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Development of a subchannel analysis code for SFR wire-wrapped fuel assemblies

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ABSTRACT

Sodium cooled fast reactor (SFR) exhibit some unique thermo-hydraulic characteristics in comparison to thermal reactors, considering that SFR have a higher core power density, special configuration of assemblies, wire-wrapped spacers, special thermo-physical properties of sodium and so on. Some design limits related to fuel, cladding and coolant temperature must be met in both the normal and transient condition. Thus, it is of great importance to obtain accurate thermal-hydraulic performance for the design and safety assessment of the SFR reactor. Therefore, a subchannel analysis code SUBAC specially for SFR wire-wrapped assemblies is developed in the present study. Sensitivity analysis of different model combinations is carried out by using ORNL 19-pin benchmark test. The calculated results show that the normalized temperature at the exit of the rod bundle calculated by the Cheng-Todreas correlation combined with Cheng-Tak turbulent mixing model is consistent with the experimental data. Subsequently, the 19-pin and 37-pin bundle benchmark problems are calculated to validate the accuracy and applicability of the code. All the calculated results agree well with the experimental data, indicating that the code is competent for steady-state calculations. Finally, SUBAC code is used for performing transient analyses of EBR-II SHRT-17 XX09 subassembly. A good agreement between calculated and experimental values of the core top sodium temperature is achieved, but slightly underestimated. In conclusion, the SUBAS program is capable of being used for subchannel analysis of SFR and some suggestions for future research are given.

1. Introduction

Sodium-cooled fast reactors (SFR) and two other fast reactors, lead-cooled fast reactors (LFR) and gas-cooled fast reactors (GFR), have been identified as the focus of future generation IV reactor development by the Generation IV International Forum (GIF) (Bell et al., 1979). Among them, the SFR system is by far the most tested and is extensively regarded as the “best” Gen IV reactor. SFR exhibit some unique thermo-hydraulic characteristics in comparison to thermal reactors, considering that SFR have a higher core power density, special configuration of assemblies, wire-wrapped spacers, special thermo-physical properties of sodium and so on. Some design limits related to fuel, cladding and coolant temperature must be met in both the normal and transient condition. Thus, it is of great importance to obtain accurate temperature and velocity distribution within assemblies or the core for the design and safety assessment of the SFR reactor.

In general, there exits three main methods widely used to the study of the thermal hydraulic characteristics of fuel assemblies. They are 1D system code, subchannel analysis code and CFD method, respectively. The system program is able to model the whole-plant system, and has the advantage of fast calculation speed and good convergence. But the drawback, however, is that it cannot capture the detailed characteristics due to the fuel subassembly is regarded as a single channel. In addition, the CFD numerical calculation, it is characterized by a fine modeling and meshing, which has a high accuracy but also provides high computational cost, time-consuming, not easy to convergence and other shortcomings. However, the subchannel program just compromises the characteristics of both of them, on the one hand it can be more sophisticated modeling to obtain local thermal hydraulic characteristics, on the other hand has a faster solution speed and good convergence. According to the characteristics of these three methods, subchannel analysis is the most practical method of SFR fuel assembly thermal hydraulic analysis, especially in transient condition or parameter study.

In the past few years, much effort has been devoted to the experimental and theoretical studies on the thermal hydraulic characteristics

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of reactors, and a series of subchannel analysis codes have been developed, e.g., TRIO-U, SUPERENGY-2, SABRE4, MATRA-LMR and COBRA-IV. As pointed out in the paper (Wu et al., 2013), each of these codes has its own characteristics and limitations. TRIO-U is a CFD 3D code developed in France and the disadvantage is time consuming. SUPERENGY-2 was developed in the United States and designed for analyzing thermal hydraulic performances in supercritical water cooled fast reactor. However, it can only be used for steady-state calculations and the number of fuels that can be solved is limited. SABRE4 is a 3D subchannel analysis program originally developed for thermal hydraulic analysis of fast reactors in UK. MATRA-LMR (Kim et al., 2002) is a modified liquid mental reactor version of MATRA that was developed at KAERI for thermal hydraulic analysis of pressurized water reactors. COBRA-IV is widely used in pressurized water reactors, and its computational accuracy and convergence has been recognized by many researchers, but it cannot be used for thermal hydraulic calculation of sodium-cooled fast reactor owing to lack of corresponding properties and models. However, some researchers have modified the COBRA-IV code to allow it to be used in fast breeder calculations in recent years.

Subchannel analysis code for SFR wire-wrapped fuel assemblies is still lacking, and there are many problems with the existing codes, such as outdated models, its applicability and accuracy cannot meet the current circumstance. Therefore, in the present study, a subchannel analysis code SUBAC (Subchannel Analysis Code) specially for SFR wire-wrapped assemblies was developed. A lot of thermal–hydraulic and safety analysis codes have been developed by Nuclear Thermal-hydraulic Research Laboratory of Xi'an Jiaotong University (XJTU-NuTHeL). SUBAC is a derivative of these in-house codes (Cui et al., 2017). Some new models embedded in this code are selected from research results in recent years through extensive literature survey. A lot of verification work is done to verify the applicability and accuracy of SUBAC code by using the existing benchmark problems. The calculation results show that a reliable tool for subchannel analysis of SFR is developed and some suggestions for future research are proposed.

2. Subchannel analysis code

SUBAC was developed to calculate the flow and temperature distributions in wire-wrapped fuel assembles of SFR. Compared with the conventional subchannel codes developed for light water reactor, the most typical characteristic of this program is that the working fluid is a kind of liquid metal-sodium, which has high boiling point and large thermal conductivity under atmospheric pressure. Therefore, the lateral heat conduction and turbulent mixing of different control volumes contribute greatly to the flattening effect of core temperature distribution, which cannot be neglected any more. In addition, the liquid metal sodium has a high boiling point and is always in a single-phase state under normal operating conditions, so we treat it as an incompressible medium and do not consider its phase change.

The sodium cold fast reactor uses hexagonal fuel assemblies in which the fuel pins are arranged in a triangular fashion as shown in Fig. 1. And the fuel element is positioned using a metal wire rather than a grid which is widely used in a light water reactor. According to the relative position and shape features of subchannels in the subassembly, it is divided into three categories: interior channel, edge channel and corner channel. Obviously, the flow area and wetted perimeter of each type of channel have changed because of the existence helical wires. In this study, the wire wrap is still assumed to be smeared uniformly around fuel rods, so the actual flow area and wetted perimeter of each channel can be obtained using axially averaging method. For the interior and edge subchannels, the flow area is equal to the bare rod flow area minus half of the area occupied by the helically wound wire, and the wetted perimeter is equal to the bare rod wetted perimeter plus 50% of the wire perimeter. For the corner subchannel, the flow area is the bare rod flow area minus 1/6 of the cross-sectional area of wire wrap, and the wetted perimeter is equal to the bare rod wetted perimeter plus1/6 of the wire perimeter.

2.1. Governing equations

It is necessary to divide the axial and lateral control volumes in order to establish and discrete control equations. Fig. 2 shows the axial and lateral control volume diagrams for the establishment and discrete control equations. The control volume specified by the solid line in Fig. 2 is applicable to the mass, energy and axial momentum conservation equation, while the separate control volume between two adjacent subchannels marked by the dashed lines is suitable for lateral momentum equation. Hence the solution of the lateral momentum equation is based on the staggered scheme. The following will describe the basic governing equations and associated auxiliary models used in the development of the code.

2.1.1. Mass conservation equation

According to the law of conservation of mass, the rate of mass increase over time in control volume (i, n) is equal to the sum of the mass flow rates flowing into the control volume through the various interfaces. Therefore, the equation can be formulated as follows:

$$A_i\frac{\partial \rho_i}{\partial t}+\frac{\partial m_i}{\partial z}+\sum_{j\in i}w_{ij,k}+\sum_{j\in i}w'_{ij,k}=0 \tag{1}$$

where subscripts i, j are the indexes of adjacent subchannels; A_i is the flow area; ρ_i is the density of coolant; m_i is the coolant axial flow rate; w_{ij} is the coolant lateral mass flow rate per unit axial length of channel; w'_{ij} is the turbulent flow rate.

Considering that turbulent mixing between adjacent control volumes has no net mass transfer, the discrete expression of the mass conservation equation for control volume (i, n) can be written as:

$$A_i\frac{\Delta z}{\Delta t}(\rho-\rho^n)_{in}+m_{in}-m_{in-1}+\Delta z\sum_{j\in i}w_{ij,k}=0 \tag{2}$$

where Δz is the space step size, Δt is the time step, and superscript n represents the last time step, while the subscript n is the axial control volume number.

2.1.2. Axial momentum conservation equation

The momentum balance equation of the axial control volume (i, n) can be obtained by applying the momentum theorem to the control volume. According to Fig. 3, the axial momentum expression can be written as:

$$\begin{aligned}&-F_i dz-g\rho_i A_i\cos\theta dz+p_i dA_i+p_i A_i-p_i A_i-\frac{\partial}{\partial z}(p_i A_i)dz\\&=\frac{\partial}{\partial t}(m_i)dz-m_i u_i+m_i u_i+\frac{\partial}{\partial z}(m_i u_i)dz-w'_{ji}u_j dz+w'_{ij}u_i dz\\&\quad+w_{ij}u^* dz\end{aligned} \tag{3}$$

Since $w'_{ij}=w'_{ji}$, $A=\text{constant}$, $F_i=\left(\frac{A_i f_i\phi_i^2}{2\rho_i D_i}+\frac{A_i K_s v_i}{2}\right)\left(\frac{m_i}{A_i}\right)^2$ and equation (1), equation (3) can be written as:

$$\begin{aligned}\frac{1}{A_i}\frac{\partial m_i}{\partial t}-2u_i\frac{\partial \rho_i}{\partial t}+\frac{\partial p_i}{\partial z}=&-\left(\frac{m_i}{A_i}\right)^2\left[\frac{f_i\phi_i^2}{2D_i\rho_i}+\frac{K_s v_i}{2}+A_i\frac{\partial}{\partial z}\left(\frac{v_i}{A_i}\right)\right]\\&-g\rho_i\cos\theta-f_T\sum_{j\in i}(u_i-u_j)\frac{w'_{ij,k}}{A_i}\\&+\sum_{j\in i}(2u_i-u^*)\frac{w_{ij,k}}{A_i}\end{aligned} \tag{4}$$

where u_i is the axial velocity; f_i is the friction factor; ϕ_i^2 is friction multiplier (only single phase exists in the present study, $\phi_i^2=1$); D_i is the hydraulic diameter; K_s is the form loss coefficient per unit length; f_T is the transverse momentum factor; w'_{ij} is the turbulent flow rate; u^* is the momentum velocity on gap k between adjacent channels, which is