SPACE Analysis for Safety Injection Line Break Concurrent with TLOSHR in SMART-ITL Facility

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1. Introduction

After Fukushima Daiichi accident, a new accident category was established and called design extension condition (DEC). It includes accidents involving multiple failures of safety features, which were historically referred to as beyond design basis accidents. Therefore, according to recent IAEA safety standard for the design of nuclear power plants (NPPs) [1] and the notice of Nuclear Safety and Security Commission (NSSC) No. 2016-02 [2], the new designs of NPPs shall consider those accident conditions during the design phase. However, since the accident scenarios of DECs include complicated two-phase flow conditions, the transient responses are not easy to simulate properly using a simplified safety analysis code. Moreover, the various inherent and passive design features of SMART100, which are different from the conventional loop-type PWRs, add further complexity to the simulation.

The Safety and Performance Analysis Code for nuclear power plants (SPACE) has been developed for the safety analysis of loop-type Pressurized Water Reactors (PWRs) and the design of advanced water reactors. The SPACE adopts advanced physical modeling of two-phase flows, mainly two-fluid three-field models which consists of gas, continuous liquid, and droplet fields. In 2017, the Nuclear Safety and Security Commission (NSSC) approved the use of the SPACE for licensing applications of Korean PWRs. However, if it is intended to be used in the licensing application of SMART-100, further development and validation is required.

In general, the prediction results of system analysis codes may be inconsistent with the experimental results due to various uncertainties in numerical schemes, empirical correlations, and user errors [3]. To enhance the reliability of the simulation results, validation work for many kinds of separate effect tests and integral effect tests is required. Therefore, the purpose of this paper is to evaluate SPACE code capability by validating SBLOCA scenario initiated by a break in SMART-ITL's Safety Injection Line (SIL) concurrent with Total Loss Of Secondary Heat Removal (TLOSHR). It first introduces SMART-ITL facility and SPACE code nodalization then presents the accident scenario, and finally discuss the comparison of the simulation results and the experimental results.

2. Methodology

2.1 Overview of SMART ITL

The SMART-ITL is a thermal-hydraulic integral effect test facility for SMART. It is designed based on the volume scaling methodology at which the height of the individual components is conserved, and the flow area and volume are scaled down to 1/49. It has the same integral features as SMART except for the externally installed Steam Generators (SGs). The main objective of the SMART-ITL are to investigate and understand the integral behavior and the thermal-hydraulic phenomena occurring in the reactor systems and components during the normal, abnormal, and emergency conditions [4]. The integral-effect test data are also used to validate the related thermal-hydraulic models of the safety analysis codes, which can be used for a performance, and accident analysis of the SMART design. A simplified schematic diagram of SMART-ITL facility is shown in Fig 1.

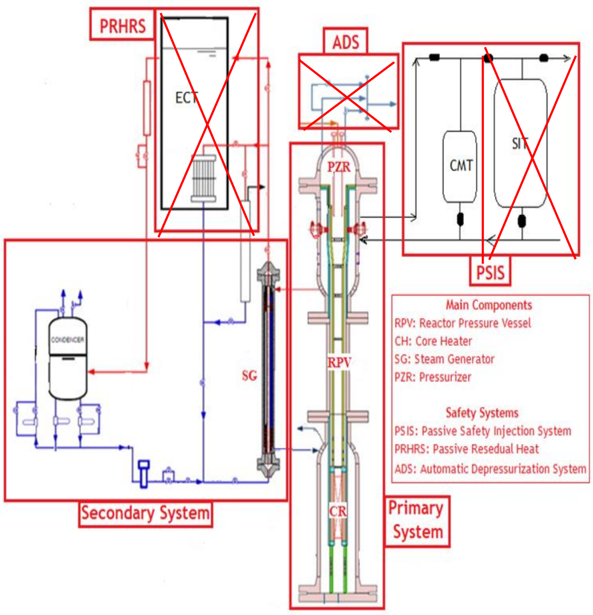

Fig. 1. Simplified schematic diagram of SMART-ITL facility

The fluid system of SMART-ITL consists of a primary system, a secondary system, Safety related systems, a break simulating system (BSS), a break measuring system (BMS), and auxiliary systems. The primary system is composed of reactor pressure vessel (RPV), reactor coolant pumps (RCPs), SGs, and primary connecting piping between Reactor Pressure Vessel (RPV) and SGs.

The secondary system of the SMART-ITL is simplified to be of a circulating loop-type and is composed of a condenser, feed water and steam lines, and related piping and valves.

The safety related systems includes four trains of the Passive Residual Heat Removal System (PRHRS), four trains of the Passive Safety Injection Systems (PSIS), and two trains of the Automatic Depressurization Systems (ADS). The PRHRS designed to removes the decay heat by natural circulation in emergency situation while the PSIS was designed to inject borated water into the RCS by gravity head to prevent core uncover in case of LOCA scenarios. The ADS helps to rapidly depressurize the RCS to activate SIT (Safety Injection Tanks) earlier during LOCA accident. In this accident scenario, all trains of PRHRS, SITs, and ADS were isolated from the system.

2.2 Nodalization of the SMART-ITL

A simplified nodalization of SPACE code for SMART-ITL is presented in Fig. 2. The RCS, secondary system, Safety Injection Tanks (SITs), Core Makeup Tanks (CMTs) and the PRHRS are modeled with cells and faces. The RCS consists of the heater for the core simulator, upper plenum, RCPs, SGs primary side, downcomer, core bottom region, and the PZR. In order to simulate the heat loss, heat structures with proper geometries, material properties and outer boundary conditions are attached to the outer cells. In this accident scenario, the PRHRS, ADS, and SITs were not actuated.

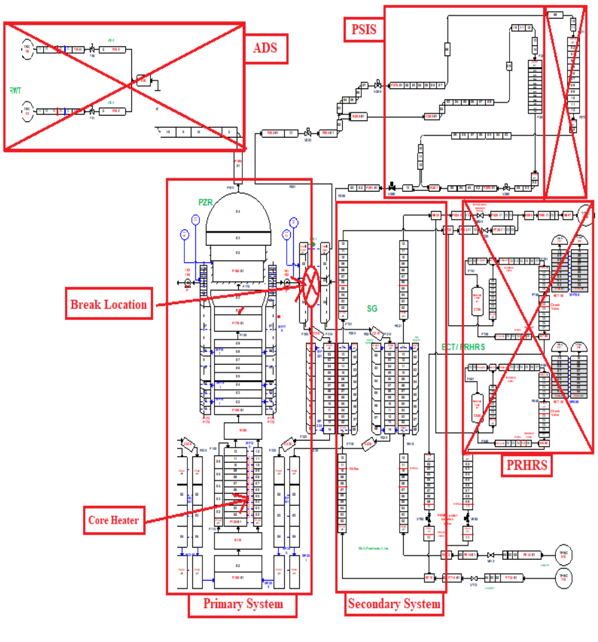

Fig. 2. Simplified nodalization for single train of SMART-ITL facility

2.2 Accident scenario

The SIL LOCA is caused by a break of the safety injection line connecting to the RCS pressure boundary. As the break occurs, reactor coolant is discharged through the break area and the Pressurizer (PZR) pressure decreases.

When the PZR pressure reaches the low PZR pressure (LPP) reactor trip setpoint (10.26 MPa), the reactor trip signal is generated and the heater power follows a decay curve (1.2×ANS-73 residual heat curve required on 10CFR50 Appendix K) [5]. The loss of offsite power (LOOP) is considered as a coincidence occurrence and the power to the RCPs and the feedwater pumps is lost simultaneously with the turbine trip. Then, the PRHRAS is generated by the low feedwater flow rate, but the PRHRS fails to operate. The RCS pressure decreases continuously due to the loss of the coolant mass and energy through the break flow.

As the CMT actuation signal (CMTAS) is generated by the LPP signal, the CMT isolation valves are opened. Consequently, the water in the CMT is injected into the RPV by the gravitational force after the empty of the pressure balance line in the PSIS.

The subcooled water is discharged through the break at the beginning of the transient. As the PZR pressure decreases to the saturation pressure and the water level in the RPV decreases to the break location, the phase of the break flow changes to the two-phase mixture and then steam. With the injection of the water from the CMT and the RPV, the water level inside of the RPV is recovered. Throughout the transient, the core is covered with water and thus the coolant temperatures as well as the fuel temperature are monotonically decreased. The sequence of events for the SB-PSIS-F101 test are shown in Table I.

Table I: Sequence of events for SB-PSIS-F101 test [5]

Sequence of Events	**Set point / Trip signal**	**Time (s)**
Steady-state	-	-754
Accident start	Break in SIL	0
Reactor trip setpoint reached	LPP=10.26 MPa	630
Reactor trip signal generation	LPP+1.1 s	631
Turbine trip		
RCP coastdown start		
Feedwater stop		
CMTAS generation		
Control rod insertion	Decay heat table	632
CMT injection	LPP+2.2 s	632
Generation of PRHRSAS (PRHRS failed to operate)	LPP+5.2 s	632
FIV/MSIV close	LPP+10.2	641
Experiment termination	-	42708

2.4 Steady-State Condition

The steady-state calculation is performed to verify the input nodalization of SPACE code for the SB-PSIS-F101 test. For the steady-state calculation, the averaged test results of the thermal hydraulic parameters of the RCS, secondary system, PSIS, and PRHRS are used. The major thermal hydraulic parameters of the SB-PSIS-F101 test at steady-state are listed in Table II. During the steady-state, the measured RCS flow rate was maintained at 10.397 kg/s while the calculated flow rate is 11.52kg/s. s. This is because the calculated flow rate is adjusted in order to match the temperature difference between the core inlet and outlet with the experimental measurements. The SG inlet and outlet temperatures are 594.3 K (321.2 °C) and 571.9 K (298.8 °C), respectively. In the secondary system, the subcooled feedwater is supplied to the SG to remove the heat from the primary system and becomes superheated steam. The feedwater flow rate is 0.778 kg/s and the steam pressure is 5.63 MPa.

Table II shows a comparison between SMART-ITL major thermal hydraulics parameters with the calculation results of SPACE. The steady-state results of the SPACE calculation for the selected parameters were in a very good agreement with experimental values.

Table II: Steady-state calculation results of SB-PSIS-F101

Parameter	EXP	SPACE	Error (%)
Power (MW)	1.6723	1.6723	BC
Core Inlet Temp (K)	568.7	569.4	0.12
Core Outlet Temp (K)	594.1	594.7	0.1
SG Primary Inlet Temp (K)	594.3	594.5	0.03
SG Primary Outlet Temp (K)	571.9	571.7	-0.04
PZR pressure (MPa)	15	15	BC
PZR level (m)	2.972	2.973	0.03
RCS flow rate (kg/s)	10.397	11.52	Adjust
SG Secondary Inlet Temp (K)	503.15	503.15	BC
SG Secondary Outlet Temp (K)	575.45	544.5	-5.3
Feed Water Flow rate (kg/s)	0.778	0.774	Adjust
Feed Water Pressure (MPa)	5.71	5.71	BC
Mean Steam Pressure (Mpa)	5.63	5.632	0.04

3. Results and Discussion

After obtaining a good agreement between the code calculation and the experiment, the steady-state conditions are used as initial conditions for the transient calculation. However, before evaluating results of the main parameters, a sensitivity analysis to select proper critical flow models and discharge coefficients should be performed as follows:

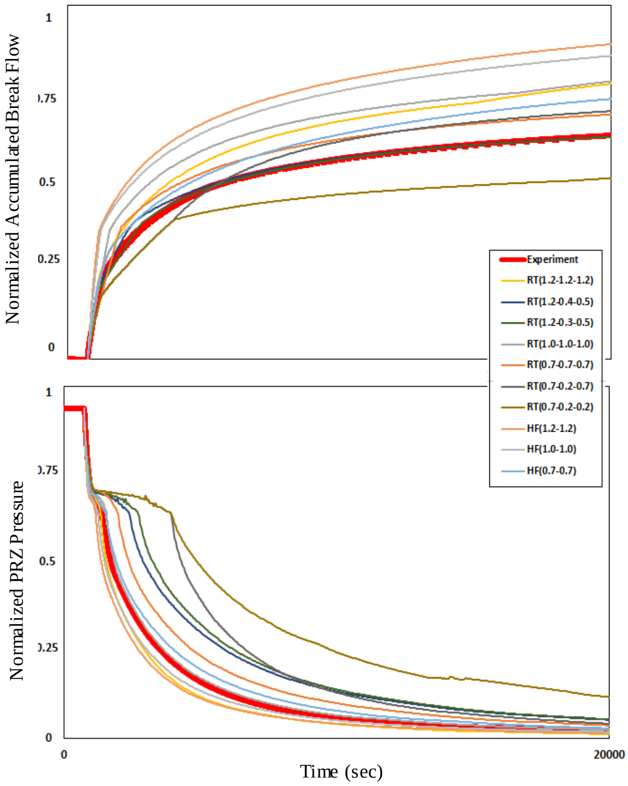

Fig. 3. Sensitivity analysis of critical flow models and discharge coefficients

Since the behavior of the primary and the secondary loop heavily depends on the accumulated break flow, it is crucial to perform a sensitivity analysis to find the effect of different critical flow models and discharge coefficients on the calculation results of SPACE code and to select the optimal settings for the simulation. As shown in Fig. 3, at equivalent discharge coefficients, the calculation results of the Henry-Fauske critical flow model resulted in higher accumulated break flows and thus higher depressurization rates compared to Ransom-Trapp model. Furthermore, a best fitting curve to the accumulated break flow resulted in a long delay of the reactor trip owing to the low depressurization rate. On the other hand, a best fitting curve to the depressurization rate resulted in an overestimation of the accumulated break flow which can lead to a core uncover. Therefore, by taking into consideration the accumulated break flow and the depressurization rate, Ransom-Trapp model with default discharges coefficients was used.

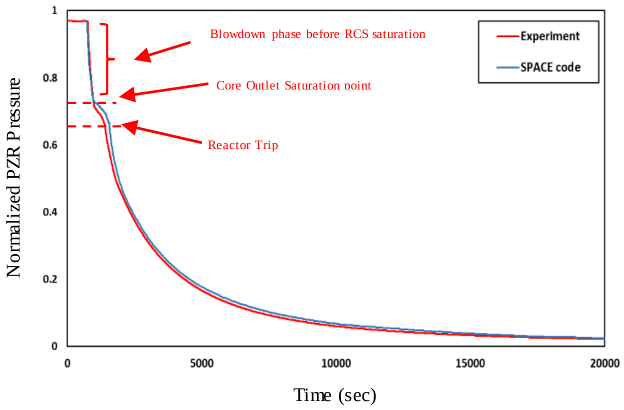

Fig. 4. Comparison of pressurizer pressure for SBLOCA

Fig. 4 shows the pressure behavior of the primary system. When the SIL break occurred, the PZR pressure decreases rapidly during the blowdown phase. Once the PZR pressure reached the saturation pressure of the core outlet temperature, the depressurization rates decreased owing to the high evaporation rates in the primary system. After a short period, the PZR pressure reached the LPP setpoint of 10.26 MPa and the reactor trip signal by the LPP was generated. Consequently, the core power started to decrease according to the simulated decay heat of the experiment. Further, with the simultaneous assumption of LOOP, the RCP started to coastdown and the forced flow circulation was terminated. After the reactor trip, the system pressure decreased continuously until the end of the scenario. The SPACE code predicts the depressurization behavior comparatively well.

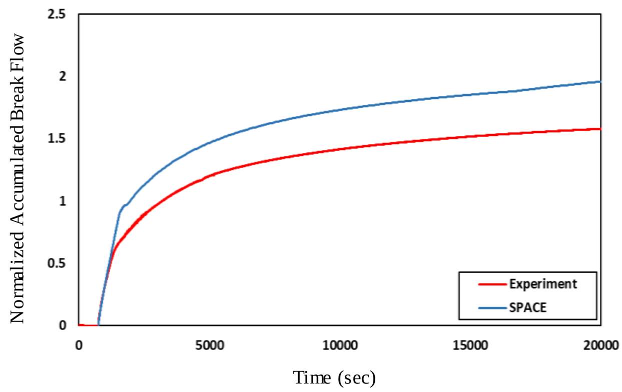

Fig. 5. Comparison of accumulated break flow

Fig. 5 shows a comparison of accumulated break flow rate between SPACE code and Experiment. The SPACE code shows a reasonable prediction for both single-phase subcooled liquid and steam . The overprediction of the code occures in the two-phase time period which can be clearly shown in Fig. 6.

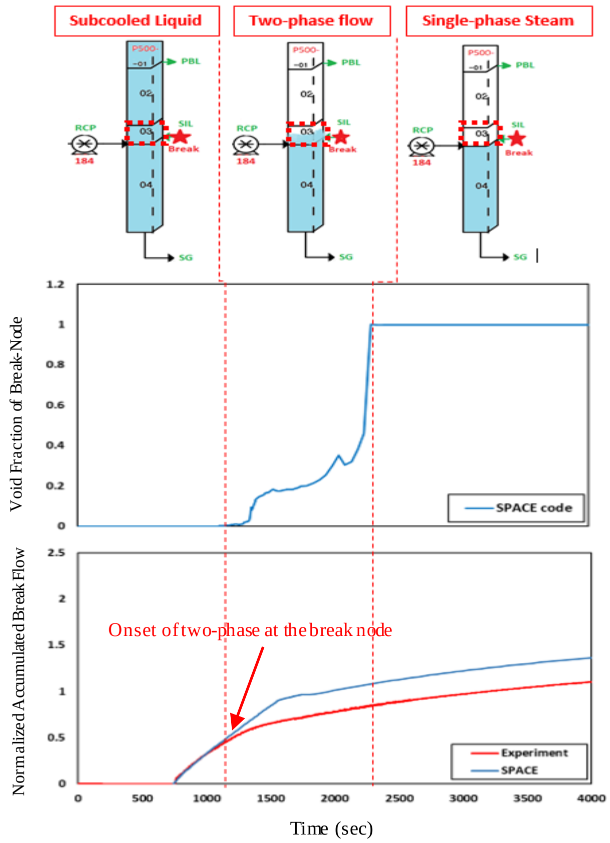

Fig. 6. Reason of the overestimation in the accumulated break flow

According to the evaluation of the accumulated break flow, the subcooled flow and the single-phase steam flow are well predicted by SPACE code as shown in Fig. 6. However, the code overpredicts the beak flow rates during the two-phase flow period. This mainly due to inaccuracy of the critical flow models during two-phase flow period. Therefore, the users of the code should always take into consideration the existance of the two-phase in the break node and their significant impact on the depressurization rate and the break flow rate.

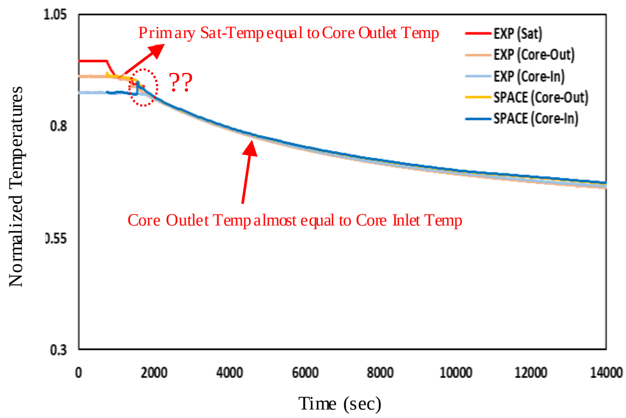

Fig. 7. Comparision of core inlet and outlet temperatures

Fig. 7 shows the fluid temperatures at the core inlet and outlet. In the experiment, the fluid temperatures decreased with the saturation temperature corresponding to the system pressure. After the reactor trip, a sudden increase in the calculated core inlet temperature was observed due to insufficient heat removal of the decay heat. The decay heat was removed continuously through the break and thus the fluid temperature at the core inlet and outlet decreased gradually until the end of the accident. The SPACE code properly predicts the overall fluid temperatures maintaining saturation condition.

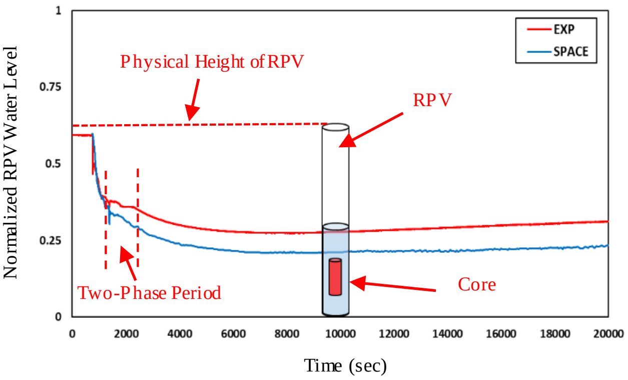

Fig. 8. Comparison of RPV collapsed water level

Fig. 8 shows the collapsed water level in the reactor pressure vessel. In the test, the water level decreased after the break and became stabilized after the actuation of CMT. The minimum collapsed water level was higher than the core top elevation. The SPACE code underpredicts the collapsed water level owing to the overprediction of the break flow rate during the existence of the two-phase flow in the break node.

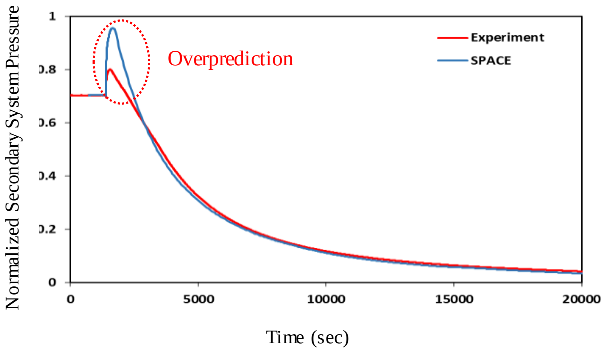

Fig. 9. Comparison of secondary system pressure

Fig. 9 shows the secondary system pressure. In the test, the SG secondary system pressure was maintained at operational pressure until the PRHRS actuation signal was operated and the MSIV and FIV were closed. The secondary system pressure increased rapidly with the actuation of the PRHRS actuation signal. Then, it decreased gradually by the heat removal to the primary side. The SPACE code properly predicts the overall behavior of the secondary system pressure but it slightly overpredicts the maximum secondary pressure at the beginning of the transient due to the over-prediction of the heat exchange between the primary and secondary systems.

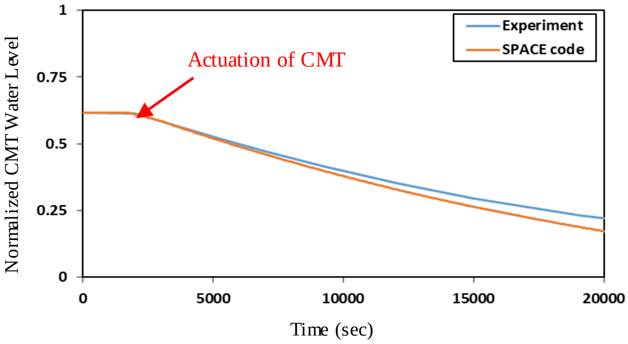

Fig. 10. CMT water level

As shown in Fig. 10, the CMT water level started at similar time with the experiment and decreases continuously until the end of the accident. The SPACE code properly predicts the measured water level of CMT.

4. Conclusion

Validation of the SPACE code was performed using the test results of SB-PSIS-F101 at SMART-ITL facility. The validation results showed that the overall thermal-hydraulic behaviors such as the primary system pressure, primary system temperatures, and secondary system pressure were properly predicted. However, SPACE code underpredicted the water level in the reactor pressure vessel because of the overprediction of the accumulated break flow. This was mainly due to the inaccuracy of the critical flow models during two-phase flow period. Therefore, the users of the code should always take into consideration the existence of the two-phase in the break node and their significant impact on the depressurization rate and the break flow rate.

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