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A Comparison Study of Neutron Sources in Active Rod Scanner by MCNPX Code

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1. Introduction

Non destructive test using neutron has been widely used in various field including harbor and aviation industries, and it is considered as an important method to measure changes in the concentration of material. The sealed nuclear fuel rod currently being used in nuclear power plant can be efficiently utilized for fission reaction in a range of concentration of ^{235}U. In order to do that, the sealed nuclear fuel rod scanning needs to be done precisely and quantitatively. The type of sealed nuclear fuel rod scanning currently used in Korea is an Active type Rod scanner which uses ^{252}Cf as a source. However, the neutron source of radioisotope shows a low reproducibility over time, as it is difficult to control the neutron flux, and has economic and technical disadvantages in maintenance and management of equipment. [1]

In this study, the neutron flux from Active Rod scanner which performs sealed nuclear fuel rod scanning using ^{252}Cf as a source was calculated and compared with that from neutron generator utilizing DD reaction which can easily control the neutron flux and thus shows reproducibility. The calculation and comparison was performed by MCNPX code, the neutron transport simulation program using Monte Carlo Method [2].

2. Methods and Results

2.1 Neutron Cross-section[3]

The fuel rod scanning using neutron measures γ-ray after the reaction of ^{235}U(n,f) and ^{238}U(n, γ). Therefore, ^{235}U and ^{238}U neutron cross section becomes an important factor. The neutron cross section of each element is shown in fig.1. and fig. 2., respectively.

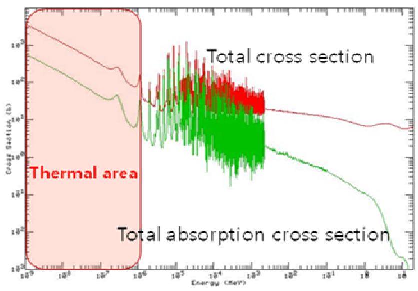

Fig.1. Neutron cross section of ^{235}U

Each element has high cross-section in the thermal neutron region. The neutron from each source should be made into thermal neutron through moderation with a moderator, because it has an average energy about 2 MeV.

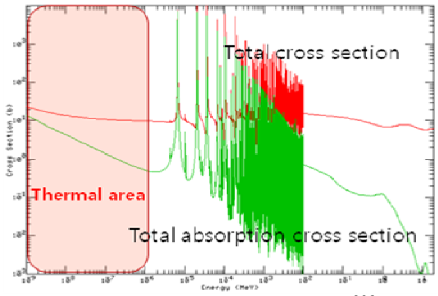

Fig.2. Neutron cross section of ^{238}U

2.2 Neutron Source

The neutrons used in this study are 2.45 MeV neutron from ^{252}Cf source and from DD neutron generator. The neutron from ^{252}Cf is being known to produce Watt fission energy spectrum[4]. This spectrum can be expressed by the following numerical formula and shown in the following fig.3.

$$p(E) = C\exp(-E/a)\sinh(bE)^{1/2} \qquad (1)$$

Where, a=1.025 MeV, b=2.926 MeV^{-1}

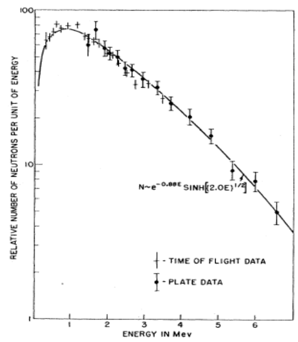

Fig.3. Experimentally determined energy spectrum of ^{252}Cf fission neutrons.

2.3 Neutron Moderator

The neutron can be more easily decelerated by a element of low atomic number. Therefore, it is good to use water to decelerate the neutron. However, considering an equipment production, it is much easier to use solid material. In this study, the simulation was done using polyethylene ($[CH_2]_{poly}$). The polyethylene is effective in neutron moderation, because it has a carbon to hydrogen ratio of 1:2.

The result of simulation is shown in fig.4. The thermal neutron converged on an electron energy of 0.05 eV and have a maximum value at a distance of 1 cm.

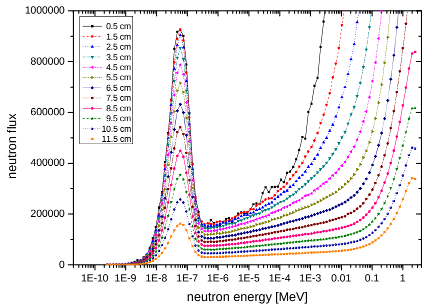

Fig.4. The distribution of linear velocity of thermal neutron

2.4 MCNP Geometry

The geometry of Fuel Rod Scanner is shown in fig.5. The polyethylene was used for the deceleration material. Table I shows source locations and fuel rod dimensions.

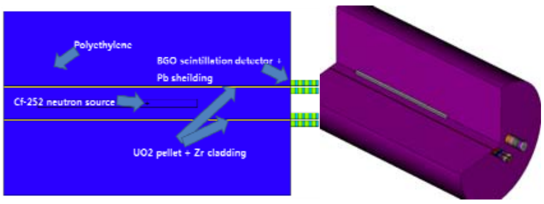

Fig.5. Fuel Rod Scanner concepts and MCNP geometry

The thermal neutron flux versus the distance from source was calculated using tally which utilized f4 tally. In order to confirm thermal neutron energy, the energy region was divided into 100 segments at log units. The reliability in this simulation was secured with the error of R<0.03±0.01 by each segment.

Table I : Source locations and fuel rod dimensions

	Diameter	Height	^{238}U Degree of enrichment
Value	0.82 cm	0.99 cm	5%
Nuclear fuel rod distance	3~9 cm in the direction of normal,1 cm interval from the location of radiation source		

2.5 Result

Fig.6. and 7. show the result calculated by the method in which the energy region was divided into 100 log segments. The neutron formed the spectrum at the electron energy of 0.05 eV, and thermal neutron flux is 3~4×10^6 #/sec from 3 mg of ^{252}Cf. The neutron flux below 0.1 eV is 10^7 #/sec.

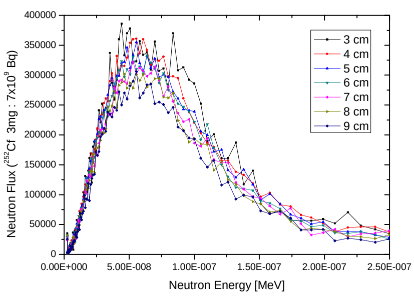

Fig.6. Neutron flux versus energy segment

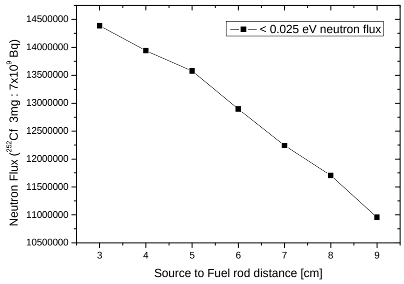

Fig.7. Thermal neutron flux versus fuel rod distance

The thermal neutron flux versus the distance between the source and fuel rod was compared in the range of thermal neutron region. As a result, the maximum of thermal neutron flux was observed at the distance of 3 cm, and the difference of neutron flux at the distance of 3 cm and 9 cm was 3.5×10^6 #/sec. This is 25% of whole flux. Fast neutron produced from France SODERN DD neutron generator is 10^8 #/sec. This result is 10 times less than the one from 3 mg of ^{252}Cf source.

3. Conclusions

Since ^{252}Cf source used as a source of fuel rod scanning equipments has physical half-life of radioisotope, it is necessary to alter or to replace the source to achieve reproducibility over time. In this study, to prevent this weakness, the availability of substitution neutron generator using DD reaction for radioisotope neutron source was investigated through thermal neutron flux. As a result, decelerated thermal neutron flux from ^{252}Cf source and from neutron generator is about 10^7#/sec, 10^6#/sec, respectively. The method through the distance control of the structure and optimization of irradiation location can allow a possibility of having a same effect by current method using radioisotope source. In the future, optimized source substitution from ^{252}Cf source to neutron generator will be accomplished through the design development considering geometry and efficiency of detector, etc.

Acknowledgement

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