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Application of nonlinear nodal diffusion method for a small research reactor

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ABSTRACT

Nodal diffusion methods are usually used for LWR calculations and rarely used for research reactor calculations. A unified nodal method with an implementation of the coarse mesh finite difference acceleration was developed for use in plate type research reactor calculations. It was validated for two PWR benchmark problems and then applied for IAEA MTR benchmark problem for static calculations to check the validity and accuracy of the method. This work was conducted to investigate the unified nodal method capability to treat material testing reactor cores. A 10 MW research reactor core is considered with three calculation cases for low enriched uranium fuel depending on the core burnup status of fresh, beginning-of-life, and end-of-life cores. The validation work included criticality calculations, flux distribution, and power distribution; in addition, a comparison between different fuel materials with the same uranium content was conducted. The homogenized two-group cross sections were generated using the TRITON–NEWT system. The results were compared with a reference, which was taken from IAEA-TECDOC-233. The unified nodal method provides satisfactory results for an all-rod out case, and the three-dimensional, two-group diffusion model can be considered accurate enough for MTR core calculations.

1. Introduction

Nodal diffusion methods are rarely used for research reactor analyses owing to the complex shape and small size of the core, which has a high leakage potential. Nodal diffusion methods have been shown to be computationally efficient and are used in practical routine core calculations which need excessive computing times. The wide application of neutron diffusion codes has been continuously conducted for criticality calculations, calculations of the neutron flux and power distribution, and in-core fuel management (Suparlina and Sembiring, 2005). In addition, important safety parameters such as excess reactivity, control rod worth, and shutdown margin can be calculated. The use of nodal diffusion theory for a small core such as in material testing reactors (MTR) maybe over optimistic, but if reasonable results can be obtained, the nodal method can be used for a fast core design and fuel management which cannot be achieved using Monte Carlo codes (de Leege and Reitsma, 2004).

Nodal methods solve multi-group diffusion equations. In most nodal methods the quantity of interest is the flux averaged over large spatial regions which are the nodes and the surface neutron currents averaged over the faces of the node. Nodal diffusion methods differ in obtaining the relationships between nodes averaged flux and surface averaged currents. Many different methods were proposed to obtain this relation. Nodal methods offer an accurate and efficient technique for solving multi-group diffusion equation.

Two principal classes of transverse-integrated methods have been developed over the years, the polynomial and the analytic methods. In nodal expansion method NEM (Finnemann, 1977) the transverse surface-integrated flux is expanded in polynomials with lower coefficients constrained so that they satisfy boundary conditions and consistency conditions and higher coefficients determined by weighted residual methods. Where in the analytic nodal method ANM (Smith, 1979), the analytic solution of the one-dimensional neutron diffusion equation is used to solve the transverse-integrated equations. Because the ANM solution to the transverse-integrated equation is exact, it provides a more accurate solution than NEM, particularly when the nodal meshing is large. The ANM has the drawback of having numerical instabilities at near-critical nodes when the core contains nearly no-net-leakage nodes. To get around this drawback (Joo et al., 1998) approximate stabilization schemes such as were proposed the linear fundamental mode approximation and hybrid ANM/NEM interface coupling technique. Since the NEM kernel involves no particular

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Nomenclature

a	node width of rectangular node
J	surface average net current
$j^{\pm}$	partial surface current
k_{eff}	eigenvalue or Multiplication factor
L	transverse leakage
$\bar{L}$	average transverse leakage
h_i	basis function, (i = 1,. . ., 4)
C_i	flux coefficient vector, (i = 1,. . ., 4)
D	diffusion coefficient
D^F	Base nodal coupling coefficient
D^N	corrective coupling coefficient
Greek symbols	
$\bar{\phi}$	node average flux
ϕ	node average flux in one direction
Σ	macroscopic cross section
χ	Prompt fission fraction
ν	average number of neutrons released per fission
Subscripts	
m	node, m = 1, 2, . . ., M
g	energy group g = 1, 2
u	direction x, y, z
r	right
l	left
t	total
s	scattering
r	removal
f	fission

solution, it does not have the numerical instability of the ANM. In fact, it is numerically stable under all conditions.

In unified nodal method UNM (Lee and Kim, 2001a,b) the ANM solution to two-group diffusion equations can be reformulated in exactly the same way as the NEM solution, and thereby, the two most popular transverse integrated nodal method formulations can be integrated into a unified nodal method UNM formulation. It was demonstrated that the numerical instabilities at the near-critical nodes can naturally be resolved by the UNM formulation itself without introducing any approximate stabilization schemes. A unified nodal method was used to make the core calculations and compare the results with the references to check the capability of this method. A brief description of the UNM with the coarse mesh finite element (CMFD) acceleration is given in the following section.

In the eighties, a safety-related benchmark problem for an idealized generic 10 MW MTR light-water pool-type reactor was defined (IAEA-TECDOC-233, 1980). The benchmark was specified under a program of research reactor core conversions from highly enriched uranium (HEU) to low enriched uranium (LEU) cores. It covers large steady state neutron kinetics and thermal-hydraulic calculations and a wide range of hypothetical dynamic accidental scenarios (Bousbia-Salah et al., 2008). The benchmark calculations cover almost all aspects of neutronic design such as criticality, feedback reactivity, and control rod worth. In the present study, for the purpose of validating the UNM for MTR, the results of the static part of the safety-related benchmark calculation proposed by the IAEA are reported and discussed. The TRITON–NEWT system was considered to obtain group constants and burnup calculations. The UNM model was used for criticality calculations, a flux distribution, and power distribution.

2. Unified nodal method

2.1. UNM formulation

The idea of the ANM reformulation is based on decoupling 2G diffusion equations into two independent equations through similarity transformation, representing the analytic solution of

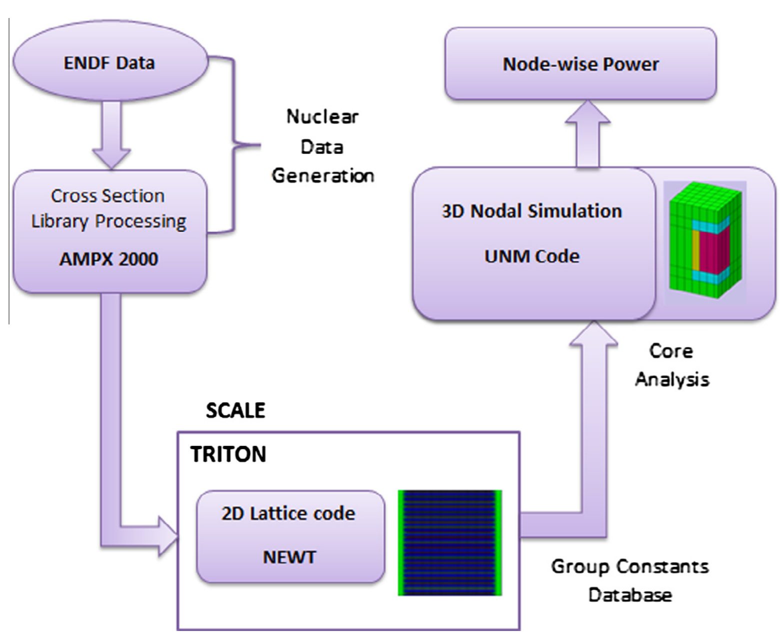

Fig. 1. Reactor core analysis with 2-D lattice code and 3-D nodal simulator.