ANDREA: ADVANCED NODAL DIFFUSION CODE FOR REACTOR ANALYSIS

J. BĚLÁČ, R. JOŠEK, L. KLEČKA, V. STARÝ, R. VOČKA
Nuclear Research Institute Řež plc., 250 68 Řež, Czech Republic
vor@ujv.cz

ABSTRACT

A new macrocode is being developed at NRI which will allow coupling of the advanced thermal-hydraulics model with neutronics calculations as well as efficient use in core loading pattern optimization process. This paper describes the current stage of the macrocode development.

The core simulator is based on the nodal expansion method, Helios lattice code is used for few group libraries preparation. Standard features such as pinwise power reconstruction and feedback iterations on critical control rod position, boron concentration and reactor power are implemented. A special attention is paid to the system and code modularity in order to enable flexible and easy implementation of new features in future. Precision of the methods used in the macrocode has been verified on available benchmarks. Testing against Temelin PWR operational data is under way.

1. INTRODUCTION

More intensive exploitation of current fuel and introduction of new fuel types cannot proceed without modernization of computer tools used for core reload optimization, core surveillance and safety analysis. A good example of this process are the regular upgrades of the SCORPIO-VVER core monitoring system, which go hand in hand with each change in fuel design [1]. Even broader revision of methods and computer tools is related to the aspects of very high burnup fuel, introduction of advanced fuel cycles, not mentioning the Generation IV program. This was the incentive for the development of a new macrocode for reactor physics calculations.

Our goal is building of a new code for neutron-physics calculations based on modern nodal methods allowing pinwise reconstruction of power distribution in fuel assemblies with monitoring of pinwise burnup. In the next stage, neutron-physics calculations will

be coupled with the thermal-hydraulics model taking into account the cross flow among neighbor assemblies, completed for the sub-channel analysis. The code should integrate new models of fuel rod thermo-mechanics suitable for application to fuels with high and very high burnup. A special attention is paid to the system and code modularity in order to enable flexible and easy implementation of new features in future.

Keeping in mind the above-mentioned framework, the first stage of development of the macrocode is oriented to PWR reactors with hexagonal fuel assemblies. In the next sections we briefly describe the current status of development of fundamental components of the macrocode, and then we outline so far obtained results.

2. MACROCODE ANDREA DESCRIPTION

2.1 NEUTRON-PHYSICAL MODULE

The neutron-physical module of the macrocode solves the diffusion equation in 3D and two energy groups on hexagonal lattice. The solver is based on advanced nodal method.

Nodal methods are based on the analytical solution of the diffusion equation in the homogenized volume of the reactor core - "node", which is usually a slice of the assembly. These nodal solutions are coupled through the boundary currents.

In general the three-dimensional diffusion equation in the node is not solved directly. The solution first proceeds by the transverse integration in order to reduce the problem to the solution of 2D [2, 3] or 1D diffusion equation. We have chosen the later option. The directions which are integrated out appear in the reduced diffusion equation as the transverse leakage terms.

In case of the hexagonal node the transverse integration approach is not straightforward, because the the final one-dimensional diffusion equation contains singular terms. This issue can be overcome by application of the conformal mapping in order to transform the homogeneous hexagonal node into the inhomogeneous rectangular node [4]. In this case, after the subsequent transverse integration, the 1D diffusion equation does not contain any singular terms, so the solution can proceed further in a standard way.

The implementations of nodal methods generally differ in the mathematical form of neutron flux expansion for the 1D diffusion equation solution and by the treatment of the transverse leakage terms. We use following flux expansion,

$$\phi(x) = A\cosh(\kappa x) + B\sinh(\kappa x) + \sum_{i=1}^{3} c_i w_i(x)$$

where the first two expansion terms form the general solution of the homogeneous diffusion equation and w_i is a set of 3 orthogonal polynomials. The 5 expansion coefficients A, B, w_1, w_2 and w_3 are evaluated from the 3 moment equations and the flux discontinuity and current continuity relations.

The expression of the transverse leakage is a delicate issue, which has a non-negligible influence on the precision of the results. From the solution of the nodal equations we have direct access only to the average leakage over the node boundaries, but it appears that in

order to achieve a good precision a linear approximation of the leakage is needed - at least for the nodes on the core periphery. Our neutronic model offers two options of transverse leakage evaluation. The first option consist in usage of a flat leakage profile in the core center while the the slope on the periphery is deduced from the crude approximation of the flux profile. If the second option is chosen, the leakage gradient is evaluated from the results of smooth nodal flux profile reconstruction (see section 2.2). This approach leads to a better precision at the expense of computation time.

2.2 PIN POWER RECONSTRUCTION

The pin power reconstruction is based on the superposition of the pin power map from the transport solution and the smooth power profile based on the results of nodal solution. The pin power maps are evaluated for each fuel type as a function of assembly burn-up at the moment of assembly homogenization. The smooth power profile is evaluated from the analytical solution of the diffusion equation in the homogeneous node using the boundary conditions resulting from the nodal solution. In order to obtain the analytical solution the set of coupled few group diffusion equations is diagonalized [2, 5] so that we obtain a set of decoupled plain wave equations,

$$\nabla^2 \xi_g - \lambda_g \xi_g = 0\,, \qquad g = 1, 2 \tag{1}$$

where ξ_g are linear combinations of the group fluxes, $\xi_g = \alpha_{1,g}\phi_1 + \alpha_{2,g}\phi_2$. The solution of the equation (1) is a superposition of plane waves. We approximate this solution by following expression,

$$\xi_g(r,\phi) = \sum_{i=0}^{5} A_{g,i}\,\mathrm{cn}\left(\sqrt{|\lambda_g|}\,r\cos(\phi - i\pi/6)\right) + \sum_{i=0}^{5} B_{g,i}\,\mathrm{sn}\left(\sqrt{|\lambda_g|}\,r\cos(\phi - i\pi/6)\right) \tag{2}$$

where r, ϕ are the coordinates, $\mathrm{cn}(x) = \cos(x)$ if $\lambda_g < 0$, otherwise $\mathrm{cn}(x) = \cosh(x)$. The function $\mathrm{sn}(x)$ is defined analogically to $\mathrm{cn}(x)$. Coefficients $A_{g,i}, B_{g,i}$ are deduced using the values of average boundary and corner point fluxes.

It should be mentioned that the nodal solution leads directly only to the boundary average fluxes. The corner point fluxes are evaluated in a self-consistent way from the current conservation condition: the sum of net currents that enter an infinitesimal volume around the corner point from the nodes surrounding this point must be equal to zero [5]. The expressions of currents entering the mentioned current conservation equation are evaluated from eq. (2).

2.3 ASSEMBLY HOMOGENISATION

The homogenized few group cross-sections for the nodal diffusion code are calculated by means of Helios 2D transport lattice code [6]. They are evaluated for each fuel type as a function of fuel burn-up, fuel and moderator temperatures, with and without introduction of control rod absorbers. The cross sections are extracted from the Helios output files in an automated way and they are transformed in a form of cross section library, which is then used by the macrocode. For the sake of easy access and portability the library is stored in a form the dbm database, which is a standard component of all versions of UNIX complying to X/Open specification.

2.4 BURNUP EVALUATION

The homogenized cross sections are tabulated as a function of burnup, so the cross section for a given burnup is obtain by interpolation. The fuel burnup is evaluated by the predictor-corrector method. During the prediction step, the prediction of burnup at the next time point is evaluated using the cross sections corresponding to the current time point. Predicted burnup and power distribution enable to evaluate the prediction for the cross sections. Follows the correction step where both the current and predicted cross sections are used for the burnup reevaluation.

During the burnup we track explicitly the number densities of ^{135}I and ^{149}Sm evaluating analytically $^{135}\mathrm{I} \rightarrow {}^{135}\mathrm{Xe}$ and $^{147}\mathrm{Nd} \rightarrow {}^{149}\mathrm{Pm} \rightarrow {}^{149}\mathrm{Sm}$ fission product chains.

2.5 FUEL ROD THERMO-MECHANICS

The temperature profile along the radius of the fuel pin is solved as a one-dimensional problem. Computation proceeds from the outside cladding of the pin inwards. The profile is evaluated as a function of coolant parameters (temperature, pressure, velocity, density) and fuel rod linear power, burn-up and material properties (thermal conductivities etc.). The burn up effects, such as swelling and gap thermal conductivity changes are also taken into account, though they are treated in a very simple way for the sake of clarity and calculation speed.

2.6 THERMAL-HYDRAULICS MODEL

At the time being the macrocode contains a basic thermal-hydraulics module where each assembly is treated as an isolated channel and the flow-rate is considered to be even in all the channels. This model is suitable for the VVER-440 reactors. For VVER-1000 reactor it is only the first approximation. That is why we are starting the development of the thermal-hydraulics model where the cross-flow among the neighbor assemblies is taken into account.

3. RESULTS

In order to verify precision of the methods applied in the neuronics model, we have evaluated several benchmarks from the literature [7]. Results obtained by the nodal neutronic model for two 2D VVER-440 model benchmarks, one 2D VVER-1000 model benchmark and one 3D VVER-440 model benchmark were compared with the reference and fine-mesh finite difference calculation results. The relative difference in the nodal powers from the reference results did not exceed 1% for the 2D benchmarks and 2% for the 3D benchmark.

We have used the Seidel's 2D VVER-440 benchmark from [7] as well to check the precision of the reconstruction of the smooth flux profile. The smooth profile was checked against reference fine mesh finite difference calculations. In order to compare the difference of the flux nodal profiles, we corrected the relative flux difference for the difference in

average nodal fluxes. Such a "corrected"relative flux difference ρ can be written as

$$\rho = 2\frac{\phi_N(x,y) - \phi_{FD}(x,y)}{\phi_N(x,y) + \phi_{FD}(x,y)} - 2\frac{\overline{\phi}_N - \overline{\phi}_{FD}}{\phi_N(x,y) + \phi_{FD}(x,y)}$$

where the subscript N stays nodal calculations and FD for finite difference calculations. In the fast group the "corrected" relative difference ρ did not exceed 1%. In the thermal group the differences are slightly higher, $\rho < 2\%$ for assemblies in the core center and ρ <5% for the assemblies at the periphery.

4. CONCLUSION

In this paper we have described the current status of the development of a new macrocode for steady state and transient reactor calculations. A good progress has been made in the development of the neutron-physics module of the macrocode. The precision of the methods applied in the nodal solution has been tested against the benchmark results published in the literature.

At the current stage the macrocode couples the neutron-physical calculations with the simple thermo-hydraulics and thermo-physics modules, so it allows the basic reactor-physics calculations with temperature feedback. The testing against the VVER-1000 operational data is under way.

The development of the macrocode is nevertheless not yet finished. We are starting the development of more involved thermal-hydraulics module with cross-flow among the neighbor assemblies. This module should also integrate the sub-channel analysis. We also foresee further development of the thermo-mechanical fuel-pin model and the development of the friendly user interface.

REFERENCES

[1] J. Běláč, J. Čížek, J. Švarný, M. Valach, R. Vočka, "New Methods Development for SCORPIO-VVER Core Monitoring Systems to Address advanced VVER Fuel Types", Proceedings EHPG Meeting 2004, MTO, Vol. 2, C5.15, (2004).

[2] N. Z. Cho and J. M. Noh, "Analytic Function Expansion Nodal Method for Hexagonal Geometry", Nucl. Sci. Eng., Vol. 121, pp. 245-253 (1995).

[3] U. Grundmann and F. Hollstein, "A Two-Dimensional Intranodal Fux Expansion Method for Hexagonal Geometry", Nucl. Sci. Eng., Vol. 133, pp. 201-212, (1999).

[4] Y. A. Chao and N. Tsoulfanidis, "Conformal Mapping and Hexagonal Nodal Methods - I: Mathematical Foundation", Nucl. Sci. Eng., Vol. 121, pp. 202-209 (1995).

[5] V. Mecir and Y. A. Chao, "Consistent Corner Flux Calculation for Pin Power Reconstruction", Transactions of the Am. Nucl. Soc., Vol. 72, pp. 346-347 (1995).

[6] Helios lattice physics V.1.8, Studsvik Scandpower (2003).

[7] M. Makai, editor, "Final report of TIC, volume 5", Akadmiai Kaid, Budapest (2000).