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Development of a code analyzer for thermal-hydraulic analysis of small integral reactor

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1. Introduction

A regional energy reactor, named REX-10, is under development at the Seoul National University for small scale electricity generation and nuclear district heating [1]. It is a small-sized integral reactor encompassing the thorium-fueled core, steam-gas pressurizer and helical coiled steam generator with natural circulation cooling at normal operation. To evaluate the thermal-hydraulic behavior and system performance of the reactor, a code analyzer is being developed based on one-dimensional momentum integral model. This is the analysis code specialized for a general integral reactor, not the all-purpose heavy code such as MARS. Thus the objective of this study is to program the simple and fast running system code. Basic assumptions, governing equations, component models and numerical solution scheme used in the code analyzer are presented in this paper. The solution method is applied to the REX-10 test facility and a set of simulation is carried out.

2. Models and numerical solution procedure

Momentum integral model eliminates the sonic effect from physical model and decouples the momentum and energy equations, enabling simple and fast calculations [2]. Mass, momentum, energy equations are applied to the one-dimensional node and flow-path network. The fluid properties are assumed to be homogeneous in each node and the coolant enthalpy is defined at flow-path.

2.1 Governing equations

The fluid is considered incompressible and thermally expandable in momentum integral model. Assumption of fluid incompressibility leads to only time-dependent flow rate from the continuity equation [3]. It indicates that the mass flow rate in the primary circuit is uniform around the loop at specific time regardless of the space coordinate.

$$W = W(t) \tag{1}$$

It is well known that the Boussinesq approximation is valid for simulating natural circulation, whereby the density is regarded as constant in momentum equation except for the gravitational term. The density is assumed to vary linearly with temperature:

$$\rho = \rho_0[1-\beta(T-T_0)] \tag{2}$$

Integration of the momentum equation over the entire flow path eliminates the convection and pressure terms and yields the next loop momentum equation:

$$\left(\sum_k \frac{L_k}{A_k}\right)\frac{\partial W}{\partial t} = -\frac{a}{2}\left(\sum_k \frac{D_{hk}^{\ b-1} L_k}{\upsilon_k^b A_k^{b+2}}\right)\frac{W^{b+2}}{\rho_0^{b+1}} - \sum_j \frac{K_j}{A_j^2}\frac{W^2}{2\rho_0} + g\beta\rho_0\sum_k T_k L_k \cos\theta \tag{3}$$

where β denotes the thermal expansion coefficient and friction coefficient is expressed in form of $f = a\,\mathrm{Re}^b$.

In addition, the enthalpy rise by viscous dissipation and pressure changes in energy equation is neglected.

$$\rho A\frac{\partial H}{\partial t} + W\frac{\partial H}{\partial z} = \dot{q}\,' \tag{4}$$

2.2 Core and steam generator models

The thorium-fueled core in REX-10 is modeled using the point kinetics model. It simulates the change of core power in time by reactivity feedback with six delayed neutron groups. The effective delayed neutron fractions and decay constants in thorium fuel are given as inputs from the dynamic characteristic analysis. The average fuel and coolant temperature determines new reactivity at each time step to achieve negative feedback effect.

In REX-10, the helical tubes in once-through steam generator coil wholly the annular space between core barrel and pressure vessel wall. Neglecting the heat conduction in axial directions and assuming the thermal equilibrium between water and steam in boiling region, the steam generator model computes the shell-side and tube-side heat transfer coefficient and the heat transfer rates from the primary to the secondary coolant. The heat transfer regions inside the tubes are divided into three parts: economizer, evaporator and superheater regions [4]. The empirical correlations used in this code analyzer are summarized in table 1.

Table 1: Empirical correlations in helical coiled S/G

Boiling region HTC	Tube-side	Shell-side
Subcooled water	Mori-Nakayama	Zukauskas
Saturated boiling x_{dryout} Mist evaporation	Chen 0.8 (Kozeki) Interpolation	
Superheated steam	Mori-Nakayama	
Friction coefficients	Mori-Nakayama	Smith-King

In order to calculate the heat transfer rates through a wall, i.e., from fuel rods to coolant in core or from the primary coolant to secondary feedwater across helical tubes in S/G, the heat conduction equation is solved in cylindrical coordinate.

$$\rho c_p \frac{\partial T}{\partial t} = \nabla \cdot \left(k \nabla T \right) + Q \qquad (5)$$

Finite difference scheme is used to advance the heat conduction solutions [5]. With the convection surface boundary condition, the discretized equations set up the tri-diagonal matrix system. Then the temperature difference between wall and the adjacent fluid is obtained.

2.3 Numerical solution method

The numerical methods utilized in the code analyzer are implicit approaches which assures the numerical stability. From the initial conditions, Eq. (4) is solved implicitly for the primary circuit to get new enthalpy distribution. Linear heat transfer rate in right hand side is obtained from the temperature of the previous time step. Energy equation is also solved for the secondary side of S/G taking the pressure drop in helical tube into account [6], and heat conduction across the tube wall is computed simultaneously with Crank-Nicholson method for time advancement. The updated temperature distributions are used to calculate the buoyancy term in Eq. (3). The implicit scheme for the integrated momentum equation leads to the nonlinear equation expressed in terms of loop flow rate, which is determined from the iterative Newton-Raphson method. The fluid properties are enthalpy-dependent and taken from steam table at each time step during all the calculations.

3. Simulation results

The models and the solution scheme described above are applied to the REX-10 test facility (RTF) designed to assess the natural circulation capability of prototype. Total 41 nodes constitute the primary circuit of RTF and the inlet temperature and pressure of the secondary feedwater are fixed to 125.0℃ and 3.0 bar for all cases. Figure 1 presents the primary mass flow rate in time and temperature distribution around the loop at steady-state under 100% rated power−200 kW.

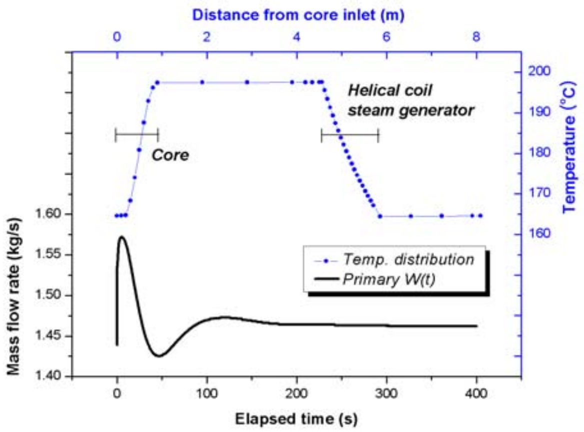

Fig. 1. Simulation results at 100% rated power

It is noted that the flow rate exhibits some oscillatory behavior before reaching the steady-state but completely converges to a specific value after about 300s have passed. At steady-state, the temperature rise through the core is 32.9℃.

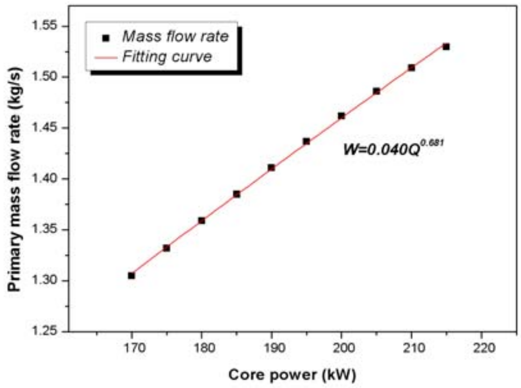

Fig. 2. Dependency of loop flow rate on core power

Figure 2 shows the variation of primary mass flow rate according to core power. From 10 numerical tests, it is revealed that the natural circulation flow is proportional to the 0.681th power of core power in this system.

4. Conclusions

A code analyzer is under development for evaluating the thermal hydraulic behavior and system performance of small-sized pressurized water reactor with an integral layout of primary circuit equipment. It is based on the momentum integral model with assumptions of the fluid incompressibility and Boussinesq approximation. The implicit method is applied for the governing equations in solution procedure to assure numerical stability. The code analyzer is able to execute some brief simulations and acquire the stabilized natural circulation flow rate, yet many activities still remain in order to utilize the program for safety analysis of the system such as an addition of safety system modeling and an extension of governing equations to two-phase flow and so on.

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