Transactions of the Korean Nuclear Society Autumn Meeting
Gyeongju, Korea, October 25-27, 2017

Applicability Evaluation of Point-Kernel Method for Gamma-ray Dose Analysis to Radiological Environmental Impact Assessment

Hae Sun Jeong*, Jo Eun Lee, Eun Han Kim, Moon Hee Han, and Won Tae Hwang
Korea Atomic Energy Research Institute., 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea
**Corresponding author: haesunin@kaeri.re.kr*

1. Introduction

With different methodologies, a variety of radiation-transport codes have been developed and used for dose analysis, radiation shielding, or designs and maintenance of facilities in the radioactive fields using gamma-ray sources. The most widely used approaches to solve gamma-ray-transport problems are classified into two physical techniques: i) point-kernel and ii) Monte Carlo methods.

Point-kernel method has been used in various areas using gamma-ray sources; since its accuracy is comparatively validated as well as the calculation procedure is easy and simple. Monte Carlo method enables the geometry and the particle behavior to be treated without approximation, and derives pre-eminently accurate and realistic results for complex problems; therefore, are extensively used for various studies even though it requires a large amount of computing time to solve problems.

Particularly, most of dose analyses in radiological environmental impact assessments such as the external exposure or the decontamination have been normally performed using Monte Carlo codes with their various advantages. In these types of assessments, however, the geometry size including a radiological source and a building structure is so considerably huge that the amount of computational time is required to achieve an enough reliable result. The radiological sources in environments are normally a type of gamma-ray, and the dominant material of buildings is simply concrete structure. In light of these characteristics, this study aims at verifying whether the point-kernel method is appropriate to be used for the dose analysis in radiological environmental impact assessments with a huge-scale geometry and source.

In this study, the calculation sensitivities of the QAD-CGGP point-kernel code [1] are evaluated by comparing to the outcomes of MCNP6.1 Monte Carlo code [2] along with the sizes of the gamma-ray source and the detection points.

2. Materials and Methods

The QAD-CGGP code uses the point-kernel ray-tracing technique for gamma-ray transport calculation. The point-kernel represents the energy transfer by the uncollided flux along a line-of-sight path, and is combined with a proper buildup factor to consider all of the contributions of scattered gamma-rays. It is integrated over the volume (V) of the source (S) for each energy (E), and the dose rate is derived as follows,

$$D(\vec{r}) = DCF\int_V \frac{S(\vec{r}\,')}{4\pi|\vec{r}-\vec{r}\,'|^2}\exp(-\mu|\vec{r}-\vec{r}\,'|)B(\mu|\vec{r}-\vec{r}\,'|,E)dV \quad (1)$$

where, DCF is the flux-to-dose rate conversion factor, $\vec{r}$ is the detection point to be calculated, $\vec{r}\,'$ is the location of the source in volume V, μ is the total attenuation coefficient at energy E, and B is the buildup factor.

Figure 1 shows two kinds of conceptual models for calculations with huge-scale geometry and sources. CASE I and CASE II have the 1×1×1 and 3×3×3 types of building with the rooms of 432 cm and the concrete walls of 24 cm thickness, respectively. In these calculations, the gamma-ray source was assumed to be a thin cylindrical volume form and have energy of 0.662 MeV. The source radius R varies from 300 cm to 10000 cm in addition to the point source in CASE I, and from 1000 cm to 10000 cm in CASE II.

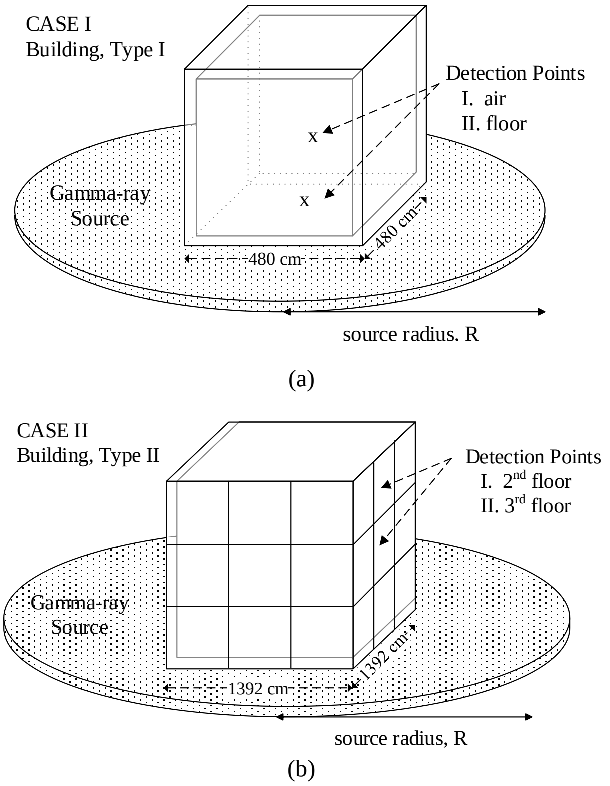

Fig. 1. Conceptual Models with the Huge-scale Geometry and Sources: (a) Case I and (b) Case II

In the CASE I, the calculation sensitivity for source volume was analyzed along with the depth of the floor using a point source and a thin cylindrical volume source with a radius of 300 cm. In addition to the detection on the floor of the room, for two types of calculation cases, the gamma-ray dose rates along with the source sizes were obtained in the air at the height of 85 cm from the floor that is the center of the body of a person who is 170 cm tall.

All series of calculations were performed using QAD-CGGP and MCNP6.1, and the flux-to-dose rate conversion factors were applied using ICRP Publication 74 [3]. The ANS 6.4.3 cross section data and the Geometric Progression (GP) buildup factor were employed in the QAD-CGGP calculations [4]. The ENDF/B-VII cross section library was used in the MCNP calculations.

3. Results and Discussion

The gamma-ray dose rates in the huge-scale modeling were calculated using the QAD-CGGP and MCNP codes. The relative errors are below 5% in all of the MCNP results.

The calculation sensitivity of QAD-CGGP against MCNP for the source types was analyzed for CASE I. **Figure 2** shows the dose rates for the point gamma-ray source along with the depth from the floor of the room. In the whole range of the floor depth, the results between two codes are almost similar, showing the maximum difference of 1.14 times only. In the volume source problem as shown in **Figure 3**, the differences of results derived from both codes are not very large (from 1.46 times to 1.88 times). However, the accuracy of QAD-CGGP decreased when the form of radiation source is the volume type even though their differences are below 2 times. This is because the mathematical error for the point source is reduced by the simple calculation form as **Equation (1)** is not a form of an integral function of volume V of the source.

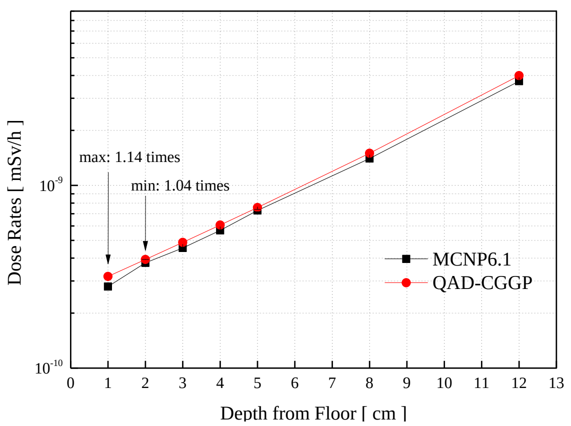

Fig. 2. Dose Rates along with the Depth from the Floor Using the Point Source (CASE I)

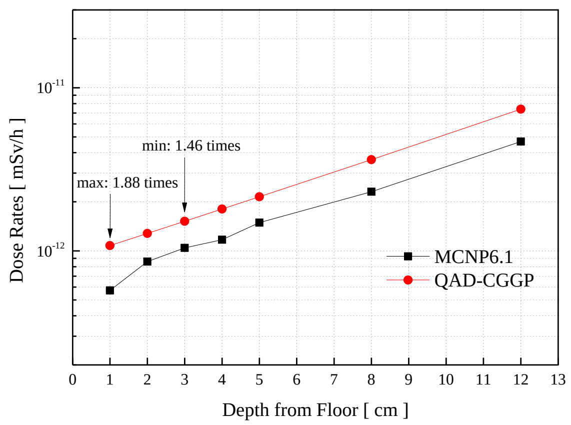

Fig. 3. Dose Rates along with the Depth from the Floor Using the Thin Cylindrical Source with a Radius of 300 cm (CASE I)

Figure 4-6 shows the dose rates detected in the air of the room in CASE I and the rooms of 2nd floor and 3rd floor in CASE II. All calculated values indicate higher and the differences between two codes also increase compared to those detected at the point under the floor. The dose rates are steadily increased and moved towards convergence as the source becomes greater in size.

In the condition of relatively simple geometry (CASE I), the differences of dose rates are approximately 2 times in the whole range of source sizes. However, the calculation sets of CASE II which has more complex and larger geometry show the differences up to 7.47 times and 4.24 times in the rooms of 2nd floor and 3rd floor, respectively.

Particularly, the outcomes on the 3rd floor derived by QAD-CGGP underestimated in the range with the relatively smaller source sizes against building sizes. As the source size is smaller, the ratio of the radiations penetrating concrete materials to those passing through air only is higher among total amount of the gamma-ray sources. In addition, the detection of the radiation dose on the 3rd floor requires more variable material changes, i.e., concrete → air → concrete → air → concrete → air, and it causes more computational error due to the buildup factor of the materials used in the QAD-CGGP code.

Therefore, the evaluation using the QAD-CGGP code in radiological environmental impact assessment with huge-scale model can be useful to obtain the rough outcomes, but it is needed to carefully use for accurate results.

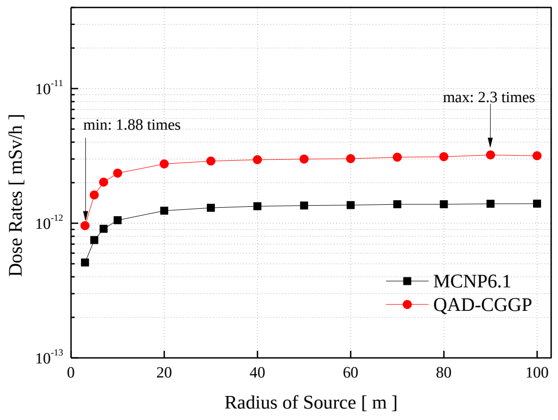

Fig. 4. Dose Rates along with the Source Size (CASE I)

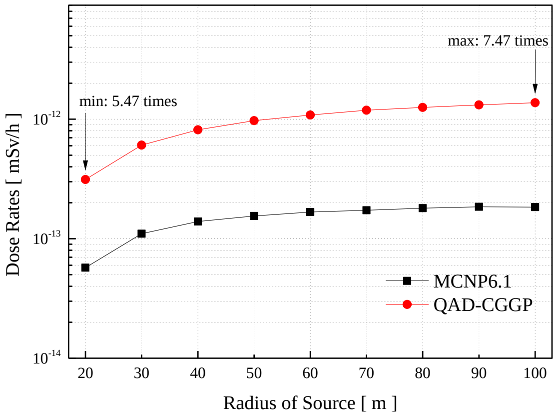

Fig. 5. Dose Rates on the 2nd Floor along with the Source Size (CASE II)

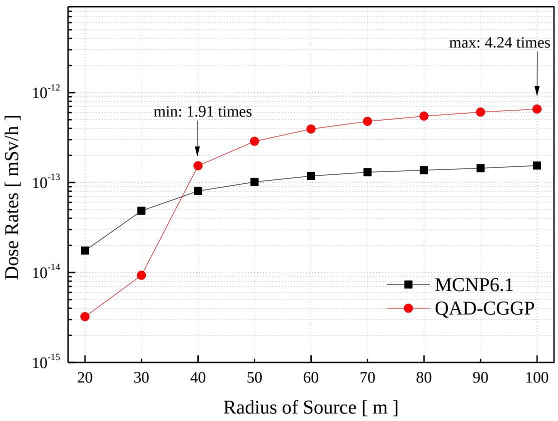

Fig. 6. Dose Rates on the 3rd Floor along with the Source Size (CASE II)

4. Conclusion

To evaluate the applicability using point-kernel method for gamma-ray dose analysis to radiological environmental impact assessments, the calculation sensitivities of QAD-CGGP against MCNP were analyzed in the condition with the huge-scale model.

In the case using the point source, the accuracy of QAD-CGGP increases compared to that using the volume source. The detection at the point under the floor shows differences smaller than results in the air. In addition, the outcomes of QAD-CGGP in the relatively simple geometry are shown to be similar to those of MCNP.

As a result, the point-kernel method in radiological environmental impact assessment with huge-scale model can be useful to obtain the results roughly, but it should be carefully used for accurate results.

REFERENCES

[1] ORNL, QAD-CGGP-A: Point Kernel Code System for Neutron and Gamma-Ray Shielding Calculation Using the GP Buildup Factor, CCC-645, Oak Ridge National Laboratory, 1995.

[2] T. E. Booth et al, MCNP – A General Monte Carlo N-Particle Transport Code," Version 5, LA-CP-03-0245, Los Alamos National Laboratory, 2003.

[3] ICRP, Conversion Coefficients for Use in Radiological Protection against External Radiation, International Commission on Radiological Protection (ICRP) Publication 74, Pergamon Press, 1996.

[4] D. K. Trubey, New Gamma-Ray Buildup Factor Data for Point Kernel Calculations: ANS-6.4.3 Standard Reference Data. Oak Ridge National Laboratory, Radiation Shielding Information Center, ORNL/RSIC-49, 1988.