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Calculation of the reactor neutron time of flight spectrum by convolution technique

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It is a very complex and time-consuming process to simulate the nuclear reactor neutron spectrum from the reactor core to the export channel by applying a Monte Carlo program. This paper presents a new method to calculate the neutron spectrum by using the convolution technique which considers the channel transportation as a linear system and the transportation scattering as the response function. It also applies Monte Carlo Neutron and Photon Transport Code (MCNP) to simulate the response function numerically. With the application of convolution technique to calculate the spectrum distribution from the core to the channel, the process is then much more convenient only with the simple numerical integral numeration. This saves computer time and reduces some trouble in re-writing of the MCNP program.

Keywords: convolution technique, neutron transportation, time of flight, response function, time spectrum

PACC: 0762, 2930, 1420C

1. Introduction

Compared with the conventional method, we find that it is an advantageous numerical way to calculate the neutron time of flight[1] spectrum transported from the reactor core to the export channel with the application of convolution technique. This paper presents a new method to calculate the neutron spectrum by using the convolution technique which considers the channel transportation as a linear system and the transportation scattering as the response function. It also applies Monte Carlo Neutron and Photon Transport Code (MCNP) to simulate the response function numerically. Based on the response function, we calculate two typical kinds of neutron source spectra and give the results.

2. Convolution technique theory

2.1. Rudimentary theory of convolution technique

In order to get the response function of the reactor channel transportation system, one adopts the $\delta(t)$ function as neutron source spectrum whose properties are

$$\delta(t)|_{t=0} = \infty, \quad \int_{-\infty}^{+\infty} \delta(t)\mathrm{d}t = 1. \tag{1}$$

Suppose that the response function of the transportation channel is $T(R,t)$ (R: the distance from the core to the measurement point). As we know, the neutrons transporting from the core to the channel are composed of direct and scattering neutrons. The following formulas express the relationships between them:

$$T(R,t) = A(R)\delta(t) + B(R)S(R,t), \tag{2}$$

$$A(R) = \Phi_{\mathrm{D}}(R)/[\Phi_{\mathrm{D}}(R) + \Phi_{\mathrm{S}}(R)], \tag{3}$$

$$B(R) = 1 - A(R), \tag{4}$$

where $A(R)$ is the direct neutron share; $B(R)$ is the scattering neutron share; $S(R,t)$ is the scattering function; $\Phi_{\mathrm{D}}(R)$ is the relative direct neutron flux; $\Phi_{\mathrm{S}}(R)$ is the measurement point total scattering neutron flux.

$$\int_{-\infty}^{+\infty} T(R,t)\mathrm{d}t = 1. \tag{5}$$

As per Eqs.(2) and (5), we have

$$\int_{-\infty}^{+\infty} [A(R)\delta(t) + B(R)S(R,t)]\mathrm{d}t = 1. \tag{6}$$

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2.2. Calculation of the relative parameters

2.2.1. Transportation channel model

In order to calculate parameters in the anterior formulae, it is necessary to do some simulations with MCNP whose input file adopts the $\delta(0)$ function as time spectrum. The model of the neutron transportation is indicated in Fig.1.

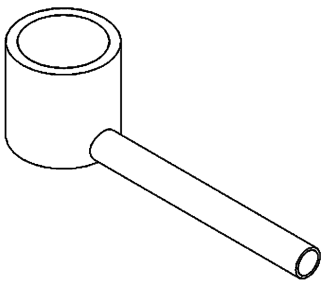


Fig.1. Model of the reactor neutron source transportation channel.

The model considers the reactor core as the Descartes coordinate system origin and the length of the channel is about 50 m. The parameters of the model are as follows: reactor core radius is 2.5 m; inner radius is 20 cm; outer radius is 50 cm. The channel is made of concrete and on the radial direction.

2.2.2. Transportation medium model

When the neutrons transport to the channel, there will be some reflections back to the core. At the same time, part of the neutrons reflecting through and through in the channel continue to transport. The atmosphere in the channel used for the calculations has an uniform, length-independent composition of 79.1 percent nitrogen, 20.4 percent oxygen, and 0.48 percent argon. These values are similar to the standard composition given in Ref.[2], which is 78.08 N, 20.95 O, 0.93 Ar, and 0.034 percent trace elements.

2.2.3. Calculation of the parameters

Basing on the space and medium models, we compile the MCNP program and get the relative parameters

$$\Phi_{\rm t}(R = 50\,{\rm m}) = 2.21555 \times 10^{-8},$$

where $\Phi_{\rm t} = \Phi_{\rm S} + \Phi_{\rm D}$ means the total neutron flux;

$$\Phi_{\rm D}(R = 50\,{\rm m}) = 1.30720 \times 10^{-8};$$

as per Eqs.(3) and (4),

$$A(R = 50\,{\rm m}) = 0.59, \quad B(R = 50\,{\rm m}) = 0.41.$$

The relationship between direct and scattering neutrons flux is illustrated in Fig.2.

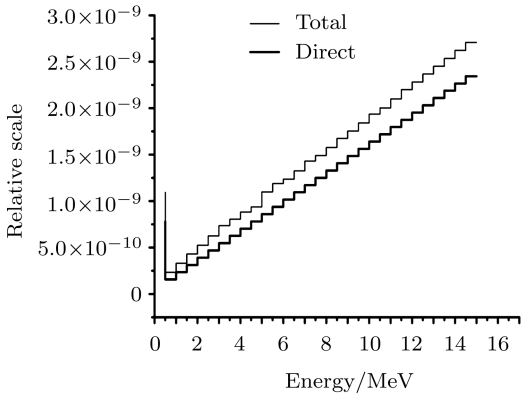


Fig.2. The relationship between direct and scattering neutrons flux at different energy points.

2.3. Calculation of the scattering function

As per the definition of the scattering function from Ref.[3], we has the following equation:

$$S(R,t) = \frac{\Phi_{{\rm S}j}(t_j, R)}{\Delta t_j\, \Phi_t(R)}, \tag{7}$$

where $\Phi_{{\rm S}j}(t_j, R)$ means the scattering neutron flux reached R point at t_j time; $\Phi_{\rm t}(R)$ means the total neutron flux at R point; Δt_j means the time step for MCNP input file. According to the numerical calculation, the scattering function is shown in Fig.3.

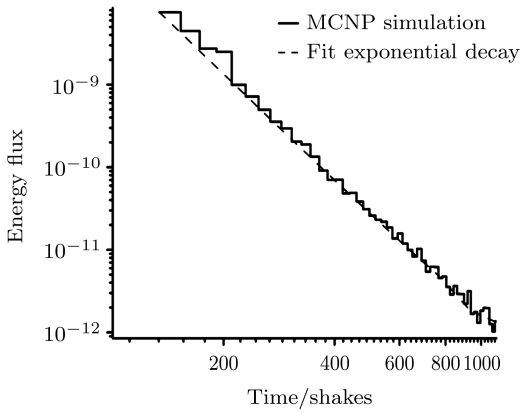


Fig.3. The scattering function of the transportation channel.

Referring to Fig.3 we can give the fit function of the scattering function.

$$S(R = 50\,\mathrm{m}, t) = (1.93347 \cdot 10^{-12} + 1.7511 \cdot 10^{-7} \cdot \mathrm{e}^{-t/42.3}). \quad (8)$$

2.4. Calculation of the response function

Considering Eqs.(2) and (8) as simultaneous equations, we get the response function expression:

$$T(R = 50\,\mathrm{m}, t) = 0.59\delta(t) + 0.41(1.93347 \cdot 10^{-12} + 1.7511 \cdot 10^{-7} \cdot \mathrm{e}^{-t/42.3}). \quad (9)$$

3. Convolution calculation of the measurement point neutron spectrum

3.1. Convolution calculation

In order to testify the feasibility of convolution technique, we adopt the positive exponential pulse and non-symmetrical kurtosis neutron source spectra and give the time spectra of the measurement point. Let the neutron source have the primary time spectrum $I(0, t)$; based on the response function, we deduce the time spectrum of the measurement point:

$$I(R, t) = \int_{-\infty}^{t} I(0, t') \cdot T(R, t - t')\mathrm{d}t'. \quad (10)$$

Combining Eq.(9) with Eq.(10), we get

$$I(R, t) = A(R)I(0, t) + B(R) \times \int_{-\infty}^{t} I(0, t')S(R, t - t')\mathrm{d}t'. \quad (11)$$

Equation (11) shows that the time spectrum of the measurement point is composed of two parts, the former is the attenuation of the source spectrum, and the latter is the attenuation of extent and the broad of the time breadth.

3.2. The positive exponential pulse

Let us suppose that the positive exponential pulse has the following expression:

$$I(0, t) = C_0\mathrm{e}^{at}. \quad (12)$$

As per Eq.(13),

$$I(R = 50\,\mathrm{m}, t) = (0.59C_0 + 0.7927 \times 10^{-12}C_0/a)\mathrm{e}^{at} + [3.0369 \times 10^{-6}C_0/(42.3a + 1)]\mathrm{e}^{\left(a + \frac{1}{42.3}\right)t}.$$

Let $a = 0.1$, $C_0 = 1$, the time spectra of source point and measurement point are sketched in Fig.4.

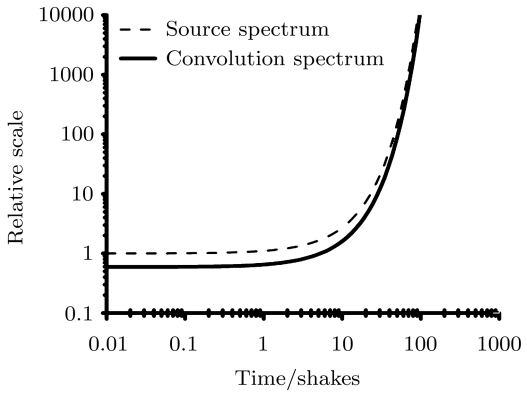


Fig.4. The relationship between source point and measurement point time spectrum.

Figure 4 shows that the measurement point time spectrum has an attenuation of extent and change of the time breadth.

3.3. The non-symmetrical kurtosis

Let us suppose that the non-symmetrical kurtosis pulse has the following expression:

$$I(0, t) = 1.87(\mathrm{e}^{-t/10} - \mathrm{e}^{-t/2}). \quad (13)$$

Similarly to Eq.(12), we can deduce the time spectrum of measurement point as follows:

$$I(R = 50\,\mathrm{m}, t) = 1.1033\mathrm{e}^{-t/10} + 2.9648\mathrm{e}^{-t/2} - 1.7573 \times 10^{-6}\mathrm{e}^{-0.0764t} - 2.8182\mathrm{e}^{-0.4764t}.$$

The time spectra of source point and measurement point are sketched in Fig.5.

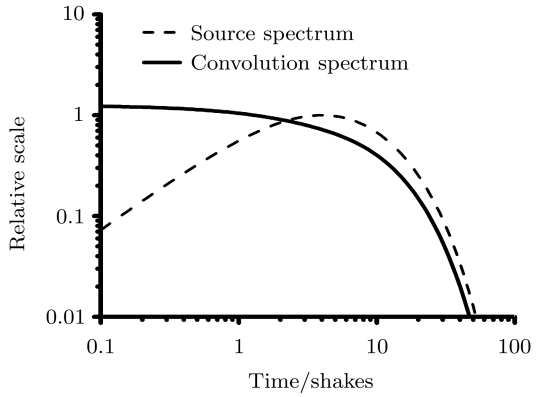


Fig.5. The relationship between source point and measurement point time spectrum.

Figure 5 shows that the measurement point time spectrum has an attenuation of extent and broad of the time breadth.

4. Conclusion

It is an advantageous numerical way to calculate the neutron spectrum transported from the reactor core to the export channel with the application of convolution technique, especially on the changes of source item and immobility of the transportation conditions. In brief, it simulates the response function with MCNP and bases on the source item to get the numerical integral. It is a good idea to calculate the neutron spectrum by applying the convolution and to choose the reasonable detector by forecast of the time of flight spectrum.

References

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