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Development and assessment of system analysis code, TASS/SMR for integral reactor, SMART

Y.J. Chung*, I.S. Jun, S.H. Kim, S.H. Yang, H.R. Kim, W.J. Lee

Korea Atomic Energy Research Institute, 1045 Daedeokdaero, Yuseong, Daejeon, 305-353, Republic of Korea

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ABSTRACT

A new requirement has motivated the development of smaller reactors since the 1980s. Integral type reactors have been highlighted as a promising option. SMART, which is an integral type reactor has been developed at KAERI, and TASS/SMR code was developed to analyze the thermal hydraulic phenomena of the SMART plant. The main purpose of the code is to simulate all relevant phenomena, processes, and conditions inside the reactor coolant system that may occur during such accidents. Development and assessment of the code is represented in detail. By means of the assessment results using experimental data, TASS/SMR code can be used for both the experiment simulation as well as the SMART analysis. The code predicts thermal hydraulic phenomena for the representative accidents for SMART reasonably.

1. Introduction

The general trend of the nuclear market has been toward larger unit sizes. However, a new requirement has motivated the development of smaller reactors since the 1980s. Integral type reactors have been highlighted as a promising option (Modro et al., 2002; OECD NEA, 2011). For this purpose, the design of a 330 MWt integral reactor, SMART (system-integrated modular advanced reactor) plant, has been developed at the Korea Atomic Energy Research Institute (KAERI) (Kim, 2010). SMART adopts a design concept containing most of the reactor coolant system (RCS) components, such as a core, reactor coolant pumps (RCPs), steam generators (SGs), and a pressurizer in a reactor pressure vessel. Also, it is a small-sized advanced integral-type pressurized water reactor with several enhanced safety features, for examples, a passive residual heat removal system (PRHRS) and an external reactor vessel cooling (ERVC). The existing proven technologies are basically adopted for the SMART design. However, SMART also adopts various innovative design features and technologies that need to be proven through experiments and analyses.

For the simulation of a design based transient and accident in an integral type nuclear power plant, it is necessary to develop a system thermal hydraulic analysis code for SMART. A thermal hydraulic analysis code, TASS/SMR, including a helical steam generator heat transfer and condensate heat exchanger models, has been developed to simulate thermal hydraulic phenomena of SMART (Chung et al., 2003). TASS/SMR code can analyze the thermal hydraulic phenomena of SMART under a full range of reactor operating and accident conditions. The basic code structure adopts a one-dimensional geometry. A node and flow-path network models the system responses. The thermal–hydraulic model is formulated with four one-dimensional conservation variables. The application scope of the code covers the analysis of operational transients, design based accidents and accidents involving partial beyond design based accidents for the SMART plant.

This paper presents an overview of TASS/SMR code development and preliminary assessment, mainly focusing on the thermal hydraulic models, a numerical solution, and validation results. Then, three types of basic transient conditions including loss of reactor coolant flow accident, steam line break accident, and feedwater line break accident are analyzed for SMART.

2. Thermal hydraulic models and numerical solutions

TASS/SMR code is developed for a comprehensive simulation of design based accidents in the integral type reactor, SMART. The main purpose is to simulate all relevant phenomena, processes, and conditions inside the reactor coolant system that may occur during such accidents. To describe the thermal hydraulic behavior of SMART, a conservative transient model is adopted in the TASS/SMR code. The governing equations are mixture mass, mixture energy, and mixture momentum, which take into account the drift flux model to consider the velocity difference between steam and liquid velocities.

* Corresponding author. Tel.: +82 42 868 8760.
E-mail address: chung@kaeri.re.kr (Y.J. Chung).

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2.1. Thermal hydraulic models

The mass, non-condensable mass, momentum, and energy conservations are

$$A\frac{\partial \rho_m}{\partial t} + \frac{\partial W_m}{\partial x} = 0 \quad (1)$$

$$A\frac{\partial}{\partial t}(\alpha\rho_n) + \frac{\partial W_n}{\partial x} = 0 \quad (2)$$

$$A\frac{\partial}{\partial t}\left(\frac{W_m}{A}\right) + \frac{\partial}{\partial x}\left(\frac{{W_m}^2}{\rho_m A}\right) + \frac{\partial}{\partial x}\left[\frac{\alpha\rho_g(1-\alpha)\rho_l}{\rho_m}{v_r}^2 A\right] = -A\frac{\partial P}{\partial x} - K_f\Phi^2\frac{W_m\left|W_m\right|}{2\rho_l A} - K_g\frac{W_m\left|W_m\right|}{2\rho_m A} + \rho_m a_{\text{ext}}A \quad (3)$$

$$A\frac{\partial}{\partial t}(\rho_m e_m) + \frac{\partial}{\partial x}(h_\rho W_m) + \frac{\partial}{\partial x}\left[\frac{\alpha\rho_g(1-\alpha)\rho_l}{\rho_m}(h_g - h_l)v_r A\right] = \dot{q}_w \quad (4)$$

$$\rho_m = \alpha(\rho_s + \rho_n) + (1-\alpha)\rho_l$$

$$h_\rho = \frac{1}{\rho_m}[\, \alpha(\rho_s h_s + \rho_n h_n) + (1-\alpha)\rho_l h_l \,]$$

$$e_m = \frac{1}{\rho_m}[\alpha(\rho_s e_s + \rho_n e_n) + (1-\alpha)\rho_l e_l]$$

For closure of the conservation equations, constitutive relations are incorporated. These include conservative models for core heat transfer and critical flow, which correspond to appendix K model (US NRC, 2003). A number of the SMART specific models reflecting the SMART's design characteristics such as a helically coiled steam generator, and a condensate heat exchanger in the PRHRS are addressed in the code. A detailed departure from the nucleate boiling ratio (DNBR) analysis is performed using a MATRA/SMR code (Hwang, 2010), which is a sub-channel analysis code, and is calculated based on local coolant conditions calculated at every time step by means of the results of the TASS/SMR for the inlet boundary conditions of the MATRA/SMR code. The relevant correlations are selected, which are as follows:

2.1.1. Core heat transfer

The core heat transfer model calculates the fuel rod temperature and the core heat transfer to the coolant. This model calculates the heat transfer based on local fluid conditions and local rod surface temperatures using a boiling curve. For a single phase liquid, Dittus–Boelter correlation (Dittus and Boelter, 1930), Collier correlation (Collier, 1994) and Churchill–Chu correlation (Churchill and Chu, 1975a) are considered according to the Reynolds number. In a nucleate boiling mode, the heat transfer coefficients are calculated by either a Chen correlation (Chen, 1963) or Forster–Zuber correlation (1955). The forced convection, natural convection, and boiling curve consist of the following correlations.

Liquid and steam forced convection:

$$h_{\text{FC}} = 0.023\frac{k_f}{D_h}Re_f^{0.8}Pr_f^{0.4} \quad \text{for} \quad Re > 10^4 \quad (5)$$

$$h_{\text{FC}} = 0.17\frac{k_f}{D_h}Re_f^{0.33}Pr_f^{0.43}\left(\frac{Pr_f}{Pr_w}\right)^{0.25}Gr_f^{0.1} \quad \text{for} \quad Re < 2000$$

where h, k, D_h, Re, Pr and Gr are heat transfer coefficient, thermal conductivity of the fluid, hydraulic diameter, Reynolds number, Prandtl number, and Grashof number, respectively. Subscripts FC, f, h, and w denote forced convection, fluid, hydraulic diameter, and wall, respectively.

Nucleate boiling:

$$h_{\text{SatB}} = Fh_{\text{SPL}} + h_{\text{NB}}$$
$$h_{\text{NB}} = 0.00122S\left[\frac{k_f^{0.79}C_{pf}^{0.45}\rho_f^{0.49}g^{0.25}}{\sigma^{0.5}\mu_f^{0.29}h_{fg}^{0.24}\rho_g^{0.24}}\right](T_w - T_f)^{0.24}(p_w - p_f)^{0.75} \quad (6)$$

where F and S denote the Reynolds number factor and the bubble growth suppression factor, and subscripts SatB, SPL, NB, and w denote saturated boiling, single phase liquid, nuclear boiling, wall, respectively (Chen, 1963).

Natural convection:

$$h_{\text{NC}} = \frac{k_f}{D_h}\left\{0.825 + 0.387Ra_f^{1/6}\left[1 + (0.492/Pr_f)^{9/16}\right]^{-8/27}\right\} \quad (7)$$

where Ra_l denotes the Rayleigh number (Churchill and Chu, 1975a). The Churchill–Chu correlation is applied to the laminar and turbulent natural convective regimes of a vertical geometry.

Transition and film boiling: MaDonough, Milich and King correlation is adopted for the transition boiling (MaDonough et al., 1958). The film boiling heat transfer model applies the minimum value between Groeneveld (1968), and Dougall and Rohsenow correlations (1963). It is recommended in the 10 CFR 50.46 Appendix K for evaluation model.

Critical heat flux: The critical heat flux model uses W-3 and MacBeth correlations. The W-3 correlation is used for a range of $6.9\,\text{MPa} < P < 16.5\,\text{MPa}$, $1350\,\text{kg/s m}^2 < W/A < 6780\,\text{kg/s m}^2$, and $x < 0.15$. The MacBeth correlation is applied for the outside conditions of the range of the W-3 correlation (Tong, 1965).

2.1.2. Steam generator heat transfer

The steam generator heat transfer model calculates the tube temperature and heat transfer rate to the coolant in the shell and tube sides. The model calculates the heat transfer based on local fluid conditions and local tube surface temperatures using forced and natural convection modes. Eqs. (8)–(10) for the heat transfer in the tube and shell sides are used for the forced and natural convection conditions.

Single-phase forced convection in tube side (Mori and Nakayama, 1967):

$$h = \frac{1}{41.0}\left(\frac{k_f}{D_h}\right)Re^{5/6}Pr^{0.4}\left(\frac{d_i}{D_c}\right)^{1/12}\left\{1 + \frac{0.061}{[Re(d_i/D_c)^{2.5}]^{1/6}}\right\} \quad \text{for} \quad Pr > 1 \quad (8)$$

where d_i and D_c denote the helical tube inner diameter and helical coil diameter, respectively. This correlation agrees well with the experimental data for the range of $Re = 2\times10^2$–4×10^4.

Single-phase natural convection in the tube side (Churchill and Chu, 1975b):

$$h = \left(\frac{k}{D_h}\right)\left\{0.6 + \frac{0.387Ra^{1/6}}{[1 + (0.559/Pr)^{9/16}]^{8/27}}\right\}^2 \quad (9)$$

where Ra is the Rayleigh number, and the Churchill–Chu correlation is reported to be valid over the full laminar and turbulent Rayleigh number range for a horizontal geometry.

Single-phase forced convection on shell side (Zukauskas, 1972):

$$h = C\left(\frac{k}{D_h}\right)Re^m Pr^{0.36}\left(\frac{Pr}{Pr_w}\right)^{0.25} \quad (10)$$

where C and m are parameters that depend on the Reynolds number, and Pr_w is the Prandtl number corresponding to the wall temperature, and this correlation can apply for all the Reynolds number ranges under a cross-flow condition.