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Assessment of the SPACE Code Using the LSTF SGTR Test

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1. Introduction

The Korea Nuclear Hydro & Nuclear Power Co.(KHNP) has developed a multipurpose nuclear safety analysis code, called the Safety and Performance Analysis Code for Nuclear Power Plants (SPACE), with other Korean nuclear industries[1]. The code is a best-estimated two-phase three-field thermal-hydraulic analysis code for the safety or performance analyses of pressurized water reactors (PWRs). The code provides sufficient functions to replace current foreign vendor code systems.

As a result of the second phase of the development, the 2.14 version of the code was released through the successive various validation and verification (V&V) programs. The topical report on the code has been prepared for license works.

In this study, the steam generator tube rupture (SGTR) test, SB-SG-06, of the Large Scale Test Facility (LSTF) were simulated to evaluate the predictability of the SPACE code on non-loss-of-coolant-accidents (non-LOCA), and the results were compared with the experimental data or those of RELAP5 code simulations.

2. LSTF SGTR Test Description

2.1 LSTF

The Rig Of Safety Assessment (ROSA)-IV Program's LSTF was designed as a 1/48 volumetric scale integral test facility to simulate the response of a typical PWR by the Japan Atomic Energy Research Institute. The reference plant to scale the facility is the Tsuruga Nuclear Power Plant Unit 2 which is a typical Westinghouse 4-loop plant of 3,423 MWt. The flow areas are also scaled by 1/48 in the pressure vessel, and by 1/24 in the two steam generators (SGs) to reflect four ones. The height of each component and relative elevations, however, were fully scaled to simulate the coolant flows under natural circulation conditions. The core power of 10 MWt was scaled by 1/48 at a power equal to 14% rated power of the reference plant power. The LSTF core consists of about 1,100 full length electrical heater rods placed in a 17X17 array. The 141 U-tubes in each SG were arranged in a square array, and they consist of nine groups of U-tubes with different heights. The inner diameter and wall thickness of each U-tube are 19.6 and 2.9 mm, respectively. The hot and cold legs are represented in two loops and scaled so as to conserve the ratio of the length to the square root of pipe diameter for the reference PWR.

2.2 SGTR Test

The SGTR test, SB-SG-06, was simulated the SGTR occurred at Mihama Unit 2 on Feb. 9, 1991, which was a Westinghouse 4-loop PWR with Model 44 SGs [2].

The test was initiated by the opening a break valve nearly at the same conditions as in the plant. The reactor trip and the safety injection (SI) signals were generated at the same setpoint pressures as the plant. After the signal, the high pressure safety injection (HPSI) flow was injected to cold legs and vessel upper plenum. The SG in faulted loop was isolated about 12 minutes. The SG in intact loop was depressurized at the same time, and terminated according to the emergency operating procedure (EOP) as in the plant. The pressure of faulted SG was regulated by the relief valves. Instead of the pressurizer power operated relief valves (PORVs), which were failed to open, the pressurizer auxiliary spray was used about 44 minutes to reduce the primary pressure during the accident. The spray was turned off after the formation of pressure equilibrium between the primary and the faulted SG. The reactor coolant pump (RCP) of intact loop was restarted about 65 minutes after reactor trip, and the RCS conditions were stabilized. The sequence of main events for the SB-SG-06 test is summarized in Table I.

Table I: Sequence of events for the SGTR test

Event	Measured	Calculated
Transient initiation	0	0
Reactor SCRAM	266	270
MFW trip	300	301
Safety injection signal	205	301
PZR Empty	331	359
RCP coastdown	348	351
MSIV close	988	992
Intact SG RV close	1,751	1,819
Intact Loop RCP restarted	4,245	4,231

3. Modeling & Simulation

3.1 SPACE Code Modeling

The SPACE code package supplies a function that converts the inputs of other system codes (*e.g.*, RELAP, MARS or RETRAN, *etc*.) into those of the SPACE code. In this study, the conversion function was used to develop premature inputs based on the RELAP5 inputs listed in the references. The deck was appropriately modified according to the SPACE code User's Manual [3, 4]. The initial conditions, assumptions and boundary conditions were provided to reflect the experiment environments.

The vessel was modeled to express control volumes and flows appropriately, such as core flow, bypass flow, upper or lower plena, *etc*. The core was divided into vertically stacked 6 sub-CELL volumes and single hydraulic channel. In the test, the core was represented by the electric heated rods, so the reactivity feedback, which was one of the primitive factors inducing the asymmetry, and asymmetry effects could be ignored. The SI lines were connected to the cold-legs or the core upper plenum. The SG tubes were modeled with 8 parallel stacked volumes and the secondary sides were divided into 13 volumes. The main or auxiliary feedwater were modeled using TFBC components, which were connected to feeder-rings. The pressurizer was modeled as vertically stacked 8 sub-CELLs. The surge-line was connected to intact loop hot-leg with 3 sub-CELLs, and the spray line was branched from intact loop cold-leg. The critical flow model used for the break was Ransom-Trapp, default model in the SPACE code, and corresponding discharge coefficients (*Cd*) were selected as 1.0. The schematic nodal diagram to simulate the test is depicted as Figure 1.

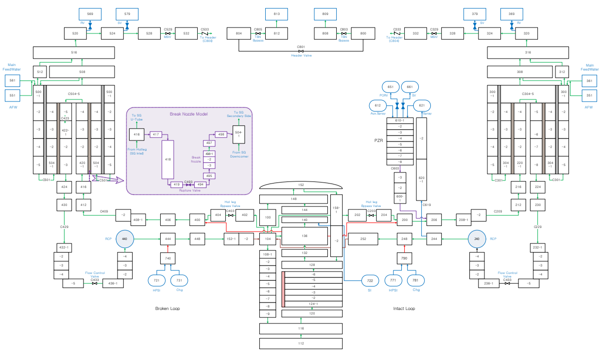

Fig. 1. LSTF model to simulate the SGTR test

3.2 SGTR Simulation

Following the tube rupture occurred in faulted loop, the break outflow from the primary side entered into the secondary side (Fig. 2), consequently the secondary side pressure increased. The results of the SPACE code showed similar trends to the experimental data except detail behaviors such as the relief valve (RV) stroke cycling, break flow amount, *etc*. It is ascertained through the integrated break flow (Fig. 3).

The coolants in the primary loops reduced to natural circulation level by the RCP trip (Fig. 4). The intact loop flow, however, maintained over 5 kg/s flow rate derived by the continuous heat removal from the primary to secondary system and kept the core in subcooling condition (Fig. 5). After the restart of RCP at 4,231 seconds, the intact loop flow was recovered.

The power trend showed acceptable agreement between the codes except the initiation of reactor trip which was caused by the pressurizer (PZR) low pressure signal (Fig. 4). The coolant temperatures at inlet, mid section and outlet of core including the saturation temperature were delineated in Figure 7. The temperatures gradually decreased by the HPSI and the steam dump in the secondary system. The trends were rapidly dropped after the restart of RCP.

The primary system pressure behavior during the transient was represented in Figure 8. After the rupture, the pressure and level (Fig. 9) rapidly decreased due to the break flow from the primary side to secondary side, which was sufficiently larger than those could be recovered by the charging pumps. During the decrease, the pressurizer was emptied at 359 seconds. After the injection of HPSI system, the pressure and level were partially recovered and formed meta-plateau up to the spray actuation, which caused the pressure decrease or level increase. Following the tube rupture, the broken loop SG was isolated by the closure of MSIV at 992 seconds, and simultaneously the intact loop SG was depressurized by the open of relief valve (RV) to maintain the heat removal from the primary to secondary system (Fig. 10). The broken loop pressure increased due to the coolant inflow from the primary system and controlled by the movement of RVs.

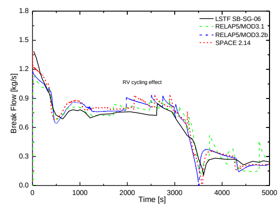

Fig. 2. Break Flow

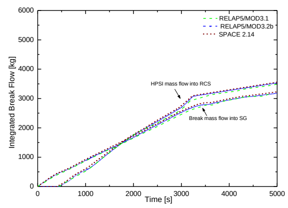

Fig. 3. Integrated Break Flow

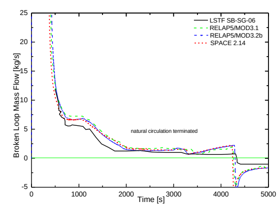

Fig. 4. BL Flow

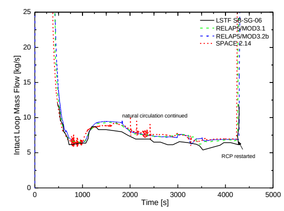

Fig. 5. IL Flow

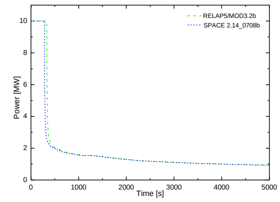

Fig. 6. Power

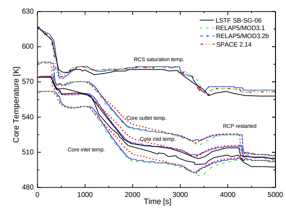

Fig. 7. Core Temperature

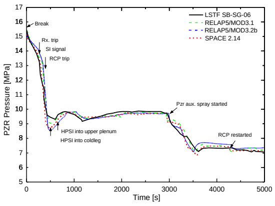

Fig. 8. PZR Pressure

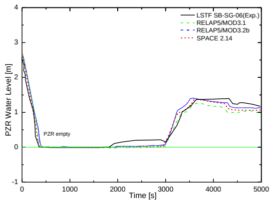

Fig. 9. PZR Level

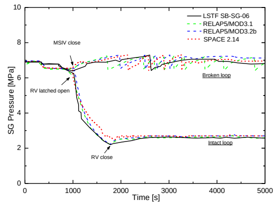

Fig. 10. SG Pressure

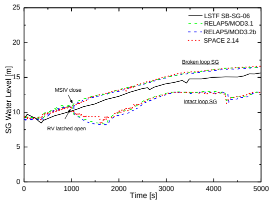

Fig. 11. SG Level

4. Conclusions

The steam generator tube rupture test of integrated test loops, the Large Scale Test Facility (LSTF), was simulated using the SPACE code as the V&V program.

The results of the simulations were compared with the experimental data and those of the other code simulations. Through the simulation, it was concluded that the SPACE code could give reliable calculation results to applicable to PWRs in the case of SGTR accidents.

ACKNOWLEDGEMENTS

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