Coupled Simulation of Component Thermal Hydraulics and Neutron Kinetics for a Nuclear Reactor Core with CUPID and MASTER

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1. Introduction

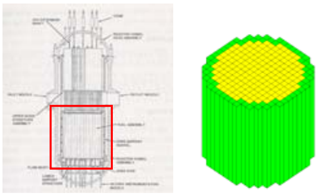
For the analysis of transient two-phase flows in nuclear reactor components such as a reactor vessel, steam generator, containment, etc., KAERI has developed a three-dimensional thermal hydraulics code, CUPID [1, 2]. It adopts three-dimensional, transient, two-phase and three-field model, and includes physical models and correlations of the interfacial mass, momentum and energy transfer for the closure. In our previous papers [3, 4], the numerical schemes and the two-phase flow models were verified and validated with various conceptual problems and experimental data.

In the present paper, the multi-physics simulation for a nuclear reactor was attempted using the CUPID code. As a first step, the CUPID code was coupled with the three-dimensional neutron kinetics code, MASTER using the dynamic link library (DLL) feature. In order to assess the overall performance of the coupled code, the thermal hydraulic and neutronic conditions of the modeled APR1400 reactor core during the normal operation were simulated. The following sections present the numerical modeling for the reactor core, applied porous media model, coupling of the kinetics code and the simulation results.

2. Numerical Methodology

2.1 Reactor Core modeling

Fig. 1(a) shows the schematic diagram of the reactor vessel [5] of APR1400. In this calculation, the fuel assemblies and their outer reflector area were merely modeled. 177 fuel assemblies, each of which includes 16x16 fuel rods, were simulated and the modeled fuel assemblies and reflector region are indicated in Fig. 1(b). In a horizontal plane, the fuel assemblies were divided into 177 regions by matching each one to a single fuel assembly and 16 cells are assigned in axial direction. A total of 3856 meshes, thus, were employed for the thermal-hydraulic analysis using CUPID including the reflector region.



(a) schematics (b) computational geometry

Fig. 1. Schematics and computational geometry of APR1400

2.2 Porous media approach

In order to simulate the two phase flow in the fuel assembly region, the porous media approach [6] was adopted. Since it is nearly impossible to model the fluid and structure regions of the reactor core exactly, the porous media approach is an effective assumption in the nuclear thermal hydraulics and it was known to give a reasonable solution for two-phase flow analysis in a complex geometry.

Porosity is a measure of the void space in an arbitrary medium. The definition of porosity in a fuel assembly is obtained as follows;

$$\gamma = 1 - \frac{(\pi D^2/4)\times No.\ of\ \ rod}{L \times L}, \qquad (1)$$

where, D is outer diameter of a fuel rod and L is width of a unit fuel assembly. Permeability is a measure of the ability for fluid to transit through cell faces and achieved with an assumption that the fluid flows freely in between cells radially but is partially constraint in the axial direction by the array of rods. Hence, Permeability in each direction is as follows;

$$\varepsilon_x = \varepsilon_y = 1\ ,$$
$$\varepsilon_z = \gamma\ . \qquad (2)$$

The reflector region is treated as another porous medium with the arbitrary porosity to control the inlet mass flux by targeting 5% of total inlet mass.

2.3 3D kinetics code coupling

For the better estimation of this simulation combined with thermal hydraulics and neutron kinetics, the 3D reactor core kinetics analysis code, MASTER, is integrated into CUPID code. Generally, it may cause any unexpected error or malfunction to combine two stand-alone programs. MASTER, to avoid such a problem, has been converted into Dynamic Link Library (DLL). For the generation of the MASTER dynamic link library (DLL), the link variables needed for incorporating thermal feedback effects were identified first and a data exchange scheme was established. The schematic diagram is shown in Fig. 2. In order to solve the energy equation in porous zone, the heat power from the fuel rod should be needed as a source term. MASTER DLL provides the distribution of heat generation to the CUPID code. Since the CUPID has a module to calculate a radial 1D conduction equation of the fuel rod by using gap conductance assumption between pellet and cladding, it produces the temperature profile of fuel rod which is used as input data for MASTER DLL as well as moderator's thermal properties such as density, temperature, etc.. The axial distribution within each computing cell is neglected because the geometry is assumed to be invariant along the axial direction.

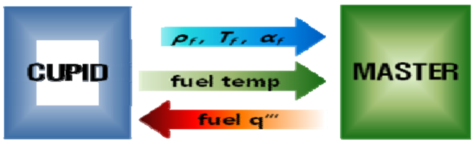
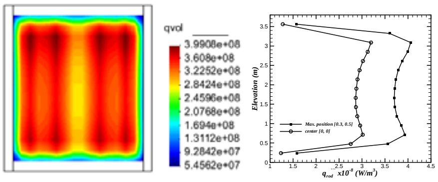


Fig. 2. CUPID-MASTER coupling

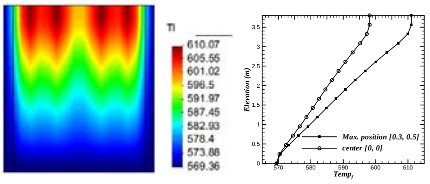
3. Result and Discussion

Fig. 3 shows the contours and axial profiles of heat generation, liquid temperature and void fraction in fuel assembly for benchmark simulation. In each profile, two different positions are considered; one is at the position of maximum heat power and the other is along the centerline. In this simulation, the heat generation from the MASTER DLL is assumed to be steady as a function of time. The result of MASTER DLL as shown in Fig. 3(a) indicates that the axial heat generation goes to the peak at upper and lower region and appears a parabolic profile in between two peaks. Besides, the radial distribution of heat source shows the annulus shape regardless of axial position. The heat power shows maximum output at about z/H=0.8125.

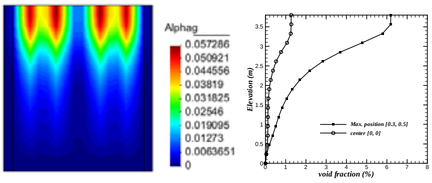
The radial temperature profile of the fuel rod and the fluid in porous zone is directly affected by the distribution of heat generation. Hence, the liquid temperature shown in the Fig. 3(b) is the highest where the output of the fuel rod shows maximum. The axial distribution, however, is almost linear except the exit region which means that the entire domain is maintained to be subcooled. And also, a subcooled boiling is observed at the upper region (Fig. 3(c)). The void fraction is up to about 6% at where the heat generation is maxima.



(a) Heat generation in reactor core



(b) Liquid temperature



(c) Void fraction

Fig. 3. Contours and axial profiles for core heat generation, liquid temperature in porous medium and void fraction

4. Conclusions

In this study, the multi-dimensional multi-physics simulation of thermal hydraulics and neutron kinetics for a nuclear reactor core was attempted. The component thermal hydraulic analysis code CUPID was coupled with the three-dimensional neutron kinetics code, MASTER using the dynamic link library (DLL) feature. The simulation result for the APR1400 reactor core showed that the multi-dimensional multi-physics analysis was successfully performed with the coupled code and this approach may be able to benefit the safety analysis for the situation when the asymmetric neutronic condition occurs accidentally. However, further improvement of the physical models such as a wall friction factor, turbulent diffusion coefficient etc. for porous media is required for more realistic simulation of the reactor core.

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