VALIDATION OF MULTIPOINT KINETICS MODEL AGAINST 3D TRIKIN CODE

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ABSTRACT

Validation of multipoint kinetics formulation for RELAP5 code has been carried out against 3D TRIKIN code. Core behavior of an asymmetric reactivity transient has been simulated through artificial tuning of lattice constants in 3D code. Individual node normalized reactivity has been conserved and power estimates from multipoint model have been compared with 3D simulation. It has been observed that localized peak power estimates from multipoint simulation are on higher side and therefore are conservative in nature. Improvements in multipoint formulation in regards to evolving coupling coefficients and involving more number of nodes can help in improving its accuracy to some extent.

Keywords: Multipoint kinetics, 3D kinetics, PHWR LOCA, RELAP5, Coupled analysis

INTRODUCTION

Safety analysis of nuclear power plant brings out the quantitative evaluation of the plant capability to control or accommodate departures from normal operation or postulated malfunctions or failures. It also demonstrates that the plant does not pose unacceptable safety hazard. For this purpose, safety analyses require use of complex computer codes which model the physics of various phenomenon relevant to the accident scenario, such as the thermal hydraulic of the associated systems, core neutronics, fuel and core component heat-up effects, etc. These codes have to be validated against several sets of experimental data. RELAP5 has been one the most popular codes adopted by nuclear industry. The code has evolved in terms of complexities and capabilities over decades. However point kinetics approach to simulate transient neutronics adapted in RELAP5 limits its use for many reactivity induced transients, which involve asymmetric core behavior. Development of full 3D core kinetics code coupled with system code RELAP is desirable for this and many initiatives have been taken up to couple 3D neutron kinetics codes with RELAP5 code across the world. However this require extensive programming, validation base and access to the source code. An intermediate approach of Multi-Point kinetics is appropriate and relatively easy to implement for analysis of several asymmetric transients in the large cores. Multi-Point kinetics formulation is based on dividing the entire core into several regions and solving ODEs describing kinetics in each region. These regions are interconnected by spatial coupling coefficients which are estimated from steady state diffusion theory. A model based on multipoint kinetics formulation was developed for RELAP5 code where associated ordinary differential equations (ODEs) governing Multi-Point kinetics formulation were solved through RELAP5 control variables [1]. The present study brings out validation of multipoint kinetics model against full 3D kinetics estimates. Core behavior of an asymmetric reactivity transient in a PHWR initiated due to voiding of one coolant loop has been simulated through artificial tuning of lattice constants in 3D code. Individual node normalized reactivity has been conserved and power estimates from multipoint model have been compared with 3D simulation. AERB 3D kinetics code TRIKIN [2, 3] has been used for 3D simulation. It has been found that power estimates of the broken loop from multipoint formulations are higher than 3D estimates whereas situation is reverse for the case of intact loop, though overall global core power prediction are in good agreement from both the methods. It has been observed that localized peak
power estimates from multipoint simulation are on higher side and therefore are conservative in nature. It can be slightly improved by modeling more number of nodes and/or updating inter node coupling coefficients in multipoint formulations.

THEORY OF MULTIPOINT KINETICS

The safety assessment of several transients in nuclear reactors requires the simulation of the dynamic behavior. However, the space independent version of the neutron balance equations, is often computationally cumbersome and unacceptable for large core coupled calculations. For this reason, several methods for the solution of the time-dependent neutronic equations have been developed over the years. Development of interconnected models for space and amplitude of neutron flux is the most sophisticated approach in this direction and is known as space-time kinetics analysis. However, computing effort associated with such formulation is very large and therefore some intermediate methods of modest accuracy to address core asymmetry in selected problems are explored. The Multi-Point kinetics is one such method. It was proposed long back in late 1950s for the kind of reactor dynamics problems in which different parts of a nuclear system evolve with little mutual influence. Under these situations the kinetic equations for each sub-domain can be derived in terms of its own amplitude function, properly connected to the others through coupling coefficients. These coefficients were mainly computed through physical considerations. In summary, among the various methods, Multi-Point kinetics methods, from which simple first order equations governing node averaged power can be cast rather easily, form a better candidate under many scenarios. Moreover, with transient simulations being an integral part of the control system design and analysis task, such models further establish their relevance in LORA and control related applications. Over the period of time, Multi-Point kinetics models have been used extensively for the analysis and simulation of light water reactors and control system design of pressurized heavy water reactors\cite{4}. If a reactor is divided into two nodes via; \textit{h} and \textit{k}, the neutron power of the two nodes is governed by:

\[
\frac{dP_h}{dt} = \left(\frac{\rho_h - \beta}{l_h}\right)P_h + \sum_{i=1}^{6} \lambda_i C_{ih} + \alpha_{kh} \frac{P_k}{l_k} - \alpha_{hk} \frac{P_h}{l_h} \\
\frac{dC_{ih}}{dt} = \frac{\beta_i}{l_h} P_h - \lambda_i C_{ih}
\]

Similarly neutron power for node \textit{k} can be written as:

\[
\frac{dP_k}{dt} = \left(\frac{\rho_k - \beta}{l_k}\right)P_k + \sum_{i=1}^{6} \lambda_i C_{ik} + \alpha_{hk} \frac{P_h}{l_h} - \alpha_{kh} \frac{P_k}{l_k} \\
\frac{dC_{ik}}{dt} = \frac{\beta_i}{l_h} P_h - \lambda_i C_{ih}
\]

where
\begin{itemize}
  \item $P=$ reactor power
  \item $\rho=$ reactivity
  \item $\beta=$ effective delayed neutron fraction
  \item $\beta_i=$ delayed neutron fraction for precursor \textquoteleft i\textquoteleft
  \item $l=$ prompt neutron life time
  \item $\lambda_i=$ decay constant of delayed neutron precursor \textquoteleft i\textquoteleft
  \item $C_i=$ concentration of delayed neutron precursor \textquoteleft i\textquoteleft
  \item $\alpha=$ coupling coefficient
\end{itemize}

Coupling coefficients between the two nodes depend on the interface area, node sizes, distance between their geometric centers and diffusion data of each node. Reactivity of each node should be defined such that net core reactivity should be conserved for the transients being analyzed.
3D Neutron Kinetics Formulation of TRIKIN Code [3]

3D neutron kinetics has been simulated by solving multi-group neutron diffusion equation in full 3D domain for Cartesian grid of PHWR. Globally space-time kinetics modeling is represented through the multi-group diffusion equations coupled with the equations for the delayed neutron precursors, as follows.

\[
\frac{1}{v_g} \frac{\partial \phi_g}{\partial t} = \nabla \cdot D_g \nabla \phi_g (r,t) - \Sigma_{fg} \phi_g (r,t) + \chi_g^{\beta} (1 - \beta) \sum_g \Sigma_{f,g} \phi_g (r,t) + \sum_{g' \neq g} \Sigma_{g' \rightarrow g} \phi_{g'} (r,t) + \chi_g^{d} \sum_i \lambda_i C_i (r,t);
\]

\[
\frac{\partial C_i (r,t)}{\partial t} = \beta_i \sum_g \Sigma_{f,g} \phi_g (r,t) - \lambda_i C_i (r,t);
\]

\[(g = 1, 2, \ldots, G)\]

where

- \(D_g\) is neutron diffusion coefficient of group \(g\).
- \(\Sigma_{fg}\) is the group removal cross-section.
- \(\Sigma_{fg}\) is fission cross-section for the group \(g\).
- \(\beta_i\) is delayed neutron fraction of \(i\)th group.
- \(\beta\) is the total delayed neutron fraction.
- \(\chi_g^{\beta}\) is fraction of prompt fission neutrons in the group \(g\).
- \(\chi_g^{d}\) is the average number of neutrons released per fission.
- \(\nu\) is the fraction of delayed neutrons in the group \(g\), and
- \(\lambda_i\) and \(C_i\) are the decay constant and the concentration of \(i\)th family delayed neutron precursors, respectively.
- \(i\) is the total number of delayed neutron group (typically 6 groups are considered).

Many approximate methods for the solution of these space-energy dependent neutron kinetics equations have been developed. In TRIKIN code, these equations have been solved using the flux factorization approach, where the total neutron flux is factored into two components, shape function (slowly varying with time) and amplitude (rapidly varying with time). Shape function solution is based on simple finite differencing scheme over square/rectangular or triangular meshes. Amplitude equations are solved by Generalized Runge-Kutta method, an efficient solver for stiff differential equations. The code has been extensively validated against a series of international benchmark problems for different types of reactors involving PHWR, VVER and PWRs.

Validation of Multipoint Kinetics Model

Simulation of loss of coolant accident (LOCA) in one loop of a PHWR540 core has been chosen as a reference problem to validate multipoint model. The core configuration of AECL three dimensional kinetics benchmark problem in a heavy water reactor [5] has been considered for validation. The core geometry is shown in Fig 1. Only the inner and outer core region was modelled excluding the reflector region (regions marked 1, 2, 3, and 4). The PHWR with this core configuration was modelled using both multi-point kinetics model and in-house 3-D kinetics code TRIKIN. System code, RELAP5 generated evolving fuel and coolant states were used to evaluate evolving core void reactivity and other feedback reactivity.
The coupling coefficients \((\alpha_{kh}, \alpha_{hk})\) for multi-point kinetics model were estimated using standard formulae [6]. The value of \(\alpha_{kh}\) was found to be 2.6498E-3. Individual node transient reactivity as a function of time, as obtained from RELAP5 multipoint formulation was superimposed in 3D formulation by artificiality tuning thermal group removal cross section of each cell. These nodal reactivity have been defined in TRIKIN as follows,

\[
\rho_h = \frac{\sum_{i=1}^{N_h} (\nu \Sigma_{f1}\Phi_1 + \nu \Sigma_{f2}\Phi_2) V_i}{\sum_{i=1}^{N_h} (\Sigma_{a1}\Phi_1 + \Sigma_{a2}\Phi_2) V_i}, \quad N_h = \text{number of cells for node 'h'}
\]

\[
\rho_k = \frac{\sum_{i=1}^{N_k} (\nu \Sigma_{f1}\Phi_1 + \nu \Sigma_{f2}\Phi_2) V_i}{\sum_{i=1}^{N_k} (\Sigma_{a1}\Phi_1 + \Sigma_{a2}\Phi_2) V_i}, \quad N_k = \text{number of cells for node 'k'}
\]

In the present calculation \(N_h = N_k\). The normalized power obtained from RELAP5 with multi-point kinetics and TRIKIN with 3-D kinetics is presented in Fig. 3 for broken node and in Fig 4 for intact node. The trends of normalized power from RELAP5 as well as TRIKIN matched each other. However, for node ‘h’, the maximum value of power from RELAP5 is 2.5965 as compared to TRIKIN of 2.03. There is more close agreement for node ‘k’. This can be attributed to more detailed modeling in TRIKIN as compared to that of RELAP5. Also the coupling coefficients are static as these are approximated on the basis of steady state parameters. In reality, as the transient progresses, due to change in neutronic parameters these coefficients should be updated accordingly. The normalized total reactor power is shown in Fig.5.
CONCLUSIONS

Asymmetric reactivity transients initiated due to loss of coolant accident has been simulated in RELAP5 code with multipoint formulation and core transient behavior has been explicitly simulated in full 3D TRIKIN code for validation purpose. Obtained transient reactivity of broken and intact loop core regions are regenerated in 3D code through numerical tuning of thermal group cross sections. 3D simulations brought out that peak power predictions in multipoint formulation differ by 25% in case of broken loop. This is deviation can be attributed to several simplifications involved in neutronic modeling of multipoint formulation viz, one energy group formulation, ignoring thermal neutron leakage, averaging of large nodal area, stationary inter node coupling coefficients and limited number of nodes. However multipoint results are conservative in nature as multipoint formulation treats a reactor core as relatively more decoupled which leads to overestimations of 3D hotspots. Improvements in multipoint formulation in regards to evolving coupling coefficients and involving more number of nodes can help in improving its accuracy to some extent.
REFERENCES


