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Neutron noise source reconstruction using the Adaptive Neuro-Fuzzy Inference System (ANFIS) in the VVER-1000 reactor core

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ABSTRACT

The neutron noise is defined as the stationary fluctuation of the neutron flux around its mean value due to the induced perturbation in the reactor core. The neutron noise analysis may be useful in many applications like noise source reconstruction. To identify the noise source, calculated neutron noise distribution of the detectors is used as input data by the considered unfolding algorithm. The neutron noise distribution of the VVER-1000 reactor core is calculated using the developed computational code based on Galerkin Finite Element Method (GFEM). The noise source of type absorber of variable strength is considered in the calculation. The computational code developed based on An Adaptive Neuro-Fuzzy Inference System (ANFIS) is used to unfold the neutron noise source. Complex neutron noise distribution (real and imaginary parts) in the detectors is considered as input data onto the developed computational code based on the ANFIS algorithm. All the characteristics of the neutron noise source, including strength, frequency and position (X and Y coordinates) are unfolded with excellent accuracy using the developed computational code.

1. Introduction

The neutron noise is the stationary fluctuation of the neutron flux around its mean value due to the induced perturbation in the reactor core. The noise source is considered as perturbation in material macroscopic cross section due to agents like absorber of variable strength or vibrating control rod in the reactor core. If the noise source is not identified well-timed, it may lead to a crucial event in the reactor core. From reactor safety analysis point of view, it is very important to recognize noise sources timely. The signal or neutron noise recorded by available detectors in the reactor core is usually used to reconstruct noise source. Diagnostic of neutron noise sources like the control rod vibrations via neutron noise analysis methods were the subject of a number of prior studies and experiments (Hosseini and Vosoughi, 2014; Itoh, 1986; Pázsit and Glöckler, 1983, 1984; Williams, 2013). Different methods such as inversion, zoning and scanning were used for identification and localization of various types of noise sources like the unseated fuel assemblies in reactor core, absorber of variable strength or vibrations of core internals in PWRs (Demazière and Andhill, 2005; Pázsit and Glöckler, 1984; Williams, 2013). The purpose from the noise source unfolding calculation is the solution of the inverse problem using different techniques. Since the matrix in the mentioned problems is usually singular or badly-scaled, the results obtained from the direct solution to the inverse problem is not so accurate. The other usual methods like the scanning and zoning require approximately high computational cost for noise source unfolding (Demazière and Andhill, 2005; Pázsit and Glöckler, 1984). Artificial Neural Network (ANN) is another approach that unfolds noise sources with high accuracy and approximately needs high computational cost (Hosseini and Vosoughi, 2014). It is the mathematical algorithm inspired by biological neural networks. In our previous published paper, the developed computational codes based on the artificial neural network and its coupling with the scanning method were used to unfold noise sources of types the absorber of variable strength and vibrating absorber. The mentioned developed computational code unfolds noise source with high accuracy. The drawback of the mentioned algorithms (ANN and scanning) is that they need high running and computational cost. The literature review of the works performed in the noise source unfolding field of study shows that the neural network or combination of neural network and the scanning method is more robust in comparison to other

Abbreviations: FEM, Finite Element Method; $\delta\phi(r,\omega)$, Neutron noise; ∇, The nabla operator; ANFIS, An Adaptive Neuro-Fuzzy Inference System.

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aforementioned algorithms. In summary, the artificial neural network is proper selection to reconstruct the instability and vibration localization (Garis et al., 1998; Tambouratzis and Antonopoulos-Domis, 2002). In the mentioned studies, the neutron noise sources of type absorber of variable strength or vibrating absorber were just localized using the neural network without any attempt to reconstruct the noise source strength and/or its frequency. In the previous published paper using ANN, detailed information about noise source characteristics including, position, frequency and its strength was reconstructed with good accuracy (Hosseini and Vosoughi, 2014). The developed dynamic simulator (DYN-FEMG) (Hosseini and Vosoughi, 2012) is used for neutron noise calculations. In the present study, a new algorithm based on Adaptive Neuro-Fuzzy Inference System (ANFIS) is proposed to unfold the noise source of type absorber of variable strength in the VVER-1000 reactor core. Because of some merits of the proposed algorithm like high accuracy of the results and need for low computational cost, the developed computational code based on the ANFIS may be introduced as a reliable tool for neutron noise identification in reactor cores.

An outline of remainder of the present paper is as follows: In Section 2, we briefly introduce mathematical formulation used for the neutron noise calculation in the reactor core. The main specifications of the VVER-1000 reactor core is presented in Section 3. The neutron noise distribution in the reactor core obtained from the calculation is presented in Section 4. In Section 5, the developed computational code based on the ANFIS algorithm and the unfolded noise source using the mentioned computational code are presented. A discussion on the results and the merits of the proposed method is presented in Section 6. Finally, Section 7 gives the concluding remarks.

2. Neutron noise calculation

In the present study, the first order approximation of the neutron noise diffusion equation in 2-energy group is considered to calculate neutron noise distribution. The general form of the mentioned equation by considering the noise source as fluctuations in the scattering, absorption and fission macroscopic cross sections is presented as Eq. (1) (Demazière and Andhill, 2005; Hosseini and Vosoughi, 2014; Itoh, 1986; Pázsit and Glöckler, 1984; Williams, 2013):

$$[\nabla.\overline{\overline{D}}(\overline{r})\nabla+\overline{\overline{\Sigma}}_{dyn}(\overline{r},\omega)]\times\begin{bmatrix}\delta\phi_1(\overline{r},\omega)\\ \delta\phi_2(\overline{r},\omega)\end{bmatrix} = \overline{\phi}_{s,1\to2}(\overline{r})\delta\Sigma_{s,1\to2}(\overline{r},\omega)+\overline{\overline{\phi}}_a(\overline{r})\begin{bmatrix}\delta\Sigma_{a,1}(\overline{r},\omega)\\ \delta\Sigma_{a,2}(\overline{r},\omega)\end{bmatrix} + \overline{\overline{\phi}}_f(\overline{r},\omega)\begin{bmatrix}\delta v_1\Sigma_{f,1}(\overline{r},\omega)\\ \delta v_2\Sigma_{f,2}(\overline{r},\omega)\end{bmatrix}, \quad (1)$$

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

$$\overline{\overline{\Sigma}}_{dyn}(\overline{r},\omega)=\begin{bmatrix}-\Sigma_1(\overline{r},\omega) & \frac{v_2\Sigma_{f,2}(\overline{r})}{k_{eff}}\left(1-\frac{i\omega\beta_{eff}}{i\omega+\lambda}\right)\\ \Sigma_{s,1\to2}(\overline{r}) & -\left(\Sigma_{a,2}(\overline{r})+\frac{i\omega}{v_2}\right)\end{bmatrix}, \quad (2)$$

$$\overline{\phi}_{s,1\to2}(\overline{r})=\begin{bmatrix}\phi_1(\overline{r})\\ -\phi_1(\overline{r})\end{bmatrix}, \quad (3)$$

$$\overline{\overline{\phi}}_a(\overline{r})=\begin{bmatrix}\phi_1(\overline{r}) & 0\\ 0 & \phi_2(\overline{r})\end{bmatrix} \quad (4)$$

$$\overline{\overline{\phi}}_f(\overline{r},\omega)=\begin{bmatrix}-\phi_1(\overline{r})\left(1-\frac{i\omega\beta_{eff}}{i\omega+\lambda}\right) & -\phi_2(\overline{r})\left(1-\frac{i\omega\beta_{eff}}{i\omega+\lambda}\right)\\ 0 & 0\end{bmatrix} \quad (5)$$

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

$$\Sigma_1(\overline{r},\omega)=\Sigma_{r,1}(\overline{r})+\frac{i\omega}{v_1}-\frac{v_1\Sigma_{f,1}(\overline{r})}{k_{eff}}\left(1-\frac{i\omega\beta_{eff}}{i\omega+\lambda}\right). \quad (6)$$

To calculate the noise source term in the right hand side of the Eq. (1) (Eqs. (3)-(5)), neutron flux distribution should be calculated from the solution of the neutron diffusion equation. To this end, the neutron flux distribution obtained from the previous developed computational code is used to calculate the neutron noise distribution (Hosseini and Vosoughi, 2012). In the present study, the neutron noise source is assumed to be an absorber of variable strength. Here, the Green's function technique is used (Demazière and Andhill, 2005) to calculate the neutron noise distribution in the reactor core. In the mentioned method, the neutron noise distribution due to 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's components due to different positions of the unit value-point noise sources are calculated via the solution of the Eq. (7):

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

where $G_{g\to1}(\overline{r},\overline{r'},\omega)$ and $G_{g\to2}(\overline{r},\overline{r'},\omega)$ are the Green's function components of the energy groups 1 and 2 in the position $\overline{r}$ induced by the noise source in group g located in the position $\overline{r'}$, respectively. 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 (the perturbation in the thermal macroscopic cross section), the Eq. (7) can be written as Eqs. (8) and (9) using the Galerkin Finite Element Method (GFEM):

$$\sum_{e=1}^{E}\left[\int_\Omega\int_{(e)} d\Omega D_1\nabla N^{(e)}(\overline{r})\nabla N^{(e)T}(\overline{r})G^{(e)}_{2\to1} - \Sigma^{(e)}_1\int_\Omega\int_{(e)} d\Omega N^{(e)}(\overline{r})N^{(e)T}(\overline{r})G^{(e)}_{2\to1} + \frac{v_2\Sigma^{(e)}_{f,2}}{k_{eff}}\left(1-\frac{i\omega\beta_{eff}}{i\omega+\lambda}\right)\int_\Omega\int_{(e)} d\Omega N^{(e)}(\overline{r})N^{(e)T}(\overline{r})G^{(e)}_{2\to2} + \int_{\partial\Omega^{(e)V}} dsN^{(e)}(\overline{r})N^{(e)T}(\overline{r})\frac{G^{(e)}_{2\to1}}{2}\right]=0, \quad (8)$$

$$\sum_{e=1}^{E}\left[\int_\Omega\int_{(e)} d\Omega D_2\nabla N^{(e)}(\overline{r})\nabla N^{(e)T}(\overline{r})G^{(e)}_{2\to2} + \Sigma^{(e)}_{s,1\to2}\int_\Omega\int_{(e)} d\Omega N^{(e)}(\overline{r})N^{(e)T}(\overline{r})G^{(e)}_{2\to1} - \left(\Sigma^{(e)}_{a,2}+\frac{i\omega}{v_2}\right)\int_\Omega\int_{(e)} d\Omega N^{(e)}(\overline{r})N^{(e)T}(\overline{r})G^{(e)}_{2\to1} + \int_{\partial\Omega^{(e)V}} dsN^{(e)}(\overline{r})N^{(e)T}(\overline{r})\frac{G^{(e)}_{2\to2}}{2}\right]=\begin{bmatrix}N^{(e)}_i(\overline{r})\\ N^{(e)}_j(\overline{r})\\ N^{(e)}_k(\overline{r})\end{bmatrix}, \quad (9)$$