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Neutron noise simulation using ACNEM in the hexagonal geometry

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A B S T R A C T

In the present study, the development of a neutron noise simulator, DYN-ACNEM, using the Average Current Nodal Expansion Method (ACNEM) in 2-G, 2-D hexagonal geometries is reported. In first stage, the static neutron calculation is performed. The neutron/adjoint flux distribution and corresponding eigen-values are calculated using the algorithm developed based on power iteration method by considering the coarse meshes. The results of the static calculation are validated against the well-known IAEA-2D benchmark problem. In the second stage, the dynamic calculation is performed in the frequency domain in which the dimension of the variable space of the noise equations is lower than the time dependent equations. Induced neutron/adjoint noise distribution due to the neutron noise source of type absorber of variable strength is calculated as well. Two different methods are used to validate the neutron noise calculation. The Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (ILOFAIP) as an important neutron noise source is investigated using ACNEM in this study.

1. Introduction

The neutron noise is defined as the difference between the time-dependent and steady state neutron flux distributions when all the processes are stationary and ergodic in time. It is a very useful concept in the core monitoring and fault detection in the reactor core (Arzhanov and Pazsit, 2002; Hosseini and Vosoughi, 2012, 2014, 2016). The neutron noise analysis method can be used to reconstruct the noise source like the absorber of variable strength, vibrating absorber and unseated fuel assemblies (Demazière, 2004; Demaziere and Pázsit, 2008; Williams, 1974). To unfold the neutron noise source, it is necessary to know the neutron noise distribution in the reactor core through the solution of the neutron noise equation in the frequency domain. The dimension of the variable space of the noise equations was reduced in the frequency domain. Namely, instead of solving the diffusion equations in both the time and space domains, one needs to solve them only in space dimension after the Fourier transform (the frequency variable acts as a parameter in such a case and there is no derivative with respect to frequency). Since the final obtained system of noise equations is not an eigenvalue problem, its solution is easier than the static ones. In the first stage of the calculation of the neutron noise distribution, the neutron noise distribution from the point-like noise source is determined. In the field study of the noise analysis, several researchers have tried to develop convenient methods for calculations of the dynamical reactor transfer function and noise analysis (Demazière, 2004; Hosseini and Vosoughi, 2012; Larsson and Demazière, 2012; Larsson et al., 2011; Malmir et al., 2010). In the aforementioned researches, the spatial discretization of noise equations was performed using the finite difference, finite element and analytical nodal methods. The results with good accuracy may be obtained using the finite difference in which high computational cost is required. In some cases, the size of the neutron noise source might be in order of a fuel assembly dimension. Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (ILOFAIP) could be an example of this type of the neutron noise sources. The neutron noise analysis of the noise sources with size of order of the fuel assembly may be done using the algorithms like the nodal method by applying the coarse meshes. The nodal balance equation involves averaged neutron fluxes on nodes and averaged neutron currents on the surface. The nodal coupling equation provides the relationships between the averaged neutron flux distributions on the nodes and the averaged neutron current on the surfaces. The analytical (Smith, 1979) and polynomial nodal (Smith, 1985; Zerkle, 1992) methods may be used for the solution of the coupling equations. A quadratic shape of transverse leakage is generally used in the analytic nodal method, and the coupling equations are solved analytically. In the polynomial nodal method, a special polynomial expansion technique is used to solve the coupling equations (Kuo, 1994). Some advanced nodal methods like Nodal Expansion Method (NEM) (Finneman, 1975), Nodal Green's Function Method (NGFM) (Lawrence and Dorning, 1980) and Analytic Function Expansion Method (AFEM) (Noh and Cho, 1993,

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Nomenclature

ACNEM	Average Current Nodal Expansion Method
DYN-ACNEM	Dynamical- Average Current Nodal Expansion Method
$\varphi_g(r)$	neutron flux in the energy group g
$\varphi_g^{\dagger}(r)$	adjoint flux in the energy group g
k_{eff}	neutron multiplication factor
$k_{eff}^{\dagger}$	adjoint neutron multiplication factor
χ_g	neutron spectrum in energy group g
$\delta\varphi_g(r,\omega)$	neutron noise in the energy group g
$\delta\varphi_g^{\dagger}(r,\omega)$	adjoint noise in the energy group g
D_g	diffusion constant in the energy group g
$\Sigma_{r,g}$	macroscopic removal cross section in the energy group g
$\Sigma_{f,g}$	macroscopic fission cross section in the energy group g
$\Sigma_{s,g'\to g}$	macroscopic scattering cross section from energy group $g'$to g
ν	fission neutron yield
∇	the nabla operator
Π^m	volume of node
Γ_{ws}^m	left (s = l)/right (s = r) w-surface of node Π^m, w = x, u, v
Φ_g^m	average flux for group g in Π^m
Ψ_{gus}^m	average flux for group g at Γ_{ws}^m
$j_{gws}^{\pm m}$	Average outing (+) and incoming (−) partial currents for group g at Γ_{ws}^m
e_{ws}^m	Unit vector in the direction of the outward normal to Γ_{ws}^m
λ_{gws}^m	Value of boundary condition at Γ_{ws}^m
A_{ws}^m	Area of Γ_{ws}^m
Π^{mus}	Node adjacent to surface Γ_{ws}^m (of node Π^m)

1994) were used for the static calculation of the reactor core. AFEM was refined using the transverse gradient basis function and interface flux moment (Woo et al., 2001). Extension of AFEM of hexagonal-z geometry was also employed by Cho et al. (1997). The development of the nodal method of hexagonal geometry using conformal mapping of the hexagon into a rectangle was presented by Chao and Shatilla (1995) and Chao and Tsoulfanidis (1995). NEM originated from Nodal Synthesis Method by elimination of the fine mesh calculation and improvement of the calculation of inter node-coupling coefficient. It presents very suitable features like simplicity, computational efficiency and acceptable accuracy for most realistic nuclear reactor simulation.

In the first stage of the present study, the 2-G neutron diffusion equation in the hexagonal geometry will be solved using the ACNEM. The calculated neutron flux distribution is used to obtain the neutron noise source. The neutron noise equation is then solved using the same utilized nodes in the solution of the neutron diffusion equation. The simulation of the neutron noise distribution due to the ILOFAIP as an actual sample of the neutron noise source was performed using the developed DYN-ACNEM computational code.

An outline of the remainder of the present paper is as follows: In Section 2, we briefly introduces the mathematical formulation used to solve neutron/adjoint diffusion equation using ACNEM. The method used for dynamical calculations is also described in Section 2. Section 3 presents the main specification of the IAEA-2D benchmark problems. The results of the static and noise calculations are given in Section 4. Also, in Section 4, we discuss about the results and considered assumptions in the noise calculations. Section 5 gives a summary and concludes the paper.

2. Methodology

2.1. The solution of the neutron diffusion equation using the ACNEM

The multigroup neutron diffusion equation is written as Eq. (1) in the absence of neutron external source (Lamarsh, 1965):

$$-\nabla.D_g\nabla\varphi_g+\Sigma_{r,g}\varphi_g-\sum_{g'\neq g}^{G}\Sigma_{s,g'\to g}\varphi_{g'}-\frac{\chi_g}{k}\sum_{g'=1}^{G}\nu_{g'}\Sigma_{f,g'}\varphi_{g'}=0$$
$$g=1,\ldots,G \tag{1}$$

Eq. (1) is a linear partial differential equation that may be solved using the different numerical methods. Here, the discretization of the neutron diffusion equation is performed using ACNEM with hexagonal nodes. A typical hexagonal node used for the spatial discretization is presented in Fig. 1. To discretize the hexagonal geometry in radial direction, the local "rectangular Cartesian axes" (x,y) and local "hexagonal axes" (x, u, v) are considered in each node. On each considered axis, the local dimensionless variable is defined as Eq. (2):

$$\xi_w=\frac{w}{H},\quad w=x,u,v \tag{2}$$

where, H denotes to the lattice length or thickness of the node. Also, rectangular and hexagonal variables in radial directions are defined as Eq. (3).

$$u=\frac{\sqrt{3}y-x}{2},\quad v=\frac{-\sqrt{3}y-x}{2} \tag{3}$$

Integration of Eq. (1) over each hexagonal node Π^m gives Eq. (4):

$$\sum_{\substack{s=r,l\\ w=x,u,v}}\frac{A_{ws}^m}{V^m}(j_{gws}^{+m}-j_{gws}^{-m})+\Sigma_{r,g}^m\Phi_g^m=\sum_{g'\neq g}^{G}\Sigma_{s,g'g}^m\Phi_{g'}^m+\frac{\chi_g}{k}\sum_{g'=1}^{G}\nu_{g'}\Sigma_{fg'}^m\Phi_{g'}^m \tag{4}$$

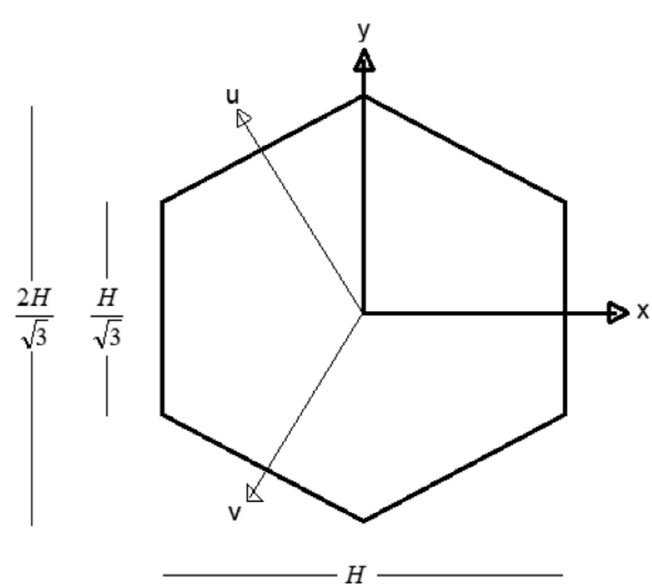

Fig. 1. The nodal coordinate system.