Development of Reactor Core Neutronics and Thermal Physics for the U-PANTHER Simulator

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INTRODUCTION

The University of Pittsburgh Advanced Nuclear Training for Higher Education Reactor (U-PANTHER) simulator is a real-time, desktop, simulation of an "AP1000 like" nuclear reactor. The simulator, currently under development at the University of Pittsburgh, is capable of dynamic response with sufficient fidelity to simulate typical PWR plant response during normal operations and accident situations. It has a graphical user interface (GUI) that mimics control panel instrumentation, including the plant displays represented similarly on the simulator as in a real control room.

The simulation models the reactor core, detailed in this paper, and two heat transfer circuits in the primary system. Each loop has one steam generator, one hot leg, and two cold legs for circulating reactor coolant for primary heat transport. The simulation models the pressurizer in one of the two loops, plus the makeup and charging systems. Two canned-motor pumps are simulated as mounted directly in the channel head of each steam generator.

The most widely used desktop simulation of an advanced PWR in nuclear engineering education is the one developed by the International Atomic Energy Agency (IAEA)[1]. However, at this stage of its development, it does not provide high fidelity simulation of a PWR when control systems are placed in manual operation mode. Further, that simulator is not "open source" and does not provide users the ability to examine or modify the internal models.

The mathematical models for the simulator have been implemented in MATLAB / Simulink™ and configured to run on a personal computer so that users can both see the underlying models and use the simulation program in laboratory-like sessions to observe plant dynamic behavior and study the effect of design changes on plant dynamic behavior. An overview of the design of the U-PANTHER simulator is discussed by Schaefer et al. [2]. This paper outlines the development of the reactor core physics (PRX) model of the core neutronics and its integration with the reactor coolant system (RCS) model.

REACTOR CORE PHYSICS

The PRX model, is a 3D space-time model that simulates the physics and reactor kinetics inside a PWR reactor core. This model is responsible for providing the spatial neutron flux and power distribution throughout the reactor core.

When computing the neutron flux and power in the reactor core, the PRX model uses control and safety rod positions from the U-PANTHER Rod Control System (PRD) model, and coolant temperature, coolant density, fuel temperature, and core pressure from the Reactor Coolant System (RCS) model. The PRX model does not account for fuel burn-up, but rather operates under constant middle-of-life conditions, and does not account for the moderating effects of boron.

Software Communication and Hierarchy

The PRX model receives core pressure, coolant density, coolant temperature, and fuel temperature from the RCS model (see Fig. 1). These values are then mapped from the coarse 5x3 RCS nodal mesh to the fine 15x15x10 PRX nodal mesh. The coolant/moderator state (pressure, temperature, and density) of the refined mesh and rod positions from the PRD model are used to calculate the macroscopic absorption cross section at each node. The cross section data and the diffusion matrix are then used to calculate the neutron flux in each node. The flux values are sent to a block that calculates the core power and maps it back onto the 5x3 RCS nodal mesh. The core power is output to the RCS model. The spatial neutron flux is also used by the Nuclear Instrumentation System (PXC) to determine flux values near ex-core detectors.

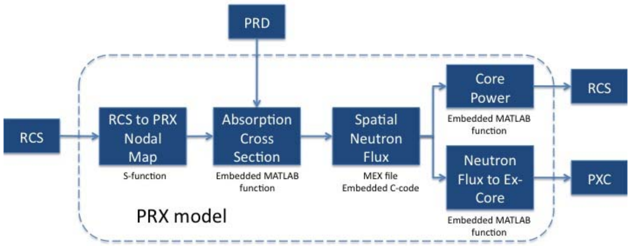

Fig. 1. Input/output block diagram of the PRX model.

Assumptions and Approximations

The PRX model employs several assumptions and approximations. We assume all neutrons in the core have the same energy (one speed approximation). We assume that each node is homogeneous and corresponds radially to one fuel assembly. The scalar flux is assumed to be spatially constant within each node, and finite-differencing is used to approximate the neutron current between adjacent nodes as a function of the average flux of each node. The delayed neutrons are modeled using only one precursor group. The node boundary conditions are approximated by imposing an albedo coefficient of reflection for each node surface that borders the outer edge of the core. The control and safety rods are all assumed to have the same rod worth. When calculating the cross section values for partially rodded nodes, we linearly interpolate between rodded and unrodded values based on control rod depth into the node. The cross sections are assumed to vary linearly with moderator density.

Mathematical Model Description

The PRX model uses the 3D steady-state neutron diffusion equation (modified one-group) to calculate the flux distribution, and a finite difference technique is used to integrate the diffusion equation in space. We integrated the modified one-group equations in time using an Euler integration scheme. The current mesh is set at 225 radial nodes and 10 axial nodes; it does not support variable mesh sizes. The model divides the core into a 15x15x10 node 3D mesh and calculates the average flux for each node. The core map, shown in Fig. 2, shows a cross section of the core, control and safety rod channels, and the location of ex-core detectors.

Because the RCS model uses a different nodal mesh than the PRX model (5x3 vs. 15x15x10), the spatial temperature, density, and power values passed between the PRX and RCS models are mapped according to the nodal map shown in Fig. 3. For PRX cells split by RCS nodes the coolant state is the average of the RCS values.

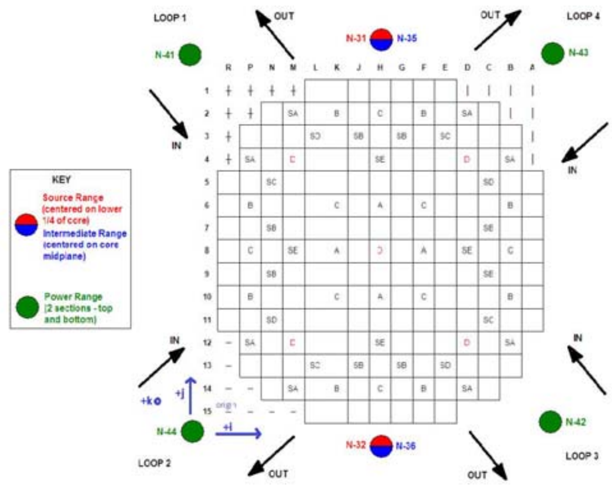

Fig. 2. PRX Core Map.

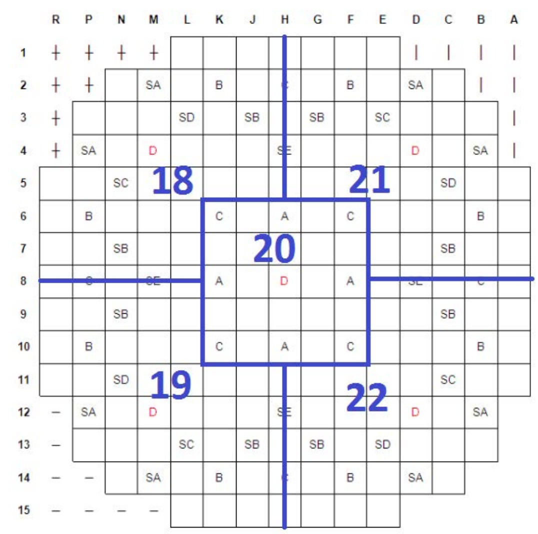

Fig. 3. Nodal map between RCS and PRX models at the bottom (core entry) layer.

Cross Section Values

The PRX model assumes the macroscopic absorption cross section, macroscopic fission cross section, and diffusion coefficient are initially uniform [3]. The "base" values of the absorption cross section are then modified based upon the control and shutdown rod positions (from the PRD model.) There are 228 rod step positions: 0 = full insertion; 228 = full removal. The PRX model uses these positions and the core map in Figure 1 to calculate the extent to which nodes will be rodded. The cross sections do not change for un-rodded nodes. The cross section for partially rodded nodes is linearly interpolated between rodded and un-rodded values. By adjusting control rod positions and measuring core power changes, we determined that the fully rodded absorption cross section $\Sigma_{rodded} = 2\Sigma_0$ is a good approximation of control rod worth. Using the cross section modified for rod position, the PRX model then modifies the absorption value for changes in fuel and moderator temperature.

Source Term Definition

The source term in the flux calculation is defined as the delayed neutron source. This is the sum of all neutrons produced by the decay of precursors. The PRX model uses a 1-group approximation for delayed neutrons.

$$q = \lambda_{eff} v \Sigma_f \phi + \lambda_{eff} C e^{-\lambda_{eff}\Delta t}$$

Here λ_{eff} is the decay constant, C is the precursor concentration, Δt is the time step, v is the number of neutrons emitted per fission, Σ_f is the macroscopic cross section, and ϕ is the neutron flux.

Flux Calculation

The PRX model solves the 3D one speed diffusion equation:

$$\frac{1}{v}\frac{\partial \phi}{\partial t} - \nabla \cdot \left(D \nabla \phi\right) + \Sigma_a \phi = v \Sigma_f \phi + q$$

Note that spatial dependence of diffusion coefficients and cross sections is included. We homogenize over equally distributed sub-volumes of the reactor so that the core is represented by a mesh of nodes. A first-order finite difference is used to write currents as functions of the average flux in neighboring nodes and the diffusion coefficient between them. The result is a time-dependent matrix diffusion equation,

$$\frac{1}{v}\frac{d}{dt}\vec{\phi}(t) = -M(t)\vec{\phi}(t) + \vec{q}(t)$$

where $M(t)$ is a matrix combining finite difference approximations of the diffusion and cross section terms at the various nodes at time t, and $\vec{\phi}$ is a vector of nodal fluxes. This matrix equation is integrated in time using Euler integration.

Power Calculation

Given the flux at each node, the power is calculated by

$$P(\vec{r}) = E_f \Sigma_f(\vec{r})\phi(\vec{r})$$

where E_f is the average energy released per fission. Because the Reactor Coolant System (RCS) model uses a coarser mesh than the PRX, the model integrates the power over a larger volume before sending the values back to the RCS.

GUI Interface and Screen Display

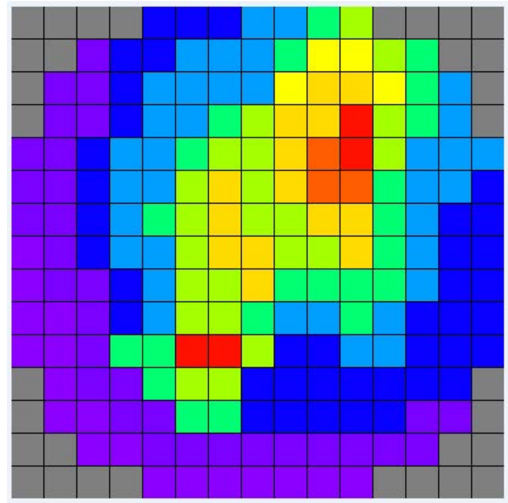

Fig. 4. Radial flux distribution display.

The GUI can display the spatial flux distribution. The full 3D flux is not displayed in detail; rather, a 2D image of the average radial flux (see Fig. 4), and a 1D image of the average axial flux are displayed (not shown).

Integration with reactor coolant system model

The heat generated by the PRX model is transferred to the RCS model (see Fig. 1). This results in the energy balance that describes the transfer of energy from fission, to the primary side coolant, and ultimately to the steam generator secondary side in the steam generator model (SSG). The change in coolant temperature results in temperature and moderator feedback when coolant temperatures feedback into the PRX model.

A fuel conductivity model describes the quasi-steady state conductivity based on thermal resistance applied from the fuel centerline to the (moderator) coolant. Resistances considered include: fuel thermal conductivity, thermal resistance at the fuel-cladding interface, thermal resistance of the fuel cladding, and resistance at the cladding-coolant interface.

Model Verification

As mentioned above, the rod absorption cross section for a fully rodded node is $\Sigma_{rodded} = 2\Sigma_0$ to account properly for control rod worth. This was chosen so that the insertion of control rod banks resulted in a proper change in core power. The prompt jump factor

$$\frac{P}{P_0} = \frac{\beta}{\beta - \rho}$$

describes this change where β is the delayed neutron fraction (700 pcm) and ρ is the change in reactivity (-2 pcm) per rod step; the desired prompt jump is 0.997.

To verify the rod worth, fuel and temperature feedback effects were temporarily removed to isolate the rod effects. The reactor power was brought to steady state and control bank D was inserted to the core at 48 steps/min (see Fig. 5). The resulting prompt jump of 0.997 demonstrates the rods were inserting the desired amount of reactivity.

Next, the control and shutdown rods were maintained at a constant position (control bank D at 220 steps, all others at 228 steps (fully removed)), and the model run to steady state (~50 seconds). A transient exists because the initial conditions are not an equilibrium condition. Trends in the reactor power follow as expected changes in reactivity due to changes in fuel and coolant temperature.

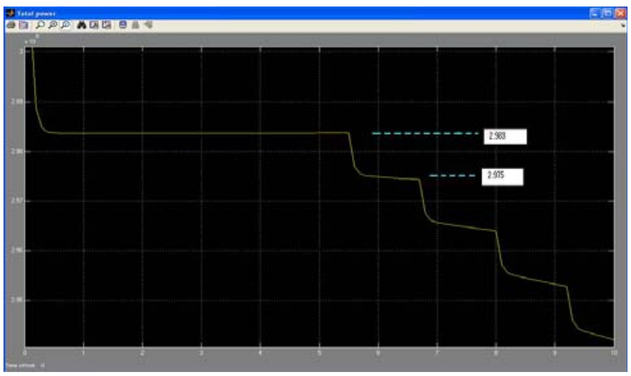

Fig. 5. Prompt jump factor for discrete rod steps.

CONCLUSION

The Reactor Core Physics (PRX) model is a 3D space-time model that simulates the physics and reactor kinetics inside a PWR reactor core. This model is responsible for providing the spatial neutron flux and power distribution through out the reactor core. When computing the neutron flux and power in the reactor core, the PRX model uses control and safety rod positions from the U-PANTHER Rod Control System (PRD) model, and coolant temperature, coolant density, fuel temperature, and core pressure from the Reactor Coolant System (RCS) model.

The PRX model outputs heat to the RCS model, which accounts for the energy balance between the reactor and primary loops. This is also the mechanism for temperature feedback and its effect on reactivity. The PRX model accounts for changes in absorption cross section due to changes in fuel and moderator (coolant) temperature.

The mathematical model of the reactor core physics (PRX) model has been implemented in MATLAB / Simulink™ and configured to run on a personal computer. One advantage of this model is that users can see the underlying models, use the simulation program in laboratory-like sessions to observe reactor dynamic behavior, and study the effect of design changes on reactor dynamic behavior.

NOMENCLATURE

PRX	Reactor Core Physics
RCS	Reactor Coolant System
PRD	Rod Control System
PXC	Nuclear Instrumentation System
SSG	Steam Generator System
GUI	Graphical User Interface

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