Transactions of the Korean Nuclear Society Spring Meeting
Jeju, Korea, May 17-18, 2012

A Polyethylene Moderator Design for Auxiliary Ex-core Neutron Detector

Hwan-Soo Lee, Ho-Cheol Shin and Seong-Man Bae
KHNP Central Research Institute, 70 1312-gil Yuseong-daero Yuseong-gu, Daejeon, Korea
**Corresponding author: leehwansoo@khnp.co.kr*

1. Introduction

The moderator of detector assembly in ENFMS (Ex-core Neutron Flux Monitoring System) plays a key role for slowing down from fast neutron to thermal neutron at outside of reactor vessel. Since neutron monitoring detector such as BF3, fission chamber detectors mostly responds to thermal neutron, moderator should be included to neutron detector assembly to detect more efficiently. Generally, resin has been used for moderator of detector in ENFMS of OPR1000 and APR1400, because resin has stable thermal resistance, availability and high neutron moderation characteristics due to the light atomic materials.

In case of an auxiliary ex-core neutron detector, the polyethylene is suggested that polyethylene has a better moderator rather than resin, then, the amounts of moderator are reduced. This is important thing for auxiliary ex-core detector equipment at reactor, because the auxiliary equipment should affect minimally to another system.

In this study, polyethylene moderator is designed for auxiliary ex-core neutron detector. To find out the optimal thickness of polyethylene moderator, preliminary simulation and experiments are performed. And sensitivity simulation for detector moderator at actual reactor is performed by DORT code [1].

2. Methods and Experiments

2.1 Preliminary Simulation and Experiments

To exam the characteristics of polyethylene moderator, preliminary simulation and experiments are conducted. High density polyethylene(HDPE, ρ =0.941 g/cm^3) is selected for moderator of auxiliary neutron detector. In preliminary simulation and experiments, various polyethylene thicknesses (15, 24, 30, 39, 45, 51, 60, 66, 75, 81mm) are simulated and experimented. The simulation and experiment are performed with same condition as shown in Figure 1.

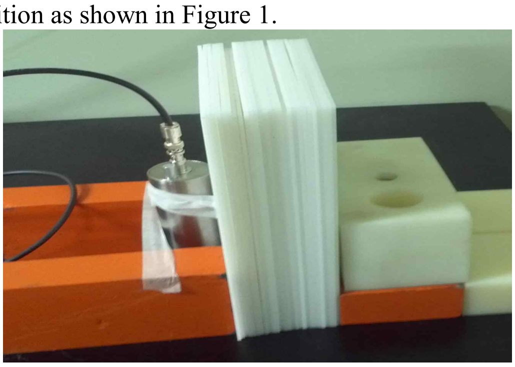

Fig. 1 Preliminary Experiment with Am-Be Neutron Source

The sealed Am-Be neutron source and BF3 detector is used for the experiments. All experiments are simulated by MCNP5 code which is Monte Carlo neutron transport code. Am-Be neutron source energy distribution is simulated as a continuous energy which has a modified Maxwell fission spectrum. BF3 detector (LND,lnc) has 5cps/nv sensitivity, and the experiment is conducted in condition with 1,350V for operating voltage and 10 for gain.

2.2 Results and Analysis of Preliminary Study

As results of preliminary simulation and experiments, the maximum thermal neutron fluence (< 0.414eV) is appeared around from 35mm to 55mm–thick polyethylene as shown in Figure 2. Unfortunately, the preliminary study did not show the sufficient fluence change through the effect of moderator, because the HDPE box contained Am-Be neutron source make the source spectrum soften (making neutron energy from high to low) prior to neutron moderation. However, the moderation patterns of polyethylene can be evaluated by this preliminary study.

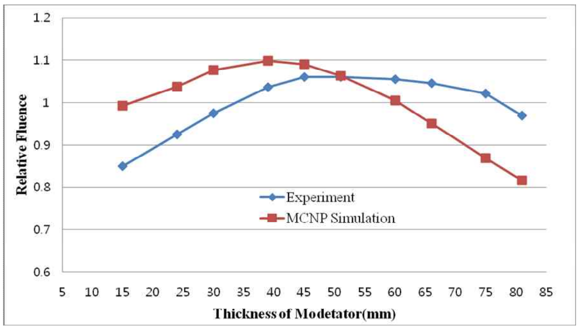

Fig. 2 Relative Fluence between Experiments vs. Simulation

3. Sensitivity Study and Results

3.1 Sensitivity Study with Reactor Condition

The auxiliary ex-core neutron detector can be used at startup range in operation mode, then, we simulated at startup of reactor operation.

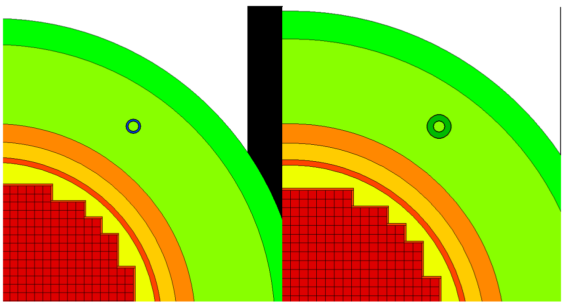

Fig. 3 Simulations for Moderator Thickness (15 and 65mm)

In order to simulate the startup range, Cf-252 neutron spectrum is used as a source, because the ex-core detector's signal by Cf-252 source in neutron source assembly (NSA) is dominant at startup of initial core.

The sensitivity simulation for calculating fluence inside of ex-core detector is performed by DORT code. DORT code is based on the discrete ordinate transport method. 47-energy-group DORT calculations were performed with P_3, S_8 approximation [2] in two-dimensional geometries. The cross section data used in this study were from the BUGLE-96 library [3]. The fluence is calculated with an R-Θ model about APR1400 as shown in Figure 3. This simulation is performed various polyethylene thicknesses (10, 15, 20, 25, 30, 35, 45, 55, 65, 75mm) to find out the optimal thickness of moderator. In DORT code simulation, Auxiliary ex-core detector is located at the same position of ENFMS, because the results of auxiliary detector are needed to compare with that of ENFMS.

3.2 Results and Analysis of Sensitivity Study

Through the sensitivity study of polyethylene thickness, the maximum thermal neutron fluence (< 0.414eV) is appeared at about 30mm-thick of polyethylene as shown in Figure 4. The neutron fluence is averaged neutron fluence in BF3 detector assembly.

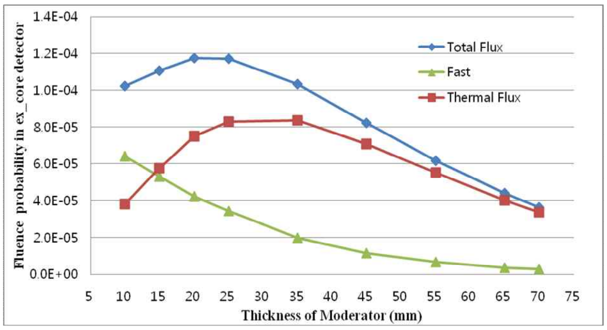

Fig. 4 Neutron Fluence in Auxiliary Ex-core Detector

The resulted optimal thickness of sensitivity study is less than that of the previous preliminary study, because the neutron energy distribution is different with the preliminary study. The average energy of Cf-252 source distribution is about 2.1 MeV and that of Am-Be source is about 4 MeV. The high energy neutron is needed to thermalize more moderator and preliminary study more soften by HDPE box.

At the same condition, maximum thermal neutron fluence of resin which gernerally used in opearting reactor is appeared at around 70-80mm thickness[4]. Compared with this study, polyethylene has better performance for slowing down than resin moderator. As it is mentioned above, it is advantage for auxnially detector to reduce the effect another system in reactor condition. Moreover, thermal neutron fluence of polyethylen is more populated than that of resin. The relative maximum thermal neutron fluence by polyethylene moderator is 3.93E+3 for 4.7E+7 source strength, but that by resin moderator is calculated as 3.93E+3.

4. Conclusions

The polyethylene moderator for auxiliary ex-core neutron detector is evaluated by computer code simulation and experiments.

In preliminary study, we can predict the characteristics of polyethylene moderator. To find out the optimal thickness of polyethylene moderator, the sensitivity study is performed by DORT code. Then, maximum thermal neutron fluence in detector is appeared at about 30mm-thick of polyethylene moderator. And polyethylene moderator is performed better slowing down rate and more thermal neutron population in detector than those of resin. Therefore, the polyethylene moderator can be suggested for auxiliary ex-core detector.

Additionally, through comparison of two studies, it is realized that changes of neutron energy distribution is dominant effect to the amount of moderator to slow down to the thermal neutron.

REFERENCES

[1] "RSICC Computer Code Collection-DOORS3.2," CCC-650, ORNL, April, 1998.
[2] J. P. Jenal et al, "The Generation of a Computer Library for Discrete Ordinate Quadrature Set," TM-6023, ORNL, October, 1977.
[3] "RSIC DATA LIBRARY COLLECTION BUGLE-96," DLC-185, ORNL, March 28, 1996.
[4] H.S.Lee "A Evaluation of Sensitivity of Advanced ENFMS of OPR1000," Transactions of KARP, Autumn, 2011.