Journal of
Radiation Protection and Research 2016;41(4):384-388
https://doi.org/10.14407/jrpr.2016.41.4.384

pISSN 2508-1888 | eISSN 2466-2461

A Proposal on Evaluation Method of Neutron Absorption Performance to Substitute Conventional Neutron Attenuation Test

Jae Hyun Kim[1], Song Hyun Kim[2,*], Chang Ho Shin[1], Jung Hun Choe[3], In-Hak Cho[3], Hwan Seo Park[3], Hyun Seo Park[4], Jung Ho Kim[4], Yoon Ho Kim[4]

[1]Department of Nuclear Engineering, Hanyang University, Seoul, Korea; [2]Nuclear Fuel Cycle Waste Treatment Research Division, Research Reactor Institute, Kyoto University, Osaka, Japan; [3]Ionizing Radiation Center, Nuclear Fuel Cycle Waste Treatment Research Division, Korea Atomic Energy Research Institute, Daejeon, Korea; [4]Ionizing Radiation Center, Korea Research Institute of Standards and Science, Daejeon, Korea

Technical Paper

Received July 17, 2015
Revision August 22, 2016
Accepted September 12, 2016

Corresponding author: Song Hyun Kim
Nuclear Engineering Science Division, Research Reactor Institute, Kyoto University, Kumatori, Sennan, Osaka 590-0494, Japan
Tel: +81-80-8047-1609,
Fax: +81-72-451-2604,
E-mail: kim@rri.kyoto-u.ac.jp

ABSTRACT

Background: For a verification of newly-developed neutron absorbers, one of guidelines on the qualification and acceptance of neutron absorbers is the neutron attenuation test. However, this approach can cause a problem for the qualifications that it cannot distinguish how the neutron attenuates from materials.

Materials and Methods: In this study, an estimation method of neutron absorption performances for materials is proposed to detect both direct penetration and back-scattering neutrons. For the verification of the proposed method, MCNP simulations with the experimental system designed in this study were pursued using the polyethylene, iron, normal glass and the vitrified form.

Results and Discussion: The results show that it can easily test neutron absorption ability using single absorber model. Also, from simulation results of single absorber and double absorbers model, it is verified that the proposed method can evaluate not only the direct thermal neutrons passing through materials, but also the scattered neutrons reflected to the materials. Therefore, the neutron absorption performances can be accurately estimated using the proposed method comparing with the conventional neutron attenuation test.

Conclusion: It is expected that the proposed method can contribute to increase the reliability of the performance of neutron absorbers.

Keywords: Neutron absorber, MCNP, Criticality control, Vitrified form, Neutron absorption ability, Neutron attenuation test

Introduction

Spent nuclear fuels, which are high level waste (HLW), have been generated with lots of radiation and heat from nuclear power plants. According to statistics information in Korea, in temporal storage, the number of spent fuel assemblies is about 11,000, and predicted number of spent fuels is about 71,000 assemblies until 2030. It is predicted that lots of spent fuel will be generated in the future; therefore, disposal and storage problems of spent fuels have gotten increased. Generally, spent fuel storage is designed to improve storage efficiency in limited area. To design spent fuel storages, criticality safety, radiation shielding, heating removal and structural integrity should be considered. Especially, criticality safety is an important safety guideline to prevent criticality accident. To satisfy criticality safety criteria and design dense storage, neutron absorber has been used. Existing neutron absorbers such as BORAL^{TM}, NeutroSorb^{TM}, and the others, which are metal-based materials, have been used. These neutron absorbers have high neutron absorber ability and mechanical strength. However, the cost of these absorbers is high. Also, it can be easily corroded with salt water (especially aluminum-based absorber). In previous study [1], to solve the disadvantages of existing neutron absorber, newly-designed neutron absorber based on vitrified form is proposed. The vitrified form which is one of radioactive wastes is generated from vitrified form facility. And then, vitrified form is reused as criticality control material in storage facility. Using the neutron absorber and storage concept, it can simultaneously store both spent fuel and vitrified form in storage. Therefore, it can increase the storage efficiency of the radioactive wastes as well as to reduce the disposal cost.

The vitrified form technique is a method to eternally prevent radioactive nuclides composed in glass structure into environment. Thus, the radioactive nuclides in vitrification form are enclosed in physically and chemically strong glass formation. Boron is contained in vitrified forms to increase thermal resistance and mechanical strength. Boron has high absorption cross section; therefore, it can be used for criticality control material in nuclear-related facilities. In addition, vitrified form has high corrosion resistance and good thermal property. The brief description on the properties of vitrified form is given in Table 1.

In previous studies [1, 2], the applicability of vitrified form as neutron absorber was verified. For the application of the vitrified forms as neutron absorber, the neutron absorption ability should be properly verified. To verify their neutron absorption ability, some problems can be occurred. Vitrified form is radioactive waste; therefore, it is difficult to perform chemical analysis. Another problem is a difficulty on analyzing neutron absorption ability with lots of impurities. The vitrification form has lots of impurity. Thus, it is impossible to distinguish capture and scattering rate with conventional attenuation test. As a result, it is required to develop a new testing method of the neutron absorption ability for the absorber. In this study, a new testing method is proposed to verify neutron absorption ability of vitrified form.

Table 1. Properties of Vitrified Form as Neutron Absorber

	Borated Glass	BORALTM
Density	2.23 g·cm^{-3}	2.66 g·cm^{-3}
Compressive Strength	20-50 MPa	~175 MPa
Thermal Property	TEC: 3-5 × 10^{-6} k: 1.14 W·m^{-1}·K^{-1} Temperature Limit: ~500ºC	TEC: 23.1 × 10^{-6} k: 237 W·m^{-1}·K^{-1} Temperature Limit: 1,000ºC
SiO_2	71.0	-
Na_2O	4.5	-
B_2O_3	~12.0	-
Al_2O_3	2.5	-
B_4C	-	~37
Al	-	~63

Materials and Methods

1. Overview of conventional attenuation test

For verification of neutron absorbers, some guidelines on the qualification and acceptance of neutron absorbers should be satisfied. One of the guidelines in American Society for Testing and Materials (ASTM) is a neutron attenuation test [3]. For neutron attenuation test, thermal neutron beam is induced to absorber, and the attenuation rate passing through the material is estimated. Then, neutron count rate is converted to areal density or effective cross section. However, this approach can cause a problem for the qualifications when lots of unknown elements are composed in a target material. For the thermal neutron, it mainly has two kinds of reaction types, which are the scattering and absorption reactions. In the neutron attenuation test, if a material has high scattering cross section for thermal neutrons, it will have a high attenuation value although it has small absorption cross section of the neutrons. This leads that the performance of neutron absorbers cannot be properly estimated for the criticality control aspect.

2. Proposed test method design

To increase the reliability of the neutron absorption performance, an estimation method is proposed. Basic principle of proposed test method is shown in Figure 1B. The proposed method is consisted to measure both direct penetration and back-scattering of thermal neutron at a same time. Polyethylene block allows increasing scattered fluxes. Hence, if scattering cross section of material is higher, the counting rate will be increased. Also, if absorption cross section of material is higher, detection rate is relatively decreased although it has same total cross section. While the attenuation coefficient using the conventional attenuation tests can only esti-

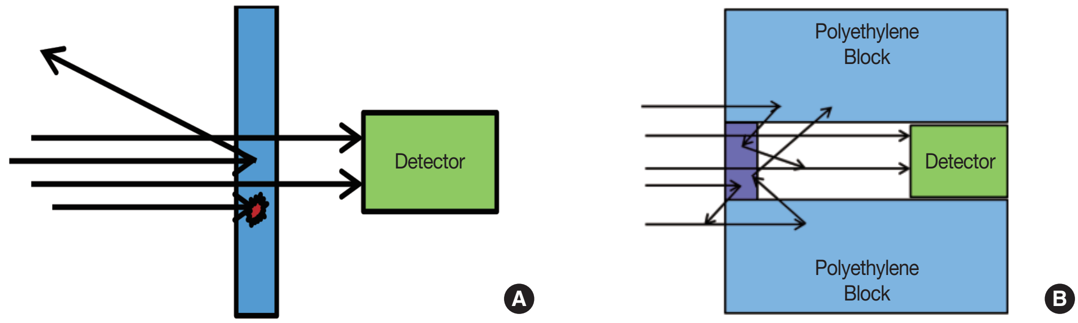

Fig. 1. Basic principle of proposed test method (A) Conventional neutron attenuation test and (B) proposed test method.

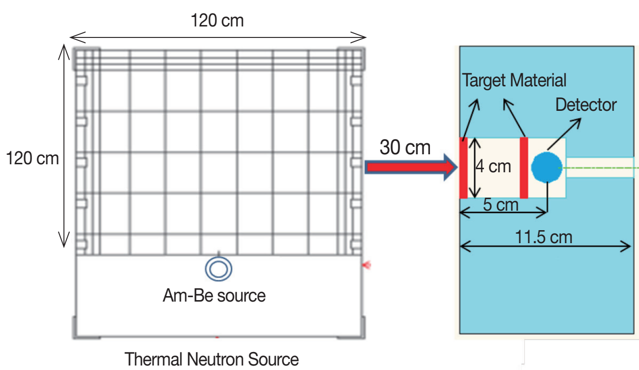

Fig. 2. Overview of performance tester.

Table 2. Tester Information of Equipment

	Value
Polyethylene Block Size	14×19×11.5 cm
Target Shape	Plate
Target Size	4×4×0.5 cm
Detector Type	SP9 Proportional Counter (He-3 Counter)

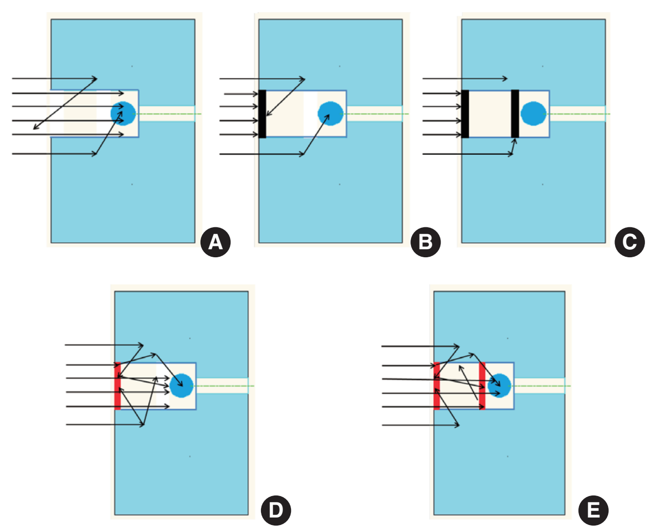

Fig. 3. Tester with proposed test method for each step. (A) Step 1: Simulation of fluxes in detector (C1) without absorber, (B) Step 2.1: Simulation of fluxes in detector (C2.1) with a perfect absorber, (C) Step 2.2: Simulation of fluxes in detector (C2.2) with two perfect absorbers, (D) Step 3: Simulation of fluxes in detector (C3) with single absorber, and (E) Step 4: Simulation of fluxes in detector (C4) with two absorbers.

mate the total attenuation; it can evaluate relative neutron absorption performance with using the proposed method.

To measure absorption ability, a proposed performance tester was designed as shown in Figure 2. A thermal neutron generator including Am-Be source with a graphite moderator, was used for the neutron test. The SP9 proportional counter was located in the polyethylene block. The polyethylene block is rectangular shape and has a square hole to insert the target material. The two targets can be located inside of the square hole, and each is used to detect relative response of the detector. Details of the tester are described in Table 2.

The overview of the experiment is shown in Figure 3. To test relative neutron absorption ability of the target materials, the simulations were performed with following steps:

Step 1) Fluxes in detector are estimated without absorber (C1). This value is used to compare the attenuation ability of the target materials.

Step 2) Thermal neutrons, which do not pass by target materials, are called background. To suppress the background fluxes, perfect absorbers are assumed that they can absorb all neutrons passing by perfect absorbers. Fluxes in detector are estimated with a perfect absorber (C2.1) and two perfect absorbers (C2.2) absorber as shown in Figures 3B and C.

Step 3) Fluxes in detector are estimated with single absorber (C3). To suppress the background fluxes, the detected results are refined with Equation 1. Finally, the result suppressed by background (B3) is used to confirm the neutron performance analysis.

https://doi.org/10.14407/jrpr.2016.41.4.384

$$B3 = C3 - C2.1 \quad (1)$$

Step 4) As much as scattered neutrons by the polyethylene block in performance tester, due to the back-scattering influences, decreasing rate of fluxes counted in detector tends to be low. To maximize the efficiency of the tester and much clearly distinguish scattering and absorption reaction by target material, two absorbers model is introduced. Fluxes in detector are estimated with two absorbers (C4). Fluxes of double absorbers with background suppression (B4) are analyzed with Equation 2. This value is used to relatively compare the neutron absorption ability of target material.

$$B4 = C4 - C2.2 \quad (2)$$

Step 5)Fluxes fraction B0 is calculated using Equation 3. If B4 is relatively lower than B3, it can be guessed that the material has relatively higher absorption cross section.

$$B0 = \frac{B4}{B3} \times 100 \quad (3)$$

Results and Discussion

To verify the estimation system of absorber performance, MCNP simulations were pursued with target materials which are the polyethylene, iron, vitrified form, and normal glass. MCNPX 2.6[1)] with ENDF-VI and sab2002 cross section library was used for the analyses. The particle histories are decided that the all relative errors of the results are within 2%.

The results with thermal neutron fluxes in detector are given in Tables 3, 4, 5, and 6. Table 3 is the results of Steps 1 (C1) and 2 (C2) which are used to suppress the background.

Step 3 is results for the case using single target. It was evaluated that the normal and background suppressed fluxes with the proposed neutron absorber are 74.26 $\#\cdot cm^{-2}\cdot sec^{-1}$ and 13.15 $\#\cdot cm^{-2}\cdot sec^{-1}$, respectively. Using the results in Tables 3 and 4, absolute neutron absorption ability can be

Table 3. Simulation Result of Flux in Detector with Step 1 and Step 2

	Step 1 (C1)	Step 2.1 (C2.1)	Step 2.2 (C2.2)
	Bare	Single Perfect Absorber	2 Perfect Absorbers
Flux in Detector	191.98 $\#\cdot cm^{-2}\cdot sec^{-1}$	61.11 $\#\cdot cm^{-2}\cdot sec^{-1}$	25.26 $\#\cdot cm^{-2}\cdot sec^{-1}$

Table 4. Simulation Result of Flux in Detector with Step 3

	Polyethylene	Iron	Absorber	Normal Glass (Without Boron)
Flux in Detector	146.33 $\#\cdot cm^{-2}\cdot sec^{-1}$	154.57 $\#\cdot cm^{-2}\cdot sec^{-1}$	74.26 $\#\cdot cm^{-2}\cdot sec^{-1}$	181.06 $\#\cdot cm^{-2}\cdot sec^{-1}$
Flux in Detector with Background Suppression B3 (C3 – C2.1)	85.21 $\#\cdot cm^{-2}\cdot sec^{-1}$	93.46 $\#\cdot cm^{-2}\cdot sec^{-1}$	13.15 $\#\cdot cm^{-2}\cdot sec^{-1}$	119.95 $\#\cdot cm^{-2}\cdot sec^{-1}$

Table 5. Simulation Result of Flux in Detector with Step 4

	Polyethylene	Iron	Absorber	Normal Glass (Without Boron)
Flux in Detector	114.21 $\#\cdot cm^{-2}\cdot sec^{-1}$	120.25 $\#\cdot cm^{-2}\cdot sec^{-1}$	34.59 $\#\cdot cm^{-2}\cdot sec^{-1}$	169.90 $\#\cdot cm^{-2}\cdot sec^{-1}$
Flux in Detector with Background Suppression B4 (C4 – C2.2)	88.95 $\#\cdot cm^{-2}\cdot sec^{-1}$	94.99 $\#\cdot cm^{-2}\cdot sec^{-1}$	9.33 $\#\cdot cm^{-2}\cdot sec^{-1}$	144.65 $\#\cdot cm^{-2}\cdot sec^{-1}$

Table 6. Flux in Detector with Background Suppression

	Polyethylene	Iron	Absorber	Normal Glass (Without Boron)
Ratio B0 (B4/B3)	104.39%	101.64%	71.01%	120.59%
Thermal Macroscopic Cross Section (Absorption | Scattering)	0.03 | 2.65	0.22 | 1.17	Abs. >> Scat.	Abs. << Scat.

1) D.B Pelowitz, et al. MCNPXTM User's manual, version 2.7.0. LA-CP-00438. Los Alamos national laboratory, 2011.

quantified. As shown in Table 4, the estimated flux using the glass absorber is much lower than normal glass. Therefore, it can be simply distinguish that the neutron absorber is well manufactured.

However, in the absorber based on vitrified form, lots of impurities can be included. To confirm the neutron absorption ability in the situation, Step 4 simulations were pursued. The results are given in Table 5. Using the results in Table 5, flux ratio was calculated using Equation 3, and the results are given in Table 6. In our sensitivity study, the fraction B0 in Table 6 is generally lowly evaluated than 100% when a material has high absorption cross section. Using the property, the absorption ability can be easily quantified with the proposed test system even though there are lots of impurities in an absorber.

Conclusion

In this study, to quantify the neutron absorption performance for neutron absorbers based on vitrification form, a performance test method was proposed. The proposed method can measure not only the direct thermal neutrons passing through materials, but also the scattered neutrons reflected to the materials. Therefore, it can efficiently quantify the neutron absorption performance for the materials which has lots of compositions. To verify the test system, MCNP simulations were performed using the proposed scheme. The results show that the neutron absorption performances can be effectively estimated using the proposed method comparing with the conventional neutron attenuation test. It is expected that the proposed method can contribute to increase the reliability of the performance test for newly developed neutron absorbers.

Acknowledgements

This work was supported by the Energy Efficiency & Resources of the Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by Korea government Ministry of Knowledge Economy (20121620100070), and Innovative Technology Center for Radiation Safety (iTRS).

References

1. Kim SH, Kim JH, Shin CH, Kim JK, Park HS, Kim SY. A feasibility study on the criticality control method using radioactive vitrified forms for spent fuel storage. Nucl. Eng. Des. 2014;280:644-650.
2. Kim JH, Kim SH, Shin CH, Choe JH, Cho IH, Park HS. A study on applicability of a new neutron absorber based on radioactive vitrified forms for PLUS7 and WH17x17 spent fuel storages. Korean Radioactive Waste Society. Incheon, Korea. May 27-29, 2015.
3. American Society Testing and Material. Standard practice for qualification and acceptance of boron based metallic neutron absorbers for nuclear criticality control for dry cask storage systems and transportation packaging. C1671-07. 2013;1-2.

 https://doi.org/10.14407/jrpr.2016.41.4.384