

Effect of Detailed Geometric Modeling on Activation and Dose Assessment of a Transportable Micro Reactor

Hyuk Han*, Beom Kyu Kim, Suk-Hoon Kim, Jae-Seon Jo
FNC TECHNOLOGY CO.

*Corresponding author: hyuk1357@fnctech.com

***Keywords :** *Transportable Micro Reactor, Molten Salt Reactor, Activation, MCNP, ORIGEN-S*

1. Introduction

Transportable Micro Reactors (TMRs) have been investigated as a flexible power generation option for meeting regional and remote electricity demands. In particular, TMR concepts based on Molten Salt Reactor (MSR) technology offer advantages in terms of compactness and mobility.[1][2] During reactor operation, internal structural components are continuously exposed to neutron irradiation, resulting in material activation. Therefore, the residual radiation field from activated structures must be evaluated to ensure radiological safety during transport.[3]

In a previous study, the activation characteristics and post-shutdown dose rates of a TMR were evaluated using an MCNP–ORIGEN-S coupled methodology.[4] As the TMR design was further developed, the computational model was refined to more detailed representation of the updated geometry and individual internal structural components. Accordingly, the number of SS316L structural cells considered in the activation analysis was increased from 64 in the previous model to 89 in the present model. The same calculation methodology was used to evaluate the effect of the refined geometric representation on the radiological characteristics of the reactor internal structure.

The fuel salt is assumed to be completely drained and stored separately after shutdown. Therefore, the analysis focuses on gamma radiation from activated SS316L-based structures. Neutron fluxes calculated by MCNP are used in ORIGEN-S to determine activation isotopes and gamma-ray source terms, which are subsequently employed in MCNP dose calculations. The objective of this study is to evaluate the post-shutdown radiological characteristics of the refined TMR model and assess its implications for transport safety.

2. Methodology and computational analysis

The radiological assessment is performed using a coupled MCNP–SCALE calculation framework. MCNP is used to determine the neutron flux in the reactor core and surrounding structural components with KCODE calculation. The calculated neutron flux is then applied to SCALE-COUPLE to generate problem dependent cross sections for the ORIGEN activation analysis. ORIGEN calculates the time dependent isotope and gamma-ray source terms of the activated structures. These gamma-ray source terms are used as fixed sources

in a secondary MCNP calculation to evaluate shutdown dose rates at selected surfaces and locations.

2.1 Neutron flux characterization

The MCNP code, developed by Los Alamos National Laboratory, was used to calculate the neutron flux during reactor operation.[5] A refined three-dimensional TMR model was developed to provide a more detailed representation of the reactor core, internal structures, and surrounding components. Compared with the previous model, the refined geometry more explicitly represents individual structural components while retaining the cell-averaged flux approach for activation calculations. Cell-averaged neutron fluxes were calculated for each SS316L structural component to maintain computational efficiency and adequate statistical convergence. The resulting neutron flux values were subsequently used as input for the ORIGEN-S activation analysis.

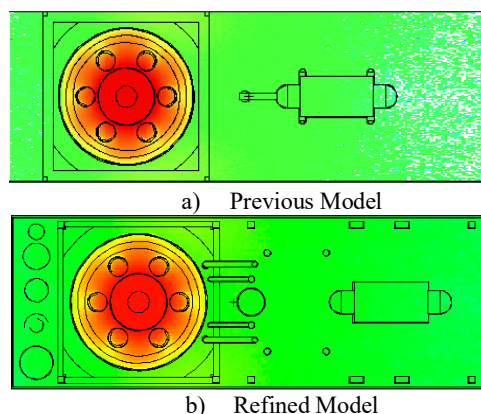


Fig. 1. Neutron flux distributions of the previous (a) and refined (b) TMR models during reactor operation at $Z = 0$

Figure 1 presents the refined TMR computational model and the corresponding neutron flux distribution during reactor operation. The neutron mesh flux distribution was calculated with a spatial resolution of 1 cm to visualize the neutron field throughout the reactor system. For the activation analysis, cell-averaged neutron fluxes were obtained using the F4 tally.

The TMR was assumed to operate at a thermal power of 1.5 MWth for 20 years. As part of the geometric refinement, the number of SS316L structural cells considered for activation analysis was increased from 64 in the previous model to 89 in the refined model. This expansion provides a more detailed representation of the

individual internal structures while maintaining the cell-averaged flux methodology. The neutron flux calculated for each of the 89 structural cells was used as input for the activation calculations.

Table I. Calculated neutron flux for each cell

1.5 MWth, 20 Year, 64 Cells		1.5 MWth, 20 Year, 89 Cells	
Cell Number	Neutron Flux (#/cm2)	Cell Number	Neutron Flux (#/cm2)
84	2.89E+09	84	2.35E+09
94	2.88E+09	94	2.43E+09
47	4.52E+09	47	5.93E+09
77	4.72E+09	77	6.77E+09
...	...	...	...
849	2.49E+08	110	2.01E+13
852	2.60E+08	23	3.49E+13
855	2.69E+08	63	1.63E+13
858	2.12E+08	64	1.85E+14

Table I compares the cell-averaged neutron fluxes used for the ORIGEN-S activation analysis in the previous model and the refined model. The refinement of the reactor geometry increased the number of SS316L cells considered in the activation analysis from 64 to 89, reflecting the addition and removal of structural components in accordance with the updated configuration. In particular, SS316L components located within the core, which had been excluded from the previous analysis under the assumption that they were sufficiently shielded by the reflector materials, were newly included in the refined model. As a result of these geometric modifications, changes in neutron flux were also observed for cells retained from the previous model. The highest neutron fluxes were calculated in the SS316L canning and piping components located within the reactor core.

2.2 Activation analysis and source term derivation

The activation analysis was performed using the SCALE COUPLE and ORIGEN modules following the methodology adopted in the previous study.[4,6] Cell-averaged neutron fluxes obtained from the refined MCNP model were used to determine the activation of the SS316L structural components. A representative 238-group neutron spectrum from the SS316L structures near the inner side of the reflector was used in COUPLE to generate problem-dependent cross sections for the ORIGEN calculations.

In the refined model, the activation calculation was extended to the 89 SS316L structural cells identified in the updated reactor geometry. For each cell, the corresponding neutron flux and structural mass were used to calculate the time-dependent radionuclide inventory and gamma-ray source term. The SS316L

composition used in the calculation is summarized in Table II.

Table II. Mass composition of SS316L

Nuclide	Composition mass ratio
C	0.08%
Mn	2.00%
P	0.05%
S	0.03%
Si	1.00%
Cr	17.00%
Ni	12.00%
Mo	2.50%
Fe	65.35%

Based on the SS316L composition summarized in Table II, activation calculations were performed for each of the 89 structural cells using the corresponding cell mass and neutron flux. The mass of each cell was determined from its geometric volume, while the cell-averaged neutron flux was obtained from the refined MCNP model. ORIGEN-S was then used to calculate the time-dependent radionuclide inventory and gamma-ray source term over the specified cooling periods.

The resulting gamma-ray spectrum and source strength were determined individually for each structural cell. These cell-specific source terms were subsequently incorporated into the MCNP transport model, preserving the spatial distribution of activated structures in the refined geometry for the post-shutdown dose assessment.

2.3 Surface gamma dose calculation with MCNP

The gamma-ray source terms calculated by ORIGEN-S were used as fixed sources in a subsequent MCNP transport calculation to evaluate post-shutdown dose rates. In the refined model, the gamma-ray spectrum and source strength calculated for each of the 89 activated SS316L cells were assigned to their corresponding locations in the MCNP geometry, thereby preserving the spatial distribution of the activated structures.

The dose assessment focused on the area-averaged dose rates at the lateral surfaces of the outer transport container. The calculated gamma fluxes were converted to dose rates using the ICRP-74 flux-to-dose conversion factors. The source strengths obtained from the ORIGEN-S activation calculations were applied to determine the shutdown dose rates.

3. Evaluation of Induced Radioactivity and Surface Dose Rates

The surface dose rates were evaluated using the computational methodology described in Section 2. Considering the mobility and deployment requirements of the TMR, a cooling period of 1 day after reactor shutdown was selected as the reference condition for the dose assessment.

Table III. Gamma Dose of 1.5MWth, 20 year condition

a) Previous Model

Surface	Dose(mSv/h)
	1 Day
1	2.59E+00
2	2.61E+00
3	3.10E-01
4	3.39E+00

b) Refined Model

Surface	Dose(mSv/h)
	1 Day
1	1.05E+01
2	9.35E+00
3	5.36E-01
4	2.42E+00

Table III compares the calculated surface dose rates for the previous and refined models under the operating condition of 1.5 MWth for 20 years. The refined model shows a substantial change in the predicted surface dose distribution. The dose rates at Surfaces 1 and 2 increased from 2.59 and 2.61 mSv/h to 10.5 and 9.35 mSv/h, respectively, while the dose rate at Surface 3 increased from 0.310 to 0.536 mSv/h. In contrast, the dose rate at Surface 4 decreased from 3.39 to 2.42 mSv/h. Compared with the transport dose criterion of 10 mSv/h[7,8], the maximum dose rate of 10.5 mSv/h at Surface 1 slightly exceeded the criterion, whereas the dose rates at the other three surfaces remained below the limit.

The increase in surface dose rates is primarily attributed to the SS316L canning and piping components located within the reactor core, which were excluded from the previous activation analysis based on the assumption that they were sufficiently shielded by the reflector. In the refined model, these components were explicitly included in the activation calculation and exhibited relatively high neutron fluxes, resulting in significant gamma-ray source terms after reactor shutdown.

Consequently, the inclusion of these previously neglected activated structures substantially changed both the magnitude and spatial distribution of the calculated dose rates. The decrease observed at Surface 4 indicates that the geometric refinement also affects gamma-ray transport and attenuation, in addition to changes in the activation source distribution. Since the maximum dose rate after a 1-day cooling period slightly exceeded the 10 mSv/h criterion, a longer cooling period may be required before transport to ensure compliance with the dose criterion.

4. Conclusion

This study evaluated the radiological safety of a TMR by assessing the shutdown surface dose rates resulting from the activation of internal SS316L structures. Using an MCNP-ORIGEN-S coupled framework, the activation source terms and resulting radiation fields

were evaluated based on a refined reactor model. For a conservative assessment, the neutron flux calculated under fresh-fuel conditions was assumed to remain constant throughout the 20-year operating period.

The refined model resulted in higher surface dose rates than the previous model, primarily due to the inclusion of SS316L canning and piping components located within the reactor core that had previously been excluded from the activation analysis. These components were exposed to relatively high neutron fluxes and consequently contributed significantly to the post-shutdown gamma-ray source term. After a 1-day cooling period, the maximum surface dose rate was calculated to be 10.5 mSv/h at Surface 1, slightly exceeding the transport criterion of 10 mSv/h. Therefore, the results indicate that a cooling period longer than 1 day may be required to satisfy the dose criterion prior to transport.

REFERENCES

- [1] Seo, J. U., Yoon, S. J., & Park, T. K., 2022. "A study on the depletion behavior of molten salt reactor with OpenMC code", *Transactions of the Korean Nuclear Society*, Changwon, Oct. 20-21.
- [2] S. J. Yoon, J. U. Seo, S. K. Zee, T. K. Park & S. J. Kim, "Multi-layer Shielding Design for Small Sized Molten Salt Fast Reactor", *International Conference of Asian Political Parties(ICAPP)*, Gyeongju, Apr. 23-27.
- [3] K. W. Chung, S. J. Yoon, T. K. Park & J. S. Cho, "Shielding Considerations for a Transportable Microreactor", *International Conference of Asian Political Parties(ICAPP)*, Gyeongju, Apr. 23-27.
- [4] H. Han, B. K. Kim, S. H. Kim & J. S. Jo "Radiological Safety Evaluation of Internal Structure Activation in Transportable Micro Reactors" *Transactions of the Korean Nuclear Society Spring Meeting* Jeju, 2026
- [5] Kulesza, Joel A., Adams, Terry R., et al. MCNP® Code Version 6.3.0 Theory & User Manual. United States: N. p., 2022. Web. doi:10.2172/1889957.
- [6] W. A. Wieselquist, R. A. Lefebvre, Eds., SCALE 6.3.2 User Manual, ORNL/TM-2024/3386, UT-Battelle, Oak Ridge National Laboratory (February 2024).
- [7] Enforcement Decree of the Nuclear Safety Act, Republic of Korea, Presidential Decree No. 2017 27972.
- [8] Nuclear Safety and Security Commission, NSSC Notice No. 2019-7 (2019).

Acknowledgment

This work was supported by Korea Research Institute for defense Technology planning and advancement(KRIT) grant funded by the Korea government(DAPA(Defense Acquisition Program Administration)) (21-107-F00-018(KRIT-CT-22-017), Next Generation Multi-Purpose High Power Generation Technology(Liquid Fuel Heat Generator Transportation and Safety Assessment Technology), 2022)