Code Assessment of SPACE 2.19 using LSTF Steam Generator Tube Rupture Test

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1. Introduction

The Safety and Performance Analysis Code for Nuclear Power Plants (SPACE) has been developed in recent years by the Korea Hydro & Nuclear Power Co. through collaborative works with other Korean nuclear industries and research institutes. The SPACE is a best-estimated two-phase three-field thermal-hydraulic analysis code used to analyze the safety and performance of pressurized water reactors. As a result of the development, the 2.19 version of the code was released through the successive various verification and validation works.

The present work is on the line of expanding the work by Kim et al. [1]. In this study, results produced by the SPACE 2.19 code were compared with the experimental data from JAERI's LSTF Test Run LSTF SB-SG-06 experiment simulating a Steam Generator Tube Rupture (SGTR) transient.

2. Experimental Facility Description

The Rig Of Safety Assessment (ROSA)-IV Program's Large Scale Test Facility (LSTF) is a test facility for integral simulation of thermal-hydraulic response of a pressurized water reactor (PWR) during small break loss-of-coolant accidents (SBLOCAs) and plant transients. The PWR core nuclear fuel rods are simulated by using electrical heater rods in the LSTF. The LSTF experimental facility was designed to model the thermal-hydraulic phenomena in a PWR during postulated small break LOCAs and plant transients.

Because the Japan Atomic Energy Research Institute (JAERI) carried out the integral simulation experiments on the SGTR incident that occurred at the Mihama Unit 2 power station, the experiment (SB-SG-06 test simulated an accident with SG single tube rupture) was initiated by opening a break valve nearly at the same RCS pressure and temperature as in Mihama Unit 2. The overall event sequences of the SGTR transient can be shown in Table 1.

3. Modeling and analysis

3.1 SPACE code modeling

The LSTF facility for experimental run SB-SG-06 is modeled with 177 fluid cells and 186 connections. The system nodalization is illustrated schematically in Figure 1. A total of 166 heat structures were used in the model to represent heat transfer in the steam generator, reactor, primary system piping, and pressurizer.

The single-ended break nozzle model is used to simulate the double-ended break based on the LSTF SB-SG-06 test configuration.

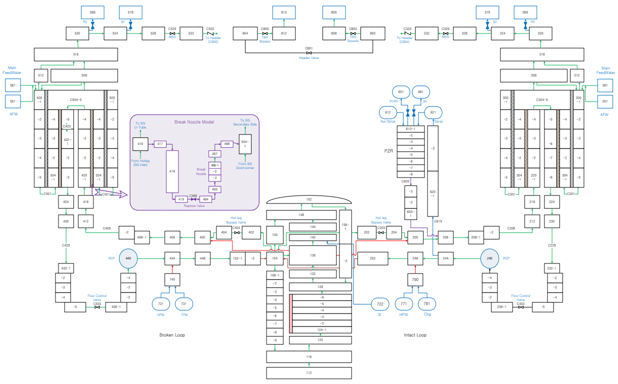

Fig. 1 Nodalization diagram of LSTF SGTR test

Table 1 Sequence of events for Experiment SB-SG-06

Event (sec)	*Experiment*	*SPACE 2.19*
Tube Break	0.0	0.0
Reactor Trip	266	242.4
Main Feedwater Trip	300	269.6
Safety Injection Signal	305	272.9
Pzr. Empty	331	353.0
RCP Coastdown (stop)	348	272.9
Aux. Feedwater Start	342	309.9
HPSI into Cold legs	403	370.9
HPSI into Core Upper Plenum	605	572.9
Affected SG Isolated (MSIV close)	988	964.4
Intact SG Depressurized (RV open in the BL SG)	988	964.4
Intact SG Depressurization Terminated (RVclose)	1751	1904.8
Affected SG RV Opened	2635(once)	cycling
Pzr. Aux. Spray Start	2932	2908.4
HPSI Turned off (stop)	3390	3172.9
Pzr. Aux. Spray Turned off	3617	3711.8
Intact Loop RCP Restarted	4245	4101.0
Termination	5000	5000

3.2 Results and analysis

The steady state calculation was performed in order to obtain appropriate steady state system conditions prior to the initiation of SG tube rupture. Table 2 represents the comparison of initial conditions between the LSTF SB-SG-06 test and the calculation. It is indicated that the major calculated parameters of the primary and secondary coolant systems agree well with the measured values in experiments.

Table 2 Comparison of initial conditions

Parameters	***Experiment***		***SPACE 2.19***	
	Intact Loop	***Broken Loop***	***Intact Loop***	***Broken Loop***
Primary Coolant System				
HL temp. (K)	587.4	585.9	586.9	586.9
CL temp. (K)	560.5	560.0	560.6	560.7
Loop flow rate (kg/s)	34.65	33.84	34.65	34.45
Pump speed (rad/s)	128.3	124.3	128.3	124.3
Core power (MW)	10		10	
Pressurizer				
Pzr. press. (MPa)	15.38		15.38	
Pzr. water level (m)	2.64		2.64	
Secondary Coolant System				
SG steam dome pressure (MPa)	6.89	6.89	6.86	6.86
SG steam flow rate (kg/s)	2.68	2.58	2.73	2.71
SG downcomer water level (m)	9.22	9.19	9.13	9.13

Figure 2 ~ figure 9 represent the comparison between experimental data and computed data on the thermal-hydraulic behavior. RELAP5/MOD 3.1 code is also used to identify the code predictability of SPACE 2.19.

Figure 2 shows the break flow rate through the break nozzle. The break flow rate gradually decreases as reduction of the pressure difference beween primary and secondary systems. After the HPSI actuation, the break flow rate slightly increases again. When the Relief Valve (RV) in the broken loop SG is opened, the rapid increase of the break flow rate appears several times periodically due to RV cycling. The break flow rate sharply decreases after the RCS depressurization using the pressurizer auxiliary spray.

Figures 3 and 4 show the mass flow rates in the intact and broken loop. The mass flow of intact loop decreases after the break, and increases rapidly at about 4000 seconds after reactor scram by restarting the RCP. The mass flow in broken loop reduces to approximately zero after the RCP trip, the natural circulation flow through the intact loop is maintained more than 5 kg/sec.

The similar trends are shown in the hot leg fluid temperatures in intact loop and broken loop, as shown in Figures 5 and 6. The overall transient behavior agree well with the experimental data.

Figure 7 represents fluid temperatures of reactor vessel core and RCS. When RCP trips, the RCS fluid temperatures decrease rapidly and the core outlet fluid temperature increases slightly. Because the atmospheric steam dump, feedwater of the secondary side, and the HPSI of the primary side are actuated, the RCS temperatures decrease gradually. In the intact loop, the fluid temperature decreases at about 4300 sec due to the forced convection according to the RCP restart of intact loop.

Figure 8 shows the primary system pressure behavior. According to the pressurizer pressure decreases monotonically, the reactor scram and safety injection and RCP trip occurs sequentially. The HPSI water continues to inject into primary side, and the pressurizer auxiliary spray system is used to increase a depressurization rate. When the RCS pressure becomes identical to the broken loop SG pressure, the pressurizer auxiliary spray is turned off. Finally, the RCP in intact loop is restarted at about 4000 sec.

Figure 9 represents the secondary pressure behavior during the SGTR transient. The broken loop SG is isolated by closing the MSIV and the intact loop SG is depressurized by opening the SG relief valve. The pressure of broken loop SG increases due to the primary coolant inflow and is controlled by opening and closing the SG relief valve. The overall trend is similar to the experiment, however there is a difference in the number of the RV cycling of broken loop SG. This difference may come from the insufficient nodalization and the modelling on the heat transfer in the secondary side [2].

Figures 10 and 11 show the water level of pressurizer and SG. the pressurizer is emptied completely at about 335 seconds and the water level is recovered from about 3000 seconds. After MSIV closure of the broken loop SG, the water level increases gradually by the primary coolant inflow. The water level in intact loop SG rapidly decreases due to the atmospheric steam dump. The SG inventory is maintained by the auxiliary feedwater system.

Based on the experimental and numerical comparisons, it is observed that the overall transient response of SPACE 2.19 agrees well with the LSTF SB-SG-06 test data and the RELAP5/MOD3.1 data, and SPACE 2.19 code has sufficient capability to simulate SG tube rupture.

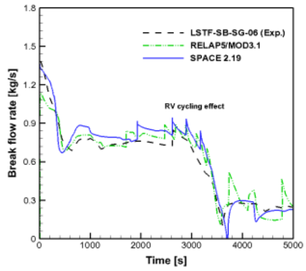

Figure 2 Break flow

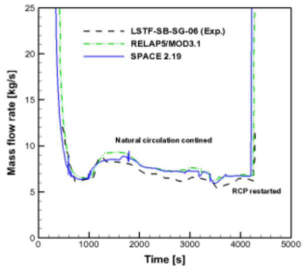

Figure 3 IL flow rate

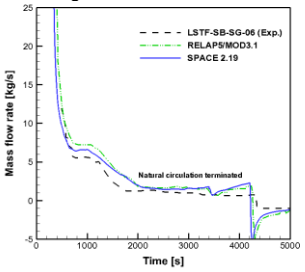

Figure 4 BL flow rate

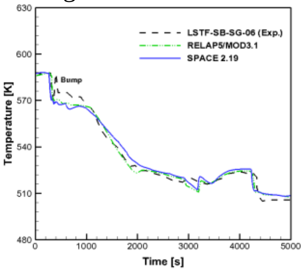

Figure 5 HL temp.e of IL

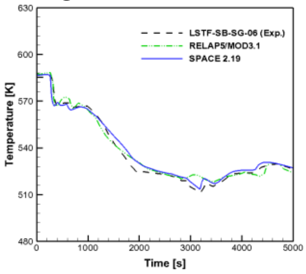

Figure 6 HL temp.e of BL

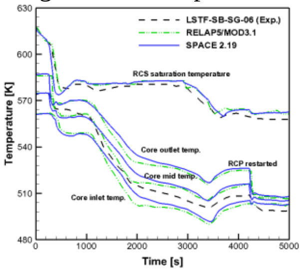

Figure7 Core fluid temp.

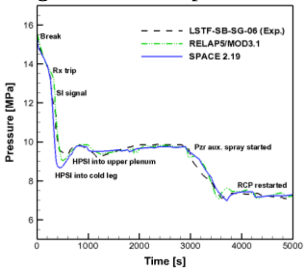

Figure 8 PZR pressuree

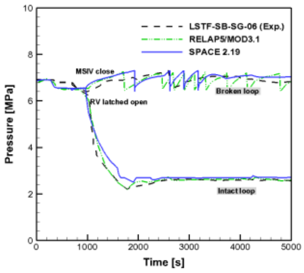

Figure 9 SG pressure

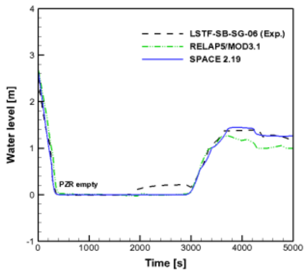

Figure 10 PZR level

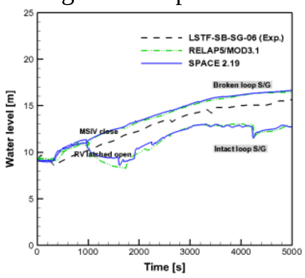

Figure 11 SG level

4. Conclusions

In order to identify the predictability of SPACE 2.19, the LSTF steam generator tube rupture test was simulated.

To evaluate the computed results, LSTF SB-SG-06 test data simulating the SGTR and the RELAP5/ MOD3.1 are used. The calculation results indicate that the SPACE 2.19 code predicted well the sequence of events and the major phenomena during the transient, such as the asymmetric loop behavior, reactor coolant system cooldown and heat transfer by natural circulation, the primary and secondary system depressurization by the pressurizer auxiliary spray and the steam dump using the intact loop steam generator relief valve.

ACKNOWLEDGMENTS

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