Preliminary Estimation of the Modeling Error in the PGSFR Core Neutronics Design

Sunghwan Yun*, Jong Hyuck Won, Donny Hartanto, and Jae-Yong Lim

Korea Atomic Energy Research Institute (KAERI)
989-111 Daedeok-daero, Yuseong-gu, Daejeon 305-353, Republic of Korea
**Corresponding author: syun@kaeri.re.kr*

Abstract - *In this paper, a preliminary verification of the modeling error in a criticality, sodium void reactivity worth, control rod worth, radial expansion reactivity worth, and Doppler reactivity worth of the PGSFR core were performed by comparing core neutronics design code system (multi-group homogeneous MC2-3/TWODANT/DIF3D-VARIANT) and continuous-energy heterogeneous MCNP6 results based on the ENDF-B/VII.0 library. The error of the core neutronics design code system is as follows: errors from -102.5 pcm to +168.6 pcm in the criticality estimation, about 10 % underestimation in sodium void reactivity worth estimation, 3.5 % maximum error in control rod worth estimation (1-D control rod cross-section case), within 5 % error in radial expansion reactivity worth estimation, within 2σ error in Doppler reactivity worth estimation. In addition, modeling error of criticality and sodium void reactivity worth in the PGSFR core is originated from the non-fuel region homogenization.*

I. INTRODUCTION

The Korea Atomic Energy Research Institute (KAERI) has been developing a metallic fueled blanket-free SFR design to aim at the start-up of Prototype Gen-IV Sodium-cooled Fast Reactor (PGSFR) up to 2028 [1]. One of the essential works to achieve our goal is validation of the core neutronics design code system, which will be submitted to the Korean regulation body (KINS) as a topical report form.

Validation procedure of the core neutronics design code system can be divided by two parts: 1) validation of the cross-section and 2) verification of modeling errors and combined biases [2]. Validation of the cross-section will be finalized at 2017 based on the physics experiments for the metal fueled blanket-free core using continuous-energy library. However, modeling errors of physics experiments are considerably differ from modeling errors of the PGSFR due to significant difference in geometry. Hence verification of a modeling error for the target core, i.e., PGSFR is an essential work to validate of the core neutronics design code system.

In this paper, a preliminary verification of the modeling error in a criticality, sodium void reactivity worth, control rod worth, radial expansion reactivity worth, and Doppler reactivity worth of the PGSFR core were performed by comparing core neutronics design code system (multi-group homogeneous MC2-3/TWODANT/DIF3D-VARIANT) and continuous-energy heterogeneous MCNP6 results based on the ENDF-B/VII.0 library [3-7]. Hence, in this paper, the modeling error includes not only homogenization error but also multi-group approximation error.

II. DESCRIPTION OF THE CORE NEUTRONICS DESIGN CODE SYSTEM AND MODELING

1. MC^2-3/TWODANT/DIF3D Code System

The calculation procedure of the MC^2-3/TWODANT/DIF3D-VARIANT code system for PGSFR is shown in Fig. 1. First, a 0-D slowing down calculation is performed with a critical buckling search to generate an 1041G Ultra Fine Group (UFG) homogenized cross-section using the MC^2-3 code. Second, a TWODANT R-Z SN transport calculation is performed to consider the global spectrum change based on the generated UFG cross-section. Third, 33 group homogenized cross-sections are generated using both TWODANT global flux distribution and 0-D MC^2-3 calculations. Finally, a 3-D hexagonal whole-core calculation is performed using the VARIANT option of the DIF3D code. The depletion calculation is performed by the REBUS-3 code, in which DIF3D-VARIANT module was employed as a neutron flux solver.

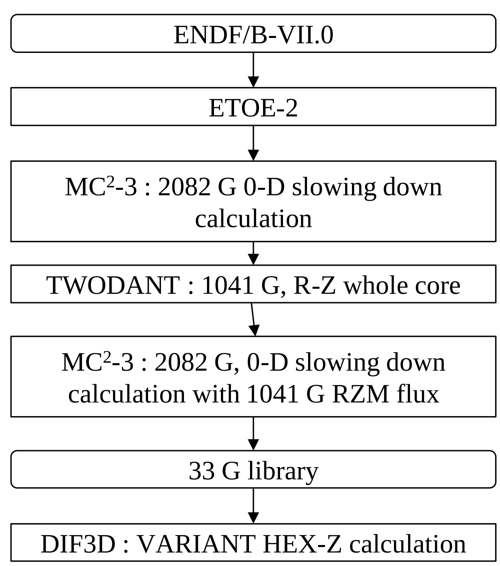

Fig. 1. MC^2-3/TWODANT/DIF3D-VARIANT calculation procedure

2. MC^2-3/TWODANT/DIF3D Model for PGSFR Analysis

Fig.2 shows the radial core configuration of PGSFR uranium core. The difference between recent works on PGSFR core [3] is in-vessel storage (IVS) modeling in core neutronics analysis. There are 66 IVS positions. 41 of them are modeled as dummy assembly (only hexagonal duct is in it), and remaining 25 IVS positions are filled by discharged fuel assembly. Other design information of the PGSFR core is listed in the reference [3].

The representative equilibrium core is selected by explicit depletion analysis. In that, all fuel assemblies except fresh one have different isotope composition and contain many fission product isotopes

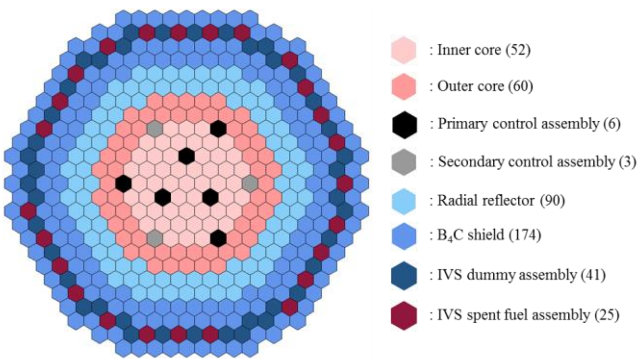

Fig. 2. Radial core layout of PGSFR core

3. Reference MCNP Model for PGSFR Analysis

The configuration of the heterogeneous MCNP model is shown in Figs. 3 and 4. To describe the BOC and EOC condition of the representative equilibrium core, total 896 different materials are assigned for fuel description (8 axial zone for 112 fuel assemblies) and 187 isotopes are assigned for each fuel material including most of fission product isotopes. Number densities of isotopes are based on results of the REBUS-3 code calculation. In the BOC model, primary control rods are inserted by 26 cm to make core critical.

Axially, active core is surrounded by the lower reflector, sodium plenum, and gas plenum. To describe lower reflector using the MCNP code, equivalent lower reflector model was developed and adopted to conserve neutron leakage [8].

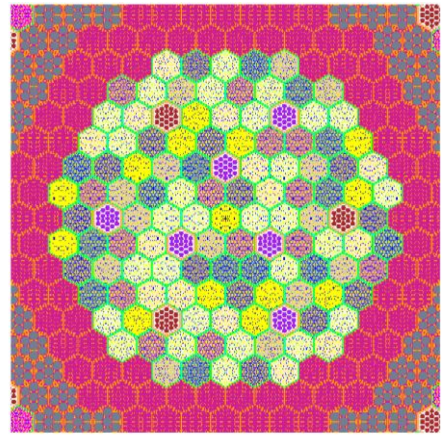

Fig. 3. Radial configurations of the heterogeneous MCNP model for the representative equilibrium core

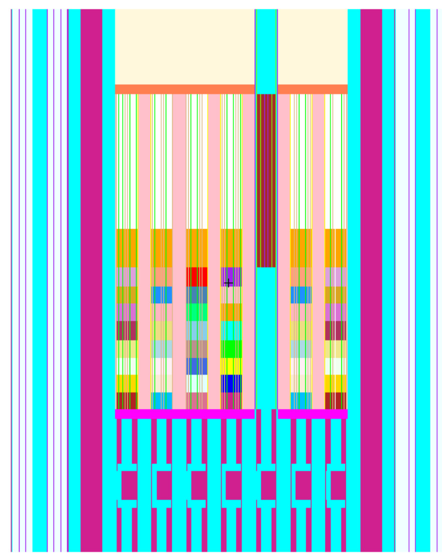

Fig. 4. Axial configurations of the heterogeneous MCNP model for the representative equilibrium core

III. NUMERICAL RESULTS

1. Criticality

Results of the criticality in the representative equilibrium core of the PGSFR are shown in table I. To make consistent condition with BOC ARO, primary control rods are inserted by 26 cm.

The difference in the clean core is similar order compared to results in the reference [2]. The remarkable reduction of difference in the BOC CRP case is observed comparing to the BOC ARO case. This is due to the overestimation of the primary control rod worth in the MC^2-3/TWODANT/DIF3D-VARIANT model which will be discussed in the section 3.

Table I. Results of criticality

Calculation model	Heterogeneous MCNP	MC^2-3/TWODANT /DIF3D-VARIANT	Difference, pcm
BOC (CY13), ARO[a)]	1.01975±0.00006	1.01975	0.0±6.0
BOC (CY13), CRP[b)]	1.00135±0.00003	1.00032	-102.5±3.0
EOC (CY13), ARO	0.99836±0.00003	1.00004	+168.6±3.0

[a)] ARO : All Rod Out.
[b)] CRP : Critical Rod Position, 26 cm insertion for representative equilibrium core.

For clear understanding of these errors, we considered following three partially-homogeneous models which are shown in Figs. 5 through 7. Table II shows results of non-fuel region homogenization effect. The largest error is coming from the homogenization of lower reflector. Homogenization effects of sodium plenum and gas plenum are relatively small, i.e., ~34 pcm.

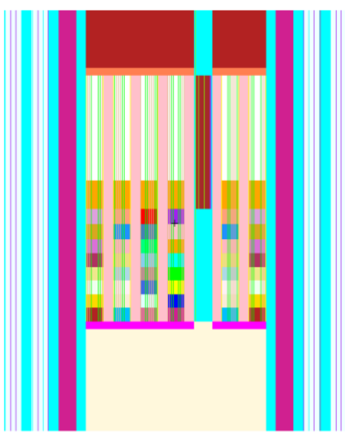

Fig. 5. Axial configurations of the heterogeneous MCNP model with homogenized lower reflector

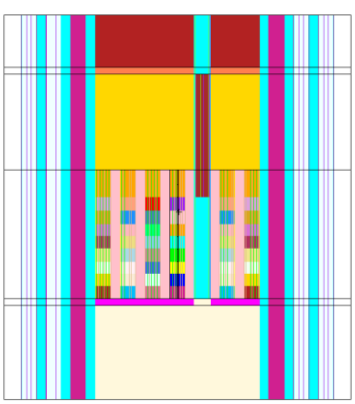

Fig. 6. Axial configurations of the heterogeneous MCNP model with homogenized lower reflector and gas plenum

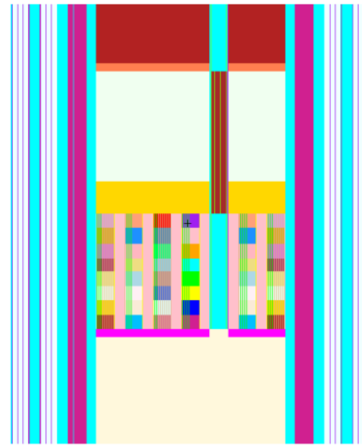

Fig. 7. Axial configurations of the heterogeneous MCNP model with homogenized lower reflector, sodium plenum, and gas plenum

Table II. Homogenization effect of axial non-fuel regions at EOC(CY13)

Code	Fuel	Lower reflector	Sodium plenum	Gas plenum	k_{eff}	Difference, pcm
MCNP	Hete	Hete	Hete	Hete	0.99836 ±0.00003	reference
	Hete	Homo	Hete	Hete	0.99955 ±0.00008	119.2±8.6
	Hete	Homo	Hete	Homo	0.99979 ±0.00007	143.3±7.6
	Hete	Homo	Homo	Homo	0.99989 ±0.00007	153.3±7.6
DIF3D	Homo	Homo	Homo	Homo	1.00004	168.6±3.0

2. Sodium Void Reactivity Worth

Modeling errors of sodium void reactivity worth are shown in Fig. 8 [9]. In describing sodium void phenomena, coolant sodium is voided in both of active core region and its upper regions, i.e., sodium plenum and gas plenum.

In both of BOC and EOC, the MC^2-3/TWODANT/DIF3D-VARIANT model shows about 10 % underestimation in whole core sodium void reactivity. Similar underestimation is also observed in both of inner core (IC) region and outer core (OC) region.

Typically, the sodium void reactivity worth is composed of a) neutron spectrum hardening effect, b) radial neutron leakage increasing, and c) axial neutron leakage increasing. We assume that an underestimation of sodium void reactivity worth at inner core region may be originated from the underestimation of the axial neutron leakage and an underestimation of sodium void reactivity worth at outer core region may be originated from the underestimation of the radial neutron leakage. Since these underestimation of neutron leakage is coming from the homogenization of the non-fuel region, we considered following partially-homogenized MCNP model for sodium void reactivity worth calculation as shown in Fig. 9, in which sodium plenum and gas plenum are homogenized.

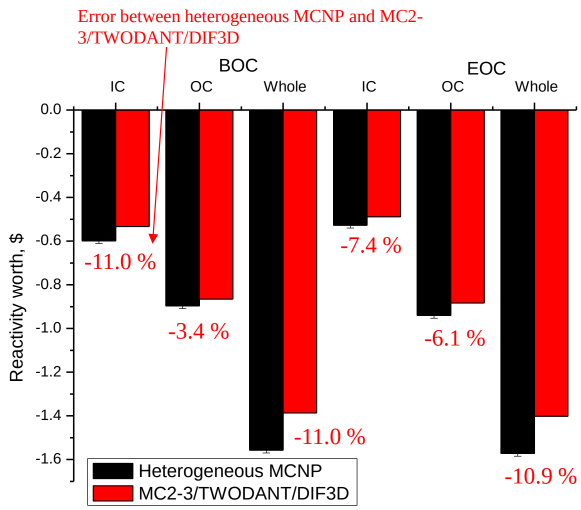

Fig. 8. Modeling error of sodium void reactivity worth

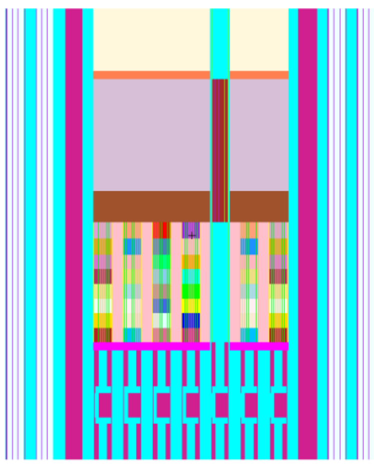

Fig. 9. Axial configurations of the heterogeneous MCNP model with homogenized sodium plenum and gas plenum

Fig. 10 shows modeling errors of sodium void reactivity worth at partially-homogenized MCNP model at EOC. Both of the MC^2-3/TWODANT/DIF3D-VARIANT model and partially-homogenized MCNP model shows very similar sodium void reactivity worth results at inner core region. However, at outer core region, the partially-homogenized MCNP model still shows similar results comparing to the heterogeneous MCNP model. In other words, axial non-fuel region homogenization effect is negligible in estimating sodium void reactivity worth at outer core region. Consequently, we can conclude that underestimation of sodium void reactivity worth at inner core region is induced by homogenization of sodium plenum and gas plenum region, which causes underestimation of the axial neutron leakage.

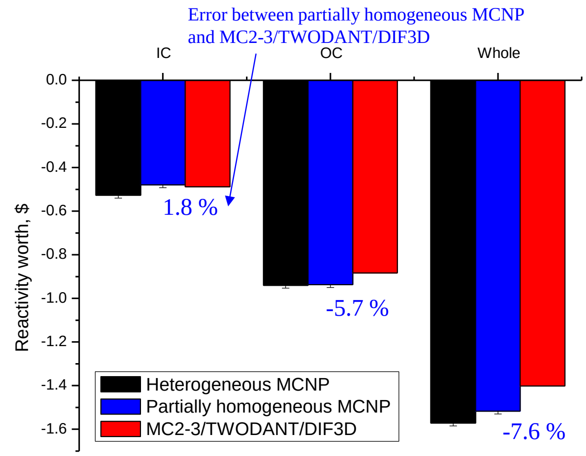

Fig. 10. Axial configurations of the heterogeneous MCNP model for the representative equilibrium core

3. Control Rod Worth

In control rod modeling, usually 0-D model has been used for generating 33G cross-section. Recently 1-D model as shown in Fig. 11 was developed for the control rod cross-section generation [10].

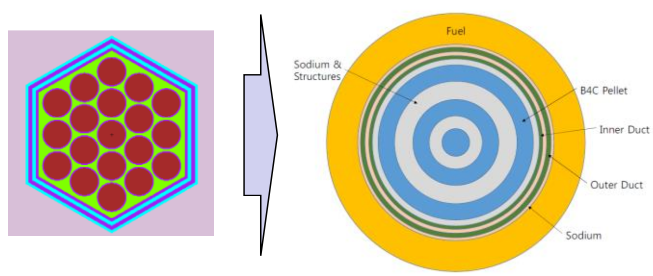

Fig. 11. 1-D MC^2-3 model for 33G control rod cross-section generation

Fig. 12 shows results of the control rod worth calculations. Maximum 9.8 % error is resulted in case of 0-D control rod cross-section, while maximum 3.5 % error is resulted in case of 1-D control rod cross-section.

Results reported at section 1 is based on the 0-D control rod cross-section, hence significant underestimation of the criticality at BOC CRP is resulted. This criticality estimation will be improved when 1-D control rod cross-section is used, and this work is on-going now.

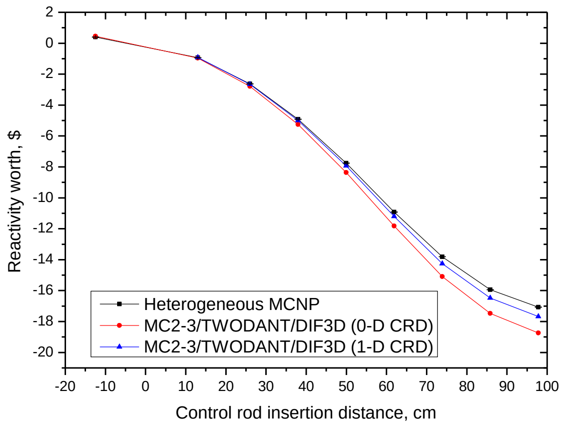

Fig. 12. Results of the primary control rod worth

4. Radial Core Expansion Reactivity Worth

The radial core expansion reactivity worth is based on the thermal expansion of a subassembly support grid as shown in Fig. 13. When radial expansion by thermal expansion of the subassembly support grid occurs at the PGSFR core, the distance between fuel subassemblies is increased and more coolant sodium is inserted into the inter-subassembly gap.

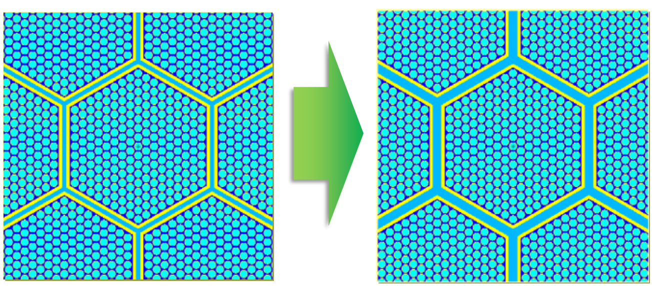

Fig. 13. Radial core expansion reactivity worth model

Table III shows modeling error of the radial core expansion reactivity worth. The MC^2-3/TWODANT/DIF3D-VARIANT model shows good agreement with heterogeneous MCNP model within 5 % error range.

Table III. Modeling error of radial expansion reactivity worth

	Heterogeneous MCNP, $	MC2-3/ TWODANT/ DIF3D, $	Error, %
BOEC	-0.758±0.010	-0.784	3.4±1.4
EOEC	-0.787±0.010	-0.822	4.5±1.4

5. Doppler Reactivity Worth

In this paper, Doppler reactivity worth is calculated based on changing temperature of fuel, i.e., U-Zr. Fig. 14 shows Doppler reactivity worth results at BOC and Fig. 15 shows Doppler reactivity worth results at EOC. In both cases, considerable error (about 10 % underestimation) was caused at 373.15 K. However, the MC2-3/TWODANT/DIF3D-VARIANT model shows good agreement with heterogeneous MCNP model within 2 σ error from 573.15 K to 1773.15 K. Since the operating fuel temperature of the PGSFR is 890.65 K, we can conclude that modeling error of the Doppler reactivity worth is negligible at range between 573.15 K and 1773.15 K.

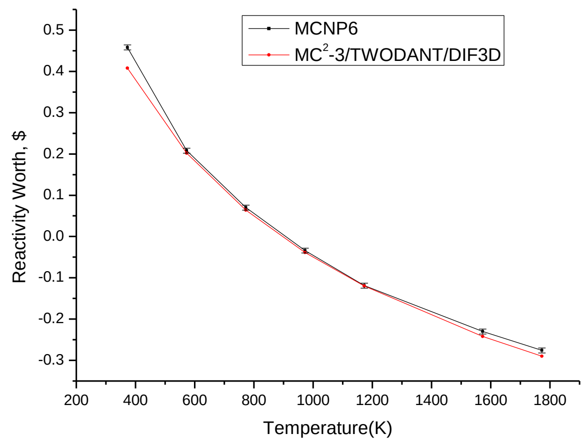

Fig. 14. Doppler reactivity worth at BOC

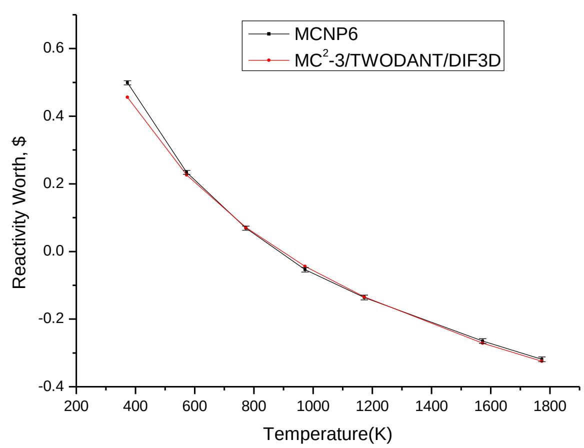

Fig. 15. Doppler reactivity worth at EOC

IV. CONCLUSIONS

In this paper, the modeling error of criticality, sodium void reactivity worth, control rod worth, radial expansion reactivity worth, and Doppler reactivity worth in the MC^2-3/TWODANT/DIF3D-VARIANT code system was studied for PGSFR by comparison with heterogeneous MCNP results.

In the criticality, the MC^2-3/TWODANT/DIF3D-VARIANT code system shows errors from -102.5 pcm to +168.6 pcm. Underestimation of criticality is due to the usage of 0-D control rod cross-section and hence it will be improved by adopting 1-D control rod cross-section. The overestimation of criticality is mainly due to the lower reflector homogenization, which induces +120 pcm error.

In the sodium void reactivity worth, the MC^2-3/TWODANT/DIF3D-VARIANT code system shows about 10 % underestimation in both of inner core and outer core regions. The underestimation of sodium void reactivity worth at inner core region is originated from the homogenization of sodium plenum and gas plenum. The underestimation of sodium void reactivity worth at outer core region may be originated from the homogenization of reflector and B_4C assemblies which induces underestimation of radial neutron leakage. This work is planned as a future study.

In control rod worth, the MC^2-3/TWODANT/DIF3D-VARIANT code system shows 9.8 % error in maximum when 0-D control rod cross-section is employed. The modeling error of control rod is reduced to 3.5 % when 1-D control rod cross-section is adopted.

In the radial expansion reactivity worth and Doppler reactivity worth, the MC^2-3/TWODANT/DIF3D-VARIANT code system shows good agreement with the heterogeneous MCNP results.

ACKNOWLEDGMENTS

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korea government (MSIP). (No. NRF-2013M2A8A2078239)

REFERENCES

1. H. K. Joo, et.al., "Status of the fast reactor technology development in Korea," The 48th TWG-FR Meeting, Obninsk, May 25–29 (2015).

2. OECD/NEA, "Benchmark for Neutronic Analysis of Sodium-cooled Fast Reactor Cores with Various Fuel Types and Core Sizes", NEA/NSC/R(2015)9, OECD/NEA(2016).

3. J. Y. Lim, S. R. Choi, and S. J. Kim, "PGSFR Core Design and Performance Characteristics," *Trans. Am. Nucl. Soc.*, **114**, 700 (2016).

4. C. H. Lee and W. S. Yang, "MC^2-3:Multigroup Cross Section Generation Code for Fast Reactor Analysis," ANL/NE-11-41 Rev.2, ANL (2013).

5. M. A. Smith, et. al., "DIF3D-VARIANT 11.0: A Decade of Updates," ANL/NE-14/1, ANL(2014).

6. D. B. Pelowitz, et.al., "MCNP6^{TM} USER'S MANUAL," LA-CP-13-00634, LANL (2013).

7. M. B. Chadwick, et. al., "ENDF/B-VII.0: Next Generation Evaluated Nuclear Data Library for Nuclear Science and Technology," *Nuclear Data Sheets*, **107**, 2931 (2006), doi: 10.1016/j.nds.2006.11.001.

8. J. Lim and D. Hartanto, "Shielding Analysis for the Lower Reflector Block of a PGSFR Fuel Assembly" Transactions of the Korean Nuclear Society Autumn Meeting 2016, Gyeongju, Korea, October 27-28, 2016.

9. J. H. Won, et.al., "Investigation of the homogenization effect in sodium void reactivity in PGSFR," International Conference on Fast Reactors and Related Fuel Cycle:Next Generation Nuclear Systems for Sustainable Development(FR17), Yekaterinburg, Russian Federation, June 26-29 (2017). (will be presented)

10. M. J. Lee, "Optimization of Non-Fuel Assembly Design", SFR-11-DR-486-039, KAERI (2016).