Analysis of Reactor Pressure Vessel Upper Head SBLOCA+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 been operating 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 experiments for transient and design basis accidents (DBAs) simulation as shown in Figure 1 [1]. In addition, KAERI has operating the domestic standard problem (DSP) program based on the experimental data from the selected experiments in order to encourage the verification and validation of system codes. Recently sixth phase of the DSP (DSP-06) blind calculation had been proceeded, the DSP-06 aims at evaluating the physical behavior during a small break loss of coolant accident with the loss of safety injections (SBLOCA +LSI) using various system codes. In this study, SBLOCA analysis is performed using MARS-KS 1.5 [2] with improved model.

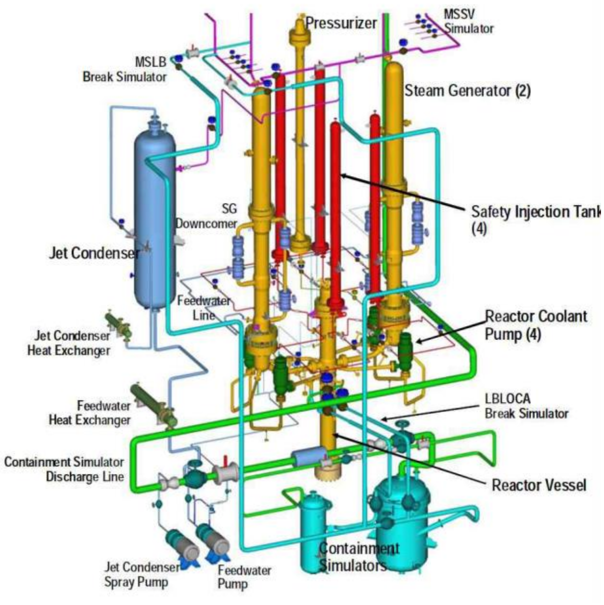

Fig.1. Schematic diagram of ATLAS facility

2. Modeling Information

In this section introduces the modified model of ATLAS based on the design drawing of the technical report reflect the correct geometry information as the actual. In addition, a new heat loss correlation is suggested by fitting the result of heat loss tests, break line modeling is introduced briefly.

2.1 Modified ATLAS Input Model

To reflect the real design data for the ATLAS, the following geometry information were modified: 1) hydraulic volume of the downcomer in the reactor pressure vessel (RPV), 2) heat structure of the primary piping, 3) heat structure of the downcomer and economizer in each steam generator (SG). Figure 2 illustrated the MARS-KS nodalization of the ATLAS facility, and the components highlighted in red are modified ones.

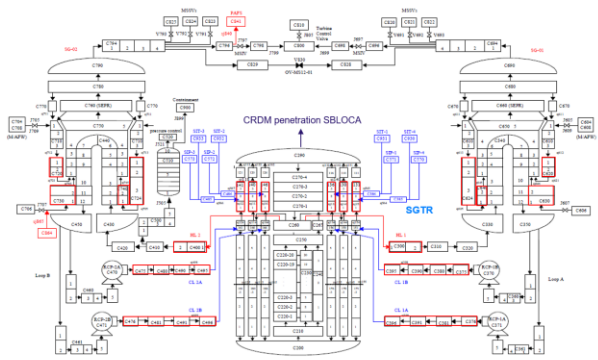

Fig.2. Nodalization of ATLAS facility

2.2 New Correlation for Secondary Heat loss

The results of calculation using the heat loss correlation for secondary system presented in the technical report [1], indicated that a large difference in heat loss from the target value was obtained. It is because the correlation presented in the report does not represent the general behavior of the heat loss well as the temperature difference between the wall and atmosphere increases. As a result, a lager deviation from the expected heat loss at the normal operation conditions of ATLAS is obtained with the same temperature difference. In order to improve the correlation, a new curve with 4^{th}-order polynomial was developed by fitting the data from heat loss tests. The new correlation followings the general trend of the heat loss successfully and has higher adjusted R^2 than the one from the original correlation, as depicted in Figure 3.

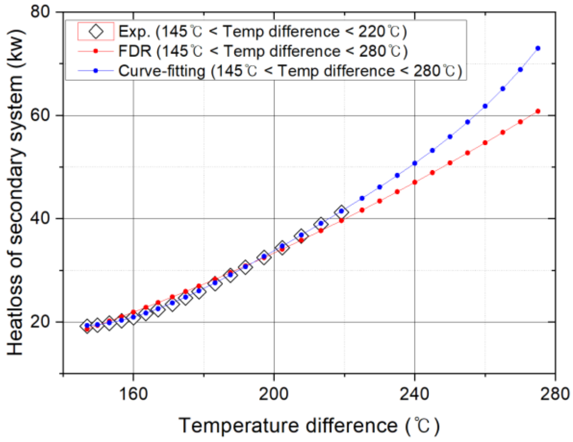

Fig. 3. Comparison of secondary system heat loss

2.3 Break line modeling

The break modules was implemented in ATLAS RPV upper head to simulate the control rod drive mechanism (CRDM) penetration during SBLOCA. The break modules consisted of break nozzle (I.D = 7.12mm), break valve, and break line. It is implemented using the pipe and valve components based on the design specifications. The critical flow model applied was Henry-Fauske option, minor loss of break line was referencing Crane [3].

3. SBLOCA Analysis of ATLAS

This chapter explains that analysis of SBLOCA with LSI at RPV upper head using the improved model.

3.1 Steady-state Calculation

Steady state calculation was conducted for 5000sec. Table II shows the steady state calculation results. Overall, there is no significant difference from the experimental data, except cold leg mass flow rate and SG pressure. The reason for this difference comes from the followings: 1) The lower core inlet temperature was predicted when the SG pressure was targeted to the experimental value. 2) In order to control the core inlet temperature, the SG pressure was controlled on the basis of the steam temperature. 3) The cold leg mass flow rate was calculated lower than the experimental value, because the temperature difference between the core inlet and outlet was lager than the experimental value.

3.2 Transient-state Calculation

Additional transient calculation for 5000sec was performed from the steady state, and the accident was initiated by opening break valve in the RPV upper head. At the beginning of the accident, the pressure of primary system decreases rapidly as shown in Figure 4, and generates pressurizer low pressure signal (LPP signal), reactor trip. Also, the main steam isolation signal (MSIS), main feedwater isolation signal (MFIS) and decay heat signal occurs in some delayed time after the LPP signal. The decay heat curve was implemented by using measured power of test specification as shown in Figure 5. Figure 6 shows break line mass flow rate. During the initial blowdown phase, single phase liquid is expelled through the break line. Afterwards, the plateau region is formed because of the two phase flow of break modules, then depressurization became less steep.

Table II: Comparison of steady state conditions with experimental and MARS-KS 1.5 code calculation

Parameter	Exp.	Case I	Error (%)
Primary System			
Core Power (MW)	1.66	1.666	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
Cold leg flow rate (kg/s)	2.0	1.9114	-0.09
Core inlet Temp (K)	565.35	564.45	-0.16
Core outlet Temp (K)	600.95	600.95	0.00
Secondary System			
Steam flow rate (kg/s)	0.415	0.416	0.24
Feedwater flow rate (kg/s)	0.435	0.416	-4.37
SG Pressure (MPa)	7.83	8.0795	3.18
Steam Temp (K)	568.85	568.85	0.00
SG water level (m)	4.99	4.99	0.00
Heat loss (kw)	70.0	69.9	-0.14

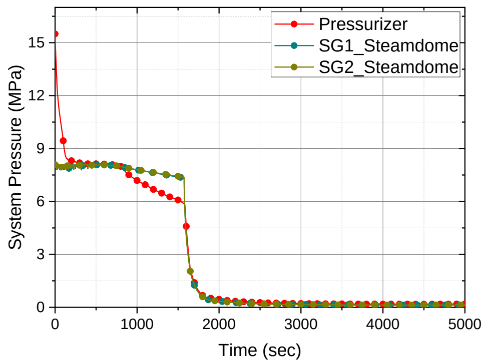

Fig.4. Pressure behavior of Primary and Secondary System

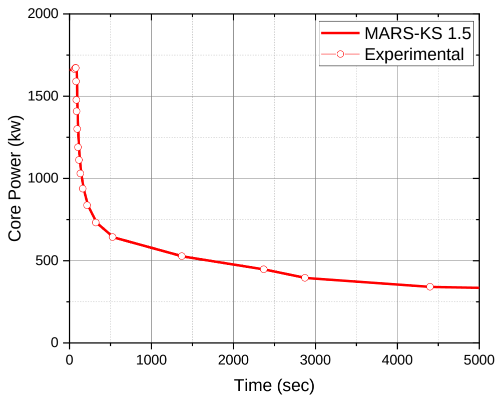

Fig.5. Core decay heat

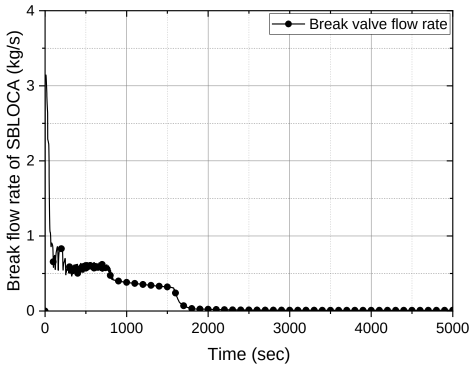

Fig.6. Mass flow rate of Break nozzle

Subsequently, the break modules void fraction increased and the break flow switch from two phase flow to single phase vapor the results of the break mass flow was significantly reduced.

The pressure in primary system decreased due to continuous break flow, safety injection pump does not operate assuming loss of safety injection. After that, when the core collapsed water level continues to decrease (Fig.7) and the peak cladding temperature (PCT) steeply exceeds 623.15K (Fig.8), the atmospheric dump valve (ADV) in the SG was opened by operator action (Fig.9) and depressurization of the primary system occurred, and the safety injection tank (SIT) was injected.

Figure 10 shows the SIT mass flow rate. In order to simulate fluidic device (FD), the flow control valve (FCV) was controlled using the valve stem position rate. After injection of SIT, it was confirmed that the core collapsed water level was recovered and the PCT was stabilized. Approximately 3500 sec later at core water level oscillation was occurred, it was confirmed that boiling occurs because temperature at the core outlet is saturated.

Figures 11 ,12 show the SG water level and auxiliary feed water (AFW) 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%. Then, until end of the calculation, the accident is terminated without further rise of peak cladding temperature and the chronology of SBLOCA is shown in Table. II.

4. Sensitivity Analysis of Discharge coefficient

This chapter explains that sensitivity analysis of discharge coefficient break nozzle. In general, the Henry-Fauske option is used as the critical flow model at NPPs, however ATLAS is an experimental facility with a volume ratio of 1/288 of the APR1400 nuclear power plant, so an appropriate discharge coefficient must be applied. In addition, considering that it is the blind calculation, a sensitivity analysis of according to the discharge coefficient was required for accurate prediction since the experimental data of integrated discharge break mass flow was not disclosed.

As the discharge coefficient decreases, the break flow of the primary system reduces slowly, and the depressurization is also delayed. Because of the effect, the reduction of the collapsed water level in the core and the occurrence of PCT are delayed, the operation of safety system is also delayed. The overall sequence of event is summarized in Table III.

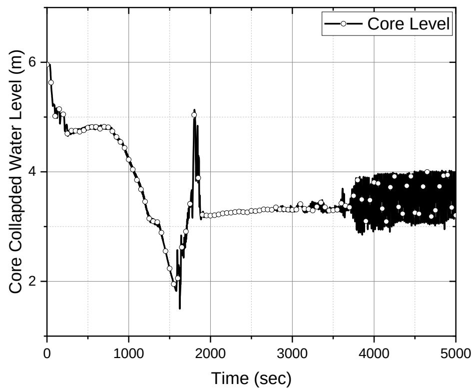

Fig.7. Core collapsed water level

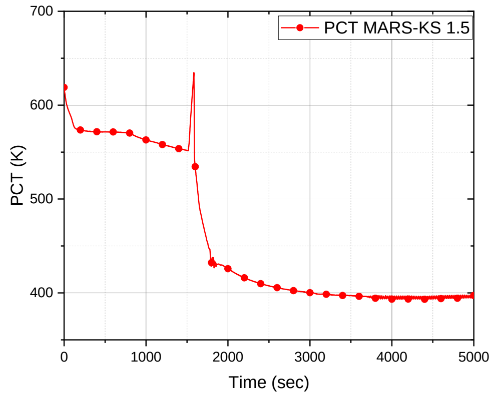

Fig.8. Peak Cladding Temperature

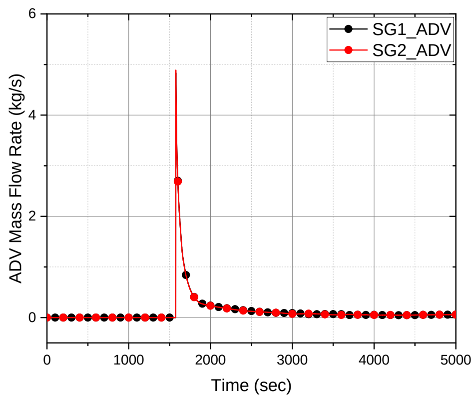

Fig.9. Mass flow rate of SG ADV

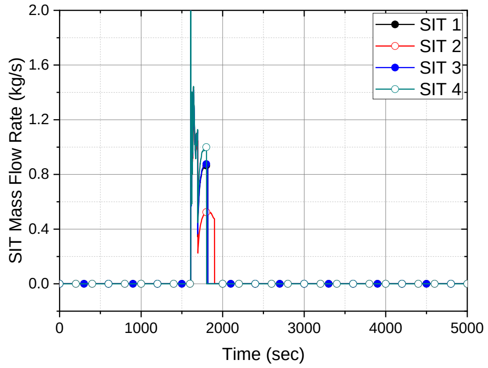

Fig.10. Mass flow rate of SIT

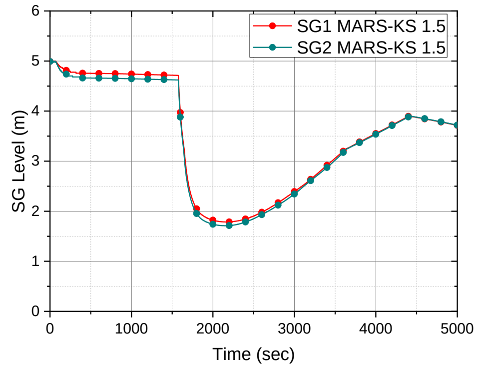

Fig.11. Leve of steam generator

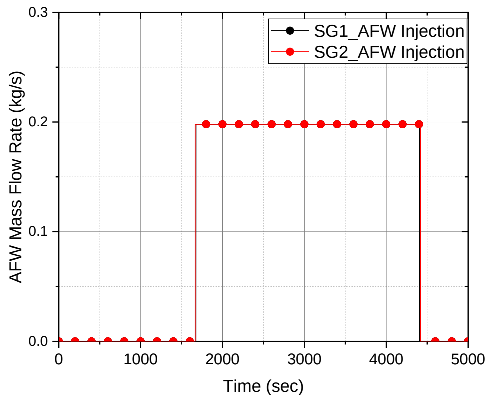

Fig.12. Mass flow rate of Auxiliary feedwater System

Table II: : Chronology of SBLOCA

Event	Set - point	Cal. (s)
Break	@t=0	0.0
LPP (Rx, RCP trip)	PZR P < 10.72MPa	61.8
MSIS	LPP +3.54s delay	65.4
MFIS	LPP+7.07s delay	68.9
Decay Heat start	LPP+12.07	73.87
SIP Injection	Not available	-
AM action	PCT > 623.15K	1574.7
ADV open		
SIT Injection	D.C P < 4.03MPa	1608.8
SIT_(Low Flow)	SIT Level < 2.0m	1693.0
SIT_Termination	SIT Level < 0.1m	1818/1901/ 1818/1803
AFW Injection start	SG Level < 25%	1676 / 1666
AFW Injection stop	SG Level > 40%	4405 / 4418

Table III: Summary table of sensitivity analysis discharge coefficient

Event	Time (sec)			
Break	Cd = 1.0	Cd = 0.9	Cd = 0.8	Cd = 0.7
	0.0	0.0	0.0	0.0
LPP Signal	61.8	67.9	73.2	82.6
MSIS	65.4	71.4	76.7	86.2
MFIS	68.9	75.0	80.3	89.7
Decay Heat Start	73.9	80.0	85.3	94.7
AM action	1574.7	1602.2	1834.3	2143.5
ADV open	1574.7	1602.2	1834.3	2143.5
SIT Start	1608.8	1636.1	1868.5	2178.0
SIT_FD (low flow)	1693	1720	1953	2262
SIT_Stop	1819/1901 1818/1803	1844/1925 1842/1827	2077/2157 2075/2060	2385/2466/ 2384/2369
AFW start	1676/ 1666	1704/1694	1933/1924	2241/2229
AFW stop	4405/ 4418	4422/4433	4638/4612	4933/4898

5. Conclusion and Further Work

This study presents the analysis result of the SBLOCA+ LPI, one of the multiple failures accident by using MARS-KS 1.5. In order to reflect the experimental conditions realistically, input was improved, and a newly secondary system heat loss correlation was introduced to complete the steady state calculation. In addition, the components necessary to simulate the transient state were modeled based on the test specifications. From the transient analysis, it was confirmed that the primary system was effectively cooled using the SG-ADV opening and AFW injection for 5,000 sec. In addition, the core water level was recovered by SIT operation. Also, from the sensitivity analysis, it was confirmed that the opening time of SG-ADV, SIT depends on the break flow. In open calculation, the system behavior will be evaluated with correct system inventory using break flow data from the experiment data. In addition, a sensitivity analysis for the RPV upper head node size will be performed to confirm the effect of the void fraction the at break line.

REFERENCES

[1] KAERI/TR-8106/2020, Description report of ATLAS facility and instrumentation, 2018.
[2] Korea Institute of Nuclear Safety(KINS), MARS-KS Code Manual Volume I: Theory Manual, KINS/RR-1882 Vol.1, 2018
[3] Crane Co., Flow of Fluids through Valves, Fittings, and Pipe, Metric Edition – SI Units, 1999.