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Development, validation and application of multi-point kinetics model in RELAP5 for analysis of asymmetric nuclear transients

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HIGHLIGHTS

- A multi-point kinetics model is developed for RELAP5 system thermal hydraulics code.
- Model is validated against extensive 3D kinetics code.
- RELAP5 multi-point kinetics formulation is used to investigate critical break for LOCA in PHWR.

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ABSTRACT

Point kinetics approach in system code RELAP5 limits its use for many of the reactivity induced transients, which involve asymmetric core behaviour. Development of fully coupled 3D core kinetics code with system thermal-hydraulics is the ultimate requirement in this regard; however coupling and validation of 3D kinetics module with system code is cumbersome and it also requires access to source code. An intermediate approach with multi-point kinetics is appropriate and relatively easy to implement for analysis of several asymmetric transients for 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 diffusion theory approximation. This model offers an advantage that associated ordinary differential equations (ODEs) governing multi-point kinetics formulation can be solved using numerical methods to the desired level of accuracy and thus allows formulation based on user defined control variables, i.e., without disturbing the source code and hence also avoiding associated coupling issues. Euler's method has been used in the present formulation to solve several coupled ODEs internally at each time step. The results have been verified against inbuilt point-kinetics models of RELAP5 and validated against 3D kinetics code TRIKIN. The model was used to identify the critical break in RIH of a typical large PHWR core. The neutronic asymmetry produced in the core due to the system induced transient was effectively handled by the multi-point kinetics model overcoming the limitation of in-built point kinetics model of RELAP5 and standalone 3D core kinetics codes.

1. Introduction

Assessment of the safety of an NPP requires that behaviour of the plant following a Postulated Initiated Event (PIE) be analyzed. Also, the plant, its' systems and equipment should be designed to ensure that under normal operation, during operational transients and accident conditions, acceptance criteria are not exceeded. One of the most important PIEs for design of Pressurized Heavy Water Reactor (PHWR) based Nuclear Power Plants (NPP) is the Loss of Coolant Accident (LOCA). For a particular range of breaks in Reactor Inlet Header (RIH), gross flow stagnation may occur in coolant channels in the reactor core leading to rise in fuel temperature. The size of the break that leads to maximum rise in clad surface temperature is called critical break. The critical break in RIH imposes most stringent requirement on Emergency Core Cooling System (ECCS) as the clad surface temperature reaches maximum. Class IV power supply is the main power output supply and it is directly connected to main grid power supply and main plant

Abbreviations: ASDV, atmospheric steam dump valve; ECCS, emergency core cooling system; IRV, instrumented relief valve; IQS, improved quasi-static; LOCA, loss of coolant accident; LORA, loss of regulation accident; LWR, light water reactor; NPP, nuclear power plant; PCP, primary circulating pump; PCT, peak clad temperature; PHT, primary heat transport; PHWR, pressurized heavy water reactor; PIE, postulated initiated event; RIH, reactor inlet header; ROH, reactor outlet header; SG, steam generator; SRV, safety relief valve; TDV, time dependent volume.

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Nomenclature

$\emptyset$	neutron flux
Σ_a	macroscopic absorption cross section
Σ_{12}	macroscopic scattering cross section (from fast to thermal)
Σ_{21}	macroscopic scattering cross section (from thermal to fast)
β	effective delayed neutron fraction
ν	number of neutrons produced per fission
Σ_f	macroscopic fission cross section
λ_i	decay constant of ith delayed neutron group
β_i	delayed neutron fraction of ith group
ρ	reactivity
α	coupling coefficient
v	neutron velocity
t	time
D	diffusion coefficient
$\tilde{C}_i$	precursor concentration of ith delayed neutron group
J	neutron current density
A	interface area between nodes
V	volume of a node
P	power of a node
C_i	delayed neutron concentration of ith group expressed in terms of power
Z	number of nodes
m_d	number of delayed neutron groups
l	prompt neutron life time
K	infinite multiplication factor
∇	del operator
Δ	distance between nodes

generator. Class-IV power supply is derived from grid through start-up transformer and from the turbo-generator system through generator transformer and unit transformer. The loads connected to this system can tolerate prolonged power supply interruption. The size of critical break at RIH will also depend on whether class-IV power supply is available or not. In the present paper, the analysis to identify the critical break in RIH with class-IV power supply unavailable in a typical large PHWR is presented.

The considered PHWR consists of two figure of eight loops and the postulated break occurs in one RIH of one loop leading to neutronically asymmetric behaviour. Considering the core characteristic size (dimensions expressed in terms of neutron migration length) beyond 30, the core is categorized as neutronically loosely coupled (Obaidurrahman and Singh, 2010). As the reactor is loosely coupled, the development and use of multi-point kinetics model to analyse the unsymmetrical core behaviour event mentioned above is required. In the present development, the in-built point kinetics model of RELAP5 was put off and user defined multi-point kinetics models were introduced. The neutron balance equation along with concentration of delayed neutron pre-cursor was solved using Euler's method for every time step. The calculated power was fed into the fuel bundles modelled as heat structures. Diffusion of neutron from one core region to the other was accounted through coupling coefficients in multi-point kinetics equations. The results were verified against inbuilt point-kinetics model of RELAP5 and validated against in-house 3D kinetics code TRIKIN. A range of break sizes ranging from 5% to 100% of double ended guillotine rupture of RIH were analysed using this multi-point kinetics model and the critical break was identified.

Though multi-point kinetics approach is in use for control studies involving core asymmetry (Shimjith et al., 2010), it has not been incorporated with system thermal hydraulic codes to analyse neutronic asymmetric transients generated due to asymmetric thermal hydraulic behaviour or feedback. In this paper an innovative approach is adopted to develop multi-point kinetics coupled with system thermal hydraulics code and apply the same to analyse a neutronic asymmetry transient with thermal hydraulics feedbacks.

2. Point kinetics model of RELAP5

The best estimate system thermal hydraulics code RELAP5/MOD3.4 has been developed by the Idaho National Engineering Laboratory and being extensively used for analysing postulated accidents and transients in water cooled reactor systems. The detailed description of code structure, system models and correlations, solution methods, user guidelines, input requirements, etc. are provided in different volumes of RELAP5/Mod 3.4 Code Manuals (Information Systems Laboratories Inc, 2001). The principal feature of the RELAP5/MOD3.4 is the use of a two-fluid, non-equilibrium and non-homogeneous hydrodynamic model for transient simulation of the two-phase system behaviour. The field equations are coupled by point kinetics model to permit simulation of feedback between neutronics and thermal hydraulics. The point kinetics ODEs are solved using the modified Runge–Kutta method of Cohen and are advanced with the same time step as the thermal fluids and heat conduction equations, and reactivity is assumed to vary linearly between time step values. The data exchange between the point kinetics calculation and the other calculations are explicit. The point kinetics calculations lag the thermal fluids, heat conduction and heat transfer calculations. The reactor power used in thermal fluids and heat conduction is the value at the beginning of the time step. The end of time step values from thermal fluids and heat conduction calculations are used to compute the reactivity used in the point kinetics calculations. A schematic of the data exchange between reactor

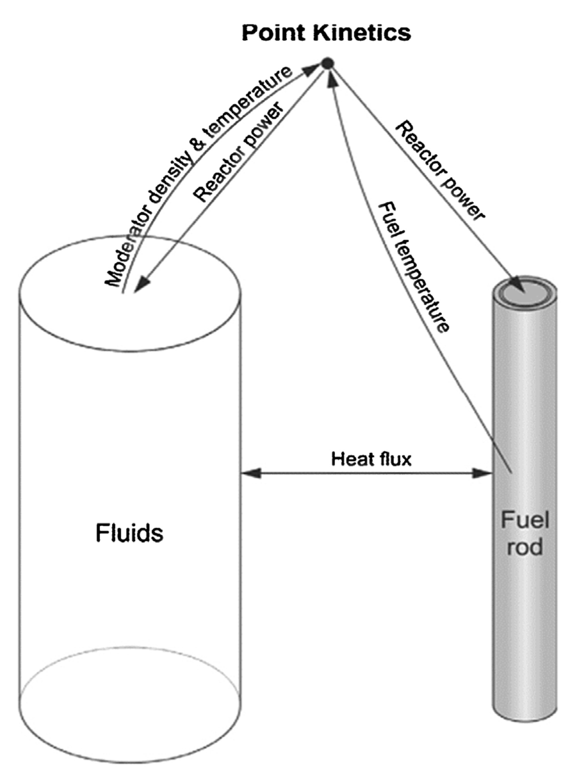

Fig. 1. Schematic of data exchange in RELAP5 in coupled point kinetics calculations (Zhang et al., 2013).