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A RAST-K Module for Multiphysics Simulation and Accident Analysis of Pressurized Water Reactor Cores

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1. Introduction

The nuclear researchers and engineers have long dreamt of a safe and cost-competitive nuclear reactor designs. In order to enhance the nuclear reactor safety and economics, the methods providing a less conservative estimation of safety parameters are needed. Such high-fidelity methods always attract a lot of attention of the nuclear researchers worldwide.

The continuous evolution of reactor core numerical methods and computer technologies has approached the high-fidelity neutron transport simulation with pin-by-pin resolution [1,2]. The further improvement of the nuclear reactor simulation requires the tough coupling of neutronics, coolant thermal-hydraulics, fuel rod thermal-mechanics and other physical models.

Notwithstanding the presence of supercomputers, massive parallel and distributive computations, the nuclear engineers still need reliable codes that consume much less time and memory resources compare with the high-fidelity transport codes. This work is devoted to the development of coupled multi-physics code designed for hypothetical accident analysis of pressurized water reactor cores.

2. Methods and Results

In this section we describe the codes using for multiphysics calculation and the scheme of their coupling.

2.1 Coupled Code Modules

The coupled code consists of four modules that consequently calculates the reactor core neutronics, coolant thermal-hydraulics, fuel rod thermal-mechanics.

RAST-K [1] is the neutron diffusion nodal code for the steady-state and transport calculation of the pressurized water reactor cores with rectangular assemblies. The code utilizes the two-group cross-section library calculated by a lattice code.

CTF [2] is a thermal-hydraulic simulation code designed for light water reactor vessel analysis. The code uses a two-fluid, three-field modeling approach and was originally developed by Pacific Northwest Laboratory.

FRAPTRAN [3,4] is a fuel performance code designed to perform the transient and hypothetical accidents simulation of light-water reactor fuel rods. The phenomena modeled by the code include heat conduction, fuel and cladding mechanical deformation, cladding oxidation, fission gas release and fuel rod gas pressure.

FRAPCON is a fuel performance code designed to calculate the steady-state response of light-water reactor fuel rods during long term burnup. The numerical methods and modeled phenomena of the code are similar with FRAPTRAN. Both codes were developed by Pacific Northwest National Laboratory and are used for certification of the light-water core designs.

The flowchart of the data exchange is depicted in Fig. 1.

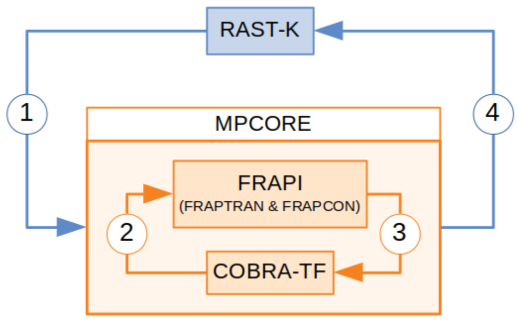

Fig.1. Flow of data between RAST-K, CTF, FRAPTRAN and FRAPCON: 1) linear power, inlet coolant mass flow, pressure and temperature; 2) coolant pressure and cladding temperature; 3) pellet-cladding gap heat transfer coefficient; 4) coolant density, fuel, cladding and coolant temperatures

2.2 Coupling Scheme

At the every time step RAST-K updates the pin-by-pin power distribution, inlet coolant mass flow, temperature and pressure. Using this data CTF performs thermal-hydraulic calculation of the considered core and forwards the cladding outside surface temperature and coolant pressure to FRAPTRAN as a boundary conditions. FRAPTRAN runs the thermal-mechanical and gas release models and update the cladding to pellet gap heat transfer coefficient. CTF receives the updated coefficient and whole iteration cycle is repeated until convergence.

The same scheme is used for the burnup calculation, with only difference that FRAPCON is used here instead FRAPTRAN. The transient initial state of the reactor core corresponds to the state at the certain step of the burnup time step iteration.

In order to adapt the FRAPCON and FRAPTRAN fuel performance codes for the time integration with a tough coupling, the Fuel Rod Analysis Program Interface FRAPI has been developed. FRAPI runs a fuel performance code and updates a fuel rod state at the each time step with respect to the data transferred by an external code. Also FRAPI restarts the previous fuel rod state when within time step iteration is repeated.

2.3 RIA Simulation

The core consists of the 4 fuel assemblies and 5 reflector blanks arranged as depicted in Fig.2. Each assembly contains 5 guide tubes and 236 fuel rods organized in 16x16 lattice. At the initial time the control rods of the rodded assembly has the elevation 315 cm from bottom. From 100 to 200 ms the control rods are gradually ejected from the core.

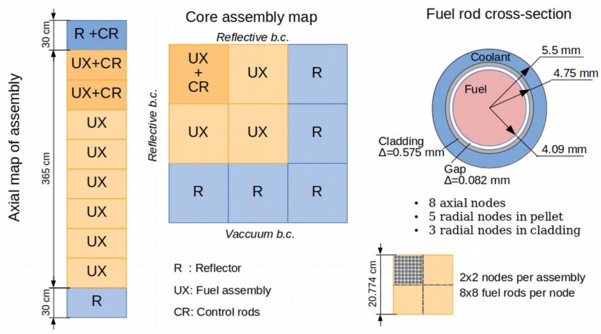

Fig. 2. Dimensions and mesh of the assemblies and fuel rods in the 3x3 core.

The control rods position change results to the positive reactivity about 0.8 $ in the core and repid increase of the total core power. The responding increase of fuel temperature and Doppler effect lead to the further decrease of the reactivity. As a result the pulse of the reactor core power is observed, Fig.3.

The control rod ejection accident is simulated with the dynamical model of pellet to cladding gap heat transfer coefficient. The pin-by-pin distribution of the linear power and fuel pellet centerline temperature are presented in the Fig.4.

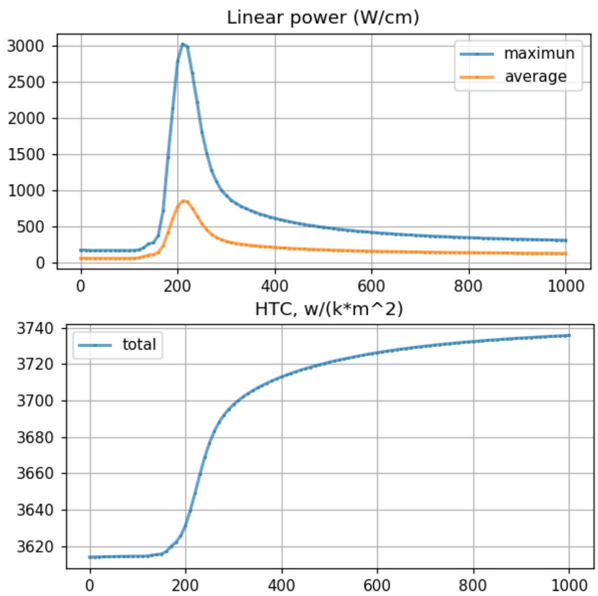

Fig.3. Linear power (*W/cm*) and heat transfer coefficient (*W/(K*m²)*) change on time (*ms*) for the control rod ejection accident in the 3x3 core.

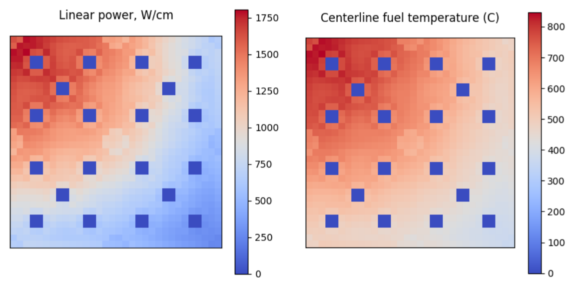

Fig.4. Linear power (*W/cm*) and center-line fuel temperature distribution (*C*) at the time 200 *ms* for the control rod ejection accident in the 3x3 core.

3. Conclusions

The coupled code for the pin-by-pin multiphysics simulation has been developed. The code includes the diffusion nodal solver for reactor core transient and burnup calculation, two-fluid thermal-hydraulic module for coolant simulation and fuel rod performance module. The developed code performes transient calculation using a dynamical change of the pellet to cladding gap size and corresponding heat transfer coefficient. The code can improve the accuracy of the hypothetical accidents simulation for light water reactor core.

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