A Study on the Applicability of Simplified GET Theory to MSFR with Local Moderator

Sungtaek Hong[ab] and Yonghee Kim[b*]
[a]*Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Republic of Korea*
[b]*Korea Advanced Institute of Science and Technology, 291 Daehak-ro, Yuseong-gu, Daejeon, Republic of Korea*
[*]*Corresponding author: yongheekim@kaist.ac.kr*

1. Introduction

The Molten Salt Reactor (MSR) is a type of reactor within the Generation IV International Forum (GIF) [1], possessing several characteristics including low-pressure operation, liquid fuel, accident resistance, and high fuel utilization efficiency. However, in order to address the limitations of conventional thermal spectrum-based MSRs, previous studies have proposed the use of the Molten Chloride Salt Fast Reactor [2].

Based on previous research [3] [4], there is a significant difference of several hundred pcm in the reactivity calculation of the Molten Salt Fast Reactor (MSFR) when comparing results from the neutron diffusion equation (hereinafter, NDE) and the Monte Carlo method. This suggests that relying solely on a general diffusion theory can lead to significant errors in the calculation of the reactivity of the MSFR. Additionally, it was found that the use of discontinuity factor [5] (hereinafter, DF) in nodal equivalence theory is effective in reducing reactivity errors. [4]

In other studies [6], a moderator was introduced to achieve long-term operation of the MSFR. In this study, the applicability of the Simplified GET-based multi-group diffusion theory to the MSFR model, in which the moderator is loaded on the reflector, was examined.

2. Methods and Results

2.1 Reactor Model

The reactor model used in this study is shown in Figure 1 *(Gray: Fuel, Green: Moderator, Yellow: Side reflector, Orange: Top/bottom reflector)*. The difference from the previous study is that the thickness of the lateral reflector has decreased from 40 cm to 20 cm, and this space is replaced by the moderator.

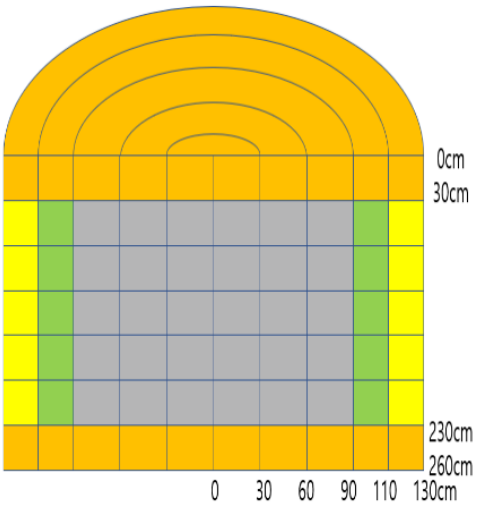

Fig 1. MSFR with Local Moderator

2.2 Reactor Materials

The materials utilized in this reactor are consistent with those from a previous investigation [2], except for the U-235 fuel concentration which is now 19.75 w%. Additionally, BeO has been added as a moderator material. The materials information is provided in Table I.

Table I. Materials Data

Materials	Data
Fuel salt	46KCl-54UCl_3
U-235 enrichment	19.75 w%
Cl-37 enrichment	99.0 a%
Moderator	BeO
Reflector	Stainless steel 304

2.3 Calculation Method

2.3.1 Monte Carlo Method

The SERPENT2 code was utilized in this study for generating reference data such as K_{eff}, group libraries, and surface current, etc. Table II provides details of the calculation settings applied to the SERPENT2 code.

Table II. SERPENT Information

Code name	SERPENT 2.2.0
Libraries	ENDF/B-VII.1
Particles	500,000
Cycle	Inactive: 200, active: 300

2.3.2 NDE Method

In this calculation, a multi-group neutron diffusion model is utilized. Since fission reactions occur in a wide energy range for fast reactors, they need to be divided into several energy groups. In this study, 9 energy groups are employed, which are detailed in Table III.

Table III. Energy Groups

Group	Upper Bound Energy (MeV)
1	20
2	6.07
3	1.35
4	4.98E-01
5	1.83E-01
6	6.74E-02
7	2.48E-02
8	9.12E-03
9	3.35E-03

The MSFR fuel is assumed to be homogeneously mixed, meaning the nuclear data is assumed to be constant regardless of its location within the fuel. To solve this equation, the Finite Difference Method (FDM) is employed.

3. Calculation Strategy and Results

3.1 Calculation Strategy

3.1.1 Determination of DF between Materials

The DF at the interface between materials is calculated only two representative values (one for each direction facing the interface) for each energy group. To determine the DF of the lateral materials of the reactor, the SERPENT2 code is utilized with an infinite cylinder model, as illustrated in Figure 2. Likewise, an infinite plate model, depicted in Figure 3, is employed to calculate the DF at the interface between the fuel and the top reflector.

The K_{inf}, group libraries, and surface current values for the infinite core model are obtained using SERPENT2, and these values are used to solve the infinite core model using the NDE method. This NDE is solved using the Finite Difference Method(FDM). The size of the computational grid for FDM calculation is 0.2cm. The NDE method solution yields the representative surface flux of each group at the interface. The DF is calculated by comparing that value to the surface flux obtained using SERPENT2 code.

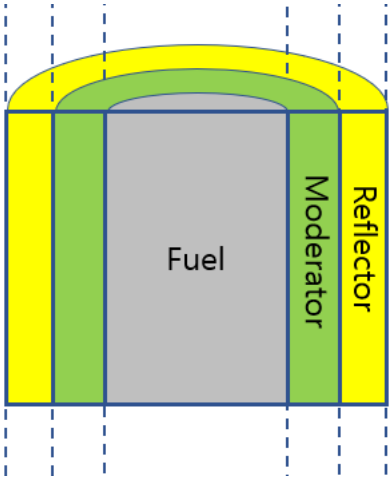

Fig 2. Infinite Cylinder Model for Side DF

An infinite cylinder model illustrated in Figure 2 was employed to calculate the DF between materials on the lateral surface of the reactor. The infinite lattice model incorporating the composition of nuclear fuel described in section 2.2 was used to obtain the DF. The interface between the materials was evaluated for surface current/flux, K_{inf}, and group libraries using the SERPENT2 code.

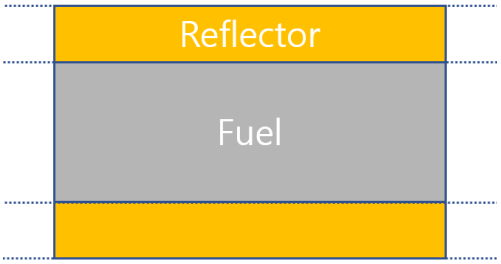

Fig 3. Infinite Plate Model for Top DF

Figure 3 illustrates the use of an infinite plate model to determine the DF between the nuclear fuel and the top/bottom reflector. The procedure for obtaining the DF is very similar to the one explained above. However, because the nuclear fuel is symmetrically positioned with respect to the top and bottom reflectors in this case, the DF obtained using the nuclear fuel and the upper reflector can be utilized for both sides.

3.1.2 DF Implementation in NDE

$$J_g^{i,i+1} = \frac{2}{\left(\frac{\Delta r_{i+1} f_g^{i+1,-}}{D_g^{i+1}}\right)+\left(\frac{\Delta r_i f_g^{i,+}}{D_g^{i}}\right)}\left(\bar{\phi}_g^i f_g^{i,+} - \bar{\phi}_g^{i+1} f_g^{i+1,-}\right) \quad (1)$$

where, Δr means the cell interval, D means diffusion coefficient, f means DF, $\bar{\phi}$ means the average neutron flux of the cell, J means the surface current. Equation 1 demonstrates the correlation between the surface current and the cell-averaged flux utilizing the DF, and it also indicates the continuity of the surface current owing to the DF. Figure 4 illustrates each parameter at the

interface between the materials, which is utilized in equation 1.

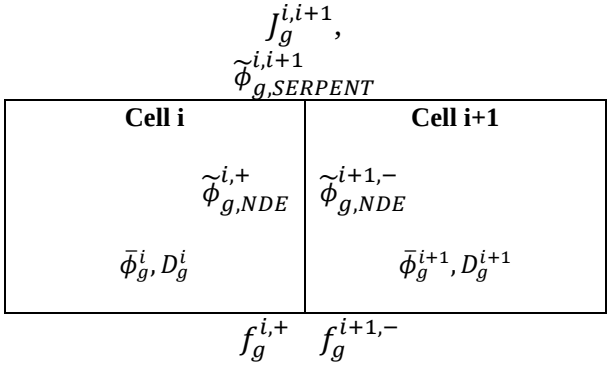

Fig. 4. Surface Flux Discontinuity

3.2 Results

3.2.1 DF Values between the materials

Tables IV and V show representative DF values between reactor materials obtained using the infinite reactor model.

Table IV. DF Values of the lateral materials

Group	Fuel-Moderator-Side Reflector			
	Fuel+	Moderator-	Moderator +	Reflector-
1	1.32210	1.39320	3.11712	0.77470
2	1.25953	1.17404	2.96795	0.87694
3	1.10366	1.06524	1.89119	0.95176
4	1.05949	1.04834	1.62098	0.93965
5	1.06039	1.03162	1.29438	0.99885
6	1.02150	1.05360	1.11212	1.01176
7	1.11610	1.03908	1.06420	0.96449
8	1.15308	0.99603	0.98232	1.04880
9	1.00102	0.93254	0.88917	0.90268

Table V. DF Values of the fuel-top reflector

Group	Fuel – Top Reflector	
	Fuel+	Reflector-
1	0.92923	1.12170
2	0.99089	1.05950
3	1.02016	1.02844
4	1.05567	0.99553
5	1.04061	1.03793
6	1.00532	1.01929
7	1.28903	0.88810
8	1.18310	0.76049
9	0.99759	0.46334

The '+' sign indicates DF looking at the right/up boundary surface of the cell, and the '-' sign indicates DF looking at the left/down boundary surface of the cell.

3.2.2 Results of Applying DF to NDE

This section presents the calculation results of the K_{eff} value according to the application of DF. Table IV, V, and Equation 1 were used in NDE to calculate the K_{eff} value of MSFR. This NDE is solved using the Finite Difference Method(FDM). The size of the computational grid for FDM calculation is 5cm.

Table VI. Result of the PMFR with Local Moderator

Method	K_{eff}	Difference[pcm]
SERPENT2 (Reference)	1.19392 ± 0.00005	0.00
Normal NDE (DF not applied)	1.20879	1030.7
Appling DF	1.19943	384.5

Table VI indicates that without using DF, the reactivity error was 1,030 pcm, but by applying DF, the reactivity error could be reduced to 380pcm. The simplified GET-based multi-group diffusion theory was effective in reducing reactivity errors. Nevertheless, the effect is weaker than the level of improvement demonstrated in previous studies.

4. Conclusion

In this study, the applicability of the Simplified GET-based multi-group diffusion theory to the Molten Salt Fast Reactor (MSFR) model with a local moderator was examined. The results obtained from the multi-group diffusion theory were compared with those obtained from the Monte Carlo method.

It was found that the simplified GET-based multi-group diffusion theory could provide accurate results for the MSFR with a local moderator, with an error in reactivity of less than 400 pcm. Additionally, the use of a discontinuity factor in nodal equivalence theory was effective in reducing reactivity errors. However, the effect was weaker than the degree of improvement shown in previous research [4].

This study contributes to the development of a simplified method for analyzing the reactivity of MSFR, which can improve the efficiency and accuracy of MSFR design and operation. Further research could focus on improving the accuracy of the multi-group diffusion theory for MSFRs with more complex geometries and materials.

REFERENCES

[1] https://www.gen-4.org/

[2] Sungtaek Hong, Seongdong Jang, Taesuk Oh & Yonghee Kim, "Preliminary Study on the Candidate Fuel Salt of Molten Chloride Salt Fast Reactor", Korean Nuclear Society Virtual Spring Meeting (2021)

[3] Eric Cervi et al., "Multiphysics analysis of the MSFR helium bubbling system: A comparison between neutron diffusion, SP_3 neutron transport and Monte Carlo approaches", Annals of Nuclear Energy, 132, 227-235 (2019)

[4] Sungtaek Hong, Seongdong Jang, Taesuk Oh & Yonghee Kim, "A Study on the Application of Equivalence Theory to Molten Salt Fast Reactor", Korean Nuclear Society Autumn Meeting (2022)

[5] K. SMITH, "Assembly homogenization techniques for light water reactor analysis," Progress in Nuclear Energy, 17, 3, 303 – 335 (1986)

[6] Eunhyug Lee, Taesuk Oh, & Yonghee Kim, "Burnable Absorber Design Study for Fast-spectrum Molten Salt Reactors", Korean Nuclear Society Autumn Meeting (2022)