Design of D-D neutron moderator for thermal neutron irradiation

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1. Introduction

The fast neutron generation facility based on a recently developed D-D neutron generator [1] is being constructed at Seoul National University. The fast neutron generation facility will be used for research and education of undergraduate and graduate students. In this study, the D-D neutron moderator for the thermal neutron irradiation is designed and analyzed.

2. Design of D-D Neutron Moderator

The D-D neutron moderator is designed for the purpose of maximizing thermal neutron flux at sample position. The distance between the sample position and neutron generation target is minimized. Considering a high voltage applied at the target, the moderator is located at 10 cm away from the front surface of the neutron generation target, as shown in Fig. 1. The basic design of the D-D neutron moderator is a well shape of which outer diameter is 18 cm. Polyethylene (PE) and water are considered as a candidate material. The detailed structure of the neutron moderator depends on whether the pneumatic transfer system is applied or not. The pneumatic transfer system is planned to transfer the sample capsule irradiated by thermal neutrons at the sample position to the γ-ray detection cage shielded by lead and copper.

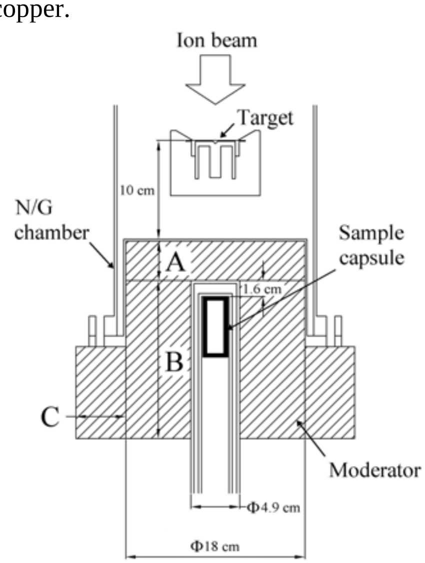

Fig. 1. Structure of the D-D neutron moderator when the pneumatic transfer system is applied.

When the pneumatic transfer system is applied, a cylindrical PE sample capsule, 6.5 cm long and 2.5 cm in diameter, is located in the aluminum transfer pipe of which diameter is 4.9 cm, as shown in Fig. 1. Using the MCNP4C code [2] and by varying the lengths of A, B and C, depicted in Fig. 1, the structure of the moderator which induces the maximum integrated neutron flux below Cadmium cutoff energy (0.55 eV) is selected. For the condition of the MCNP4C calculation, it is assumed that 2.5 MeV neutrons emit isotropically from the center of the target front surface with an intensity of 10^8 n/s and the average thermal flux at the inside of the sample capsule is calculated by using the cell flux tally (F4 tally). The interaction cross sections of the neutron are taken from the ENDF60 library by choosing the basic library of the MCNP4C code. The number of neutron histories is 10^8.

To determine the length of A, those lengths of B and C are fixed as 10 cm and 0 cm, respectively. The length of A is searched in step of 0.5 cm. The results are shown in Table 1. The thermal neutron flux for PE or water moderator reaches the maximum for A = 2.0 cm and 2.5 cm, respectively, while PE yields a higher value than water does. Hence PE is selected as the moderator material. As the length of A is fixed at 2 cm and the length of B is increased in steps of 1 cm, the thermal neutron flux gradually increases until B = 16 cm, after which the increment of the flux become insignificant. As the length of C is increased with the length of A and B fixed: A= 2 cm, B = 16 cm, the thermal neutron flux is increased up to C = 3 cm and the increment of the flux is 1.3 %. The length of C is selected as 3 cm. Applying the neutron reflector like graphite, beryllium or heavy water is considered. Even if the neutron reflector is added on the outside of the PE moderator, the thermal neutron flux

Table 1. Calculated thermal neutron flux when the length of A is changed and lengths of B and C are fixed.

A [cm]	Thermal neutron flux [$\times 10^4$ n/cm^{2}s]	
	Polyethylene	Water
1.0	2.14 ± 0.01	1.67 ± 0.01
1.5	2.20 ± 0.01	1.73 ± 0.01
2.0	**2.23 ± 0.01**	1.74 ± 0.01
2.5	2.20 ± 0.01	**1.77 ± 0.01**
3.0	2.16 ± 0.01	1.76 ± 0.01
3.5	2.09 ± 0.01	1.73 ± 0.01
4.0	2.00 ± 0.01	1.67 ± 0.01

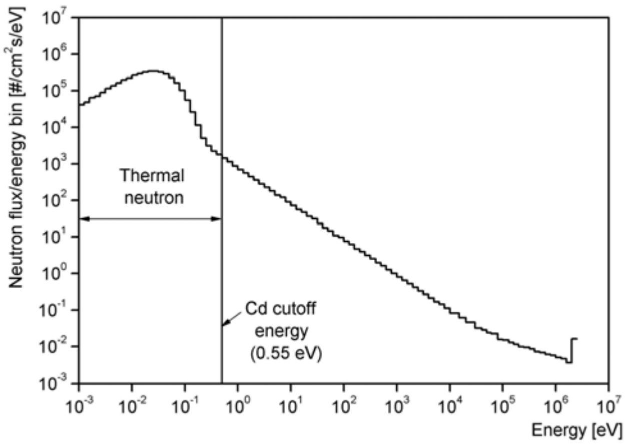

Fig. 2. Spectrum of neutron flux. Dimension of PE moderator is A = 2 cm, B = 16 cm and C = 3 cm.

is increased insignificantly. As part of the PE moderator is substituted by reflectors, the thermal neutron flux is decreased. Hence the neutron reflector is not considered in this moderator design. Finally, the calculated thermal neutron flux is $2.53(1) \times 10^4$ n/cm^{2}s when the pneumatic transfer system is applied and PE is only used as the moderator material. The neutron spectrum at this condition is shown in Fig. 2.

When the D-D neutron generator is used for in-situ research fields which require the γ-ray detection under the neutron irradiation, the pneumatic transfer system can be removed. The structure of the neutron moderator is considered with the position of detectors as shown in Fig. 3. Two γ-ray detectors are used to increase the detection efficiency and the dimension of a button shaped sample is 1 cm high and 3 cm in diameter. Lengths of A, B and C which maximize the thermal neutron flux at the sample position are determined with the same process of MCNP4C calculation for applying the pneumatic transfer system. When A = 4 cm, B = 14 cm, C = 0 cm, the thermal neutron flux is calculated as $2.46(1) \times 10^4$ n/cm^{2}s.

3. Conclusions

To maximize the thermal neutron flux at the sample position, the D-D neutron moderator is designed and analyzed by using MCNP4C code. PE is selected as the moderator material and the structure of the moderator is designed differently depending on whether the pneumatic transfer system is applied or not. The D-D neutron moderator will be produced and the thermal neutron flux will be measured using activation foils.

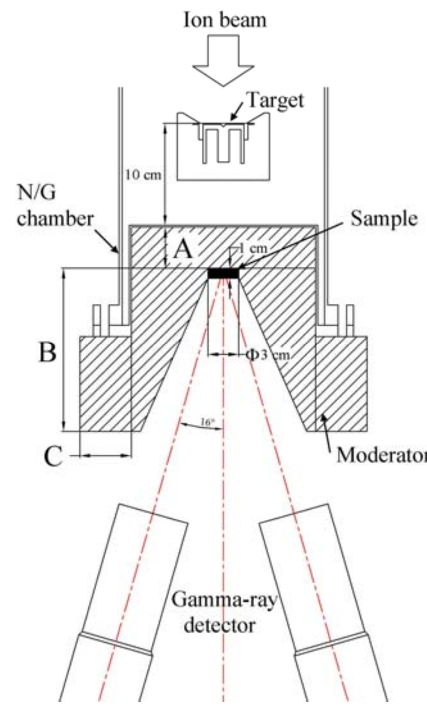

Fig. 3. Structure of the D-D neutron moderator used for in-situ research fields.

Acknowledgement

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REFERENCES

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[2] J.F. Briesmeister(ed.) MCNP - a General Monte Carlo N-particle Transport Code, Version 4C, LA-13709-M, Los Alamos National Laboratory, 2000.