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Development of a steady-state sub-channel code for small reactor on the basis of combined cross momentum and non-linear conduction

Chen Huandong, Zhang Xiaoying*

School of Electricity, South China University of Technology, Guangzhou, China

HIGHLIGHTS

- Combining equations to have a more stable and faster convergence solution.
- Taking account of non-linear conduction of fuel rods.
- Validating code with COBRA code.
- Applying code to the small reactor "MUTSU" and comparing result with its design conditions.

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ABSTRACT

For purpose of thermal hydraulic analysis in small nuclear reactors, a sub-channel code with an improved convergence has been developed based on the homogenous flowing model. A combined lateral momentum equation coupling with continuity and axial momentum equation has been used to substitute the original lateral momentum equation. The Gauss iteration method has been adopted to solve the Kirchhoff's transformation equation of nonlinear heat conduction of fuel rod, a temperature dependent conducting has been considered. The code has been validated by using experimental data from the NUPEC PWR Sub-channel and Bundle Tests (PSBT) and then applied to the "MUTSU" reactor. Results show that the code can predict the experimental data with acceptable accuracy and has ability to analyze the small PWR reactor.

1. Introduction

The severed accidents happened in nuclear plant in past decades showed danger of reactor core (Son and Suh, 2014). Design and safety operation of reactor core is still attracted focus of study in nuclear industry. The most common concern is thermal hydraulic characteristics of the core. Much research has been published about this problem, about 10% of them made numerical simulation with thermal hydraulic code (Jing et al., 2012).

In view of computation analysis of thermal hydraulic characteristics in the nuclear core, sub-channel model and code have been widely applied. The most important feature of sub-channel analysis is to simulate the transverse interchange of mass, energy, and momentum between adjacent channels in the core. The interchange has effect of reducing the enthalpy and temperature of coolant in hot channel, as well as the temperature of the fuel element. If the simulation is preciser, the critical heat flux and ratio of critical heat flux would increase with decreasing coolant temperature, thus improves safety and economy of the reactor core (Yu, 2002). The most significant difficulty of sub-channel analysis is convergence of solution, caused by the lateral flow driven by pressure difference between the adjacent flowing channels. Research about the lateral momentum equation and solution with better convergence is among the most important tasks in this study for the strong coupling effect between cross flow, axial momentum, and pressure difference of adjacent channels. It is even more difficult in thermal hydraulic sub-channel analysis of a small nuclear reactor core, as flowing rate of coolant is very small, which may cause a significant inter effect of mass flow, pressure and temperature. Heat conduction in fuel rod is another research issue need to be addressed in sub-channel analysis. One of the most remarkable characteristics in heat conduction of fuel rod is the varying material property with temperature, which has significant influence on results of rod temperature. Most present sub-channel codes use constant material properties in thermal analysis of fuel rod. That method limits the accuracy of the thermal analysis of fuel rod.

* Corresponding author at: Number 9 building in South China University of Technology, Guangzhou, China. Tel.: +86 13824401639.
E-mail address: zxy1119@scut.edu.cn (Z. Xiaoying).

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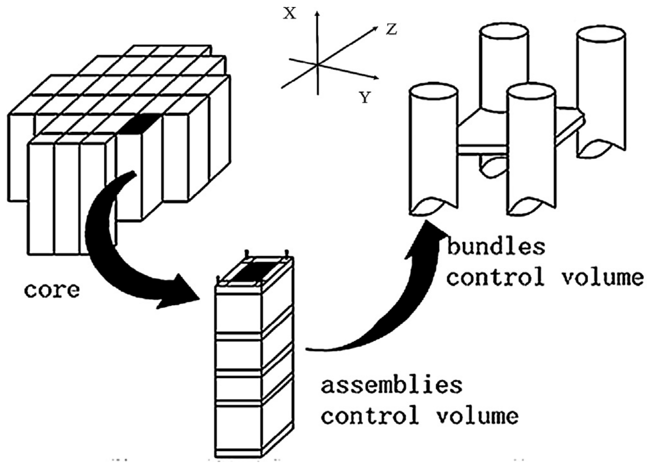

Fig. 1. Control volume of the core sub-channel analysis.

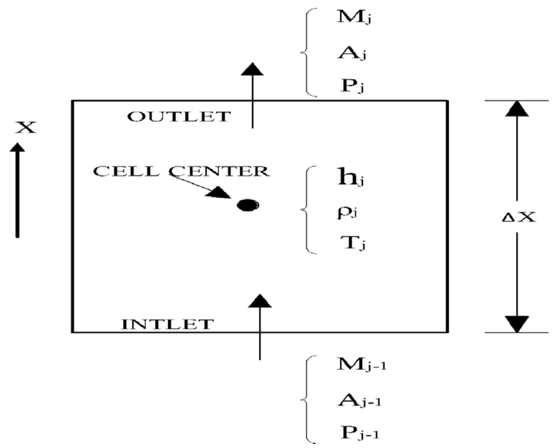

Fig. 2. Variables in control volume *j*.

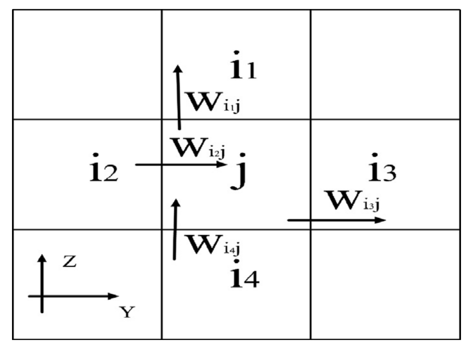

Fig. 3. Channel distribution at sub-channel model.

Many sub-channel codes have been developed to simulate the thermal hydraulic behavior of the reactor core. These codes include COBRA series (Jackson and Todreas, 1981; Stewart et al., 1977), FLICA series (Allaire, 1995; Fillion et al., 2011), SACATRI (Merroun et al., 2009), and WOSUB (Wolf et al., 1978). Most of these codes use the regular 4 basic conversation equations based on homogeneous flow model. Problem of computational divergence may come up when applies these codes in small reactor core with a low coolant flow rate. To solve problem with convergence, there were researches considering coupling the basic conversation equations to get new form equation with good convergence. For example, the COBRA code proposes the idea of a combined lateral momentum equation with continuity equation and axial momentum equation. The SACATRI code uses the SIMPLE algorithm (Patankar and Spalding, 1972), developed by Patankar and Spalding to solve an auxiliary pressure correction equation coupled with velocity and pressure (Bai et al., 2013; Jia et al., 1983). Corresponding to requirement of developing small reactor, existing research on combined conversation equation to improve convergence of sub-channel code is far from enough.

For purpose of thermal hydraulic analysis of small reactor core, this study concentrates on optimizing convergence of conversation equations and precision of heat conduction analysis for fuel rod. The combined lateral momentum equation, which is coupled with continuity equation and axial momentum equation, has been derived. A correctional algorithm of lateral velocity and pressure has been applied to solve equation of flowing, including continuity equation, energy equation, axial momentum equation and the combined lateral momentum equation. As for heat conduction of fuel rod, a Kirchhoff transformation equation has been developed and Gauss iteration method is used for solution. A steady-state sub-channel code has been put forward based on above theory.

The present paper starts with description of the steady-state sub-channel code. Subsequently, the code was validated with PSBT Benchmark. Equilibrium quality, void fraction and DNB power are calculated to compare with the experimental data. Finally, the code is applied to the “MUTSU” reactor and main thermal hydraulic parameters are calculated for analysis.

2. Sub-channel analysis theory model

The theoretic model of sub-channel analysis comes from the regular 4 conversation equations of continuity energy, lateral momentum and axial momentum. As shown in Fig. 1, sub-channel analysis of a PWR can be performed on the whole core, as well as on the hottest assembly in the core. When comes to analysis of core, due to the symmetry structure, a quarter or eighth fuel assemblies are selected for analysis and it considers the flow filed of assembly as a sub-channel. As for analysis of fuel assembly, it considers the flow filed between fuel rods as a sub-channel. Subsequently, the sub-channels are divided into a finite number of axial segments along the axial direction $\vec{X}$ as shown on Fig. 2. The basic flow variables in control volume j, such as the axial and lateral flow rates, pressure, and enthalpies, are calculated axially layer by layer from the inlet to outlet. Fig. 3 presents the channel distribution of sub-channel model. Because of pressure difference between adjacent sub-channels, there is crossflow at the gap between *i* and *j*. *j* is the studied sub-channel and $\{i\}$ are the adjacent sub-channels.

2.1. Governing equations of the subchannel analysis model

The coolant is assumed to be incompressible, pressure is equal in both phases, and the vapor phase is saturated. The solving of the equations should ensure the pressure at the outlet of sub-channels is equal to system pressure. The governing equations of steady-state sub-channel analysis model are derived based on theory of Stewart et al. (1977).

2.1.1. Mass conservation

Changes in the mass flow of the sub-channel control volumes are determined by the inflow and outflow of coolant across the control boundary. The continuity equation provides mass balance across the sub-channel.

$$\frac{\partial m_j}{\partial X} + \sum_i w_{ij} = 0 \quad (1)$$

where the first term on the left of continuity equation is the axial transport of mass flux. The second term is the sum of lateral flow