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## ARTICLE

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### Dose assessment analysis on neutron source terms of 70 MeV proton and UC<sub>2</sub> target reaction for RAON ISOL target system

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Isotope Separator Online (ISOL) facility at the Institute for Basic Science (IBS) in South Korea has been designed for producing rare isotopes. The purpose of this work was to analyze the effect of the strategy on neutron source term to evaluate effective dose for ISOL target room. Several neutron source terms were generated by varying the dimensional distribution, the number of angular divisions, and the number of energy divisions. For the dimensional distribution of neutron source, a cylindrical distribution with same shape as the target was recommended in front wall and a cylinder and a point distribution were recommended in side wall. Also, more than 32 angle divisions which had within 1.5 % discrepancy both in the front and side wall was recommended for dividing angles. In a case of dividing energies, a discrepancy within 2 % was shown in the source terms divided into more than 100 divisions. However, in case that discrete energy distribution is used, the dose was highly overestimated as the number of energy divisions decreases.

**Keywords:** Isotope Separator Online (ISOL); cyclotron; neutron source term; Monte Carlo; effective dose

#### 1. Introduction

Preliminary radiation shielding studies were carried out for RAON (it is named after Korean native word meaning joyful), accelerator under construction for Korea Rare Isotope Science Project, Isotope Separator Online (ISOL) target room [1]. ISOL target room is a facility for producing rare isotopes by collisions of a UC<sub>2</sub> target with high energy protons. The protons are accelerated up to 70 MeV by cyclotron with a current of 1 mA. Many harmful secondary radiations are generated by the reaction of the high-energy protons with the target. In a view of radiological safety, radiation dose assessment for the target room must be carried out. The regulatory body in South Korea has set an effective dose limit for radiation workers to 20 mSv per year. Assuming 2000 hours of work per year, the dose limit is converted to 10 µSv per hour. To design a structure, determining a thickness of the ordinary concrete wall was needed to keep the workers from irradiation exceeding the occupational dose limit. Secondary neutrons were considered as major concern in radiation shielding of ISOL target room because neutrons have higher penetrability and radiation weighting factor than electrons or photons.

Effective dose was calculated by MCNPX 2.7 code

on the outer surface of the concrete wall [2]. In Monte Carlo calculation, two-step calculation using neutron source terms is occasionally used for dose assessment avoiding time-consuming proton transport [3]. The accuracy of the dose estimation by the two-step method depends on how to make the neutron source terms. In this study, the appropriateness of dose assessment by the produced neutron source terms varying the angle and energy distributions of secondary neutrons was calculated. The validity of the neutron source terms was also evaluated on the RAON ISOL target room.

#### 2. Method and results

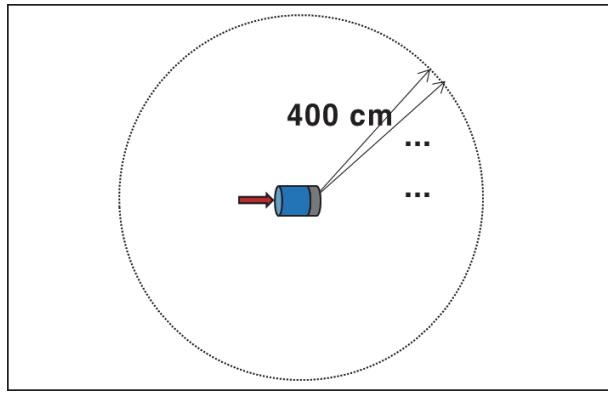
The ordinary concrete wall is in the front and side directions at 1.5 m from the target. The target geometry was simply described as bulk type of target and beam dump without any other structures. The UC<sub>2</sub> target is cylindrical shape of 3 cm radius and 3 cm long followed by 1 cm long graphite dump considering 3.77 cm range of 70 MeV proton in UC<sub>2</sub>. The calculation geometry is indicated in **Figure 1**. The effective dose was calculated on the outer wall surface of 100 x 100 cm<sup>2</sup> area. The dose conversion factor of antero-posterior direction remarked in ICRP Publication 116 was used [4]. The calculated doses according to neutron source terms were compared with the dose of one-step proton transport.

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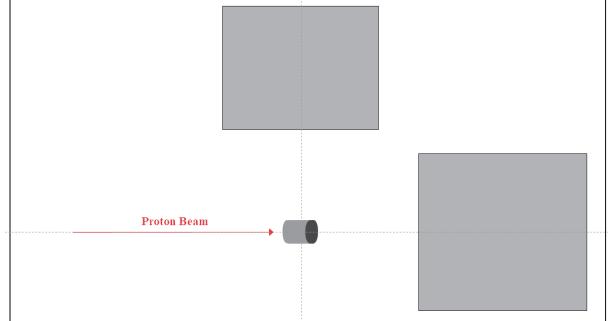
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## 2.1. Strategy on neutron source term generation

To score secondary neutrons from the reaction, a tally sphere with a radius of 400 cm, which is similar in size to the target room, was modeled around the point where the protons collide with the  $\text{UC}_2$  target. The produced neutrons were tallied on sphere surface divided 1000 energy bins and 1024 angular bins. The neutron source terms were generated by organizing the tallied data varying the number of the energy and angular divisions. There are two categories of strategies to generate neutron source terms. The first is dividing angular directions to 512, 256, 128, 64, 32, and 16 bins with fixed 500 energy bins. The second is dividing energy divisions to 500, 250, 200, 100, and 50 bins with 512 angular divisions. The energy distribution of each source term is defined in two ways: histogram and discrete value having the maximum value of each energy bins.



(a) Tally sphere to obtain neutron source term.



(b) Calculation geometry for dose calculation.

Figure 1. Schematic view of calculation geometry (For illustrative purpose, the target is expressed larger than the actual size).

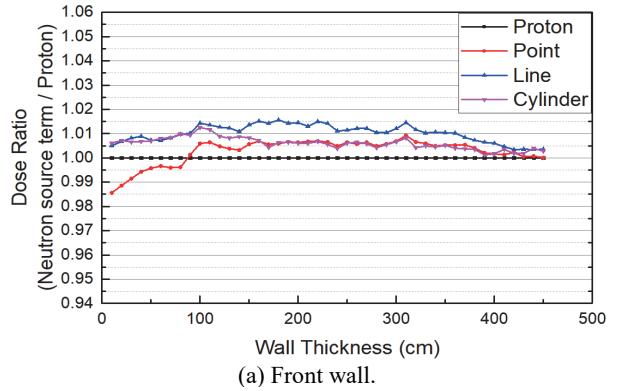
## 2.2. Source term in dimensional distribution

Prior to the analysis according to the number of angular divisions or energy divisions, dose changes were analyzed to the position of neutron source term in source definition. Since the target exists in case of the direct proton transport, the position where the neutrons are generated depends on the target shapes. There are three options to define source position and distribution: Option 1 is a point where the protons collide with the target; Option 2 is a line with length equal to the target

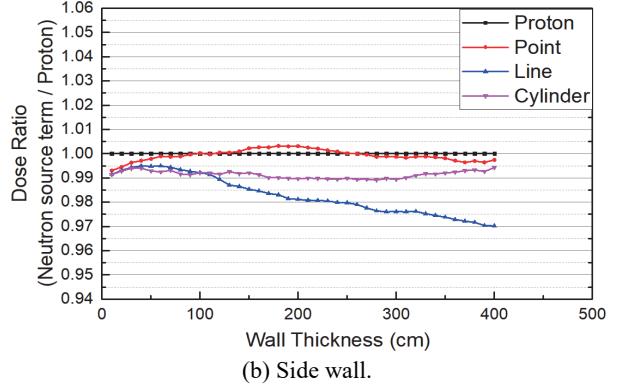
length; Option 3 is a cylinder of the same shape as the target. The calculated doses according to the neutron source term of point, line, and cylinder distribution were evaluated and compared. The source terms used here have 500 energy bins and 512 angle bins. The relative errors of MCNPX calculation were less than 1 % for dose calculation. Relative errors of some bins cannot be reduced in source term generation due to large number of angular and energy bins. However, the neutron generation probability was negligible (less than one millionth) in these bins and the dose was hardly affected by the errors.

Results of the dose evaluation by dimensional distribution options are shown in Figure 2. In the front wall, the calculated effective dose of neutron source term using three distributions had a discrepancy within 1.6 % compared with direct proton transport from 0 to 450 cm concrete thickness. The source term of cylinder distribution showed the least discrepancy among the three options.

Results of the side wall are shown in Figure 1 (b). The neutron source term of point distribution showed the least discrepancy. Since produced neutrons in the sideward direction have slight forward direction rather than vertical, the neutrons that do not enter the scoring region are increasing according to the wall thickness. As the result, the dose was gradually decreased in the case of using line distribution. The neutron source term of cylinder distribution showed a discrepancy within 1.1 %.



(a) Front wall.



(b) Side wall.

Figure 2. Dose ratio of neutron source term to direct proton transport according to source distribution.

### 2.3. Neutron source term analysis on the number of angular divisions

In this section, the dose estimation varying the number of angular divisions of neutron source terms were performed. The neutron source terms with 500 energy divisions and cylinder distribution on front wall and point distribution on side wall were produced.

The doses of neutron source terms by the number of direction divisions to direct proton transport is shown in **Figure 3**. As the number of angular divisions decreases, the calculated dose compared to the proton transport is evaluated to be lower in the front wall. For the neutron source term of 16 angular divisions, the calculated dose was estimated to be 6.2 % low and the other source terms showed the discrepancy within 1.5 %. In the side wall, as shown in Figure 2 (b), the neutron source term of 16 angular divisions had a discrepancy of up to 1.9 %, while the other source terms did not exceed 1.1 % discrepancy.

For conservative evaluation, the source terms having a maximum value of each energy bin instead of histogram distribution are used. As using maximum single value in each energy bin, the neutrons are generated with the same or higher energy than histogram distribution that makes the dose higher. The results are shown in **Figure 4**. In this case, in both front and side wall, the doses were higher than the dose using the source terms of histogram distribution.

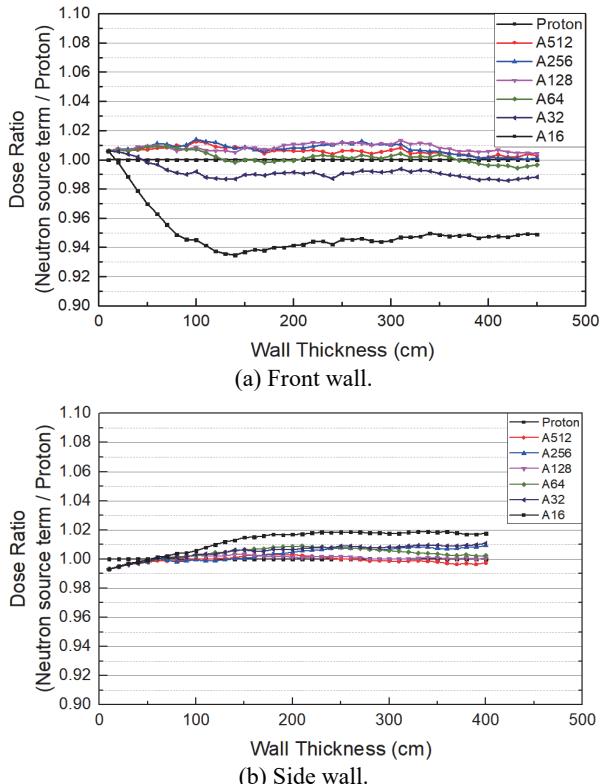


Figure 3. Dose ratio of neutron source term to direct proton transport according to the number of direction divisions using histogram energy distribution.

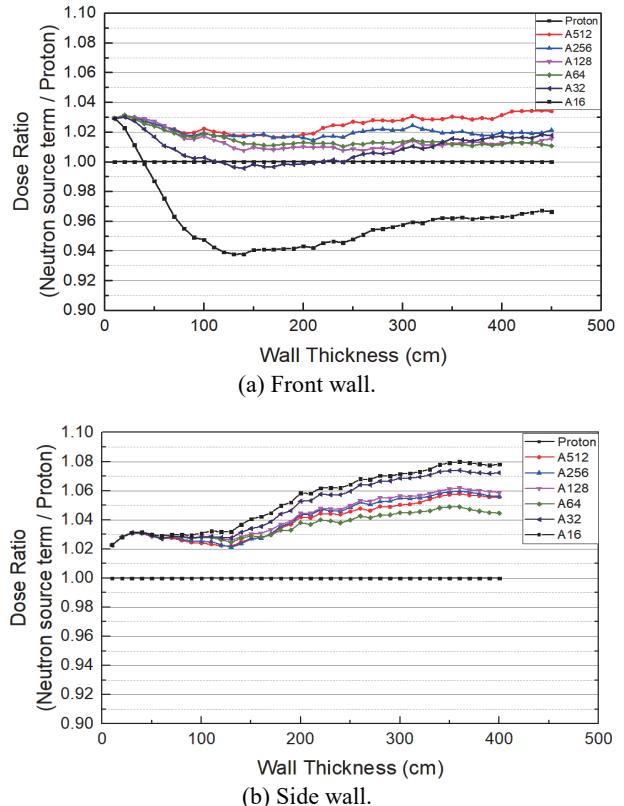


Figure 4. Dose ratio of neutron source term to direct proton transport according to the number of direction divisions using discrete energy distribution.

### 2.4. Neutron source term analysis on the number of energy divisions

In the front concrete wall, all neutron source terms except the source term divided 50 energy bins have a discrepancy of less than 1.5 %. The source term divided by 50 energy bins has maximum 3.2 % discrepancy as shown in **Figure 5** (a). In the side wall, the dose of the neutron source terms of 50 energy divisions shows a discrepancy of 5.0 % and the others show 1.7 % discrepancy.

In both front and side wall, the doses of all the neutron source terms were overestimated. As the number of the energy division decreases, the discrepancy between the dose of proton transport and the doses of two-step method increased as shown in **Figure 6**. The dose overestimation is occurred by the energy increase of the produced neutrons as the energy interval in each bin is increasing in case of the discrete energy distribution.

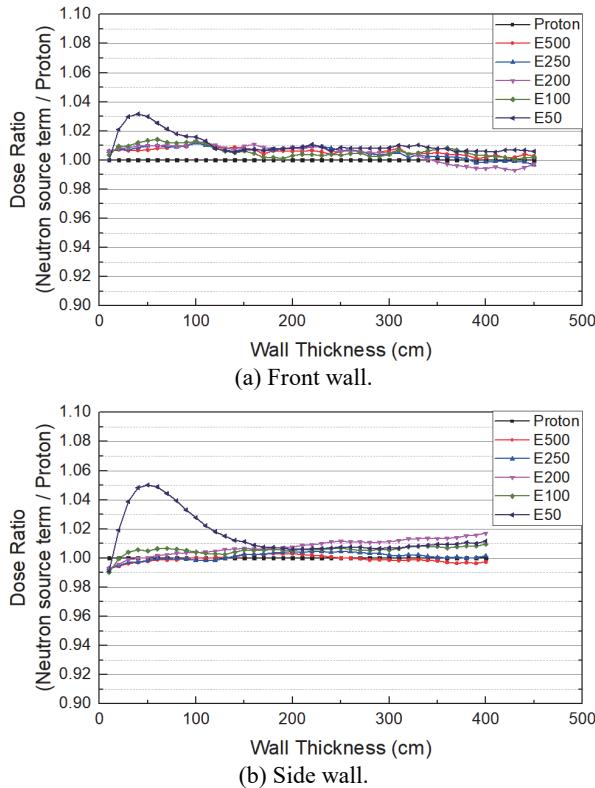


Figure 5. Dose ratio of neutron source term to direct proton transport according to the number of energy divisions using histogram energy distribution.

### 3. Conclusion

The analysis on the change of effective dose calculations according to the strategy on the neutron source term production was studied for RAON ISOL target room. The dose according to source dimensional distribution showed a discrepancy of less than 1.6 % in the front wall in case of the point, the line, and the cylinder distributions. The discrepancies on the side wall were within 1.1 % in case of the point and cylinder distribution. Neutron source term of cylinder distribution showed the least discrepancy in front wall. Neutron source term of point distribution showed the least discrepancy in side wall. The source term divided by 16 angles showed a discrepancy of about 6.2 % for the front walls and about 1.9 % for the side walls, while the other source terms showed a discrepancy within 1.5 %. Accordingly, source terms divided by more than 32 angular bins are recommended. The source term divided by 50 energy divisions showed the largest discrepancy for both front and side wall, so that neutron source terms dividing more than 100 energy divisions are recommended. In the case of using discrete energy distribution, the dose was highly overestimated as the number of energy divisions decreases. In this case, therefore, it was required to use a source term that has 500 energy divisions. The recommendations can be used for dose assessment of RAON ISOL system and the neutron source term generation process in this study can

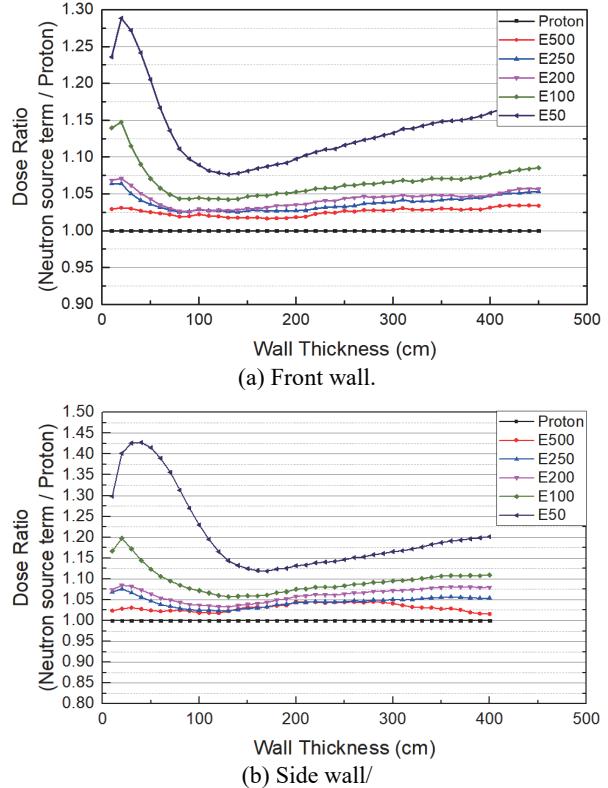


Figure 6. Dose ratio of neutron source term to direct proton transport according to the number of energy divisions in discrete energy distribution.

be utilized for other facilities having different proton energy, target material, and calculation geometry.

### Acknowledgements

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