Predictability Evaluation of SPACE for DVI Line Break Test with ATLAS

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1. Introduction

Based on the experience and the database for operating the nuclear power plant during several tens of years, the accident classification has been re-considered. Especially, there were many studies to consider realistically the loss of coolant accident in terms of the occurrence frequency. The nuclear industry and the regulatory organization in France have excluded the Large-Break Loss Of Coolant Accident (LBLOCA) in the Design Basis Accident (DBA) [1]. In Korea, the research project is on-going to exclude the LBLOCA in the DBA and simultaneously develop the safety analysis methodology for the Intermediate Break Loss Of Coolant Accident (IBLOCA) as the DBA instead of the LBLOCA. Therefore, the phenomenological understanding for the IBLOCA is needed and the phenomena identification and ranking table (PIRT) is now developing. As an effort to develop the IBLOCA PIRT, it is necessary to investigate the integral effect tests (IET) for the IBLOCA and validate the predictability of the safety analysis code for those tests. As one of the IET IBLOCA tests, the Korea Atomic Energy Research Institute (KAERI) conducted the DVI line break test (ATLAS B3.2) with the Advanced Thermal-Hydraulic Test Loop for Accident Simulation (ATLAS). The purpose of this study is to evaluate the predictability of the safety and performance analysis code for nuclear power plants (SPACE) for the ATLAS B3.2 test.

2. Description of ATLAS B3.2 test

The ATLAS B3.2 test simulated a DVI line break corresponding to 8.5-inch break in APR1400 [2]. To preserve similarity in a double-ended guillotine break of a DVI line of APR1400, the break area was designed to be 1.798×10^{-4} m^2 corresponding to 8% of the cold leg.

After the break initiation, the pressure of the reactor coolant system (RCS) was rapidly decreased and the safety injection pump (SIP) was actuated at the early period of the transient. Also, the main steam safety valve (MSSV) was actuated in the secondary system owing to feedwater and main steam isolations. The RCS pressure was continuously decreased and the injection of the safety injection tank (SIT) was initiated at the late period of the transient (~240 seconds after break initiation).

The core heat-up and quenching was observed at the early phase. The core was quenched via the residual coolant injection driven by the loop seal clearance (LSC).

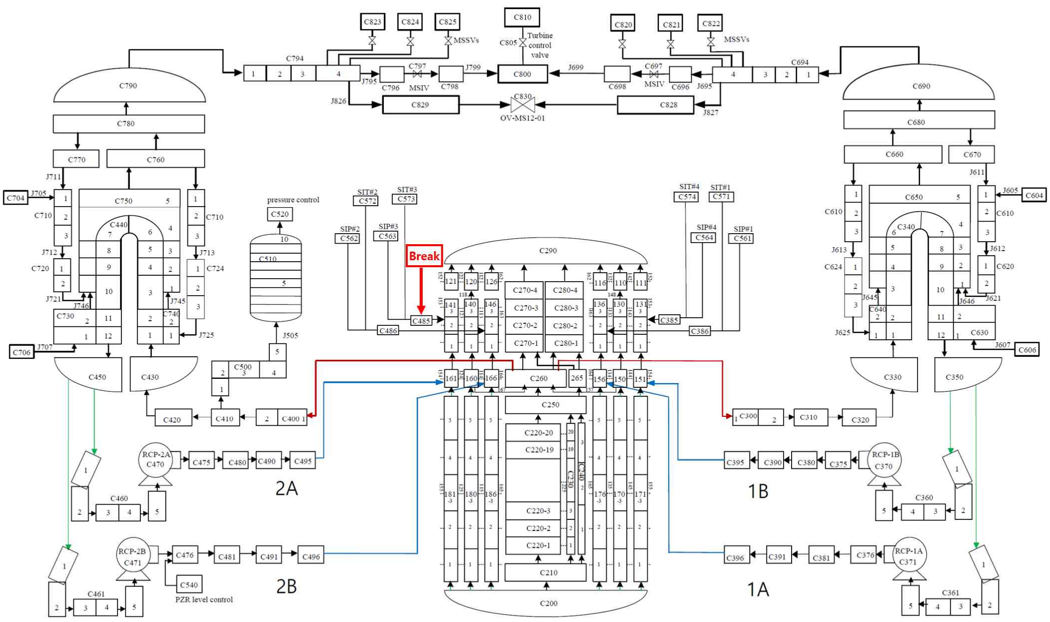

Fig. 1. Nodalization of the SPACE input model for ATLAS test facility

3. SPACE input model

The original ATLAS standard input was developed for the validation of the MARS code [3]. The ATLAS standard input for the MARS code was converted into that for SPACE. Recently, the ATLAS input model was modified for the update of main component such as design modifications of core and downcomer, the volume measurement of SG and etc. The nodalization of the ATLAS input model is shown in Fig. 1.

In the SPACE input model, a DVI break was simulated via connecting the time dependent volume having the atmospheric condition.

Using the SPACE input model described above, the steady state and the transient analyses were conducted for ATLAS B3.2. The detailed analysis results are explained in the section 4.

4. Results and discussion

Using SPACE, the steady state analysis was conducted and the initial condition for the ATLAS B3.2 test was obtained, as shown in Table I. The initial condition for the transient analysis was well agreed with the experimental conditions.

Based on the initial condition, a double-ended guillotine break of the DVI-3 line was simulated and the Henry-Fauske model was used with a discharge coefficient of 1.0 for the critical flow in the break location. In the ATLAS B3.2 test, the accumulated break flow was measured and compared with the SPACE analysis results, as shown in Fig. 2. The accumulated break flow of the SPACE analysis was well matched with that of the experimental results. Also, SPACE predicted well the break flow which is an inclination of accumulated amount in Fig. 2.

The RCS pressure during the transient is strongly dependent on the break flow. Consequently, the RCS pressure for SPACE analysis was quite well matched with the ATLAS B3.2 results, as shown in Fig. 3.

As shown in Fig. 4, the actuation time of a SIP and 3 SITs for the SPACE analysis was also well matched with the experimental results owing to the setpoints of the safety injections determined by the RCS pressure.

Table I: Steady state for B3.2 tests and SPACE analysis

System parameter	SPACE	ATLAS B3.2
Primary system		
Power (MWt)	1.641	1.641
PZR pressure (MPa)	15.54	15.54
Core T_{in} (K)	562.9	562.3
Core T_{out} (K)	598.9	601.3
Cold leg flowrate (kg/s)	1.95	1.95
Secondary system		
Steam flow rate (kg/s)	0.428/0.427	0.387/0.410
FW flow rate (kg/s)	0.427/0.427	0.428/0.426
Steam pressure (MPa)	7.85	7.85
Steam temperature (K)	566.9/566.9	569.5/569.1
SG level (m)	5.23/5.23	4.99/4.99

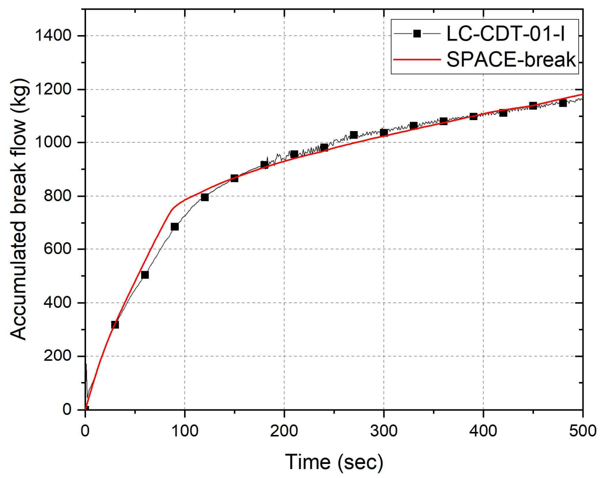

Fig.2. SPACE prediction of the accumulated break flow

Table II: Sequence of event for B3.2 tests and SPACE analysis

Event	Time (sec)		Remarks
	ATLAS B3.2	SPACE	
Break initiation	0	0	
Reactor trip	19	18	PT-PZR-01<10.72 MPa
Decay heat start	31	30	~12 sec after core trip
SIP initiation	48	46	Reactor trip + 28.28 sec delay
Minimum water level in the reactor core	85	236	
Loop seal clearing	90	223	
PCT	105	284	
Core quenching	110	375	
SIT injection	238	235	PT-DC-01<4.03 MPa
Activation of fluidic device	725/785/752	546/546/546	SIT level<2.0 m
Termination of SIT injection	4171/4520/4328	-	SIT level<0.1 m

Although SPACE predicted that the loop seal clearing was began in the same period with the ATLAS B3.2 test results, the clearing rate at the low water level in all intermediate legs was relatively slow in the SPACE analysis.

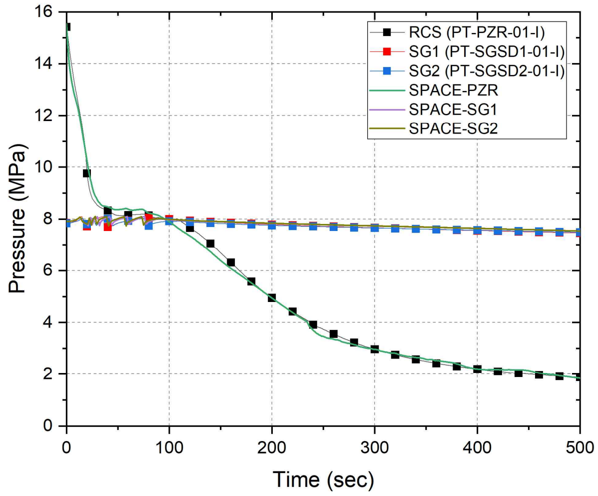

Fig. 3. SPACE prediction of system pressure

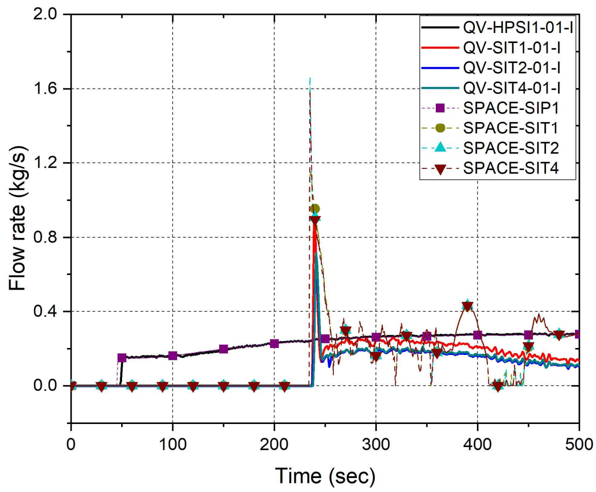

Fig. 4. SPACE prediction of safety injection

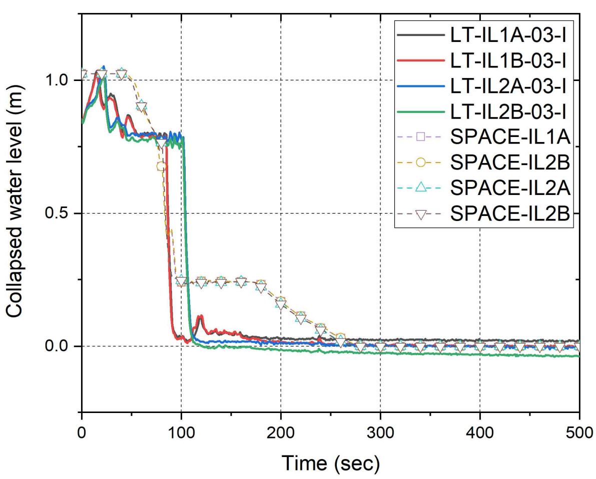

Fig. 5. SPACE prediction of LSC

The water level in the core is determined by the combined effect of the break flow, the safety injection and the LSC. The collapsed water level in the core is shown in Fig. 6. Right after the break initiation, the core water level was rapidly decreased owing to a large break flowrate. During the short period of the loop seal, the core water level was decreased near the bottom of the active core and the increase of the cladding temperature was observed as shown in the experimental results (Fig. 6 and 7). SPACE predicted well the decrease of the core water level due to the loop seal. However, the heat-up of the core was not appeared in the SPACE analysis due to less decrease of the core water level than the experimental results.

After the LSC, the core water level was increased due to the injection of the residual coolant driven by LSC. The core water level was decreased again owing to the boil-off. The decrease of the core water level was relatively small in the experimental results. However, SPACE has much overestimated the water level decrease of the core. Depending on those behaviors of the core water level due to the boil-off, the heat-up of core was appeared only in the SPACE analysis results.

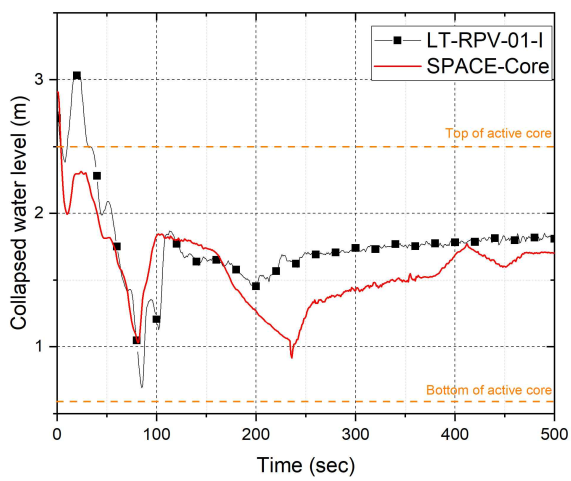

Fig. 6. SPACE prediction of core collapsed level

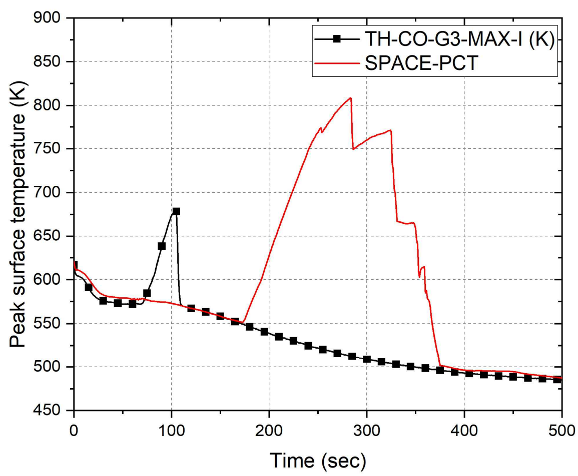

Fig. 7. SPACE prediction of peak surface temperature

5. Conclusion

To investigate the IBLOCA characteristic and evaluate the predictability of SPACE, the SPACE analysis was conducted for the ATLAS B3.2 test which simulated a DVI line break.

The main system parameters such as the break flow, the system pressure and the safety injection were well predicted in the SPACE analysis results. However, there is a discrepancy for the local distribution of the coolant in the RCS between the SPACE analysis results and the ATLAS B3.2 test results. Especially, the minimum water level in the active core due to the loop seal was different and the core heat-up observed in the experiment during the loop seal period was not shown in the SPACE analysis. Moreover, the core heat-up due to the boil-off predicted by SPACE was not observed in the ATLAS B3.2 test results.

Those mismatching of the PCT occurrence may result from the local coolant distribution in the RCS which can be affected by the counter current flow limit between the core and the upper plenum, the flow resistance in the downcomer and etc. The further study for the effect of those phenomena on the PCT is necessary in the future.

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