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Steady-state Thermal-hydraulic Analysis of PWR Whole Reactor Core in Subchannel Scale Using CUPID

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1. Introduction

With increasing computing power and improved numerical algorithm, high-fidelity and multi-physics analysis for the reactor safety evaluation is becoming feasible gradually and many attempts have been made to apply the latest codes and methods as an advanced safety analysis tools. They may allow the coupled 3D whole core transient analysis, which is considered as the most suitable tools for transient analysis under asymmetric power distribution conditions. These methods can assure higher safety margin and minimize the economic uncertainty by optimizing fuel cycle costs and fuel assembly design [1].

Considering the computational power necessary for a whole core pin-by-pin analysis, a subchannel scale is desired to achieve both required accuracy and endurable computational time. Subchannel indicates the imaginary flow area which is surrounded by fuel rods. In the subchannel scale analysis, one computing cell represents one subchannel in a reactor core.

Recently, in the CASL (Consortium for Advanced Simulation of Light water reactor) project [2], subchannel scale T/H analysis code COBRA-TF [3] has been utilized to attempt for 3D whole core transient simulation coupled with neutronics codes. Also, AREVA has developed ARCADIA code system [4] which enables the coupled T/H and neutronics analysis for whole reactor core.

In the present study, feasibility test for subchannel scale whole core T/H analysis using CUPID [5] was performed. CUPID is a component scale T/H analysis code developed by KAERI (Korea Atomic Energy Research Institute). It adopts three-dimensional two-fluid model for solving the governing equations. It has highly parallelized numerical solver and the code performance was tested with various simulations. These features of CUPID would be advantageous to realize whole core T/H simulation.

In this paper, required key subchannel models were implemented to CUPID and feasibility test results for subchannel scale whole core T/H analysis for the APR1400 reactor core was introduced.

2. Implementation of Key Subchannel Models

The fluid transfer between adjacent subchannels can be explained by three mechanisms, i.e. diversion cross flow, turbulent mixing and void drift [6]. These mechanisms are modeled as a closure terms to solve the mass, momentum and energy conservation equations. These models are presented by Todreas and Kazimi [7] and Hwang et al [8].

2.1 Pressure Drop Model

Diversion cross flow can occur due to the lateral pressure difference between adjacent subchannels. The pressure drop model consists of the wall friction and form loss models with consideration of flow direction in subchannel. For axial direction, wall friction loss on the fluid-rod interface and grid spacer model formulated with form loss were included. These models were added to the axial momentum equation of CUPID as a type of pressure drop as follows,

$$\Delta P = -\frac{1}{2}\left(\frac{f}{d_{hy}} + K\right)\left(\frac{G^2}{\rho}\right) \tag{1}$$

where, f and K are the wall friction factor and the form loss coefficient for a grid spacer, respectively. Wall friction factor is the function of Reynolds number and it has different value with laminar and turbulent flow conditions.

To consider the fuel gap change by rod arrangement in the transverse direction, the form loss model was implemented to the transverse momentum equation of CUPID as follows,

$$\Delta P_L = -\frac{K_G}{2}\left(\frac{W_{IJ}\left|W_{IJ}\right|}{l_{IJ}\rho_{IJ}s_{IJ}}\right) \tag{2}$$

where, W_{IJ} is the mass flow which flows subchannel I to J and l_{IJ} is the length between the center of subchannel I and J and s_{IJ} is the gap size between two fuel rods. K_G means transverse form loss coefficient and the default value is 0.5.

2.2 Turbulent Mixing Model: Equal Volume Exchange Model

The fluid exchange can occur due to the turbulent mixing which caused by turbulent fluctuation and flow disturbance by structure in reactor core. In single-phase incompressible flow with heated condition, mass, momentum and energy can be transferred between adjacent subchannels due to the density difference at each subchannel. These fluid mixing mechanisms are modeled by Equal Volume exchange model (EV model) and the model is added as a source term in mass, momentum and energy conservation equations. Following the Todreas and Kazimi [7], turbulent mixing of mass, momentum and energy for liquid phase between adjacent subchannels *I* and *J* can be captured as follows,

- Turbulent mixing of mass transfer

$$\vec{M}_e^T = \frac{\beta \bar{G} s_{IJ}}{\bar{\rho}} \left[\rho_{l,j}\alpha_{l,j} - \rho_{l,i}\alpha_{l,i} \right] \tag{3}$$

- Turbulent mixing of momentum transfer

$$\vec{M}_k^T = \frac{\beta \bar{G} s_{IJ}}{\bar{\rho}} \left[\frac{\dot{m}_{l,j}}{A_j} - \frac{\dot{m}_{l,i}}{A_i} \right] \tag{4}$$

- Turbulent mixing of energy transfer

$$\vec{M}_h^T = \frac{\beta \bar{G} s_{IJ}}{\bar{\rho}} \left[\rho_{l,j}\alpha_{l,j}h_{l,j} - \rho_{l,i}\alpha_{l,i}h_{l,i} \right] \tag{5}$$

where, $\dot{m}$ is mass flow rate, A is flow area, β is mixing parameter which is determined by user input, $\bar{G}$ and $\bar{\rho}$ are area-averaged axial mass flux and density between adjacent subchannels, respectively.

3. Feasibility Test of Subchannel Scale Whole Core T/H Analysis for APR1400

The subchannel scale whole core T/H analysis for APR1400 (Advanced Power Reactor 1400 [MWe]) was conducted using CUPID. The simulation was carried out against hot full power steady state of APR1400. For high precision simulation, detailed geometry features of reactor core and individual rod power distributions from the neutronics code, nTRACER [9] were considered in this simulation.

3.1 Subchannel Scale Whole Core Modelling

The reactor core of APR1400 consists of 241 fuel assemblies and each fuel assembly includes 236 fuel rods and 5 guide tubes. In the simulation, the core shroud and inter-assembly water gap were considered. In accordance with the detailed geometry features of reactor core, all computing cells were characterized into 13 different types as indicated in Fig. 1 and Table I. The subchannel types include fuel assembly, guide tube, water gap and core shroud. Thereafter, geometry information which are required for the calculation such as porosity, permeability, hydraulic diameter and gap size were input following subchannel types. By this classification method, desired local T/H information at the specific parts in reactor core could be obtained.

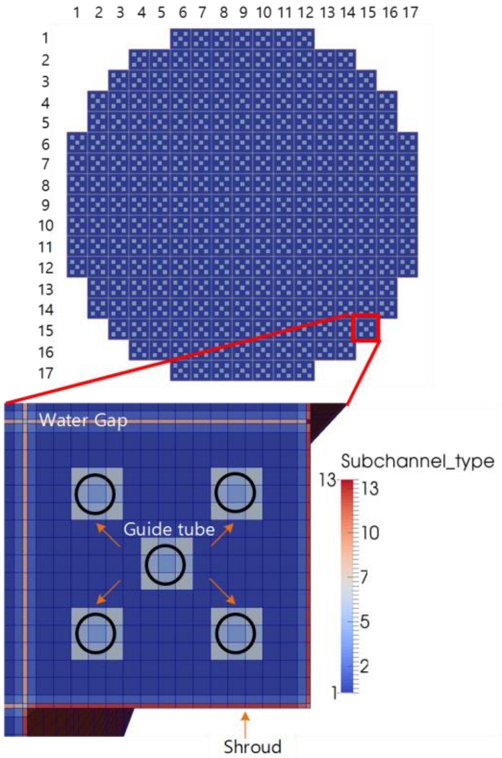

Fig. 1. The definition of subchannel type in reactor core

Table I: Subchannel type definition

Type	Name
1	Assembly (center)
2	Assembly (side)
3	Assembly (corner)
4	Guide tube (center)
5	Guide tube (side)
6	Guide tube (corner)
7	Water gap (center)
8	Water gap (corner)
9	Water gap (side)
10	Shroud (edge)
11	Shroud (near edge)
12	Shroud (side wall)
13	Shroud (between gap)

3.2 Implementation of Individual Rod Power to Subchannel

Individual rod power distributions from the calculation results of neutronics code, nTRACER, were applied to each subchannel for describing the detailed whole core pin-by-pin power distributions. For applying the individual rod power to each subchannel, it was necessary to designate the subchannel to rod connectivity. Afterwards, the loaded power to each subchannel could be obtained from the connectivity and applied to CUPID.

3.3 Simulation Results of Whole Core T/H Analysis for APR1400

The simulation was performed at normal operation conditions for hot full power steady state of APR1400. The coolant enters the reactor core with averaged mass flux of 12.6 million Kg/h-m^2 and the pressure is set to 15.5 MPa. In the calculation, 2,675,698 cuboidal meshes with non-uniform height of 34 meshes were used. The number of computing cells was nearly evenly distributed with the maximum difference less than 15 to each processor by METIS program [10] in CUPID. Therefore, it was possible to perform efficient calculation with achieving good computational load balancing. The CUPID runs were made on a LINUX cluster built with Intel® Xeon® E5-2600 CPUs. The calculation was conducted until a steady state was achieved and the total wall clock time to finish the calculation with utilizing 50 processors was 1h 29 min. The assigned number of computing cells to each processor is shown in Fig. 2.

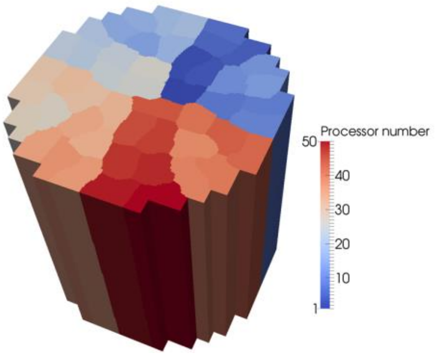

Fig. 2. Number of computing cells to each processor

The 3D axial contour maps of the implemented core power density distributions and the calculated coolant temperature distributions are presented in Fig. 3 and Fig. 4, respectively. Reasonable core power distributions including cosine shape for axial direction and increased coolant temperature were obtained.

Also, the 3D view of calculated coolant velocity distributions is presented in Fig. 5. The acceptable calculation results including low velocity at guide tube and high velocity at water gap could be obtained.

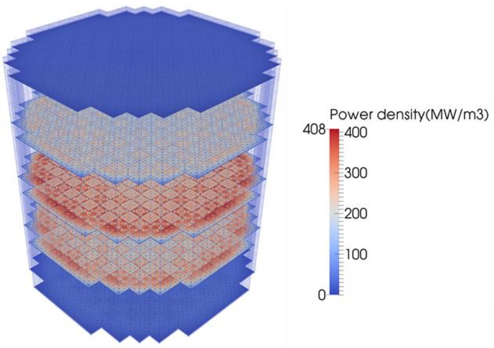

Fig. 3. Implemented power density contour map

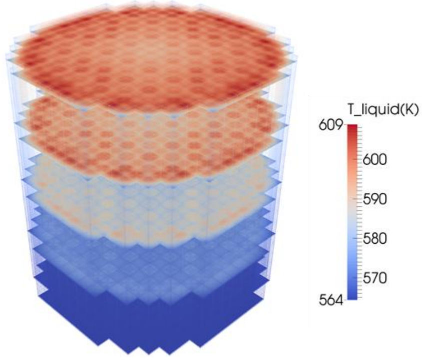

Fig. 4. Coolant temperature contour map

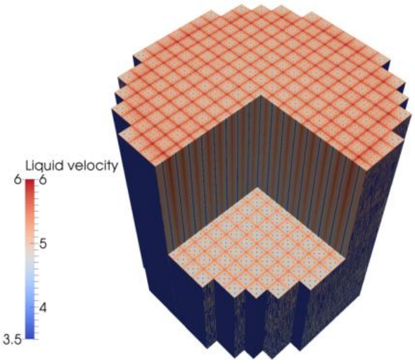

Fig. 5. Coolant velocity distributions

Additionally, preliminary calculation to evaluate the capability of CUPID for predicting the DNBR (Departure from Nucleate Boiling Ratio) was carried out. It was required to use proper CHF correlation to predict the DNBR. However, in this calculation, one of the simple forms of CHF correlation, Biasi correlation [11] was used because the CHF correlation for APR1400 is confidential. The calculated DNBR at the middle elevation of reactor core is presented in Fig. 6. The minimum DNBR was found at the assembly corner subchannel and the value was 2.21.

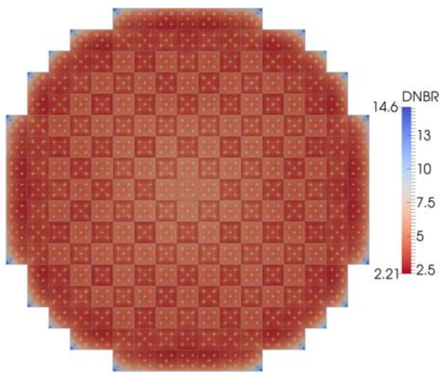

Fig. 6. Calculated DNBR at the middle of reactor core

4. Conclusions

In this paper, required key subchannel models such as pressure drop and turbulent mixing models were implemented to CUPID. Thereafter, feasibility test of subchannel scale whole core T/H analysis for APR1400 reactor core was carried out. From the test results, the capability of CUPID for the high precision whole core pin-by-pin T/H analysis was confirmed. Also, CUPID showed its capability of predicting the minimum DNBR for reactor safety analysis.

In the future, more validations for the two-phase flow will be performed and required T/H models will be implemented to CUPID for an accident analysis.

ACKNOWLEDGEMENT

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