Analysis of Neutron and Photon Dose Rate Reduction in i-SMR with Enhanced Radiation Shielding Structures Using ADVANTG and MCNP6

Hyun Chul Roh, Minki Kim, Chang Ho Shin and Ser Gi Hong*
Departments of Nuclear Engineering, Hanyang University, 222, Wangsimni-ro, Seongdong-gu, Seoul, 04761, Republic of Korea
Corresponding author: hongsergi@hanyang.ac.kr

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1. Introduction

Small Modular Reactor (SMR) is an integrated reactor designed based on reactor system simplification and modularization, representing a new type of nuclear power plant with enhanced safety and flexibility compared to the conventional large-scale nuclear power plants.

The innovative SMR (i-SMR), currently developed in Korea [1], is a novel type of SMR composed of four reactor modules based on light water reactor technologies. The i-SMR features different system designs and equipment arrangements compared to existing commercial reactors, providing a smaller electrical output of 170 MWe per module in contrast to the APR-1400 and APR-1000. Due to its novel design characteristics, licensing, maintenance, and construction technologies differ from those of conventional commercial reactors; consequently, the behavior of radiation sources and radioactive nuclides is expected to differ as well.

Therefore, considering the unique design features of the i-SMR, it is important to calculate dose rate in order to confirm the reduction of the occupational radiation exposure.

In this study, one module of the i-SMR was modeled in actual size using the MCNP6. MCNP is a general-purpose, continuous-energy, generalized-geometry, time-dependent, Monte Carlo radiation-transport code designed to track many particle types over broad ranges of energies [2]. Neutron and photon dose rates were calculated at ground level (0 m) for three different shielding designs and the reduction in dose rates resulting from additional shielding structures was analyzed. To calculate dose rates, neutron and photon dose conversion coefficients from ICRP-116 [3] were applied.

In this work, the Automated Variance Reduction Parameter Generation (ADVANTG) code [4] was employed to generate weight window file to reduce variances in the MCNP6 calculations.

2. Methods and Results

2.1 Calculation Methodology

The MCNP6 code was used to calculate the dose rate at ground level (0 m). For dose rate calculation, ENDF/B-VII.1 continuous energy cross-section library was used. To precisely calculate the dose rate at ground level (0 m), the calculation region was divided into meshes in 1 m × 1 m × 50 cm size, resulting in a total of 17 × 18 × 2 tally regions. The effective dose conversion coefficients in ICRP-116 published in 2010 were used to the FMESH tally for dose calculations.

The isotropic source distribution in the RV (Reactor Vessel) region was used for the transport calculations to reduce the computing cost. Actually, KEPCO E&C provided the flux distributions from the core to the CV (Containment Vessel) region for three energy bins for 100% power operation i-SMR. So, we interpolated the fluxes for three energy bins in the RV region from these flux distributions.

A variance reduction technique known as the weight window method was applied before performing tally calculations with MCNP6 to effectively reduce variance in the tally results. The weight window file was generated using Automated Variance Reduction Parameter Generation (ADVANTG) code. Specifically, Forward-Weighted Consistent Adjoint Importance Sampling (FW-CADIS) method was applied to generate weight window covering the region from the radiation source term to the mesh tally calculation area.

2.2 MCNP6 modeling and source term definition

To calculate the dose rate, one module of the i-SMR was modeled using the MCNP6 code. The modeling was performed based on actual design data of the i-SMR, including major structures such as the Reactor Vessel (RV), Steel Containment Vessel (CV), Steam Generator, and other internal steel structures. Specifically, a 50 cm-thick concrete shielding wall surrounding the CV was modeled

To evaluate the changes in dose rates at ground level (0 m) according to different shielding structures, the following structures were additionally modeled: 1) a 2 m-thick concrete shielding wall located at the intermediate region between upper and lower parts of the CV , 2) a 10 cm-thick concrete shielding wall at the worker platform area, and 3) a 5 cm-thick SS304 plate covering the top of the 50 cm concrete shielding wall at the upper CV region.

Fig. 1 and Fig. 2 show the dimensions of the shielding structures for one module of the i-SMR modeled using MCNP6 based on actual measurements. Fig. 3 shows the considered calculation cases for evaluating dose rates at ground level (0 m) as the additional shielding structures. The Case 1 represents only the presence of a 2 m-thick concrete shielding wall in the intermediate region between upper and lower CV regions, the Case 2 adds a 10 cm-thick concrete shielding wall at the worker platform area to Case 1, and the Case 3 adds a 5 cm-thick SS304 steel plate covering the top of CV to Case 2.

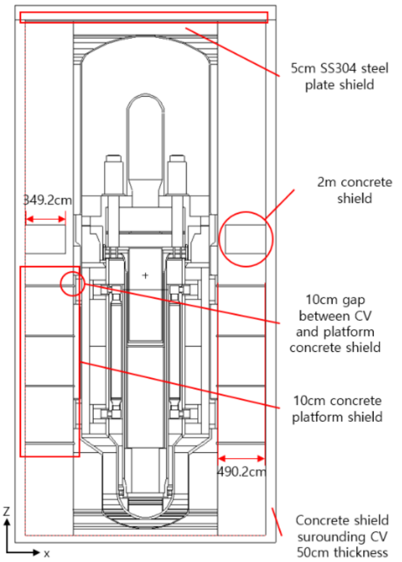

Fig. 1 i-SMR radiation shielding structure dimensions in xz direction

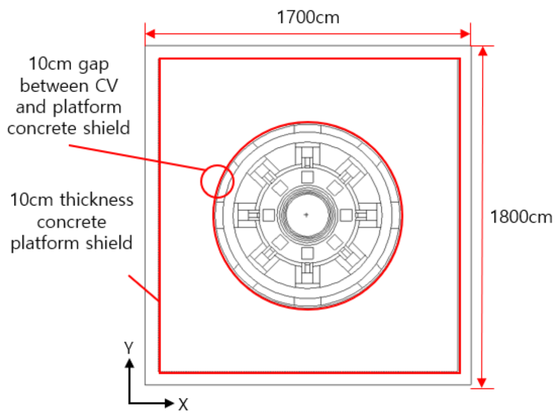

Fig. 2 i-SMR radiation shielding structure dimensions in xy direction

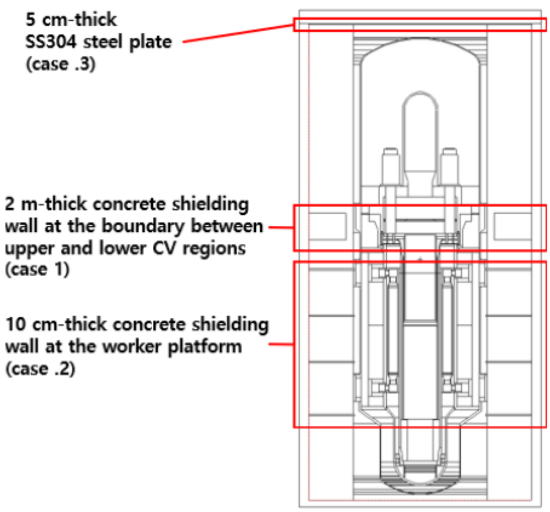

Fig. 3 Additional radiation shielding structures for each calculation case

Fixed source was defined to perform transport calculations for neutron and photon using the MCNP6 code. For the energy spectrum of the source, neutron fluxes in the RV region for three energy bins (0~0.6250 eV, 0.6250eV~1.00 MeV, and 1.0MeV20.00 MeV) are used as the sources, which were interpolated the flux distribution provided KEPCO E&C. The source term was defined as a cylindrical shape at the RV outer wall with the same height as the active core, assuming uniform and isotropic emission within this cylindrical region.

Fig. 4 shows the neutron flux distributions for each energy bin, which were provided by KEPCO E&C.

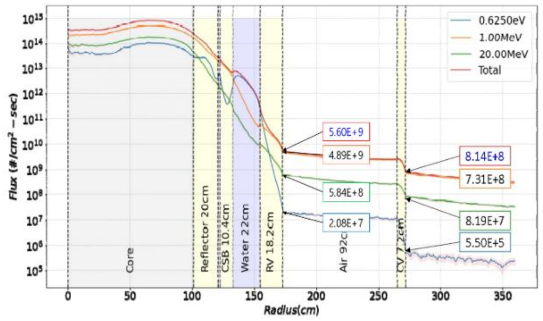

Fig.4 Neutron flux in three energy (0.6250 eV, 1.00 MeV, and 20.00 MeV), bin at RV and CV outer wall surface for 100% power operation i-SMR

2.3 Variance reduction using ADVANTG

The weight window method, one of the variance reduction techniques, was applied to effectively reduce variance in the tally region during transport calculations using the MCNP6 code. The ADVANTG code calculates the adjoint flux based on the MCNP input file to generate a weight window file. The generated weight window file is applied during execution of the MCNP input file, effectively reducing variance by ensuring that the particles are efficiently transported to the tally region.

To generate the weight window, the FW-CADIS option in the ADVANTG code was used. The entire i-SMR geometrical model was divided into 10 segments along the x-axis, 10 segments along the y-axis, and 400 segments along the z-axis to generate the weight window.

2.4 Calculation result

Dose rate calculations at ground level (0 m) were performed for three cases described in Section 2.2. The effective dose conversion coefficients from ICRP-116 were applied using the "DE" and "DF" cards in the MCNP6 code to evaluate the neutron and photon dose. Dose rates were calculated by multiplying the tally result by source intensity and area.

Fig. 5, Fig. 6 and Fig. 7 show the neutron dose calculated for each mesh region, indicating the regions with the highest and lowest neutron dose rates. In each case, the region with the highest neutron dose rate is located at the upper region of the CV, while the lowest neutron dose rate occurs at the corner of 50cm concrete shield surrounding CV region.

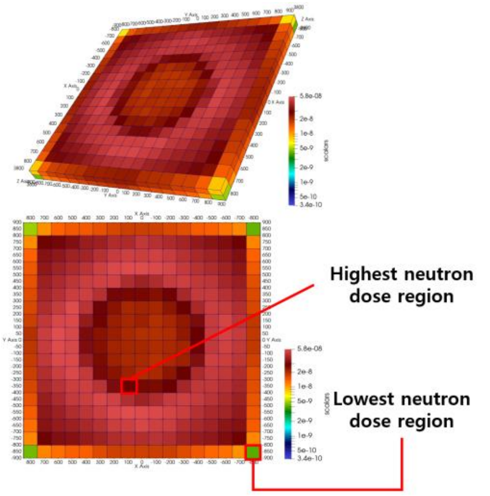

Fig. 6 Neutron dose in 0m level mesh tally region with highest and lowest dose for Case 1

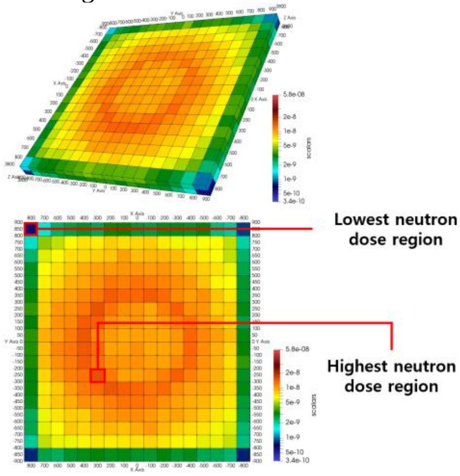

Fig. 7 Neutron dose in 0m level mesh tally region with highest and lowest dose for Case 2

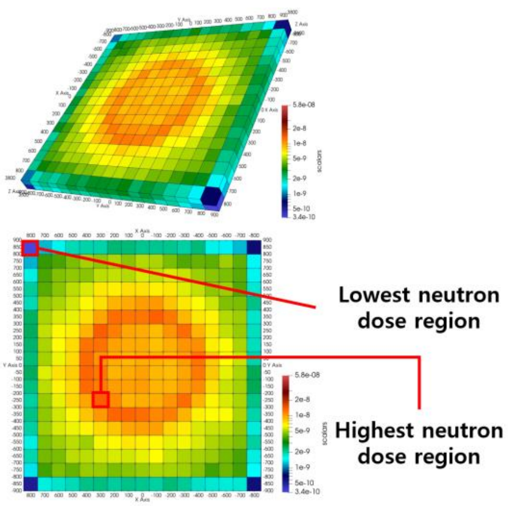

Fig. 8 Neutron dose in 0m level mesh tally region with highest and lowest dose for Case 3

Table I shows the maximum, the minimum and the average neutron dose rates including their statistical errors (%) at the edge of the tally region. As shown in Table I, the statistical errors for the tally values are acceptably low and the dose rates are quite low.

Table. I Maximum, minimum and edge average neutron dose rates and statistical error for each case

	Neutron dose rate		
	Max	Min	Edge average
$\mu Sv/hr$ (case 1)	4.63E+05	3.15E+04	1.28E+05
Error (case 1)	1.73%	6.14%	-
$\mu Sv/hr$ (case 2)	1.46E+05	9.84E+03	3.15E+04
Error (case 2)	3.15%	1.18%	-
$\mu Sv/hr$ (case 3)	1.09E+05	2.72E+03	1.58E+04
Error (case 3)	3.82%	1.64%	-

Table II shows the percentage reduction in neutron dose rates for the Cases 2 and 3 compared to the Case 1. In the Case 2 where, 10 cm-thick concrete shielding wall was added to the worker platform in Case 1, the maximum, minimum, and edge average neutron dose rate decreased by 68.4, 68.9, and 75.3%, respectively, compared to the Case 1. In the Case 3, where a 5 cm steel plate was added to Case 2 on top of the CV, the maximum, minimum, and edge average neutron dose rates were decreased by 76.6 91.4, and 87.6%, respectively, compared to the Case 1.

Table. II The difference in neutron dose rates between Case 2 and Case 3 compared to Case 1

	Neutron dose rate vs. Case 1		
	max	min	Edge average
Case 2	-68.39%	-68.91%	-75.29%
Case 3	-76.61%	-91.42%	-87.57%

Fig. 8, Fig. 9, and Fig. 10 show the photon dose rates calculated for each mesh region and the regions with the highest and lowest photon dose rates. In each case, the regions with the highest and lowest photon dose rates are like those observed in the neutron dose rate calculations, with the highest photon dose rate occurring at the upper region of the CV and the lowest photon dose rate at the corner of 50cm concrete shield surrounding CV region.

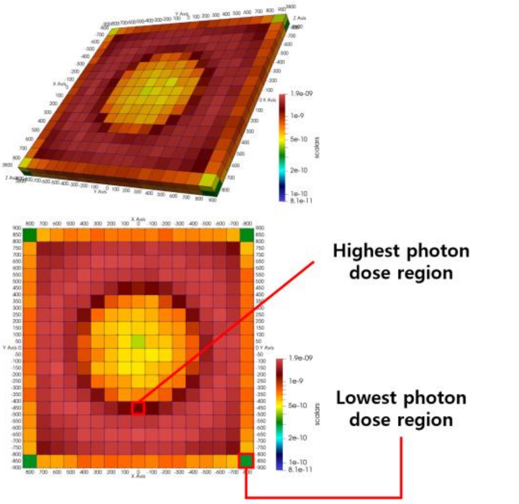

Fig. 9 Photon dose in 0m level mesh tally region with the highest lowest dose for Case 1

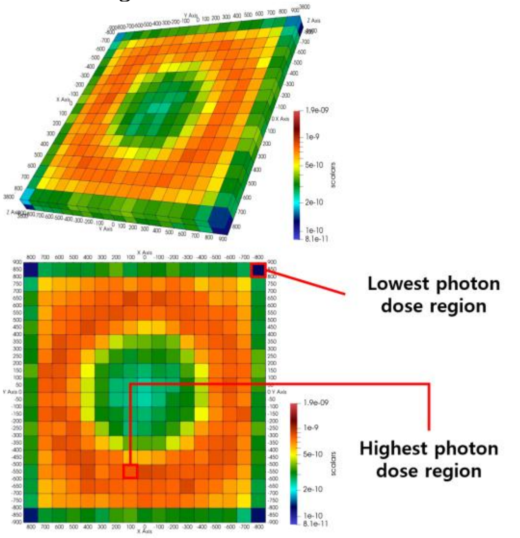

Fig. 10 Photon dose in 0m level mesh tally region with the highest lowest dose for Case 2

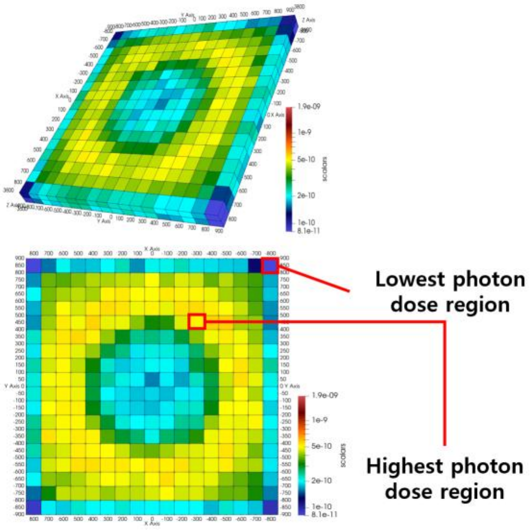

Fig. 11 Photon dose in 0m level mesh tally region with the highest lowest dose for Case 3

Table III shows the maximum and minimum dose rates and the average photon dose rate at the edge of the tally region, and the corresponding statistical error values for each calculation. Similar to the neutron dose rate results, it can be observed that as additional shielding structures are introduced from Case 1 to Case 3, the photon dose rate decreases accordingly.

Table. III Maximum, minimum and edge average photon dose rates and statistical error for each case

	Photon dose rate		
	Max	Min	Edge average
$\mu Sv/hr$ (case 1)	1.52E+04	2.30E+03	6.01E+03
Error (case 1)	2.44%	6.92%	-
$\mu Sv/hr$ (case 2)	5.55E+03	6.40E+02	1.91E+03
Error (case 2)	4.17%	1.27%	-
$\mu Sv/hr$ (case 3)	4.46E+03	5.83E+02	1.55E+03
Error (case 3)	3.82%	1.34%	-

Table IV shows the percentage reduction in photon dose rates for the Cases 2 and 3 compared to the Case 1. In the Case 2, where a 10 cm-thick concrete shielding wall was added to the worker platform in the Case 1, the maximum, minimum, and edge average photon dose rates decreased by 63.5, 72.3, and 68.2%, respectively, compared to the Case 1. In the Case 3, where a 5 cm steel plate was added to the Case 2 on top of the CV, the maximum, minimum, and edge average photon dose rates decreased by 70.5, 74.5, and 74.3%, respectively, compared to the Case 1.

Table. IV The difference in photon dose rates between Case 2 and Case 3 compared to Case 1

	Photon dose rate vs. Case 1		
	max	min	Edge average
Case 2	63.47%	72.23%	68.17%
Case 3	70.47%	74.49%	74.31%

3. Conclusions

In this study, neutron and photon dose rates at ground level (0 m) for the i-SMR were calculated using the MCNP6 code. To achieve more detailed calculations, the FMESH tally feature of MCNP6 was utilized, and effective dose conversion coefficients from ICRP-116 were applied to the FMESH tally to calculate dose rates.

The results of the analysis confirmed significant reductions both in neutron and photon dose rates as additional shielding structures were introduced. From the comparison with the Cases 1 and 2, it was observed that adding a 10 cm-thick concrete shielding wall to the worker platform resulted in a significant dose rate reduction of approximately 70%. On the other hand, the addition of a 5 cm-thick steel plate on top of the CV did not lead to a substantial reduction compared to Case 2.

In the case of Case 3, where the shielding structure was applied the most, the neutron and photon dose rates were decreased significantly by more than 70% compared to Case 1.

In the future work, we will calculate the dose rates in the actual operator available by applying the source term generated from the structure of the i-SMR in the shutdown state.

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