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Finite difference applied to the reconstruction method of the nuclear power density distribution

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In this reconstruction method the two-dimensional (2D) neutron diffusion equation is discretized by finite differences, employed to two energy groups (2G) and meshes with fuel-pin cell dimensions. The Nodal Expansion Method (NEM) makes use of surface discontinuity factors of the node and provides for reconstruction method the effective multiplication factor of the problem and the four surface average fluxes in homogeneous nodes with size of a fuel assembly (FA). The reconstruction process combines the discretized 2D diffusion equation by finite differences with fluxes distribution on four surfaces of the nodes. These distributions are obtained for each surfaces from a fourth order one-dimensional (1D) polynomial expansion with five coefficients to be determined. The conditions necessary for coefficients determination are three average fluxes on consecutive surfaces of the three nodes and two fluxes in corners between these three surface fluxes. Corner fluxes of the node are determined using a third order 1D polynomial expansion with four coefficients. This reconstruction method uses heterogeneous nuclear parameters directly providing the heterogeneous neutron flux distribution and the detailed nuclear power density distribution within the FAs. The results obtained with this method has good accuracy and efficiency when compared with reference values.

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1. Introduction

The finite difference method is one of the first methods used in the neutron diffusion equation discretization to obtain the detailed power density distribution within the reactor. However, this method requires an excessive number of meshes to an acceptable accuracy, resulting in a high cost of storage and computational time. In the 70's, the development of fast and economical methods to calculate the power density distribution in reactor FAs was initiated. The coarse mesh nodal methods were proposed to solve the multi-group neutron diffusion equation and nowadays is currently used in reactor core analysis codes.

Methods that use the coarse mesh nodal calculation are often found in literature as the coarse mesh finite difference (Aragones and Ahnert, 1986). The finite differences in nodal calculation are mainly intended to reduce the processing time when compared to the Nodal Expansion Method (Finnemann et al., 1977). The NEM method uses the transversely integrated diffusion equation, which these integrations generate a set of coupled one-dimensional equations by the transverse leakage terms and their solution provides a relation between the node average flux and average net current on surfaces of the node. Another important method using polynomial expansions in the source term and scattering is known as semi-analytical (Kim et al., 1999). These methods are able to calculate with great accuracy the effective multiplication factor, average fluxes and power densities in the nodes with dimensions of a FA. The mesh dimensions turn the coarse mesh nodal methods more economical, but the determination of local values within the FAs it's complicated since only averages values can be calculated. Thus, various methods to reconstruct the detailed power density distribution within the FA were developed based on the average values provided by the nodal calculation.

In the past four decades, several researches based on the reconstruction methods of power density distribution were developed (Koebke and Wagner, 1977; Koebke and Hetzelt, 1985; Rempe et al., 1988; Boer and Finnemann, 1992; Joo et al., 2009; Dahmani et al., 2011; Pessoa et al., 2015). These methods are distinguished by the way they represent and generate the homogenous flux distribution. A methodology known as modulation method (Mattila, 1999) is used to estimate heterogeneous distribution within the FAs by the product of a reconstructed homogeneous distribution for a local heterogeneous form function coming from the homogenization process. In all reconstruction methods cited the determination of the homogeneous flux distribution and corners fluxes should

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be based on the results of nodal calculation performed for homogeneous FAs. Corner fluxes of the node can be calculated using finite difference techniques, procedures interpolation, approximations with polynomial solutions or a combination of these techniques. Note that each reconstruction method mentioned has a different way to calculate these corner fluxes.

An important reconstruction method uses finite difference combined with the Maximum Entropy Method (Jung and Cho, 1991), which Lagrange multipliers are obtained by using probability distributions and average values provided by the nodal calculation to calculate flux distributions on surfaces of FAs. These distributions are used as boundary conditions to determine heterogeneous flux distributions inside the FAs from the discretization of the 2D neutron diffusion equation by finite differences. Corner fluxes and the modulation method are not used in this reconstruction method because the solution is discretized in meshes with dimensions of a fuel-pin cell, which heterogeneous nuclear parameters is needed.

In this paper is presented a reconstruction method that uses discretization by finite differences of 2D and 2G neutrons diffusion equation in homogeneous meshes with dimensions of a fuel-pin cell, which the heterogeneous neutron flux distribution is provided directly and so the power density distribution within the FAs. This discretization is combined with flux distributions on four surfaces of the node with dimensions of a FA, while is used as boundary conditions for solution. A fourth order 1D polynomial expansion with five coefficients is used to determine the flux distribution on four surfaces of the node separately. The coefficients for each solution of each surface are determined with a centralized solution in one direction and three consecutive surfaces of three nodes. In this way, the five conditions necessary to determine the coefficients are average fluxes on the three consecutive surfaces and two corner fluxes between these surface fluxes.

The NEM method (Finnemann et al., 1977) is used in this work for the coarse mesh nodal calculation and to provides the surfaces average fluxes and the effective multiplication factor for the reconstruction method. Corner fluxes of the nodes are determined using a third order 1D polynomial expansion with centralized solution in four node consecutive surfaces at one direction. This solution is applied to the four corners of the node separately and the coefficients are determined using average fluxes in four consecutive surfaces. Note that the surface discontinuity factor of the FAs is used in coarse mesh nodal calculation. The modulation method is not used. The nodes have dimensions of a FA for calculating the surface flux distribution and these distributions represent the minimum set of boundary conditions to be used by the reconstruction method proposed here. Thus, the original contribution of this work is the development of a new reconstruction method that combines the discretization by finite differences of the 2D neutron diffusion equation with a flux distribution on surfaces of the nodes, the method also innovates in the way the surfaces flux distribution and corner fluxes of the node are calculated.

The discretization of the 2D neutron diffusion equation is shown in the Section 2. In Section 3, the determination of the flux distributions on the four node surfaces is shown. Section 4 details how the corner fluxes of the node are determined. Section 5 describes the procedures for determining the nuclear power density distribution. Section 6 presents and specifies the benchmarks. Section 7 contains the numerical results obtained with the reconstruction method and final conclusions are presented in Section 8.

2. Discretization of the 2D neutron diffusion equation

The discretization by finite differences of the 2D and 2G neutrons diffusion equation is implemented in the domain of fuel-pin cell and using flux distributions on the four surfaces of the node as boundary conditions. These fluxes are located on surfaces of the meshes with fuel-pin cell dimensions and these surfaces becoming the node boundary. Thus, if the surface flux distribution and the effective multiplication factor on the problem are known, the heterogeneous flux distribution in each mesh of the node or FA may be completely reconstructed. With this in mind, the boundary conditions are calculated using a fourth degree 1D polynomial expansion and the five coefficients of this expansion are determined based on the average quantities provided by coarse mesh nodal calculation. The NEM method provides these average values along with the effective multiplication factor.

Then, once known the heterogeneous flux distribution on surfaces of the node or FA, the neutrons diffusion equation is discretized by finite differences in mesh with dimensions of a fuel-pin cell and the average neutron flux in each mesh of the node can be calculated. Fig. 1 shows the meshes in the domain of a node, where the areas represented by the pair (i,j) have the same size of a fuel-pin cell with uniform nuclear parameters. The pair (i,j) represents a mesh with area $A = a_x^j a_y^i$.

Starting from the 2G continuity equation, with Cartesian geometry:

$$\sum_{u=x,y}\frac{\partial}{\partial u}J_{gu}(x,y)+\Sigma_{Rg}(x,y)\phi_g(x,y) = \frac{\chi_g}{k_{eff}}\sum_{g'=1}^{2}\nu\Sigma_{fg'}(x,y)\phi_{g'}(x,y)+\sum_{\substack{g'=1\\ g'\neq g}}^{2}\Sigma_{gg'}(x,y)\phi_{g'}(x,y), \qquad (1)$$

and applying the average operator or integrating in the area A of the mesh represented by the pair (i,j) shown in Fig. 1, the Eq. (1) become:

$$\sum_{u=x,y}\frac{1}{a_u^n}\left(\bar{J}^{i,j}_{gur}-\bar{J}^{i,j}_{gul}\right)+\overline{\Sigma}^{i,j}_{Rg}\;\overline{\phi}^{i,j}_{g} = \frac{\chi_g}{k_{eff}}\sum_{g'=1}^{2}\overline{\nu\Sigma}^{i,j}_{fg'}\;\overline{\phi}^{i,j}_{g'}+\sum_{\substack{g'=1\\ g'\neq g}}^{2}\overline{\Sigma}^{i,j}_{gg'}\;\overline{\phi}^{i,j}_{g'}, \qquad (2)$$

where

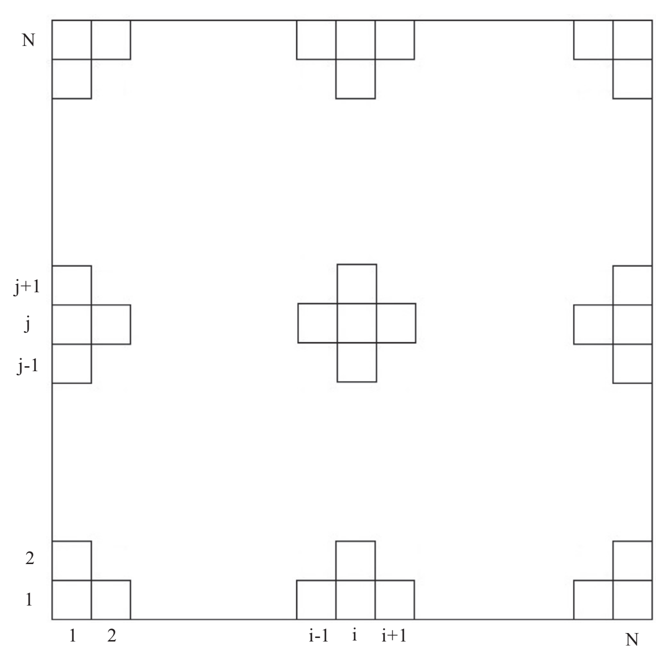

Fig. 1. Spatial mesh in the node area.

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