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Assessment and SMART application of system analysis design code, TASS/SMR-S for SBLOCA

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HIGHLIGHTS

- ▶ TASS/SMR-S code is thermal hydraulic design code for SMART.
- ▶ Thermal hydraulic design code must be assessed using ITE such as LOFT and semiscale test.
- ▶ TASS/SMR-S was assessed for SBLOCA using LOFT L3-7 with good agreement.
- ▶ It is proven that TASS/SMR-S has feasibility of application to SBLOCA.
- ▶ TASS/SMR-S was applied to simulate SBLOCA scenario of SMART.

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ABSTRACT

TASS/SMR-S is thermal-hydraulic design code for safety analysis of SMART plant. SBLOCA is considered as the design basis accident in the SMART. The capability of TASS/SMR-S for SBLOCA analysis was assessed using LOFT L3-7 as TMI action plan. Because TASS/SMR-S has been developed only for SMART, which is an integral reactor with helical steam generator heat exchanger, the steam generator model of TASS/SMR-S cannot be used directly for LOFT experiment that involves the use of U-tube steam generator. Therefore, the general heat structure model of TASS/SMR-S was used for modeling the SG heat exchanger. Nevertheless, TASS/SMR-S predicted important thermal-hydraulic parameters such as system pressure, fluid temperatures, and cladding temperature of reactor within reasonable error ranges. Further, TASS/SMR-S was applied to simulate SBLOCA in the SMART plant. In this simulation, thermal hydraulic parameters similar to those predicted in LOFT L3-7 were predicted.

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1. Introduction

Recently, advanced SMRs (small modular reactors) have garnered considerable attention globally. This is because SMRs have many advantages such as wide applicability; enhanced safety, owing to the adoption of inherent safety characteristics and passive features; and reasonable cost, because of simplification and modularization.

An SMR has multiple purposes, such as electricity generation, water desalination, district heating, and hydrogen generation, among others. For this reason, SMRs have attracted attention in many countries that do not need a large conventional nuclear power plant owing to their population distribution, insufficient power supply networks, and limited finances.

In response to the demand for SMRs, a number of these reactors have been actively developed, such as NuScale Power's NuScale; Babcock & Willcox's B&W mPower; Westinghouse's SMR; GE Hitachi's PRISM in the USA; Toshiba's 4S reactor and Mitsubishi's IMR in Japan; VBER-300, SVBR-100, and ABV6M in Russia; and KAERI's SMART in Korea. Among these SMRs, SMART (System-integrated Modular Advanced ReacTor) acquired Standard Design Approval (SDA) in July 2012. This is very meaningful to the development of SMRs worldwide, because it is the first SDA for an SMR plant (Kim et al., 2011; Laina and Subki, 2011).

SMART is one of the next-generation, integral-type SMRs developed by KAERI (Korea Atomic Energy Research Institute). The schematics of SMART are shown in Fig. 1. SMART can be used for various applications such as electricity generation, seawater desalination, and district heating. A single unit of SMART generates 330 MWth, which can provide 90 MWe of electricity and 40,000 tons of fresh water a day. Because SMART is an integral-type reactor, which houses all major components in a single vessel and eliminates the use of connecting pipes between the major components, the possibility of the occurrence of LBLOCA (large break loss of coolant accident) in SMART is eliminated (Kim et al., 2010).

Further, TASS/SMR-S (transient and setpoint simulation/small and medium reactor-safety) has also been developed by KAERI; TASS/SMR-S is a thermal-hydraulic design code developed for the

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Nomenclature

ρ	density
W	mass flow rate
P	pressure
A	area
x_f	quality
σ	void fraction
K_f	pressure loss coefficient by wall friction
K_g	pressure loss coefficient by geometry
$\mathbf{\Phi}^2$	two-phase friction multiplier
e	specific energy
h	specific enthalpy
Q	heat rate

Subscripts

m	mixture
l	liquid
s	steam
n	non-condensable gas

safety analysis of SMART. Moreover, TASS/SMR-S code uses several SMART specific models such as helical SG (steam generator) heat transfer model and PRHRS (passive residual heat removal system) model for the analysis. Furthermore, core heat transfer model, heat structure model, heat transfer model, and several critical flow models are also included in the TASS/SMR-S. The governing equations of TASS/SMR-S are homogeneous equilibrium conservation equations, which include mass, momentum, and energy equations for mixtures and a mass conservation equation for non-condensable gases. In addition, the drift-flux model can be used to simulate the two-phase flow condition that would occur during SBLOCA (small break loss of coolant accident) in SMART (Kim, 2010a).

TASS/SMR-S ought to be validated using the experimental data obtained from the SET (separate effect test) and IET (integral effect test) facilities. In fact, TASS/SMR-S has already been validated on the basis of the results obtained by performing several SETs for determining the fuel heat transfer, helical SG heat transfer, PRHRS heat transfer, void distribution, critical flow, and so on (Chung et al., 2012). However, there are not many IET facilities that simulate the conditions occurring during SBLOCA. In addition, TASS/SMR-S must be assessed according to the requirements of the TMI action plan. As per this action plan, the computer codes used for the safety analysis of a reactor should be validated using the simulated-SBLOCA IET data, specifically that obtained in LOFT (loss of fluid test) (Mitsubishi, 2009) and semiscale test facilities (NRC, 1983).

The L series of LOFT includes SBLOCA simulation experiments. Among these LOFT experiments, LOFT L3-7 was performed in this study to assess the safety analysis capability of TASS/SMR-S for SBLOCA in SMART. Additionally, SBLOCA scenario of SMART plant was predicted by TASS/SMR-S using same methodology.

2. Description of LOFT L3-7

2.1. Description of LOFT facility

The LOFT has been designed to simulate a commercial, four-loop PWR (pressurized water reactor) with 3400 MWth. A LOFT facility includes a 50 MWth PWR to measure and determine the thermal-hydraulic data.

The LOFT reactor vessel simulates the reactor vessel of a commercial PWR. It consists of an annular downcomer, a lower plenum, a nuclear core, an upper plenum, and lower core support plates. The downcomer is connected to the cold legs of the intact and broken loops and it contains two instrument stalks. On the other hand, the upper plenum is connected to the hot legs of the intact and broken loops. Further, 1300 unpressurized nuclear fuel rods arranged in five squares (15 × 15 assemblies) and four triangular fuel modules make up the core. The intact loop simulates the three unbroken loops of a commercial, four-loop PWR and it consists of a PZR (pressurizer), a venture flowmeter, a steam generator, two primary coolant pumps in parallel, and connecting piping. The broken loop consists of a hot leg and a cold leg that are connected to the reactor vessel and the blowdown suppression tank header. Each leg includes a break plane orifice, a recirculation line, a quick-opening blowdown valve, an isolation valve, and connecting piping. The recirculation line allows a small flow from the broken loop to the intact loop to maintain the loop temperatures at an approximately equal level. The hot leg of the broken loop also comprises a simulated steam generator and a simulated pump. These simulators possess hydraulic orifice plate assemblies that offer resistance to the flow in a manner similar to that of flow resistances in an active steam generator and a pump. The blowdown suppression system consists of a blowdown suppression tank header, suppression tank, suppression tank spray system, and nitrogen pressurization system. The blowdown header is connected to the downcomer of the suppression tank; this downcomer extends into the tank and discharges the fluid below the water level. A brief schematic of a LOFT facility is shown in Fig. 2 (Gillas and Carpenter, 1980).

2.2. Procedure of LOFT L3-7 experiment

The main aims of L3-7 are to produce a break flow similar to that produced by an HPIS (high-pressure injection system) at an intermediate pressure level during the transient, to determine the conditions for steam generator reflux cooling, and to analyze the data obtained in this experiment in order to investigate the associated phenomena. LOFT L3-7, which is a part of the L3 series of LOFT SBLOCA experiments, was performed and a 1 in. diameter break in the cold leg of a commercial PWR was simulated. A break diameter of 4 mm was considered for the LOFT L3-7 experiment.

The initial conditions for L3-7 are as follows: core power, 50 MW; primary system pressure, 14.86 MPa; intact loop cold leg temperature, 556.8 K; and primary coolant flow rate, 478.8 kg/s.

The transient was initiated by opening the cold leg quick-opening blowdown valve on the vessel side of the broken loop. The reactor was scrammed when a signal indicating low system pressure was received from the reactor shutdown system. The control rods reached the bottom of the core approximately 2 s after the scram signal was received, and the pump tripped and initiated a coast down 3 s after the reactor was scrammed.

When a low-pressure signal (13.16 MPa) was triggered by the PZR, borated water was injected into the cold leg of the primary coolant system through the HPIS pump. The rate at which the borated water was injected by the HPIS was nearly equal to the out flow of the fluid from the break. In order to maintain the water level of the steam generator, auxiliary feedwater was supplied to the secondary cooling system 75 s after the transient. At approximately 1800 s after the initiation of the experiment, the HPIS and auxiliary feedwater supply were terminated. After 1 h of the experiment, the steam bleeding from the steam generator cooled down the primary and secondary cooling systems. After approximately 6000 s, the HPIS was activated once again. Next, an accumulator and an LPIS (low-pressure injection system) injected borated water into the primary cooling system. Finally, the cold leg quick-opening blowdown valve and the isolated valve were closed at 7300 s. Table 1 lists the conditions for L3-7 experiment (Kee and Taylor, 1980; Lee et al., 1990).

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