Experimental investigation of thermal-hydraulic behavior and safety performance of reactor coolant system and containment during a steam line break using ATLAS-CUBE facility

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1. Introduction

The safety and integrity of nuclear power plants during accidents, such as the events at Fukushima Daiichi in 2011 and Three Mile Island in 1979, is of paramount importance in order to prevent catastrophic consequences and protect both the environment and public health. A key aspect of ensuring nuclear safety is understanding the thermal-hydraulic behavior of reactor coolant systems (RCS) and a containment during accidents. The Korea Atomic Energy Research Institute (KAERI) has recently conducted the SLB-CT-05 test using the ATLAS-CUBE facility, which links the Advanced Thermal-hydraulic Test Loop for Accident Simulation (ATLAS) and the Containment Utility for Best-estimate Evaluation (CUBE). This study presents the results of the SLB-CT-05 test, which focuses on the safety-critical role of the containment during an upward steam line break (SLB) with a total failure of the safety injection pump (SIP) and a 1-train failure of the containment spray system. This experimental setup allows for a realistic simulation of the complex, multi-dimensional thermal-hydraulic behavior inside a containment during an SLB accident. The SLB-CT-05 test aimed to generate a test database (DB) to evaluate containment pressure and temperature (P/T) buildup and validate safety analysis models [1,2].

The results of the SLB-CT-05 test provide valuable insights into the transient behavior of the RCS resulting from the containment's thermal-hydraulic behavior. These insights can contribute to the development of more accurate integrated analysis frameworks for the RCS and containment, considering multi-dimensional or multi-compartment models. Moreover, the experimental data can help regulatory bodies and industries better understand the containment's overpressure prevention performance during accidents and develop more effective strategies to ensure the safety and integrity of nuclear power plants.

2. Descriptions for the test

2.1 ATLAS-CUBE facility

The ATLAS-CUBE test facility is an advanced, integrated experimental platform designed to investigate the thermal-hydraulic behavior and safety performance of RCS and containment during various accident scenarios. This facility combines ATLAS and CUBE facility to enable comprehensive, large-scale testing of reactor safety systems.

The ATLAS facility is a versatile test loop designed to simulate the RCS of a pressurized water reactor (PWR), providing a scaled representation of key components such as the reactor pressure vessel, steam generators, pressurizer, safety injection systems, and reactor coolant pumps. The CUBE facility is an innovative containment simulation platform that complements ATLAS facility by enabling the investigation of thermal-hydraulic behavior within the containment building. CUBE features a large, cylindrical structure that represents a scaled-down containment building, equipped with various measurement instruments and compartments [3].

2.2 SLB-CT-05 test conditions

In the SLB-CT-05 test, the ATLAS facility achieved steady-state conditions, including a core power of 1.62 MW (representing 8% of the scaled-down power), a pressure of 15.5 MPa, and core inlet and outlet temperatures of 292.8 °C and 326.9 °C, respectively. The test simulated an SLB accident by utilizing the break simulation device on the steam generator # 1 (SG-1).

The break mass was discharged to the CUBE facility through a flow restrictor and a break connection pipe, as illustrated in Fig. 1. The flow restrictor's narrowest diameter was 38.6 mm, which achieved a choking condition.

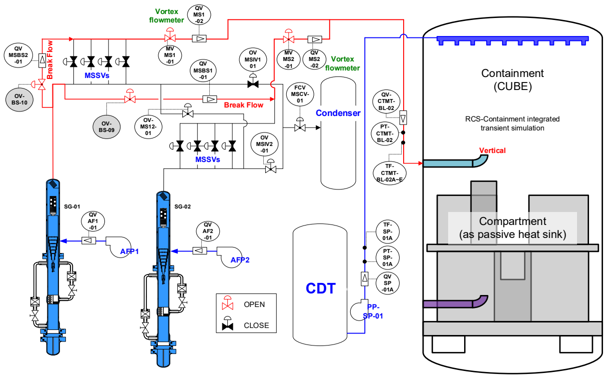

Fig. 1 A schematic diagram for the ATLAS-CUBE integrated test

Table I: Sequence of events and set-points on the SLB-CT-05 test

No	Event	Set-points	Time (-)
1	SLB Start	Break valve open (OV-BS-09, 10)	0.60
2	HCP signal	PT-CTMT-01 > 0.028	0.61
3	SCRAM	Coincidence with HCP signal	0.61
4	Reactor trip / Rx trip	Coincidence with SCRAM	0.61
5	MSCV close	SCRAM + 0.14 s delay	0.61
6	MSIS	SCRAM + 3.54 s delay	0.62
7	MFIV close	SCRAM + 7.08 s delay	0.63
8	Decay Power Start	Reactor trip + 12.07 s delay	0.64
9	Total failure of SIP	LPP (PT-PZR-01 < 2.68)	N/A
10	AFAS signal	LT-SGSDRS1/2-01 < 3.475	0.64
11	Auxiliary feedwater injection start	AFAS + 43.45 s delay	0.73
12	Auxiliary feedwater injection stop	LT-SGSDRS1/2-01 > 4.875	21.35
13	Containment spray system actuation	H-HCP (PT-CTMT-01 > 0.059)	4.82
14	SIT injection start	PT-DC-01 < 1.008	17.61
15	Containment spray system failure	H-HCP + 9.6 normalized time	14.24
16	End of test		24.00

HCP: High Containment Pressure
H-HCP: High-High Containment Pressure
LPP: Low Pressurizer Pressure
MSCV: Main Steam Control Valve
MSIS: Main Steam Isolation Signal
MFIV: Main Feedwater Isolation Valve
AFAS: Auxiliary Feedwater Actuation System

The main scenarios of the test, along with their respective set-point values, are presented in Table I. Taking into account the confidentiality concerns related to the test data, all results in this manuscript have been normalized using an arbitrary value.

3. Experimental results

During the SLB simulation, the coolant from broken SG-1 was discharged into the containment and caused rapid depressurization in both SG-1 and the primary system. The primary system became pressurized between 0.98 to 1.55 normalized time (t*), as core

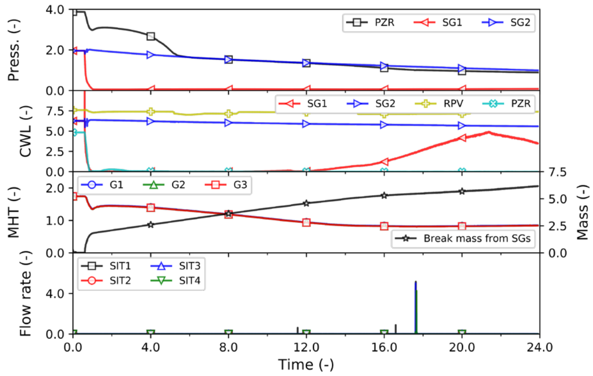

Fig. 2 Comparison of RCS thermal-hydraulics in the SLB-CT-05 test

decay heat exceeded the cooling provided by auxiliary feedwater and intact steam generator 2 (SG-2).

Intact SG-2 experienced pressurization immediately after the break, with pressure gradually decreasing after main steam safety valve (MSSV) operation. As decay heat reduced, the primary system pressure converged with the SG-2 pressure. Following the rapid discharge of coolant from SG-1 to the containment, from t* =1.2 onward, the steam was generated in SG-1 equivalent to the supplied auxiliary feedwater and transported to the containment. Subsequently, at t* = 12, the auxiliary feedwater supplied to SG-1 was not fully vaporized, leading to an increase in the water level in SG-1 and a decrease in the break mass transferred from SG-1 to the containment.

Figure 2 displays the major thermal-hydraulic behavior in the RCS during the SLB-CT-05 test. The SLB resulted in overcooling and a decrease in primary system pressure. Despite the total failure of the SIP, cooling was maintained due to the overcooling caused by the SLB and the cooling effect of auxiliary feed water injection into the broken SG-1. Although the pressure decreased to the set-point pressure for actuation of safety injection tank (SIT), minimal water injection from the SIT occurred due to the lack of pressure difference between the SIT and the primary system.

Figure 3 illustrates the thermal-hydraulic behavior within the containment during the SLB-CT-05 test. The first row of Fig. 3 presents the fluid temperature field inside the containment. Following the break, before the containment spray system operation, the upward steam supply occurred above the internal compartments, resulting in a significant temperature difference between the fluid in the upper dome and the lower region, leading to thermal stratification. The second row of Fig. 3 shows the containment vessel wall temperature (TW_UDOME) at the upper dome and the wall temperatures of the internal compartments in SG-1, the primary shield wall (PSW), and the secondary shield

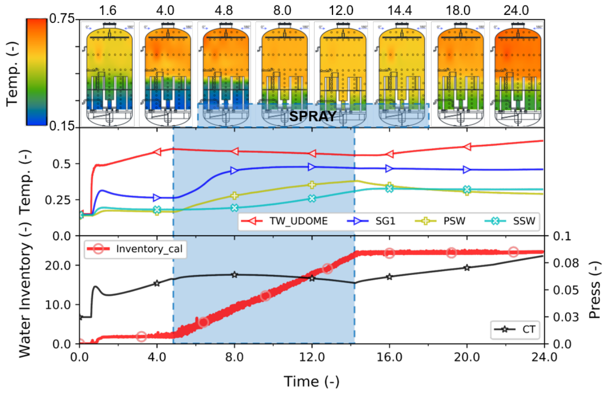

Fig. 3 Comparison of containment thermal-hydraulics in the SLB-CT-05 test

wall (SSW). Before the operation of containment spray system, there was a significant temperature difference between the upper dome and the internal compartment wall temperatures, similar to the fluid temperature. Upon the activation of the containment spray system, however, the wall temperatures of the lower internal compartments increased due to thermal mixing caused by the spray droplets. The last row of Fig. 3 shows the change in the containment coolant inventory and the pressure behavior due to the operation of the spray system using externally supplied water.

In the SLB-CT-05 test, a limited flow rate of spray was provided, assuming the 1-train failure of the spray system. As a result, there was no rapid depressurization effect during spray operation; however, a gradual reduction in containment pressure was observed. This behavior is believed to be due to the lower internal compartments as the passive heat sink, which resulted from thermal mixing caused by the spray operation.

4. Conclusions

The SLB-CT-05 test conducted using the ATLAS-CUBE facility has provided valuable insights into the thermal-hydraulic behavior and safety performance of the reactor coolant system and containment during a steam line break. The study has highlighted the importance of understanding the complex interactions between the RCS and containment during such an event.

The test results have demonstrated that even under design extension conditions, including the total failure of the SIP and a limited flow rate of the containment spray system, cooling was maintained, and the system pressure was gradually stabilized. This indicates the robustness of the reactor safety systems in managing an overcooling accident scenario.

Furthermore, the test revealed the crucial role of thermal mixing in the containment, which contributed to the containment pressure reduction over time. This finding emphasizes the importance of containment spray systems and their effectiveness in mitigating the consequences of steam line break transients. Overall, the findings from the SLB-CT-05 test contribute to a better understanding of the complex thermal-hydraulic behavior during an SLB and can be used to improve safety analyses, emergency response procedures, and the design of future nuclear power plants.

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