Simulation of neutron shielding performance of concrete material in different incident directions

Yinghong Zuo¹*, Jinhui Zhu¹, Shengli Niu¹, Peng Shang¹ and Honggang Xie¹

¹Northwest Institute of Nuclear Technology, Xi'an, China

*Corresponding author e-mail: zuoyinghong@nint.ac.cn

Abstract. The calculation model of isotropic neutron shielding by concrete with different thickness had been established to investigate the neutron shielding performance of concrete in different incident directions. The transmission coefficients of fission neutrons and mono-energetic neutrons at energy of 14 MeV penetrating through the concrete in different directions were calculated by using Monte Carlo simulation method. The results show that the neutron attenuation effects in different incident directions for the same concrete shield at a certain thickness are very different, and the attenuation of concrete to neutrons with perpendicular incident direction is weakest, and the attenuation effect on neutrons with oblique incident direction increases with the increase of the angle between the incident direction and the normal of the concrete shield surface; the tenth-value attenuation thickness in which the intensity of primary neutron will be reduced to 1/10 is approximately proportional to the cosine of the angle between the incident direction and the normal of the shield surface, but also related to the energy of neutron.

1. Introduction

Neutron shielding is one of the important research topics in the field of radiation protection, and the parameters of neutron shielding performance of materials are basis for radiation protection and shielding design [1-4]. Concrete is the most commonly used neutron shielding material because it is inexpensive and generous in the construction design. A lot of experiments and numerical simulations have been carried out on the neutron shielding performance of concrete at home and abroad [5-9]. Some of the parameters now available describing the neutron shielding properties of materials, such as neutron attenuation thickness, are given for a unidirectional source, and some are given for an isotropic point source [10-12]. The neutron shielding performance of concrete in different incident directions has not been reported.

For isotropic neutron point source, the angle between the incident direction of the neutron and the normal of the concrete shield surface is not the same at different points on the surface of the concrete shield. When the angle between the incident direction of neutron and the normal of the shield surface is zero degree, the neutron will experience the shortest linear interaction distance in the shield. The distance traveled by the ray in the shield is different when the angle between the incident direction and the normal of the shield surface is different. The attenuation effect of the shield on the incident neutrons perpendicular to the shield is different from that of the oblique incident neutrons. In this paper, a shielding calculation model of isotropic neutron point source incident on concrete of different thickness is established to simulate the neutron shielding performance of concrete in different incident directions. It provides a theoretical reference for radiation protection practice.
2. Physical model and method

The shielding calculation model established is shown in figure 1. The isotropic neutron source is located in front of the concrete shield, and the shield body is a circular plate with the radius \( R \) is 8m, which ensures that only a small part of the oblique incident neutron can transmit from the side of the circular plate. The thickness of the shield is \( d \), and \( d \) is chosen in the range of 0–80cm. The shielding performances of concrete with different thickness to neutrons with different energy in four typical directions were calculated. The angles between these four directions and the normal of shielding body are 0°, 30°, 45° and 60°, respectively.

Figure 1. Schematic diagram of shielding calculation model.

Without considering the difference of the incident angle, the traditional definition of neutron transmission coefficient \( I_t \), which characterizes the shielding performance, is defined as the number of neutrons penetrating through the shield \( N_t \), divided by the neutron number \( N_0 \) at the same exit surface when the shield is not set, i.e. \( I_t = N_t / N_0 \). In order to characterize the shielding performance in different directions of incidence angle, a transmission coefficient related to incidence angle is defined. The calculation method is as follows.

\[
I_{t,\theta} = \frac{\Phi_{\theta}}{\Phi_{\theta,0}}
\]  

Among them, \( \Phi_{\theta} \) is the neutron fluence at the measuring point when the shield is not set in the model, which is mainly related to the distance from the source and the air density, reflecting the geometric attenuation and the scattering and absorption of air; \( \Phi_{\theta,0} \) is the neutron fluence at the same measuring point after the concrete is set in the model.

In order to evaluate the intensity of secondary gamma rays associated with neutron shielding in concrete material, the secondary gamma rays produced by the neutron-concrete interaction in the shielding process due to \((n, \gamma)\) capture reaction is also given. Two types of secondary gamma ray intensity have been given. One is the ratio of the number of secondary gamma rays produced in the shielding process to the number of neutrons penetrated the concrete shield, and the expression is as follows.

\[
p_{\gamma} = \frac{I_{\gamma,\theta}}{I_{\gamma,\theta,0}}
\]

Where, \( p_{\gamma} \) is the ratio of secondary gamma rays to penetrated neutrons; \( I_{\gamma,\theta,0} \) is the number of secondary gamma rays produced in the shielding process; \( I_{\gamma,\theta,0} \) is the penetrated neutrons. The other is directly giving the secondary gamma fluence \( \Phi_{\theta} \) at the measuring point after setting the concrete shielding material.

Monte Carlo simulation is carried out by MCNP program [13]. The neutron and gamma ray fluence at the measured points on the axis are recorded by the F5 card which is a point detector; the fluence in the incident direction with different angles to the normal of the shield surface are recorded by the FZ5
card which is a ring detector; the neutron and gamma ray fluence at the transmission surface of the shield are recorded by the F1 card which is a surface current counter. The number of simulated particles is $1 \times 10^8$, which will make the statistical error of simulation results less than 5%.

The material composition of the concrete used in the simulation is shown in Table 1. The main component of the concrete is SiO$_2$, and it also contains a certain amount of H and C elements.

| Element | H  | C  | O  | Na | Mg | Al | Si | K  | Ca | Fe  |
|---------|----|----|----|----|----|----|----|----|----|-----|
| Wt/ %   | 2.21 | 0.248 | 57.5 | 1.52 | 0.127 | 1.99 | 30.5 | 1.004 | 4.29 | 0.644 |

### 3. Results and discussion

Based on the above model, the shielding performances of concrete for fission neutron and mono-energetic neutron of 14 MeV were calculated. The energy spectrum of fission neutron source obeys the Maxwell distribution [14-15], and the average energy of the fission neutron is 2.12 MeV

#### 3.1. Fission neutron source

Figure 2 is the conventional transmission coefficient of fission neutrons shielding by concrete with different thicknesses. The conventional transmission coefficient of fission neutron is obtained by using F1 card, which reflects the average transmission coefficient of the whole circular plate concrete shield, and it reflects the comprehensive shielding effect, which is an integral result in different incident directions. Figure 2(a) shows that the concrete thicknesses corresponding to the transmission coefficient of 0.1 and 0.01 are 19.2 cm and 36.8 cm, respectively. When the thickness of concrete exceeds 20 cm, the thickness of the concrete required to decay one-tenth of the neutron intensity is about 18.5 cm. Figure 2(b) is the ratio of number of secondary gamma rays emitted during the shielding process to the number of transmitted neutrons. The results show that the ratio increases with the increase of concrete thickness. When the thickness of concrete exceeds 20 cm, the total number of gamma rays emitted during the shielding process exceeds the total number of neutrons transmitted the concrete, that is, $p_\gamma > 1$. With the further increase of thickness, $p_\gamma$ tends to be stable.

**Figure 2.** Shielding performance of concrete with different thickness to fission neutron. (a) The traditional neutron transmission coefficient versus thickness of concrete; (b) The number ratio of the secondary gamma rays produced in the shielding process to the transmitted neutron.

From the calculation results showing in figure 3(a), it can be seen that the attenuation effect of neutron with different incident direction in the concrete shield with same thickness is very different. The thicknesses of concrete required to reduce the fission neutron intensity to 0.1 of the original are about 34.3 cm, 28.9 cm, 22.4 cm and 15.1 cm in the direction of 0°, 30°, 45°, and 60° from the normal of the shield surface, respectively. The thicknesses of concrete required to reduce the neutron intensity to 0.01 of the original are about 55 cm, 48 cm, 39 cm and 29.5 cm in the direction of 0°, 30°, 45°, and 60° from the normal of the shield surface, respectively. In different angles of direction from the...
normal, the thickness of concrete required to attenuate the same neutron intensity approximately satisfies the relationship of $d_\theta = d_0 \cos(\theta)$.

When the thicknesses of concrete are in the range of 20~80 cm, the thickness of concrete required for neutron tenth-value attenuation is 19.8 cm, 18.5 cm, 15.6 cm and 14.7 cm respectively in the directions of 0°, 30°, 45°, and 60°. When neutron is transported in concrete, it mainly interacts with the nucleus of concrete material nuclides. For neutrons with a same energy, the macroscopic cross section of the concrete is the same in different incident directions, but the interaction distance increases with the angle between the incident direction and the normal of the shield. Supposing that the interaction distance in the direction of 0° is $d_0$, the interaction distance in the direction of $\theta$ is $d_0 / \cos(\theta)$, that is, the interaction distances in the directions of 30°, 45°, and 60° are 1.155$d_0$, 1.414$d_0$ and 2$d_0$, respectively. The greater the angle between the incident direction and the normal of the concrete shield, the longer the interaction distance, the better the shielding performance of concrete to neutron.

![Figure 3](image-url)  
*Figure 3. Shielding performance of concrete to fission neutrons in different incident directions. (a) The transmission coefficient of fission neutrons varies with the thickness of concrete; (b) the secondary gamma ray fluence produced during the shielding process.*

Figure 3(b) shows the variation of fluence of secondary gamma ray generated during the neutron shielding process with thickness of concrete in different directions. As can be seen from figure 3 (b), the secondary gamma ray fluence generated in different directions increases firstly and then decreases with the increase of the concrete shield thickness. When the concrete shield is thinner, the number of gamma rays generated is larger than the number of gamma rays absorbed. As the thickness of the concrete shield increasing, the rate of gamma rays produced is lower than that absorbed by the concrete shield. From the absolute amount of secondary gamma rays produced in the neutron shielding process, the number of secondary gamma rays produced in the normal direction of the shield is largest. For the concrete with a same thickness, the larger the angle between the neutron incident direction and the normal of the concrete shield, the smaller the amount of secondary gamma rays produced in the neutron shielding process.

### 3.2. Neutron with energy of 14 MeV

Figure 4 is a simulation result for mono-energetic neutron with energy of 14 MeV. From figure 4(a), it can be seen that the corresponding concrete thickness is 30.2 cm and 55 cm when the neutron transmission coefficient is 0.1 and 0.01, respectively; when the concrete thickness exceeds 30 cm, the thickness of the concrete required for neutron tenth-value attenuation is about 25.7 cm. To achieve the same shielding performance, the thickness of concrete required for neutron with energy of 14 MeV is thicker than that required for fission neutron. Figure 4(b) also shows that the ratio of the number of secondary gamma rays generated in the shielding process to the number of transmitted neutrons increases with the increase of concrete thickness.
Figure 4. Shielding performance of concrete with different thickness to 14MeV neutrons. (a) The variation of the traditional neutron transmission coefficient with thickness of the concrete; (b) the number ratio of secondary gamma ray to transmitted neutron during the shielding process.

From the calculation results shown in figure 5(a), it can be seen that the thicknesses of concrete required to decrease the intensity of neutron with energy of 14 MeV to one-tenth of the original are about 53 cm, 45.1 cm, 36 cm and 24.8 cm in the direction of 0°, 30°, 45°, and 60° from the normal of the shield surface, respectively. When the thickness of concrete is in the range of 20~80 cm, the thickness of concrete required for neutron tenth-value attenuation is 27.6 cm, 24.6 cm, 21.9 cm and 18.3 cm in the directions of 0°, 30°, 45°, and 60°, respectively. Figure 5(b) shows the relationship between the secondary gamma ray fluence generated during shielding process of 14 MeV neutrons and the thickness of the concrete shield. The generated secondary gamma ray decreases with the increase of the thickness of the concrete shield, and the secondary gamma ray fluence decreases most rapidly with the increase of the thickness in the direction of 60° from the normal of the shield surface.

Figure 5. Shielding performance of concrete for 14 MeV neutrons in different directions. (a) The neutron transmission coefficient varies with the thickness of concrete; (b) the secondary gamma ray fluence produced during shielding.

Although the thickness of the concrete shield required for neutron tenth-value attenuation is approximately equal to \(d_0\cos(\theta)\) in different incident direction of angle \(\theta\) from the normal, in which \(d_0\) is the interaction distance between neutron with concrete in the direction of 0°, the shield thickness is also related to the neutron energy due to the complexity of the neutron transportation process. The detailed design of radiation shielding should be based on the results of Monte Carlo simulation. For convenience of application, the above results are listed in table 2 to provide a reference for the application of concrete to be used as neutron shielding material in practice.
Table 2. The tenth-value attenuation thickness of concrete for neutrons with different angles between incident direction and normal of the shield surface.

| Neutron source      | Thickness required for surface current reducing to 1/10 /cm | Angle between incidence and normal       |
|---------------------|-------------------------------------------------------------|------------------------------------------|
|                     |                                                             | 0° | 30° | 45° | 60°                                      |
| Fission neutron     | 19.2                                                        | 19.8 | 18.5 | 15.6 | 14.7                                     |
| 14 MeV              | 30.2                                                        | 27.6 | 24.6 | 21.9 | 18.3                                     |

4. Conclusions

In this paper, the calculation model of isotropic neutron point source shielding by concrete material was established. The transmission coefficients of fission neutron and 14MeV neutron penetrating through a concrete shield with various thicknesses at different incident directions were calculated out by using Monte Carlo simulation method. The results show that the macroscopic total cross section of neutron interacting with concrete is determined for applied neutron energy, but the interaction distances between neutrons and concrete shield material are not equal in different incident directions of neutron, so the neutron attenuation coefficient of shield in different incident directions for the concrete with a same thickness is different. The attenuation of concrete to the neutron with perpendicular incident angle to the shield is weakest, and the attenuation of neutrons with oblique incident angle increases with the increase of the angle between the incident direction and the normal of the shield surface. When the thickness of concrete is in the range of 20–80 cm, the thicknesses of concrete required for tenth-value attenuation of fission neutron are 19.8 cm, 18.5 cm, 15.6 cm and 14.7 cm, respectively, in the direction of 0°, 30°, 45°, and 60°, while the thickness of concrete required for tenth-value attenuation of neutron with energy of 14 MeV are 27.6 cm, 24.6 cm, 21.9 cm and 18.3 cm, respectively. In the practice of radiation protection, fine radiation shielding design can be carried out according to the neutron shielding attenuation performance of concrete in different incident directions.

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