Annals of Nuclear Energy 115 (2018) 494–501

Contents lists available at ScienceDirect

Annals of Nuclear Energy

journal homepage: www.elsevier.com/locate/anucene

Implementation of a frequency-domain neutron noise analysis method in a production-level continuous energy Monte Carlo code: Verification and application in a BWR

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ARTICLE INFO

Article history:
Received 21 November 2017
Received in revised form 31 January 2018
Accepted 2 February 2018

Keywords:
Monte Carlo
Neutron noise
Frequency domain
BWR

ABSTRACT

A Monte Carlo algorithm for solving the transport equation of the neutron noise in the frequency domain has been newly implemented into a production-level continuous energy Monte Carlo code, MCNP. Using the continuous energy Monte Carlo code, accurate neutron noise calculations can be performed with fewer approximations. The implemented algorithm is based on a method that was previously developed by the author of this paper. The modified code is currently applicable to neutron noise calculations only within the plateau frequency range. The modified code is applied to the neutron noise calculations in a one-dimensional homogeneous multiplying system to verify its effectiveness through comparison with a two-energy group in-house research-purpose code. The neutron noise calculations for a benchmark model of a BWR core are performed using the modified MCNP code. The spatial and frequency characteristics of the neutron noise propagation in the BWR core are investigated through the calculations. The neutron noise near the noise source differs largely from the point kinetics because of the higher order mode effect.

1. Introduction

Measuring neutron noise (i.e., the difference between a time-varying neutron flux and its steady state value) in a reactor core is an effective means for core monitoring or diagnoses, such as an unseated fuel assembly, abnormal vibrations of the internal core structures, or flow blockage (Jonsson et al., 2012; Viebach et al., 2017). Research activities on the core monitoring technique are being pursued by many institutions. Recently, the CORTEX project, a research and innovation action aiming at developing an innovative core monitoring technique using the fluctuations in neutron flux, started in the Euratom 2016–2017 work program (Demazière et al., 2017). The reconstruction or unfolding of the noise source is an important process to identify the anomalies occurring within the reactor core. Various reconstruction or unfolding techniques have been developed to date (Glöckler and Pázsit, 1987; Demazière and Andhill, 2005; Hosseini and Vosoughi, 2013; Hosseini and Vosoughi, 2014). Regardless of the technique used for the reconstruction or unfolding process, the calculation of the neutron noise at the detector positions caused by an anticipated noise source is required to solve an inversion problem to identify the anomalies in a core. The response of the neutron noise induced by a noise source can be calculated by an analytical formula (Behringer et al., 1977; Pázsit and Analytis, 1980; Jonsson et al., 2012) or by solving the neutron noise diffusion or transport equation in the frequency domain. Thus, developing a noise calculation method with high accuracy is an important task towards enhancing nuclear safety through the use of reactor noise diagnostics.

A variety of calculation tools for obtaining the neutron noise responses have been developed thus far; they are mostly based on diffusion theory (Demazière, 2004; Demazière and Pázsit, 2009; Pázsit and Demazière, 2010; Larsson et al., 2011; Demazière, 2011; Larsson and Demazière, 2012a,b; Hosseini and Vosoughi, 2012). The extension of the noise calculation to transport theory was attempted by (Yamamoto, 2013) and (Rouchon et al., 2017). In these studies involving transport theory, the neutron noise transport equation was solved using the Monte Carlo method. In (Yamamoto, 2013), the neutron noise distributions were calculated both using diffusion theory and using the Monte Carlo method. The comparison highlighted the unsatisfactory results of diffusion theory, particularly near the noise source position, suggesting the necessity of introducing the transport method for neutron noise calculations. Unlike the diffusion equation for neutron noise, which can be solved using the conventional numerical techniques, the Monte Carlo method is confronted with some

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https://doi.org/10.1016/j.anucene.2018.02.008
0306-4549/

difficulties, especially for low- and high-frequency noise. If a conventional Monte Carlo algorithm (i.e., implicit capture and Russian roulette, etc.) is employed, a very large number of particles having positive or negative weights are produced for low- and high-frequency noise, resulting in abnormal termination of the computation. The positive and negative weights need to be canceled to prevent the exploding number of particles, but a special technique must be introduced for the cancellation because no two particles exist at the same position. In contrast, the difficulty of the exploding number of particles does not occur for the plateau frequency range (Rouchon et al., 2017), which is approximately between 0.01 Hz ($\approx\lambda/(2\pi)$) and 40 Hz ($\approx(\lambda + \beta/\Lambda)/(2\pi)$) for light water reactors. To circumvent this difficulty that occurs outside the plateau frequency region, Yamamoto (2013) and Yamamoto and Sakamoto (2015) introduced a weight cancellation technique and a power iteration-like algorithm. However, the countermeasures for managing the exploding number of particles demand that the whole calculation domain should be discretized into many small subdomains, where positive and negative particle weights are cancelled. Installing the weight cancellation technique into a production-level Monte Carlo code would cause the code to be less versatile. Thus, the numerical application in (Yamamoto, 2013) was limited to a homogeneous simple geometry that was modeled using an in-house research-purpose code. Rouchon et al. (2017) developed a new Monte Carlo method that does not require the weight cancellation technique. The new method removes the implicit capture technique and adds a pseudo total cross section to the neutron noise equation.

The advantage of introducing the continuous energy Monte Carlo method for neutron noise calculations is that the Monte Carlo method can model the three-dimensional fine structures in a power reactor core without spatial homogenization and it is free from the inaccuracy involved in energy group collapsing. The implementation of the frequency domain Monte Carlo method into a production-level continuous energy Monte Carlo code is important in terms of expanding the availability of the method. Remarkable improvement in the accuracy of the neutron noise calculation can be expected by using the continuous energy Monte Carlo code. Neutron noise Monte Carlo calculations in the frequency domain have thus far been performed using an in-house research-purpose code. Thus, to expand the implementation of the Monte Carlo method in neutron noise calculations, in this paper, the frequency domain Monte Carlo method is implemented into a production-level continuous energy Monte Carlo code, MCNP 4C (Briesmeister, 2000). The Monte Carlo technique adopted in this paper is based on the previously developed complex-valued weight Monte Carlo method in (Yamamoto, 2012; Yamamoto, 2013). The method used in this paper is available only for the plateau frequency range, where the number of particles can be controlled without introducing a new method. The implementation of the methods developed in (Rouchon et al., 2017) or some other useful techniques will be left as a future work.

As the first step, the verification of the modified MCNP code is performed by comparing the code with the in-house research-purpose Monte Carlo code (Yamamoto, 2013) for a very simple multiplying system. Next, neutron noise transport calculations are performed for a BWR benchmark model. The properties of the noise propagation within the BWR core structure are discussed.

2. Neutron noise equation in the transport theory

The fundamental theory on the propagation of neutron noise in the frequency domain has been presented in previously published literature reports. Within the diffusion approximation, the neutron noise propagation has been discussed in many publications (Behringer et al., 1977; Behringer et al., 1979; Demazière, 2004; Demazière and Pázsit, 2009; Larsson et al., 2011; Demazière, 2011; Larsson and Demazière, 2012a,b). Transport theory has been recently introduced in some papers (Yamamoto, 2013; Yamamoto and Sakamoto, 2015; Rouchon et al., 2017). In this section, the transport equation to be solved for neutron noise propagation in a reactor core is only briefly explained.

The neutron noise in the time domain is defined by

$$\delta\phi(\mathbf{r},\boldsymbol{\Omega},E,t) \equiv \phi(\mathbf{r},\boldsymbol{\Omega},E,t) - \phi_0(\mathbf{r},\boldsymbol{\Omega},E) \tag{1}$$

where $\phi(\mathbf{r},\boldsymbol{\Omega},E,t)$ = the neutron flux at position $\mathbf{r}$ with energy E and direction $\boldsymbol{\Omega}$ at time t, and the subscript "0" denotes the mean value. The neutron noise is induced by the fluctuation of the macroscopic cross sections around their mean value:

$$\delta\Sigma_x(\mathbf{r},E,t) \equiv \Sigma_x(\mathbf{r},E,t) - \Sigma_{x0}(\mathbf{r},E) \tag{2}$$

where x = t, s, or f. The neutron noise in the frequency domain is obtained by Fourier transforming the neutron noise in the time domain:

$$\delta\phi(\mathbf{r},\boldsymbol{\Omega},E,\omega) \equiv \int_{-\infty}^{+\infty} \delta\phi(\mathbf{r},\boldsymbol{\Omega},E,t)e^{-i\omega t}dt \tag{3}$$

where ω = the angular frequency and $i = \sqrt{-1}$. The neutron noise transport equation in the frequency domain is derived from the time-dependent neutron transport equation and the equation for the delayed neutron precursor density using a linear approximation and a Fourier transformation. The equation finally obtained is shown below:

$$\begin{aligned}
&\boldsymbol{\Omega}\cdot\nabla\delta\phi(\mathbf{r},\boldsymbol{\Omega},E,\omega) + \Sigma_{t0}(\mathbf{r},E)\delta\phi(\mathbf{r},\boldsymbol{\Omega},E,\omega) \\
&\quad = \int_{4\pi} d\boldsymbol{\Omega}' \int dE'\Sigma_{s0}(\mathbf{r},\boldsymbol{\Omega}'\to\boldsymbol{\Omega},E'\to E)\delta\phi(\mathbf{r},\boldsymbol{\Omega}',E',\omega) \\
&\qquad + \frac{\chi(E)}{4\pi k_{eff}}\left(1-\frac{i\omega\beta}{i\omega+\lambda}\right)\int_{4\pi} d\boldsymbol{\Omega}' \int dE'\nu\Sigma_{f0}(\mathbf{r},E')\delta\phi(\mathbf{r},\boldsymbol{\Omega}',E',\omega) \\
&\qquad - \frac{i\omega}{v(E)}\delta\phi(\mathbf{r},\boldsymbol{\Omega},E,\omega) + S(\mathbf{r},\boldsymbol{\Omega},E,\omega)
\end{aligned} \tag{4}$$

where Σ_t = the macroscopic total cross section, Σ_s = the macroscopic scattering cross section, Σ_f = the macroscopic fission cross section, χ = the fission neutron spectrum, ν = the number of neutrons per fission, v = the neutron speed, β = the fraction of the delayed neutrons, λ = the time decay constant of the delayed neutron precursors, and S = the noise source. For simplicity, the fission neutron spectrum of the delayed neutrons is assumed to be the same as that of the prompt neutrons. Furthermore, one delayed neutron group is assumed. The noise source, which is induced by the temporal fluctuations of the macroscopic cross sections, is defined by

$$\begin{aligned}
S(\mathbf{r},\boldsymbol{\Omega},E,\omega) \equiv& -\delta\Sigma_t(\mathbf{r},E,\omega)\phi_0(\mathbf{r},\boldsymbol{\Omega},E) \\
&+ \int_{4\pi} d\boldsymbol{\Omega}' \int dE'\delta\Sigma_s(\mathbf{r},\boldsymbol{\Omega}'\to\boldsymbol{\Omega},E'\to E,\omega)\phi_0(\mathbf{r},\boldsymbol{\Omega}',E') \\
&+ \frac{\chi(E)}{4\pi k_{eff}}\left(1-\frac{i\omega\beta}{i\omega+\lambda}\right)\int_{4\pi} d\boldsymbol{\Omega}' \\
&\times \int dE'\nu\delta\Sigma_f(\mathbf{r},E',\omega)\phi_0(\mathbf{r},\boldsymbol{\Omega}',E'),
\end{aligned} \tag{5}$$

where $\delta\Sigma_x(\mathbf{r},E,\omega)$ is the Fourier transform of the fluctuation of the reaction x:

$$\delta\Sigma_x(\mathbf{r},E,\omega) \equiv \int_{-\infty}^{+\infty} \delta\Sigma_x(\mathbf{r},E,t)e^{-i\omega t}dt. \tag{6}$$

In most cases, our interest is focused on a neutron noise propagation in a critical state reactor core. Thus, the neutron production terms in Eqs. (4) and (5) are divided by k_{eff} to eliminate the bias in the calculated k_{eff} that usually deviates from unity.