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Original Article

Adaptive group of ink drop spread: a computer code to unfold neutron noise sources in reactor cores

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ABSTRACT

The present paper reports the development of a computational code based on the Adaptive Group of Ink Drop Spread (AGIDS) for reconstruction of the neutron noise sources in reactor cores. AGIDS algorithm was developed as a fuzzy inference system based on the active learning method. The main idea of the active learning method is to break a multiple input–single output system into a single input–single output system. This leads to the ability to simulate a large system with high accuracy. In the present study, vibrating absorber-type neutron noise source in an International Atomic Energy Agency-two dimensional reactor core is considered in neutron noise calculation. The neutron noise distribution in the detectors was calculated using the Galerkin finite element method. Linear approximation of the shape function in each triangle element was used in the Galerkin finite element method. Both the real and imaginary parts of the calculated neutron distribution of the detectors were considered input data in the developed computational code based on AGIDS. The output of the computational code is the strength, frequency, and position (*X* and *Y* coordinates) of the neutron noise sources. The calculated fraction of variance unexplained error for output parameters including strength, frequency, and *X* and *Y* coordinates of the considered neutron noise sources were 0.002682 #/cm^{3}s, 0.002682 Hz, and 0.004254 cm and 0.006140 cm, respectively.

1. Introduction

In the safety analysis of reactor cores, it is very important to recognize the symptoms of an impending accident that may lead to an accident in the reactor core. One of the methods of identification of such symptoms is neutron noise analysis, in which fluctuations are identified through the neutron noise recorded in detectors. These fluctuations may be induced by small variations of the absorption, scattering, or fission cross sections of the materials in the reactor core. Neutron noise is the result of small variations of the neutron flux distribution due to the mentioned fluctuations. The obtained neutron noise distribution of the detectors in the reactor core may be utilized for reconstruction of neutron noise sources. Diagnosis of neutron noise sources such as control rod vibrations via neutron noise analysis methods was the subject of a number of prior studies and experiments. Various algorithms such as inversion, zoning, and scanning have been used for identification and localization of neutron noise sources such as unseated fuel assemblies in the reactor core, absorbers of variable strength, or vibration of core internals in pressurized water reactors [1–3]. The unfolding of a neutron noise source through the inverse method includes the solution of the inverse problem, in which the coefficient matrix is usually singular or badly scaled. The direct solution of the inverse problem leads to results with low accuracy (the accuracy of the localization of the neutron noise source is approximately 15 cm). Because just a limited number of the detectors are present in the reactor core, data on neutron noise are not available at all considered points (meshes) when using the inverse method. Therefore, to match the size of the measured neutron noise distribution and the calculated coefficient matrix in the inverse problem, interpolation is performed to obtain neutron noise values at all considered meshes. This interpolation leads to more error in the unfolding of the neutron noise source. The zoning method is another algorithm that may be used to unfold neutron noise source. In the zoning method, the reactor core is divided into certain zones, and the inverse method is used for determination of the neutron noise source in each zone. A comparison between the

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reconstructed neutron noise sources in each zone gives the actual neutron noise source. The error induced from the interpolation is slightly less than that when using the inverse method; however, the accuracy of the unfolded neutron noise source when using the zoning method is low. The scanning method is developed based on using a comparison of the neutron noise recorded in the detectors and the calculated responses (neutron noise) due to the possible locations of the neutron noise source in the reactor core. Scanning of all possible locations of the neutron noise source and minimizing of the difference between the detector readings and the calculated neutron noise due to possible sources gives the actual neutron noise source. Dividing the reactor core into certain zones (in the zoning method) and scanning of the reactor core (in the scanning method) lead to high running time (a few hours) for the reconstruction of the neutron noise source [1,2]. An artificial neural network (ANN) is another approach that may be used to unfold noise sources with acceptable accuracy (accuracy of the localization of the neutron noise source is between 5 cm and 10 cm) [4]. This is a mathematical algorithm inspired by biological neural networks. In the reported studies, using the developed algorithm based on ANNs, the neutron noise source was localized with accuracy close to 10 cm [5,6]. In the mentioned studies, the neutron noise sources of a type of absorber of variable strength or of a vibrating absorber were localized simply using the neural network, without the identification of other characteristics of the neutron noise source such as the strength and frequency. In a paper previously published by the first author of the present paper [4], computational codes developed based on the ANN were used for reconstruction of noise sources (all characteristics of neutron noise source) for types of absorber of variable strength and for a vibrating absorber with good accuracy (accuracy of 0.1–10 cm in the localization of the neutron noise source). A literature review of the studies performed on noise source unfolding shows that a neural network or a combination of a neural network and the scanning method are more accurate in comparison to the other aforementioned algorithm [4].

In the present study, a new algorithm based on the Adaptive Group of Ink Drop Spread (AGIDS) is proposed to unfold, with high accuracy, the noise source of a type of vibrating absorber in the IAEA (International Atomic Energy Agency)- two dimensional (2D) reactor core (accuracy between 0.001 cm and 9 cm in the localization of the neutron noise source). The input data (neutron noise distribution in the detectors) of the developed computer code have been calculated using the previously developed DYN-FEMG computational code [7].

An outline of the remainder of the present paper is as follows. In Section 2, we briefly introduce the mathematical formulation used for the calculation of the neutron noise distribution in the reactor core. The main specifications of the IAEA-2D reactor core are presented in Section 3. In Section 4, the developed computational code based on AGIDS and the unfolded neutron noise source using the mentioned computer code are presented. The results of the neutron noise calculation and a reconstruction of the neutron noise source is presented in Section 5. A discussion of the results and the merits of the proposed method is presented in Section 6. Finally, Section 7 gives the concluding remarks.

2. Methodology for simulation of neutron noise distribution

In the present study, a first-order approximation of the neutron noise diffusion equation in two energy groups is considered to calculate the neutron noise distribution due to the neutron noise source. The general form of the mentioned equation, obtained by considering the neutron noise source as fluctuations in the scattering, absorption, and fission macroscopic cross sections, is presented as Eq. (1) [1–4,8]:

$$\begin{aligned}&\left[\nabla.\overline{\overline{D}}(\overline{r})\nabla+\overline{\overline{\Sigma}}_{\mathrm{dyn}}(\overline{r},\omega)\right]\times\begin{bmatrix}\delta\varphi_1(\overline{r},\omega)\\ \delta\varphi_2(\overline{r},\omega)\end{bmatrix}=\\&\overline{\varphi}_{s,1\to2}(\overline{r})\,\delta\Sigma_{s,1\to2}(\overline{r},\omega)+\overline{\overline{\varphi}}_a(\overline{r})\begin{bmatrix}\delta\Sigma_{a,1}(\overline{r},\omega)\\ \delta\Sigma_{a,2}(\overline{r},\omega)\end{bmatrix}+\\&\overline{\overline{\varphi}}_f(\overline{r},\omega)\begin{bmatrix}\delta\nu_1\Sigma_{f,1}(\overline{r},\omega)\\ \delta\nu_2\Sigma_{f,2}(\overline{r},\omega)\end{bmatrix},\end{aligned}\tag{1}$$

where all quantities are defined as usual and the matrices and vectors are expressed as Eqs. (2–5):

$$\overline{\overline{\Sigma}}_{\mathrm{dyn}}(\overline{r},\omega)=\begin{bmatrix}-\Sigma_1(\overline{r},\omega) & \dfrac{\nu_2\Sigma_{f,2}(\overline{r})}{k_{\mathrm{eff}}}\left(1-\dfrac{i\omega\beta_{\mathrm{eff}}}{i\omega+\lambda}\right)\\ \Sigma_{s,1\to2}(\overline{r}) & -\left(\Sigma_{a,2}(\overline{r})+\dfrac{i\omega}{\nu_2}\right)\end{bmatrix}\tag{2}$$

$$\overline{\varphi}_{s,1\to2}(\overline{r})=\begin{bmatrix}\varphi_1(\overline{r})\\ -\varphi_1(\overline{r})\end{bmatrix}\tag{3}$$

$$\overline{\overline{\varphi}}_a(\overline{r})=\begin{bmatrix}\varphi_1(\overline{r}) & 0\\ 0 & \varphi_2(\overline{r})\end{bmatrix}\tag{4}$$

$$\overline{\overline{\varphi}}_f(\overline{r},\omega)=\begin{bmatrix}-\varphi_1(\overline{r})\left(1-\dfrac{i\omega\beta_{\mathrm{eff}}}{i\omega+\lambda}\right) & -\varphi_2(\overline{r})\left(1-\dfrac{i\omega\beta_{\mathrm{eff}}}{i\omega+\lambda}\right)\\ 0 & 0\end{bmatrix}\tag{5}$$

The coefficient $\Sigma_1(\overline{r},\omega)$ used in Eq. (2) is defined as Eq. (6):

$$\Sigma_1(\overline{r},\omega)=\Sigma_{r,1}(\overline{r})+\frac{i\omega}{\nu_1}-\frac{\nu_1\Sigma_{f,1}(\overline{r})}{k_{\mathrm{eff}}}\left(1-\frac{i\omega\beta_{\mathrm{eff}}}{i\omega+\lambda}\right)\tag{6}$$

To calculate the neutron noise source term on the right-hand side of Eq. (1) (Eqs. 3–5), the neutron flux distribution should be calculated from the solution of the neutron diffusion equation. To this end, the neutron flux distribution obtained from the previously developed computational code is used to calculate the neutron noise source term [7]. In the present study, a vibrating absorber type neutron noise source is assumed. The Green function technique is used [2] to calculate the neutron noise distribution, in which the neutron noise distribution due to the unit value of the point noise source in the reactor core is calculated. The point source may be located in any considered triangle element. Therefore, the Green components due to different positions of the unit value-point noise sources are calculated via the solution of Eq. (7):

$$\begin{aligned}&\left[\nabla.\overline{\overline{D}}(\overline{r})\nabla+\overline{\overline{\Sigma}}_{\mathrm{dyn}}(\overline{r},\omega)\right]\times\begin{bmatrix}G_{g\to1}\left(\overline{r},\overline{r'},\omega\right)\\ G_{g\to2}\left(\overline{r},\overline{r'},\omega\right)\end{bmatrix}=\\&\begin{bmatrix}\delta\left(\overline{r}-\overline{r'}\right)\\ 0\end{bmatrix}_{g=1}\ or\ \begin{bmatrix}0\\ \delta\left(\overline{r}-\overline{r'}\right)\end{bmatrix}_{g=2}\end{aligned}\tag{7}$$

where $G_{g\to1}(\overline{r},\overline{r'},\omega)$ and $G_{g\to2}(\overline{r},\overline{r'},\omega)$ are the Green function components of energy groups 1 and 2 at position $\overline{r}$, induced by the noise source in group g located at position $\overline{r'}$. It is possible to consider the neutron noise source in the fast or thermal energy group.

If the noise source is considered to be in the thermal energy group (a perturbation in the thermal macroscopic cross section), Eq. (7) can be written as Eqs. (8) and (9) using the Galerkin finite element method (GFEM):