Analysis of SBLOCA for CRDM Nozzle Rupture with LSI at the ATLAS Experimental Facility using the MARS-KS 1.5

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1. Introduction

Korea Atomic Energy Research Institute (KAERI) has operated an integral effect test facility, the Advanced Thermal-Hydraulic Test Loop for Accident Simulation (ATLAS), with reference to the APR1400 (Advanced Power Reactor 1400) for transient and design basis accident (DBA) simulations as shown in Fig. 1 [1]. In addition, KAERI has operated the domestic standard problem (DSP) program using the experimental data from the selected experiments at ATLAS in order to encourage the verification and validation of system codes. The sixth DSP (DSP-06) aims at evaluating the importance of the accident management (AM) action during a small-break loss-of-coolant-accident (SBLOCA) with loss of safety injection (LSI). The rupture of a control rod drive mechanism (CRDM) nozzle at the upper head of the reactor pressure vessel (RPV) was selected as a postulated accident for DSP-06.

In this study, the analysis of the SBLOCA with LSI has been conducted by using MARS-KS 1.5 [2] and an improved model from the reference steady-state input distributed by KAERI [3].

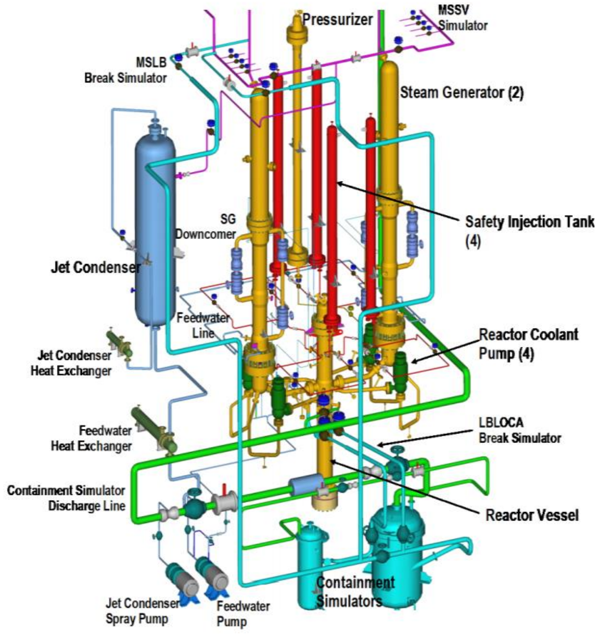

Fig.1. Schematic diagram of ATLAS facility [1]

2. Test Condition

In the SBLOCA test with the CRDM nozzle rupture, four safety injection tanks (SITs) were utilized as a safety injection system during the test period. However, four safety injection pumps (SIPs) were not actuated to consider the total failure of SIP. The safety systems in the secondary system such as the main steam safety valves (MSSVs) and auxiliary feedwater (AFW) system were assumed to be available. Also, the atmospheric dump valve (ADV) was implemented for the AM action, and it opened for 50% when the surface temperature of the heater rods in the core is higher than 623.15K. The initial heater power was controlled to be 1.664MW, and the decay heat was modeled by using the ANS-73 curve with a multiplier of 1.2. Detailed test condition information of DSP-06 can be found in the reference [4].

3. Modeling Information

The reference input for steady-state of ATLAS distributed by KAERI has been modified on the basis of the facility design report in order to have the correct geometry and boundary conditions. Especially, a new heat loss correlation for the secondary system was suggested by fitting the result of the heat loss tests.

3.1 Modified ATLAS Model

The detailed review of the reference model performed with reference to the actual geometry of ATLAS indicated that several geometry information in the reference model should be corrected. The corrections were implemented to the hydraulic volume of the downcomer in the RPV, heat structure of the primary piping, and heat structure of the downcomer and economizer in each steam generator (SG), as highlighted in red in Fig. 2.

3.2 New Correlation for Secondary Heat loss

Heat loss from the primary side was defined by the sum of heat loss from the RPV and primary piping. In case of the secondary side, the heat loss from the outer shells of the SGs and main steam lines were considered. The heat loss could be estimated as a function of the wall temperature, which had been developed from the separated heat loss test [1]. However, it was found that the heat loss estimated by the correlation for the secondary system given in reference showed a large difference from the experimental result when the temperature of the secondary system was high. Thus, a new curve with 4th-order polynomial has been developed by fitting the data from the heat loss tests. The new correlation follows the general trend of the heat loss successfully and has higher adjusted R^2 (0.99993), as depicted in Fig. 3.

3.3 Break line modeling

In order to simulate the CRDM nozzle rupture, the break system was implemented on the top of the upper head of the RPV. So as to predict the behavior of break flow against the experiment data correctly, the break system should describe the test configuration as realistic as possible. The break system consisted of a break nozzle, a break valve, sink volume, and break pipes. Among those components, the modeling of the break nozzle is very important in this simulation since chocking in the break line occurs at the smallest area section. In this break line, the break nozzle has the smallest inner diameter of 7.12mm, whereas the inner diameter of the break pipe line is 33.99mm. Thus, the Henry-Fauske critical flow model [5], the default model of MARS-KS, was applied to the break nozzle.

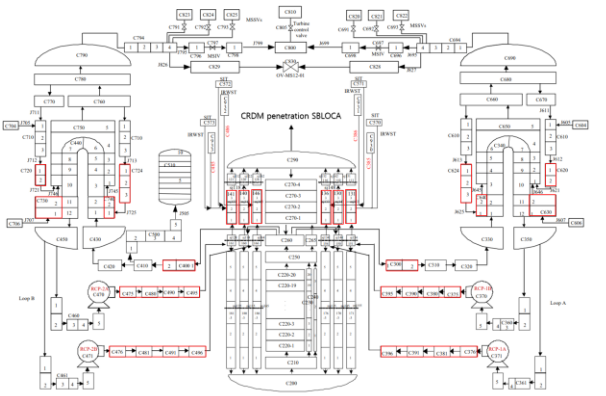

Fig.2. Nodalization of ATLAS facility

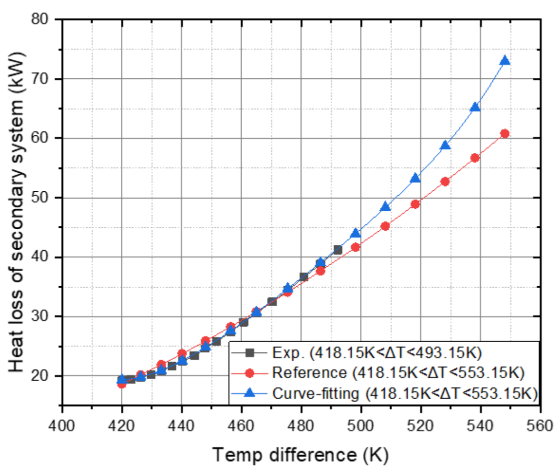

Fig. 3. Comparison of secondary system heat loss

4. Analysis Results

4.1 Steady-state Calculation

A steady-state calculation has been conducted for 5000 sec in problem time to achieve the initial conditions for the postulated accident. The results of the steady-state calculation summarized in Table I. All major parameters except for the SG pressure were well predicted within the error bands of the experimental values. The secondary system parameters indicated that the saturation pressure corresponding to the steam temperature was different from the measured SG pressure. The preliminary analysis confirmed that the utilization of the SG pressure as a parameter for the secondary control prevented the system from reaching the desired steady-state conditions. Thus, it was decided to achieve the steady-state conditions of the secondary system based on the SG temperature. The resulted SG pressure was exactly same as the saturation pressure corresponding to the steam temperature of each SG and all system parameters were predicted within acceptable error range, as aforementioned. The steady-state results for the heat loss also confirmed that the new correlation developed in this study predicted the heat loss appropriately.

Table I: Steady-state calculation results

Parameter	Exp.	Cal.	Error [%]
Primary System			
Core power [MW]	1.66	1.66	0.00
Heat loss [kW]	98.4	98.0	-0.41
PZR pressure [MPa]	15.5	15.5	0.00
PZR level [m]	3.62	3.62	0.00
Core inlet temp. [K]	565.35	564.45	-0.16
Core outlet temp. [K]	600.95	600.95	0.00
Secondary System			
Feed water flow rate [kg/s]	SG 1:0.410 SG 2:0.420	SG 1:0.416 SG 2:0.416	SG 1:1.46 SG 2:-0.95
Feed water temp. [K]	506.45	506.45	0.00
Steam pressure [MPa]	7.83	8.0795	3.18
Steam temp. [K]	SG 1:569.35 SG 2:568.35	SG 1:568.85 SG 2:568.85	SG 1:-0.09 SG 2:0.09
SG level [m]	4.99	4.99	0.00
Heat loss [kW]	70.0	69.9	0.01
Primary Piping			
Cold leg flow [kg/s]	2.0	1.9114	-0.09

4.2 Transient-state Calculation

An analysis of the CRDM nozzle rupture with LSI has been conducted from the steady-state conditions. Table II shows the sequence of major events occurred during the accident, comparing the results from experiment and transient calculation. MARS-KS predicted the overall trends of the major sequence observed in the ATLAS test successfully.

Table II: Chronology of the transient main events

Event	Exp.(s)	Cal. (s)	Remarks
Break	0	0	@t=0
LPP (Rx, RCP trip)	68	66	PZR P < 10.72MPa
MSIS	72	70	LPP +3.54s delay
MFIS	75	73	LPP+7.07s delay
Decay Heat	80	78	LPP+12.07s delay
AM Action	2181	2206	PCT > 623.15K
ADV Open	2181	2206	ADV 50% open
SIT Injection	2301	2279	D.C P < 4.03MPa
SIT FD (Low flow)	2671	2600	SIT Level < 2.0m
SIT Termination	2998	2987	SIT Level < 0.1m
AFW Injection	2648	2505	SG Level < 25%
AFW Stop	4565	4424	SG Level > 40%

The CRDM penetration nozzle break accident is initiated by opening the break valve at 0.0 seconds. At the beginning of the accident, the primary system is rapidly depressurized and the reactor trip signals are generated by the low pressurizer pressure (LPP). Also, the main steam isolation signal (MSIS) and main feedwater isolation signal (MFIS) occurs in some delay time after the LPP signal. The decay heat curve is implemented by using measured power described in the test specifications and it is activated with a delay of 12.07 seconds from the reactor trip considering the scaled nominal power of ATLAS, as shown in Fig. 4. The behavior of the primary pressure is depicted in Fig. 5. During the initial rapid depressurization, the safety injection should be started when the setpoint of the SIP is reached. However, the SIP does not operate in this test because of loss of safety injection. Thus, the primary system keeps depressurized slowly until the ADV is opened.

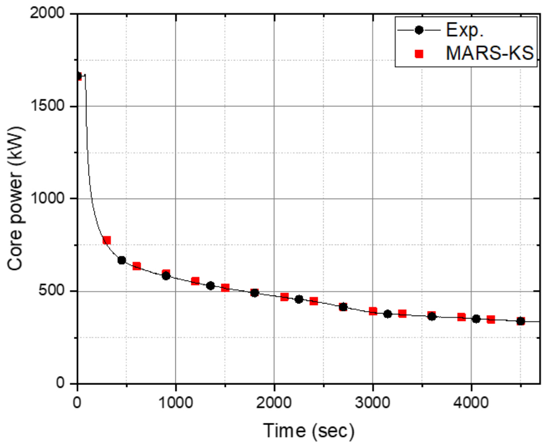

Fig.4. Power of core heater

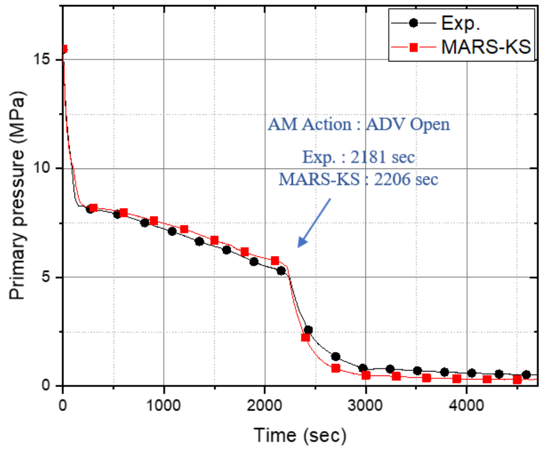

Fig.5. Pressure of primary system

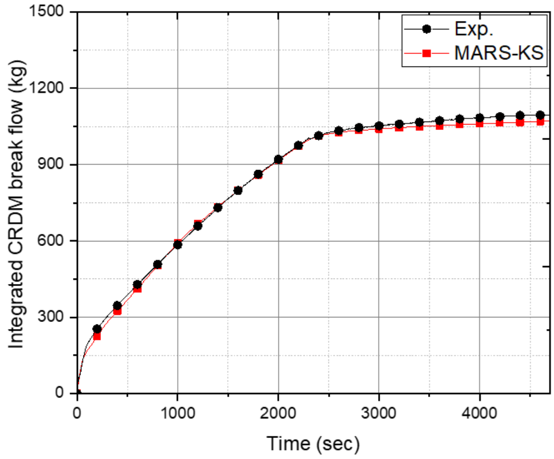

Fig.6. Integrated break flow

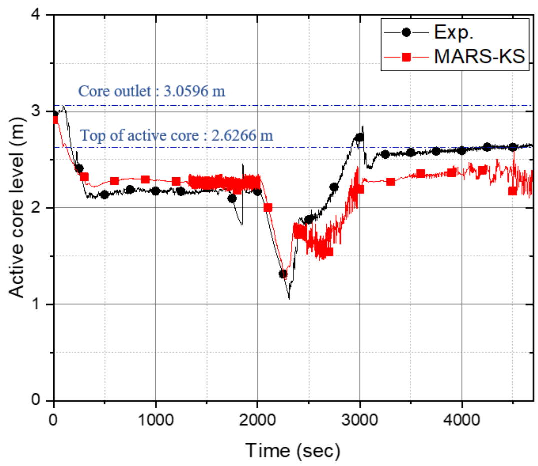

Fig.7. Active core collapsed water level

Figure 6 shows the behavior of the integrated break flow through the break valve. During the initial blowdown phase, single phase liquid is expelled through the break line. Afterwards, a less steep depressurization region is formed both experiment and calculation up to approximately 2200 seconds because of the two-phase flow at the break system. Subsequently, void fraction of the break modules increased and the break flow switch from two phase flow to single phase vapor. In order to predict the critical flow at break nozzle, the Henry-Fauske critical flow model with a discharge coefficient of 0.8 was applied to the break nozzle based on the preliminary analysis results.

Figures 7 and 8 show the active core collapsed water level and peak cladding temperature (PCT), respectively. It was found that the behavior of overall level and temperature in the calculation were similar to those in the experiment. When the PCT exceeds 623.15K, the AM action to open 50% of ADV is performed to increase cooling by the secondary system. Thus, the pressure of the primary system is decreased, and the SIT is actuated when the primary system is depressurized to the setpoint as shown in Fig. 9. It is confirmed that the core collapsed water level was recovered and the PCT was stabilized after the injection of the SIT in both experiment and calculation.

Figure 10 shows the SG collapsed water level and AFW mass flow rate. The AFW system was activated when water level of the SG decreased since the SG ADV opening and reached 25% of SG level. Also, hysteresis trip logic has been controlled to stop injection of the AFW if the SG level exceeded 40%.

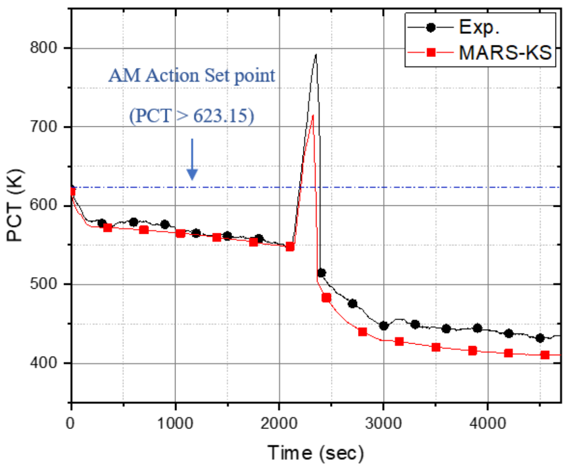

Fig.8. Peak cladding temperature

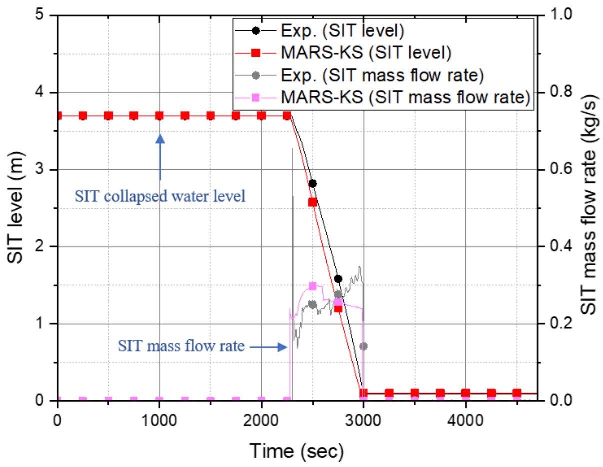

Fig.9. SIT collapsed water level and mass flow rate

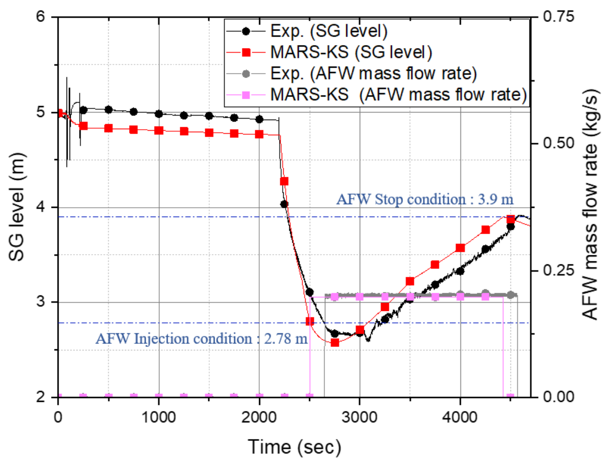

Fig.10. SG collapsed water level and AFW mass flow rate

5. Conclusion

The ATLAS test for the CRDM nozzle rupture SBLOCA was analyzed with the improve model using the MARS-KS 1.5 code in order to resolving the safety issue on the multiple failure accident and AM action. The objectives of this study also include the investigation of the predictability of the system code for the AM action reported by the experiment. The analysis results indicated that the impact of AM action and SIT injection were reasonably predicted by the system code.

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