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Computational investigation of the neutron shielding and activation characteristics of borated concrete with polyethylene aggregate

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ABSTRACT

This paper presents the result of a computational study to investigate the neutron shielding and activation characteristics of concretes containing boron carbide and polyethylene. Various mixes were considered with changes in the contents of boron carbide and polyethylene aggregate. The Monte Carlo simulation code MCNP-5 was utilized to determine the transmission of neutron through concrete at different energies from 0.1 eV to 1 MeV, and ORIGEN-S code was then used to predict activation characteristics of the concretes. It was shown that the replacement of polyethylene in borated concrete greatly enhanced the shielding efficiency of the concrete, and total activity levels of the concrete were considerably decreased with this replacement. Furthermore, double-layered structures having the first layer of polyethylene aggregate-replaced concrete and the second layer of 2 wt% borated concrete are shown to improve shielding efficiency more significantly than monolithic structures.

1. Introduction

Concrete is commonly used for shielding of neutron-generating facilities, such as fusion reactors, fission reactors, and accelerators facilities. The shielding characteristic and activation properties can vary and are highly dependent on the composition of the raw materials of the concrete [1]. The composition of the aggregate, which is composed of 70–80% of concrete, is the most important factor in determining the shielding properties [2] as well as the mechanical performance [3]. In the last few years, several articles have been devoted to the study of developing a special type of concrete by changing the properties of the aggregate and cement. Kinno et al. [4] proposed limestone, quartzite, colemanite, alumina-ceramic, Portland cement, and high-alumina cement as raw materials for low-activation concrete. Akkurt et al. [5] investigated the gamma shielding performance of concrete with barite aggregate and Genkel [6] studied the effect of colemanite aggregate on the shielding characteristics. As the energy level and the service life of neutron facilities are steadily increasing, the attenuation of high-energy neutrons and methods to reduce activation have become the main concerns related to shielding materials [7].

When a concrete wall is exposed to neutrons, it becomes radioactive, therefore the concrete should be categorized as a radioactive waste during the decommissioning process of a power plant due to decaying gamma-rays from the activated elements. Studies have indicated that long-term induced activity of concrete is dominated by long-lived nuclides, typically, ^{60}Co (half-life: 5.271 y), ^{152}Eu (half-life: 13.54 y) and ^{137}Cs (half-life: 30 y) [8,9]. In order to reduce the activation, the boron isotope B-10 has been introduced due to its high absorption cross-section for thermal neutrons [10]. Several studies have shown that an addition of boron increases the neutron shielding effectiveness of concrete and also attempts to capture neutrons that would lead to activated material and delayed gammas [1,11]. However, several limitations of a boron addition should be noted. Since the neutron absorption cross-section of the boron isotope is inversely proportional to the neutron energy, there is an insignificant difference in high-energy neutron attenuation, as represented in a study by Kinno et al. [12]. Additionally, the replacement amount of boron is limited as the boron compound impedes the setting of cement and has a deleterious effect on the compressive strength of concrete. It has been shown that an addition of 4 wt% of boron sludge can reduce the 90-day compressive strength of concrete by 65% [13]. With consideration of the concrete strength and economics, previous research indicates that additions of boron up to 1 wt% are acceptable [14–16].

As fast neutrons are very penetrative, shielding materials should have a high scattering cross-section [17]. Hydrogen, iron, and carbon are the most commonly used elements due to their suitable scattering cross-sections for attenuating fast neutrons

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[1]. Polymers, especially polyethylene and polypropylene, consist of hydrogen and carbon, and are highly effective as a neutron shielding material since they can reduce the energy of fast neutrons through elastic collisions [18]. Hence, the neutron shielding effectiveness of the borated concrete is expected to be improved by an addition of polyethylene. Polyethylene itself has been used for traditional neutron shielding in nuclear fields as a form of a plate; however, it degrades the durability and heat-resistance [19]. By admixing polyethylene in concrete, the durability and heat-resistance of the material might be enhanced while the neutron shielding effectiveness could be maintained.

In this study, a new shielding concrete containing boron carbide and polyethylene is proposed. In general, there are three aspects expected from a neutron shielding material: (i) high attenuation rate, (ii) capture of thermal neutrons and (iii) attenuation of secondary gamma radiation generated from interactions between neutrons and the nuclei in the shielding material [20]. The first feature was verified with linear attenuation rate, and the second and the third features were examined with total activities, total neutron flux and neutron energy distribution in each depth of the shielding concrete in the present study. These values were numerically calculated with Monte Carlo N-Particle Transport Code version 5 (MCNP-5) and ORIGEN-S codes at different neutron energies between 0.1 eV and 1 MeV by using a simplified geometry model.

2. Simulation details

Fig. 1 summarizes simulation outlines for the investigation of the neutron shielding and activation characteristics of concretes containing boron carbide and polyethylene. Ingredients of the shielding concrete are cement, water, sand, gravel, boron carbide, and polyethylene. The Eu, Co, and Cs elements, which emit long-lived radioactive rays in the concrete, were then quantified via the Inductively Coupled Plasma Mass Spectrometry (ICP-MS). Chemical compositions of the materials used in this study were referred to in the Pacific Northwest National Laboratory (PNNL) report [21]. Neutron fluxes were calculated by using the MCNP-5 code in order to estimate the neutron shielding effectiveness of the concrete. Finally, the radioactivity of the concrete was calculated by using the ORIGEN-S code in an effort to evaluate the neutron activation characteristics of the concrete.

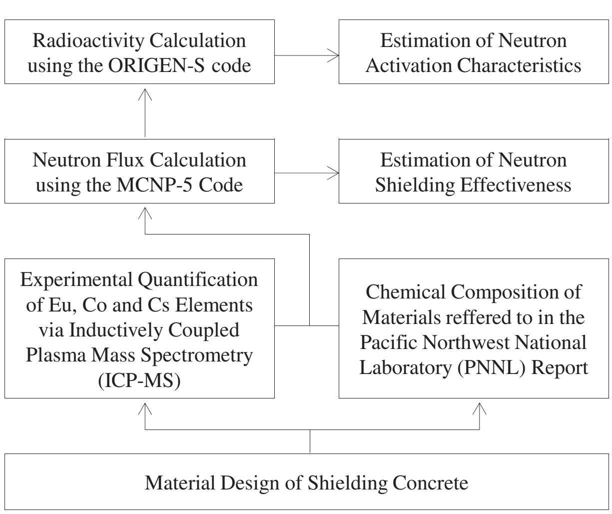

Fig. 1. Outline of simulation details.

2.1. Chemical composition of shielding concrete materials

The chemical compositions of each component, the ordinary concrete, boron carbide and polyethylene, were adopted from the PNNL report [21] as listed in Table 1. The material compositions based on the concrete mixing proportions were considered and simulated with various contents of boron and polyethylene in this study. In this paper, the label 'B1' refers to the mix with 1 wt% boron carbide only, while the labels 'B1PE4', 'B1PE10', 'B1PE20', 'B1PE50' refer to the mixes with 4 wt%, 10 wt%, 20 wt% and 50 wt% of polyethylene aggregate mix with 1 wt% of boron carbide, respectively. Note that substitution rate of the polyethylene aggregate up to 50 wt% of fine aggregate is reported to be acceptable according to the previous research [22]. Since boron can reduce the compressive strength of concrete and have deleterious effects on the setting of cement [13,20], the substituted amount of boron carbide was limited to less than 5 wt%. The boron content of the borated concrete with polyethylene aggregate was fixed at 1 wt% in all cases.

2.2. Neutron transport simulations

The Monte Carlo method is a numerical technique that, unlike other applications, offers numerical solutions to radiation transport problems that are either too complex or impractical to be solved analytically. Particle interactions in material media are assessed statistically and quantities such as energy transferred, interaction position and flight directions are estimated from probability distributions [23]. The MCNP code based on the Monte Carlo methods is generally used to solve radiation transport problems, and is widely used by engineers and researchers in the fields involving radiation shielding issues [2,24]. In the present study, the MCNP-5 [25] and the Evaluated Nuclear Data File B-VI (ENDF/B-VI) continuous energy neutron cross-section library are employed to carry out a series of neutron transport simulations.

The geometry model used for the MCNP calculation is shown in Fig. 2. The shielding concrete was designed as a rectangular parallelepiped with height 10 cm, width 10 cm and depth 30 cm. The rectangular parallelepiped was subdivided along its depth into a series of shorter 'sub-rectangular parallelepipeds' which had a thickness of 2 cm. The source was a monoenergetic neutron point source of energy E, positioned in 5 cm away from the concrete shielding wall along the $-x$ direction and emitting mono-energetic neutrons along the $+x$ direction. In this study, various energy levels (E) of 0.1 eV, 10 eV, 100 eV, 1 KeV, 10 KeV and 1 MeV were considered, with dry air defined for all spaces excluding the presence of the shielding concrete. According to the PNNL report [21], the air density of 0.001205 g/cm^3 was used and chemical compositions of the air was specified. The detector which recorded the average neutron flux was located at every 2 cm depth interval and uncertainties were maintained at less than 1%.

The attenuation efficiency of the concrete was evaluated in terms of the linear attenuation coefficient (μ, cm^{-1}). When a concrete having thickness x is placed in the path of a neutron radiation beam, the neutron intensity is attenuated according to the Beer–Lambert Law,

$$I = I_0 e^{-\mu x} \tag{1}$$

where I and I_0 are the neutron intensities with and without the shielding material, respectively, and x is the material thickness. In addition, the following half-value thickness (cm), where the intensity of the primary beam is reduced by 1/2, was calculated for comparison.

$$\text{Half-Value Thickness} = \frac{\ln 2}{\mu} \tag{2}$$