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Validation of SPACE for Steam Generator Tube Rupture Accident Using SMART-ITL Experimental Data

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1. Introduction

The SMART-ITL facility is modeled by multiple passive systems along with a single Reactor Pressure Vessel (RPV) and four Steam Generator (SG) components. While the SMART plant is designed with the module concept and SGs are encapsulated in the RPV vessel, SMART-ITL is designed such that the SGs are installed in exterior of the RPV and connected with pipe lines. The adiabatic Passive Safety Injection System (PSIS) including Core Makeup Tanks (CMTs) and Safety Injection Tanks (SITs) are connected at upper downcomer of the RPV.

The Steam Generator Tube Rupture (SGTR) of SMART-ITL is a postulated accident, where one of tubes in a SG is ruptured [1]. As a single helical tube of the steam generator is ruptured, the coolant of the Reactor Coolant System (RCS) is discharged to the secondary side of SMART-ITL through the ruptured tube, and eventually mixed with a fluid in the secondary system. The SGTR is an important accident in view of the radioactive material release to the secondary system.

In this study, the SGTR is modeled by an opening value, break nozzle, and two pipe components that directly connect the primary side of steam generator and steam line, and SPACE [2] calculations for the SGTR accident are validated using SMART-ITL experimental data.

2. Methods and Results

2.1 Major sequence of event of the SGTR for the SMART-ITL

The SGTR is initiated by opening the break valve (OV-BS6B-01) installed in the SGTR accident simulator of the break simulation system. The primary pressure and core level gradually decrease during an early phase of SGTR accident and eventually reach the Low Pressurizer Level (LPL) set point at 1471 s. A reactor trip signal is generated with an l.1 seconds of delay when the pressurizer level falls below 45% of its reference value. Control rods are inserted almost immediately right after the reactor trip signal leading to a significant decrease in core power. The Passive Residual Heat Removal Actuation Signal (PRHRAS) and Chemical and Volume Control System (CVCS) isolation actuation signal are generated by LPL with 1.45 seconds of delay. The SG secondary side is isolated from turbine by closing the main steam and feedwater isolation valves, and subsequently connected to the Passive Residual Heat Removal System (PRHRS) with 5 seconds of delay. The CMT water is passively injected by gravity head following the CMT actuation signal generated after PRHRAS with 1.45 seconds delay. The SGTR test is terminated when the RCS temperature decreases to a safety shutdown condition of 488 K.

2.2 Validation Results

The accumulated mass of the primary coolant discharged through the break nozzle depends on the critical flow models and their specific coefficients utilized for simulating break flows. Ransom-Trapp critical flow model is applied to the break nozzle with phasic discharge coefficients adjusted to be 0.9 to match the collapsed liquid level in the pressurizer.

As opening the break valve at 0.0 s, the coolant in the primary side is discharged to the secondary side through the break nozzle, and mixed with a fluid in the secondary system. The pressurizer level calculated by SPACE reaches LPL at 1345 s while the measured collapsed liquid level decreases to LPL set point in 1471 s, showing 126 seconds of deviation in time between them (see Fig. 1). As shown in Fig. 2, the pressurizer pressure decreases sharply with time owing to the significant loss of primary inventory. When the Feedline Isolation Valves (FIVs) and Main Steam Isolation Valves (MSIVs) are fully closed, steam pressure for the ruptured steam generator increases significantly, and eventually becomes equivalent to the primary pressure (see Fig. 3). The effect of the break flow becomes negligible under this condition, whereas cooldown capability of the passive safety systems including PRHRS and CMT becomes more significant. As soon as the feed water supply to the SGs is terminated at about 1400 s by closure of the FIVs, PRHRS is activated as depicted in Fig. 4. Although the PRHRS flow rate is lower than the feed flow rate measured at steady state condition, this passive system effectively cools down the primary system such that the RCS temperature continues to decrease to the safety shutdown condition of 488 K (see Fig. 5). Fig. 6 compares the CMT liquid level calculated by SPACE with experimental data. As a result of the CMT injection by head difference between the PBLs and CMT injection lines, the CMT liquid level starts to decrease at 5000 s. Actuation of the CMT injection recovers the inventory loss of the primary side, and subsequently returns the collapsed liquid level in the pressurizer back to a higher level as depicted in Fig 1.

As the system is cooled down gradually by the actuation of PRHRS and CMT, the RCS temperature as well as the primary and secondary pressure continues to decrease to the safety shutdown condition. SPACE predicts the system pressure and temperature well, which highly depends on discharge rate in earlier phase and cooldown rate in long term period. Calculations are terminated when the RCS condition reached the safe shutdown condition.

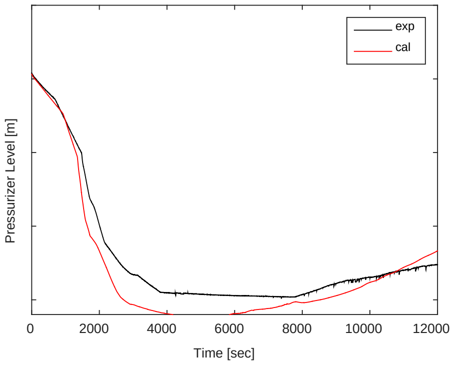

Fig. 1. Comparison of pressurizer level for SGTR

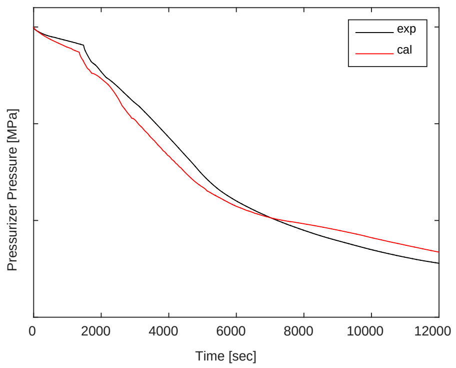

Fig. 2. Comparison of pressurizer pressure for SGTR

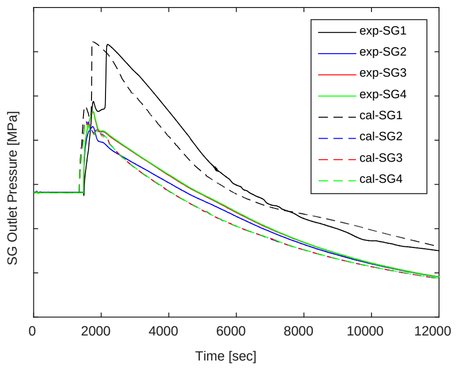

Fig. 3. Comparison of SG secondary pressure for SGTR

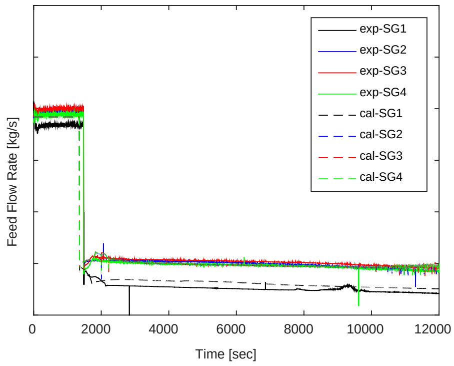

Fig. 4. Comparison of feed flow rate for SGTR

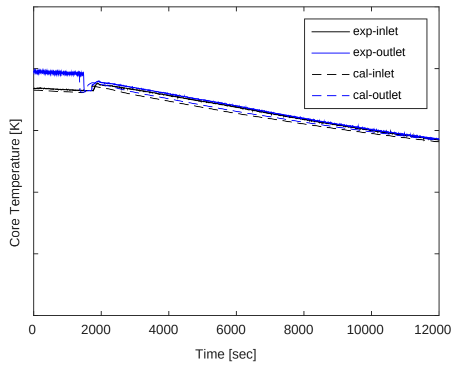

Fig. 5. Comparison of RCS temperature for SGTR

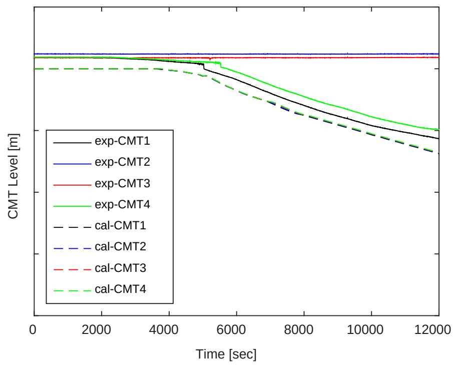

Fig. 6. Comparison of CMT level for SGTR

3. Conclusions

Validation of SPACE for the SGTR accident was performed using SMART-ITL experimental data. In this study, the SGTR was modeled by an opening value, break nozzle, and two pipe components that directly connected the primary side of steam generator and steam line. It was shown that SPACE predicted the system pressure and temperature well, which highly depended on discharge rate in earlier phase and cooldown rate in long term period.

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[2] S. J. Ha, C. E. Park, K. D. Kim, and C. H. Ban, "Development of the SPACE Code for Nuclear Power Plants," *Nuclear Engineering and Technology*, **43**, 1, 2011.