} = \mathbf{n}_{in}^{i+1} \tag{2}$$

$$\mathbf{n}_{in}^{i+1} = \frac{\mathbf{n}_{in}^{i+1} V_{in} + \mathbf{n}_{out}^{i} V_{out}}{V_{in} + V_{out}} \tag{3}$$

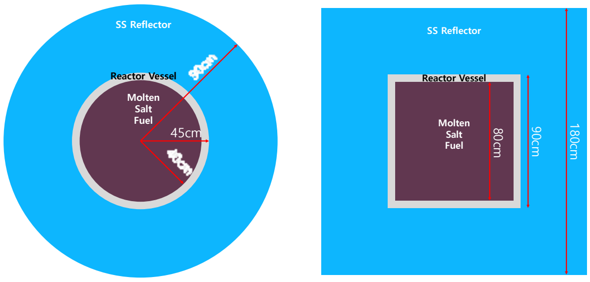
$$\mathbf{n}_{out}^{i+1} = \mathbf{n}_{tmp} \tag{4}$$

3. Validation

The composition of reactor is shown in Table 1, and Fig. 1 depicts the reactor geometry. The initial core is designed to have about a 3,000 pcm excess reactivity.

Table I: Reactor Specification

fuel	$62NaCl\text{-}8MgCl_2\text{-}30TRUCl_3$
reactor vessel	hastelloy-n
reflector	stainless steel
thermal power	$6MW_{th}$



(xy surface) (xz surface)

Fig. 1. Reactor geometry.

The density, volume and mass of each material utilized in the reactor are summarized in Table 2. The mass of the reflector occupies a significant portion of the current geometry, and it is necessary to consider reflectors made of other materials with lower density.

Table II: Material Specification

	density [g/cm^3]	volume [cm^3]	mass [kg]
fuel	3.064	402,123.9	1,232.1
reactor vessel	8.86	170,431.4	1,510.0
reflector	8.0	4,007,886	32,063.1

As the XS library, ENDF-B-VII.1 was used. Total particle number is 50,000. Total and active cycles are 300 and 200, respectively. The target core cycle is 5 years at full power operation, and burnup calculations for 10 years were performed to confirm the change in criticality caused by nuclear fuel burnup. Furthermore, to account for the effect of molten fuel circulation, the results were obtained by mixing nuclear fuel at 0.5, 1, and 2 year intervals. The criticality value, which drops as the reactor operates, is shown in Fig. 2. The condition of operation for five years at full power cannot be satisfied, as indicated in Fig. 2, but it is confirmed that this is true when the fuel circulation is taken into account. It can be seen that the flow of nuclear fuel is well represented, exhibiting a distinct difference from the result value without considering nuclear fuel mixing, even though there is a small difference in the k_{eff} decrease value when the cycle for mixing nuclear fuel is changed.

Fig. 3 shows the criticality reduction results due to nuclear fuel burnup when the in-core and ex-core volume ratios were variated. As a mix interval, 1 year condition was applied. As can be easily expected, it can be seen that the decrease in criticality is reduced when the ratio of in-core to ex-core is less.

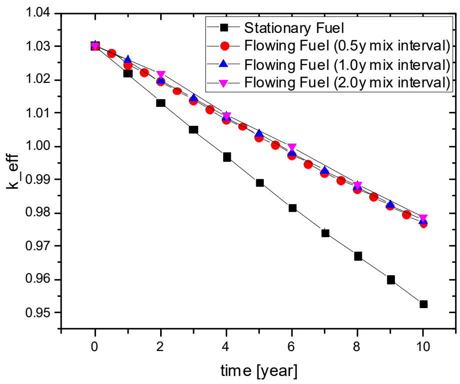


Fig. 2. Decrease in criticality (mix interval variation).

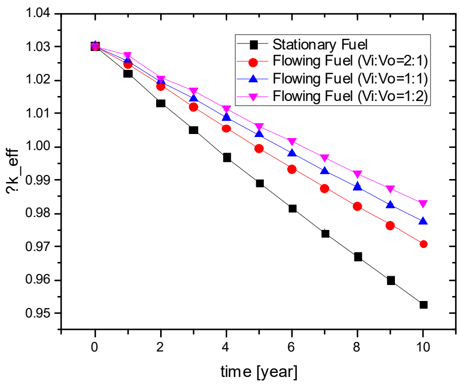


Fig. 3. Decrease in criticality (volume ratio variation).

The power fraction by nuclear reactor of each nuclide is shown in Fig. 4. The heat generated by the nuclear reaction of Pu-239 accounts for approximately 62.7% of the heat, and the heat generated by the nuclear reactions of Pu-240 and Pu-241 accounts for approximately 20.4% of the heat. Table 3 shows the energy generated per fission of each nuclide.

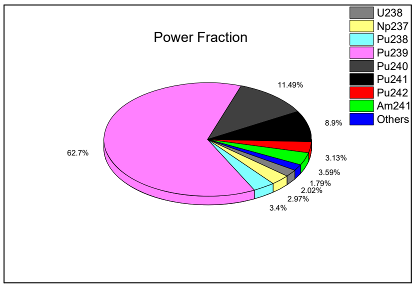


Fig. 4. Heat generation fraction for each nuclide.

Table III: Fission Energy

nuclide	Pu-239	Pu-240	Pu-241
fission energy [eV]	198.902	199.47	201.98

Fig. 5 shows the mass change of Pu-239 over time. Pu-239 is produced by causing two β-decays after U-238 undergoes a neutron capture reaction and is depleted through fission reaction and neutron capture reaction. After ten years of operation, the amount of Pu-239 in the reactor was confirmed to have decreased from 181.8 kg to 166.3 kg. This implies that the amount of Pu-239 depleted by nuclear reaction exceeds the amount of Pu-239 produced by U-238. It is clear from this that the feeding of Pu-239 may be required for the reactor to operate continuously.

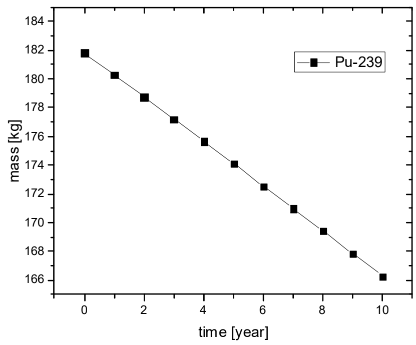


Fig. 5. Decrease in Pu-239 mass over burnup time.

3. Conclusions

In this study, a method of mixing the composition of nuclear fuel inside and outside the core after burnup calculation was introduced in order to analyze the fuel flow in MSR using OpenMC, a Monte Carlo code. The nuclear fuel mix approach for modeling the flow of nuclear fuel was proven to have a discernible impact. This shows that the assumption used to simulate the circulation of nuclear fuel is effective and accurate. Based on the amount of Pu-239 produced and depleted in the reactor by this mixing model, current reactor design requires feeding of Pu-239 for continuous operation of 10 years.

ACKNOWLEDGMENTS

This work was supported by the Korean government (MIST) [KAERI grant number 522310-22]

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