SPACE Analysis for Pressurizer Safety Valve Break in SMART-ITL Facility

Sultan Al-Faifi [a,b,c*], Eslam Bali [a,b], Kyung Doo Kim[b], Jeong Ik Lee[c]
[a]*King Abdullah City for Atomic and Renewable Energy, Riyadh 12244, Saudi Arabia*
[b]*Korea Atomic Energy Research Institute, 989-111 Daedeokdaero, Yuseong, Daejeon, 305-353, Korea*
[c]*Korea Advance Institute of Science and Technology, 291 Daehak-ro, Yudeong-gu, Daejeon 34141, Republic of Korea*
**Corresponding author: s.faifi@energy.gov.sa*

1. Introduction

Demonstrations of the capabilities of nuclear system analysis codes are required to obtain an approval for their use in various applications of nuclear power plants. The Safety and Performance Analysis CodE (SPACE) has been developed by the Korean nuclear industries and approved by the Korean Nuclear Safety and Security Commission (NSSC) to be used for licensing applications of Pressurized Water Reactors (PWRs). However, since new innovative designs such as SMART100 incorporate inherent and passive safety design features that are not used in conventional loop-type PWRs, especial models should be developed and validated to reflects the characteristics of the SMART-100 and obtain reliable predictions.

A well-known approach to validate numerical tools, which are used to analyze complex systems such as nuclear reactors, starts by selecting the scenarios that are intended to be analyzed, and their most significant figures-of-merit (i.e., maximum or minimum acceptable limits stemming from design/operational or safety requirements). After that, identify all dominant phenomena that govern or influence the selected figures-of-merit and rank them according to their importance. Finally, design a Separate Effect Test (SET) or an Integral Effect Test (IET) facility to simulate those phenomena at a reduced scale and compare the numerical results with the experimental ones.

Based on the last step of the general validation process, this paper aims to assess SPACE code capability by simulating SBLOCA scenario initiated by a break in the pressurizer safety valve of SMART-ITL (SMART-Integral Test Loop) facility. This paper begins by providing a brief description of the experimental test facility and its nodalization in SPACE code. Then, it presents the sequence of the accident scenario. Finally, it discusses the comparison of the simulation results and the experimental results for the steady-state and transient scenario.

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 [1]. 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.

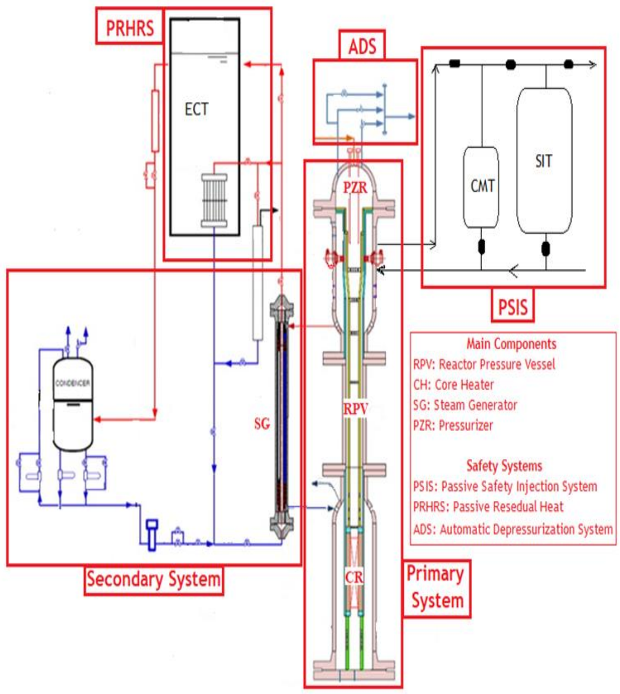

Fig. 1. Simplified schematic diagram of SMART-ITL facility

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 SITs (Safety Injection Tanks) earlier during LOCA accident.

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.

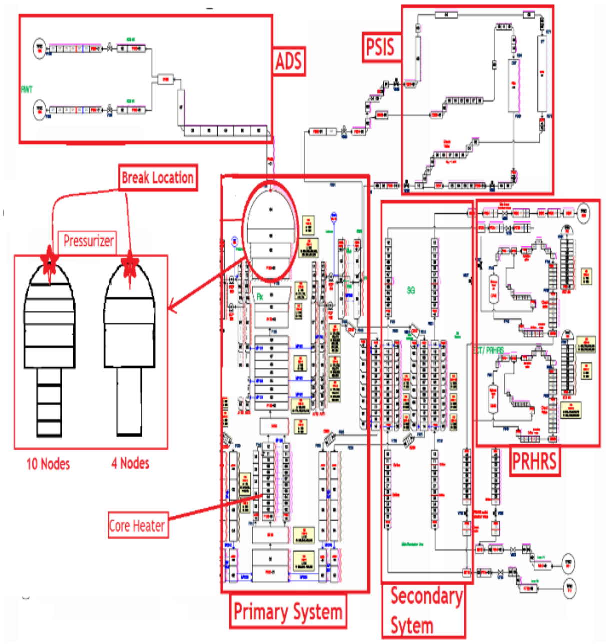

Fig. 2. Simplified nodalization for single train of SMART-ITL facility

2.3 SBLOCA scenario

The PSV LOCA is caused by a break of the pressurizer safety valve 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)[2]. 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, and the PRHRS is actuated. With the actuation of PRHRS, the residual heat of the core is removed through the PRHRS and break flow. Hence, the RCS pressure decreases continuously.

As the CMT actuation signal (CMTAS) is generated by the LPP signal, 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. When the PZR pressure decreases further to the SIT actuation signal (SITAS) setpoint, the cold water in the SIT is also injected into the RPV.

Throughout the transient, only saturated steam is discharged through the break. Therefore, with the injection of the water from the CMTs and SITs into the RPV, the water level inside the RPV is recovered and the coolant temperatures as well as the fuel temperature are monotonically decreased. The sequence of events for the SB-PSIS-F301 test are shown in Table I.

Table I: Sequence of events for SB-PSIS-F301 test [2]

Sequence of Events	Set point / Trip signal	Time (s)
Steady-state	-	-744
Accident start	Break in PSV	0
Reactor trip setpoint reached	LPP=10.26 MPa	204
Reactor trip signal generation	LPP+1.1 s	205
Turbine trip		
RCP coastdown start		
Feedwater stop		
CMTAS generation		
Control rod insertion	Decay heat table	206
CMT injection	LPP+2.2 s	206
PRHRS valve open	LPP+10.2 s	214
FIV/MSIV close	LPP+10.2	215
SITAS generation	LLPP=2 MPa	4,127
SIT injection	SITAS+1.1 s	4,131
ADS #1 open	CMT level= 35%	24,093
Experiment termination	-	261,326

2.4 Steady-State Condition

The steady-state calculation is performed to verify the input nodalization of SPACE code for the SB-PSIS-F301 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-F301 test at steady-state are listed in Table II. During the steady-state, the measured RCS flow rate was maintained at 10.46 kg/s. 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 0.774 kg/s and the steam pressure is 5.63 MPa.

The steady-state calculation was performed for 3000 sec, and the results are summarized in Table II. It was confirmed that the steady-state results of the SPACE calculation for the major parameters were in a very good agreement with the experimental values. Therefore, the transient simulation was performed by restarting from this steady-state results. Henry-Fausky critical flow model was selected with discharge coefficients of 0.75 and 0.4 for single-phase and two-phase flow, respectively.

Table II: Steady-state calculation results of PSV SBLOCA

Parameter	EXP	SPACE	Error (%)
Power (MW)	1.693	1.693	BC
Core Inlet Temp (K)	569.6	569.2	-0.07
Core Outlet Temp (K)	594.2	594.6	0.07
SG Primary Inlet Temp (K)	594.3	594.6	0.05
SG Primary Outlet Temp (K)	571.9	571.2	-0.12
PZR pressure (MPa)	15	15	BC
PZR level (m)	3.17	3.16	-0.32
RCS flow rate (kg/s)	10.46	10.46	BC
SG Secondary Inlet Temp (K)	503.1	503.1	BC
SG Secondary Outlet Temp (K)	587.9	594.5	1.12
Feed Water Flow rate (kg/s)	0.774	0.774	BC
Feed Water Pressure (MPa)	5.72	5.72	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 experimental results, the steady-state conditions were used as initial conditions for the transient calculation and the results of the main parameters are shown as follows:

Fig. 3 shows the behavior of the PZR pressure. When PSV break occurs, the PZR pressure drops rapidly during the blowdown phase until it reached the saturation pressure of the core outlet temperature. Then, the depressurization rate decreased owing to the high steam generation in the core. 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. Moreover, the simultaneous assumption of LOOP led to the coastdown of the RCPs and the flow pattern was changed from forced circulation to natural circulation. After that, the system pressure decreased continuously until the end of the scenario.

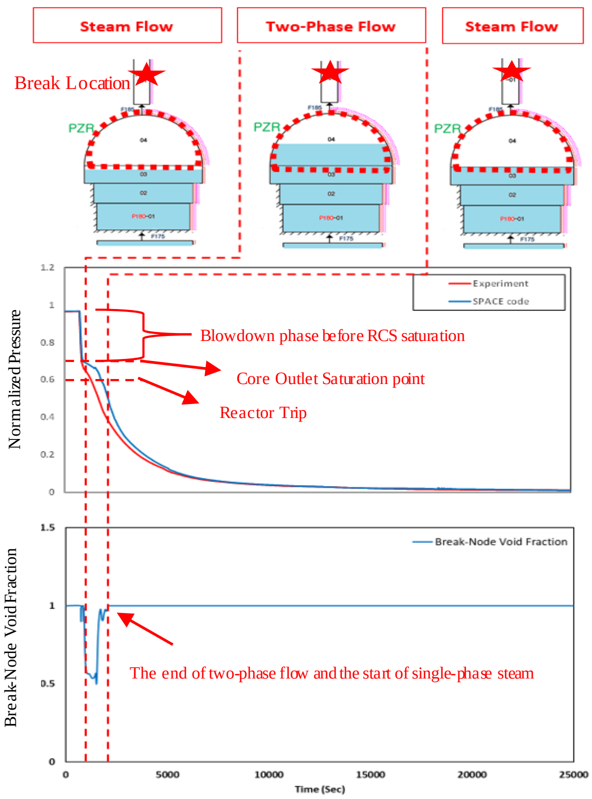

Fig. 3. Comparison of pressurizer pressure for SBLOCA

The SPACE code predicts the overall depressurization behavior comparatively well but slightly underpredicts the depressurization rate at the end of the blowdown phase which resulted in a delay of the reactor trip. The overprediction was clearly a result of the existence of two-phase flow in the break node after the swelling of the RCS. Therefore, while only a discharge of single-phase steam flow was observed in the experiment, SPACE code predicted two-phase flow at the onset of the accident. This error can have an impact not only on the depressurization rate but also on the break flow rate. That is because a release of two phase flow instead of a single-phase steam flow lead to a lower depressurization rate and higher break flow rate. Those effects can be shown in Fig. 3 and Fig. 4, respectively.

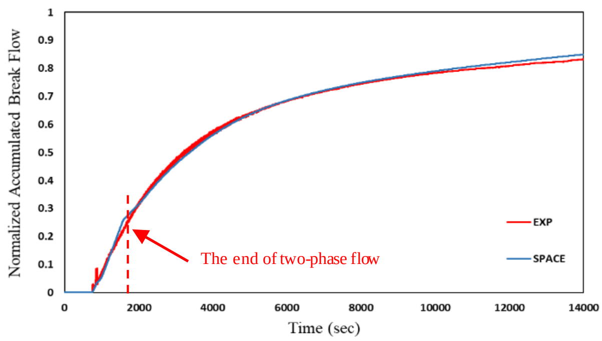

Fig. 4. Comparison of accumulated break flow

Fig. 4 shows a comparasion of accumulated break flow rate between SPACE code and the experiment. The SPACE code shows excelent prediction for the selected critical flow model and discharge coefficients.

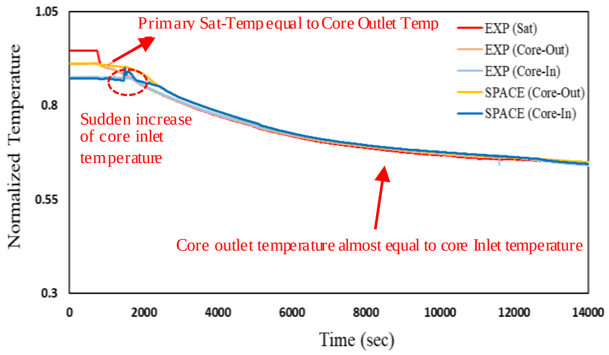

Fig. 5. Comparison of core inlet and outlet temperatures

Fig. 5 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. As the PRHRS actuated and the natural circulation established in the secondary side, the decay heat was removed continuously and the fluid temperature at the core inlet and outlet decreased gradually until the end of the accident. The SPACE code correctly predicts the overall fluid temperatures maintaining saturation condition.

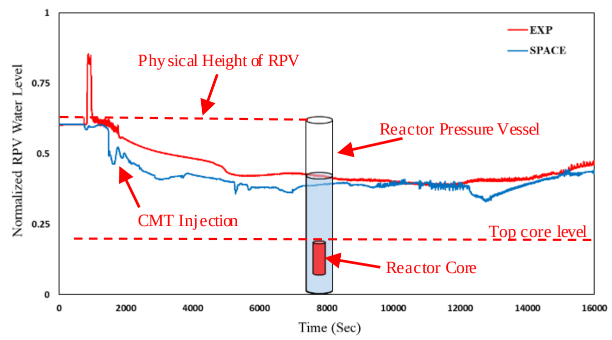

Fig. 6. Comparison of RPV collapsed water level

Fig. 6 shows the collapsed water level in the reactor pressure vessel. In the test, the water level raised suddenly after the break and then rapidly dropped. After that, the water level gradually decreased until it was stabilized by the actuation of CMT. However, during the sudden raise of the water level, which was measured based on the pressure difference, the measured value exceeds the actual height of the RPV. This error was a result of an improper mounting position of the instrument, which was installed in the break line of PSV. Because of the improper position, the sudden change of the dynamic pressure after the break resulted in a sudden reduction in the static pressure at the same position and thus a higher pressure difference and collapsed water level. The minimum collapsed water level was 6.2 m higher than the core top elevation. The SPACE code reasonably predicts the collapsed water level with slightly underprediction at the start of the transient due to the early discharge of two-phase flow.

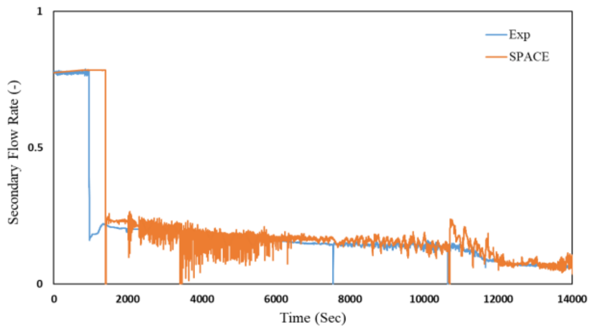

Fig. 7. Comparison of secondary flow rate

Fig. 7 represents the total flow rate in the secondary system. It is clearly shown that the normal feedwater flow rate was maintained before actuation of PRHRS. After reactor trip, PRHRS operation started and a stable natural circulation flow rate was established after few seconds. After that, a gradual decrease of the natural circulation with a constant rate was shown owing to the decay heat in the primary side. As shown from the graph, there was a delay in the actuation of the PRHRS of the code due to the delay of the reactor trip signal. However, the SPACE code properly predicts the overall natural circulation flow rate in the secondary system.

4. Conclusion

Validation of the SPACE code was performed using the test results of SB-PSIS-F301 at SMART-ITL facility. The validation results showed that the overall thermal-hydraulic behaviors such as the Accumulated break flow, Primary system temperatures, secondary flow rates, and the water level in the reactor pressure vessel were properly predicted. However, SPACE code underpredicted the depressurization rate of the primary side which resulted in a delay in the reactor trip.

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