Transactions of the Korean Niclear Society Autumn Meeting
PyeongChang, Korea, October 30-32, 2008

Development of Neutron Shielding Material for Cask and Accelerator

Hee Young Kang, Ki Seog Seo, Byung Chul Lee, Chang Jae Park, Ho Dong Kim
Korea Atomic Energy Research Institute 150 Dukjin-dong, Yuseong-gu, Taejon , Korea
Corresponding author: hykang@kaeri.re.kr

1. Introduction

The neutron shielding materials are used as a neutron shield for spent fuel shipping cask, beam accelerators and neutron generators. At early stage, the neutron attenuations of materials were evaluated with the cross sections. After that, benchmark or mock-up experiments on the multi-layer problem to confirm the shielding characteristics or to evaluate analysis accuracy were reported [1-4]. Recently, the need to transport spent nuclear fuels is increasing due to the current limited storage capacity. The on-site storage capacity at some of nuclear power plants is expected to be full in near future. With a growing inventory of spent fuels at power plants, these spent fuels need to be transported to other storage facilities.

Shipping casks have been developed to safely transport spent fuels that emit high neutrons and gamma-ray radiation. The external radiation level of the shipping cask from the spent fuel must be limited to meet the standards specified by the IAEA radioactive material package regulation [5], so it is important to develope a proper neutron shielding material for a shipping cask.

Neutron shielding experiments and analyses on the shielding effects of materials have been conducted, and some experiments have been performed to examine the shielding effects of selected materials. The shielding experiments consist of evaluating not only the shielding effects of a material alone but also the effects of the material thickness. The experimental results were compared with those obtained by using the MCNP-5c code[6].

2. Experimental setup and method

Neutron sources are more important in neutron shield attenuation measurements. The dominant source term of neutrons is spontaneous fission from Cf-252 for neutron with mean 2.14 MeV. The experimental model for the neutron shielding materials consists of a Cf-252 source, the shielding material being investigated, and a neutron detector system. Figure 1 shows a cross section of neutron measurement system model. The detection system consist of the connection of the counters and the PSR-B signal module so that the neutron count rates can be displayed. The signal module was connected with a serial port to a notebook PC. The neutron count rates using a continuous mode can be monitored and displayed by using the data processing software of the PC.

In the detection system, a long He-3 tube counter with a suitable thick polyethylene moderator was used to detect the neutrons. The polyethylene slab has 3 holes for the detector tubes. The experimental model for the Cf-252 neutron source has three He-3 gas detector tubes that can detect neutrons by using the (n, p) reaction. The neutron detector tubes are approximately 30 cm long, enough to obtain a constant response for all the shielding materials. The neutron source was encased with a Cadnium tube to absorb the back-scattered neutrons.

The shielding materials, such as polyethylene, graphite, K-resin, and paraffin examined in this study are considered. Polyethylene is the most popular solid neutron shield. Graphite is a pure carbon (more than 98%) product that is used for a high thermal resistor and electrode. K-resin is a kind of synthetic resin that contains boron to reduce the production of secondary gamma rays. Paraffin is a very useful neutron shield; however, it is melted at less than 200 ℃. The shielding materials are prepared using a solid cylinder with an 80 mm diameter and a 40, 50, or 60 mm thickness.

At first, the neutron count rate without any shielding material between the source and the detectors was measured. Then, the experimental data were collected by sequentially changing the kind of material sample and thickness. The signal module equipment supplied a +5 V power and a high voltage bias (1,750 V) to the pre-amp of the three neutron detectors. The total neutron count rates were average values with 10 cycles of 30 seconds each for each material sample.

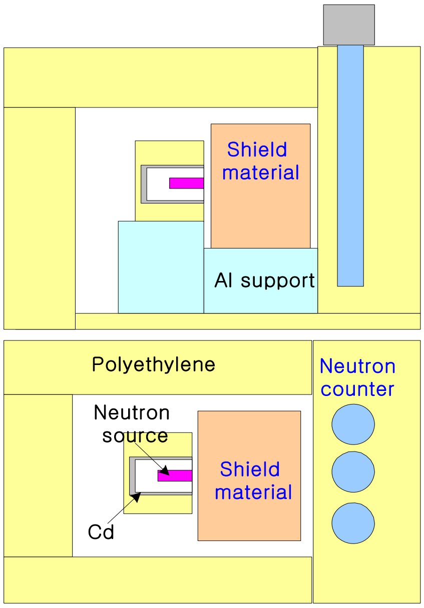

Fig. 1 Neutron shield test model

The experiments were analyzed by using the continuous energy Monte Carlo code MCNP-5c. The neutron energy spectrum of a Cf-252 source for the MCNP code is simulated as a Watt fission spectrum by using the coefficients provided with the MCNP code. That is,
where E is the neutron energy in MeV.

Among a variety of the shielding materials, the following neutron shielding materials were considered: polyethylene, graphite, K-resin, and paraffin. The neutron cross-section library FMn card was used for the reaction type of (n, p). The calculations were performed with a series of materials and thicknesses.

3. Results and Discussions

This study conducted to evaluate the neutron shielding effects of four materials, polyethylene, K-resin, paraffin, and graphite, by using a neutron source. The total neutron count rates from three neutron detectors were measured by using a Cf-252 source to investigate the shielding effects of various materials. The measured neutron count rates as a function of the material thickness for the four shielding materials are shown in Figure 2. Three different values of the thickness, 40, 50 and 60 mm, were considered for the shielding materials. According to the four attenuation curves, the count rates were observed to decrease as the thickness of material samples. The attenuation of neutron through the shielding material is generally explained by a high moderating ratio due to the light element contents of the materials composition. The count rates for the K-resin show lower than the graphite, paraffin and polyethylene attenuation curves. It shows the shielding effects of K-resin materials are slightly better than those of the other materials.

As expected, the K-resin exhibits much better shielding effects than the polyethylene material because of the higher density of 1.5 g/cc compared with polyethylene with 0.94 g/cc.

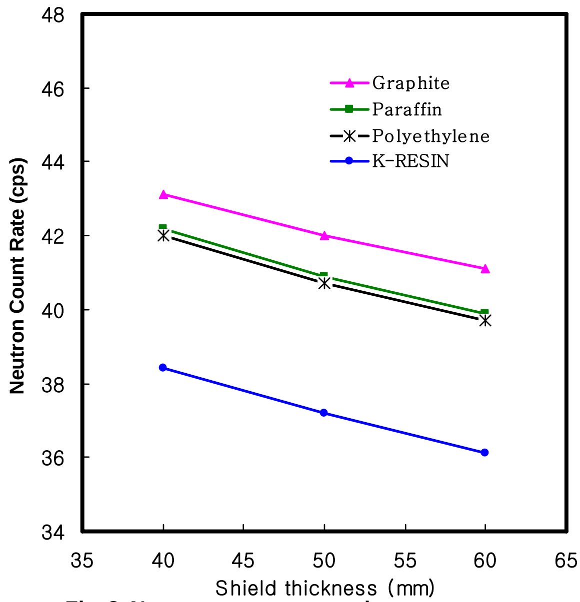

Fig. 2 Neutron count attenuation curve on shield materials and thickness

REFERENCES

[1]R. J. Adams and K. H Lokan, Health Phys. 36, 671(1979).
[2]A. S. Mollah, G. U. Ahmad, and S. R. Husain,Nuclear Eng. Design, 135,321(1992)
[3]K. Ueki, A. Ohashi, and N. Nariyma, Nuclear Sci. Eng., 124, 455(1966).
[4]A. X. da Silvia and V. R. Crispim, Radiation Protection Dosimetry, 95, 4, 333(2001).
[5] IAEA Safety Standards, "Regulations for the Safe Transport of Radioactive Material," Safety series No. 6, IAEA, Vienna, 1990.
[6] J. F. Briesmeister, Ed., "MCNP - A General Purpose Monte Carlo Code for Neutron and Photon Transport," LANL report La-12625-M, Ver. 4C(2001)