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Integral Effect Test Coupling Reactor Coolant System (RCS) and Containment for Direct Vessel Injection (DVI) Line Break Loss-of-Coolant-Accident (IBLOCA)

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1. Introduction

A best-estimate evaluation for the safety of a reactor containment can be improved by a coupled analysis with a reactor coolant system (RCS), since major thermal hydraulic phenomena in the containment including a pressure and temperature (P/T) transient are highly dependent on the behavior of a mass and energy (M/E) supply from the RCS. In order to experimentally investigate the interaction between the RCS and the containment, an integral effect test facility, ATLAS-CUBE (Advance Thermal-hydraulic Test Loop for Accident Simulation - Containment Utility for Best-estimate Evaluation) has been designed and constructed [1] at KAERI (Korea Atomic Energy Research Institute). In this study, a direct vessel injection (DVI) line intermediate-break loss-of-coolant accident (IBLOCA) was simulated by utilizing the ATLAS-CUBE facility, regarding that the IBLOCA scenario was of importance in a safety analysis according to a risk-informed regulation (RIR) [2]. Also, the characteristics of the M/E during the IBLOCA and the multi-dimensional effect in the containment according to the break simulation position were investigated with considering its impact on the P/T behavior.

2. Description of the Test Facility

ATLAS is a thermal-hydraulic integral effect test facility to simulate multiple responses of the RCS during postulated accidents, and its reference plant is APR1400 (Advanced Power Reactor 1400 MWe). The ATLAS facility is a half-height and 1/288-volume scaled test facility according to the three-level scaling methodology [3,4]. The fluid system of ATLAS consists of a primary system, a secondary system, a safety injection system, a break simulating system, a containment simulating system, and auxiliary systems.

ATLAS-CUBE test facility was designed and constructed by connecting a new containment simulation system, CUBE, to ATLAS as shown in Fig. 1. The CUBE test facility was constructed with a containment simulation vessel, compartment structure, connection pipe, and spray system. Since the P/T transient in a reactor containment is highly governed by the M/E supply from a RCS, the containment simulation vessel in the ATLAS-CUBE test facility was designed to be conserved the volumetric scaling ratio of ATLAS, 1/288.

To reflect the characteristics of the compartment as a role of a passive heat sink and also the flow path inside the containment, the ATLAS-CUBE test facility incorporated compartment structures inside the containment simulation vessel. For a compact design of the compartment structures, CT-140 (a kind of a refractory) was selected as a material of the compartment structures due to a high heat capacity and a low thermal conductivity.

The containment simulation vessel incorporated a spray system to cool down the steam-gas mixture during an accident transient. It was designed to supply the coolant according to the scaling ratio of the flow rate (1/203.6) at the equivalent temperature condition. The spray nozzles were composed of two trains with 88 nozzles in total.

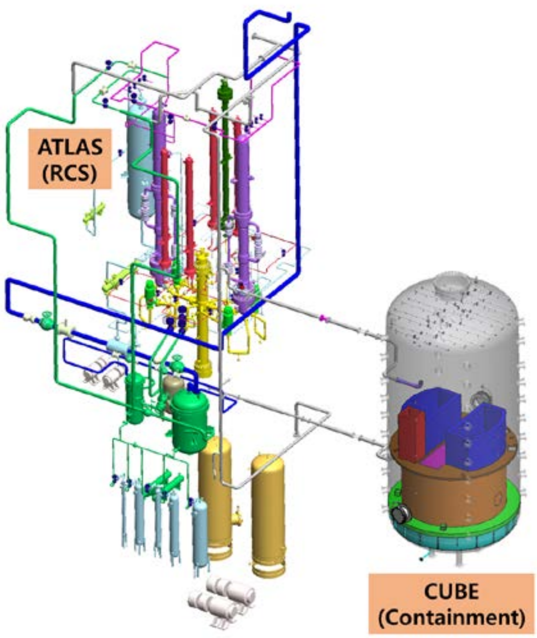

Fig. 1. ATLAS-CUBE integral effect test facility

3. Test Condition

The integral effect test, named IBDVI-CT-01, coupling a RCS and a containment was performed in the ATLAS-CUBE facility. The initial and boundary conditions for the RCS were determined equivalently to the B3.2 test in the OECD/NEA ATLAS-2 international joint project [5], which has simulated a DVI line IBLOCA by utilizing the ATLAS facility only.

The steady-state condition was achieved at 1.56 MW of the core power (about 8 % of ideally scaled core power) plus a heat loss compensation. The decay heat in the reactor core was simulated to be 1.2 times the ANS-73 decay curve from a conservative point of view. After a reactor trip signal was generated, the core heater power was controlled to follow the specified decay heat curve.

To preserve similarity in a double-ended guillotine break of a DVI line of APR1400 (8.5 inch pipe), the inner diameter and the length of the break nozzle were designed to be 15.13 mm and 191.6 mm, respectively. The break nozzle area is 1.798×10^{-4} m^2, which corresponds to 8 % of the cold leg flow area. The break flow was discharged into a containment simulation vessel of the CUBE facility. A single failure criterion was applied to an operation of a safety injection pump (SIP) and three safety injection tanks (SITs) were available through intact DVI nozzles. The injection flow rate from the SIP was determined according to a variation of the down-comer pressure of a reactor pressure vessel. The safety injection water temperature from the SIP and the SIT was set to be 50 ^oC. Decrease of the SIT flow rate by a fluidic device was simulated by controlling a valve on each pipe line from SIT to the DVI nozzle when the water level in the SIT became lower than 2.0 m.

Initial and boundary conditions for the containment system in the IBDVI-CT-01 test were determined with those of the APR1400 FSAR (Final Safety Analysis Report) [6]. In order to simulate a conservative condition of the containment pressure, the initial air temperature and the compartment wall temperature were increased up to 45 ~ 55 ^oC by utilizing a tracing heater system at the outer wall of the containment simulation vessel.

A spray system in the containment was operated at a high pressure signal of the containment to cool down the steam-gas mixture in the containment. It circulated the water in the in-vessel refueling water storage tank (IRWST) to the spray nozzles at a top region of the containment dome. According to a single failure assumption, 44 nozzles in a single train were used for the spray injection and the total flow rate was 1.66 kg/s.

4. Test Result

Considering the confidentiality of the test data, the experimental results were normalized by an arbitrary value including the time frame (t*).

Figure 2 shows the overall behavior in the RCS during the IBDVI-CT-01 test. After the initiation of the break at t*=0.03, a rapid depressurization of the primary system induced a reactor trip. The primary system pressure showed a plateau around t*=0.04, when the primary and secondary system pressures became a state of nearly equilibrium. Decrease of the coolant level in the down-comer changed the break flow from a liquid-dominant two-phase flow to a vapor-dominant two-phase flow, so that the primary system pressure decreased steeply again after the end of the plateau region.

The maximum cladding temperature showed an excursion behavior in the early period after the break. Even though the loop seal was cleared, a remained coolant in the upper down-comer prohibited an effective release of the steam flow toward the DVI line break. So that, the core water level was maintained low and the cladding temperature kept increasing until the coolant in the upper down-comer was cleared. After the clearance of the upper down-comer, quenching of the reactor core was completed resulting from an increase of the core water level. The overall behavior of the RCS was similarly simulated to the experimental result of the B3.2 test.

The break flow from the DVI line was injected into the SG-2 compartment room of the containment simulation vessel, through a break simulation pipe. Figure 3 shows the pressure of the containment simulation vessel in the IBDVI-CT-01 test. A rapid build-up of the containment pressure after the break was observed due to a large amount of the M/E supply. The initial peak pressure of the containment simulation vessel was observed at t*=0.053. The decrease of the containment pressure after the peak was related to a characteristic of the break flow. During a blowdown of the coolant inventory in the RCS, a peak flow from the break line was observed and it increased the containment pressure and temperature. After a decrease of the break flow around t*=0.04, a heat transfer at the passive heat sink surface became more dominant to reduce the containment pressure. As the internal compartment structure and the outer wall of the containment simulation vessel were heated up, the containment pressure increased again at t*=0.102 until an activation of the containment spray system at t*=1.003.

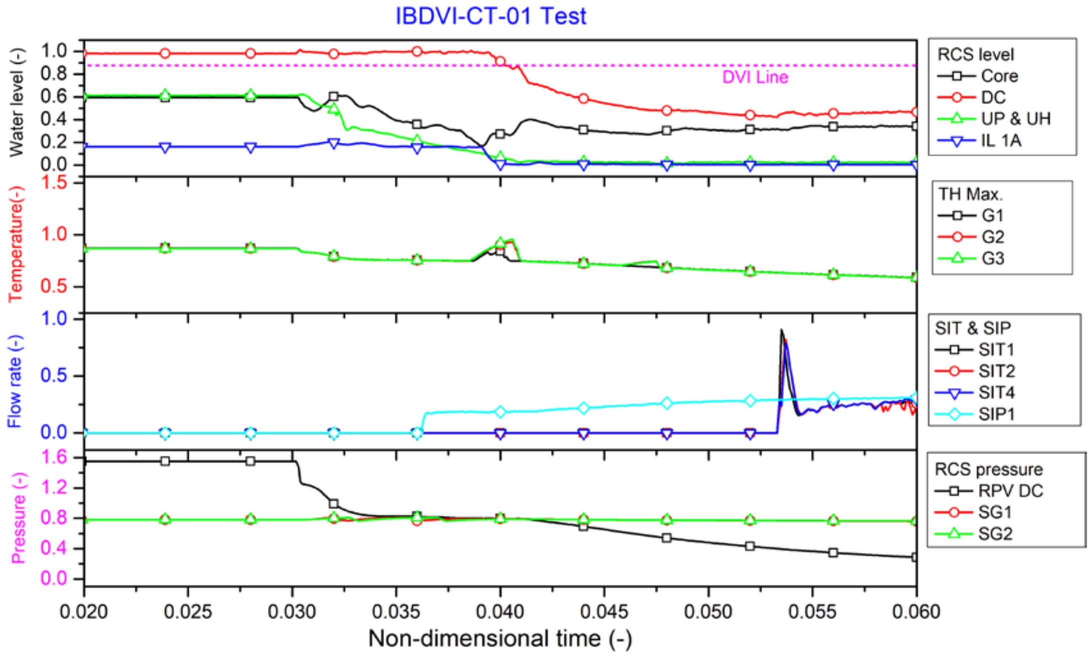

Fig. 2. Comparison of RCS thermal hydraulics in the IBDVI-CT-01 test

The steam-gas mixture temperature inside the containment was plotted in Fig. 4, Fig. 5, and Fig. 6, which compared an axial distribution at the Point A, L, I. The mixture temperature of the containment dome region above the SG compartment wall (Level 01 ~ 04) presented a well-mixed behavior and an equivalent trend to the containment system pressure. The steam-gas mixture temperature at the Level 07 showed an asymmetric behavior according to the location of the break flow injection. Since the injection nozzle for the primary system break was located on a lower part of the SG-2 compartment room, the steam-gas mixture temperature at Point L / Level 07 (TF-CTMT-07L) showed a larger value with a fluctuation behavior, rather than the temperature at Point I (TF-CTMT-07I, inside SG-1 compartment room). On the other hand, the steam-gas mixture around a refueling pool (TF-CTMT-06A) was maintained with a lower temperature before the spray injection, since an injection of the high-temperature steam flow could not sufficiently mix the fluid in the refueling pool due to a small area of the flow path. This result pointed out that a multi-dimensional distribution of the fluid temperature around the passive heat sink should be considered to realistically estimate a build-up behavior of the P/T in the containment.

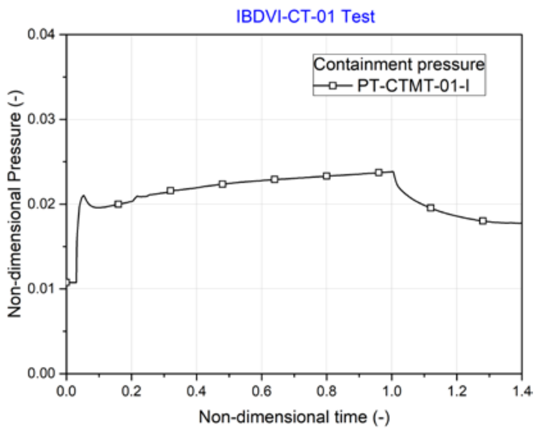

Fig. 3. Containment pressure in the IBDVI-CT-01 test

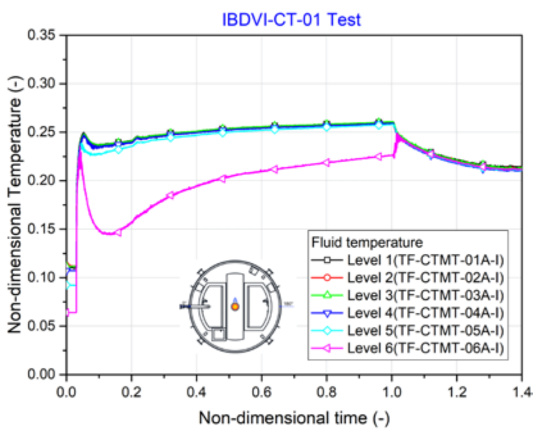

Fig. 4. Steam-gas mixture temperature of the containment (Point A)

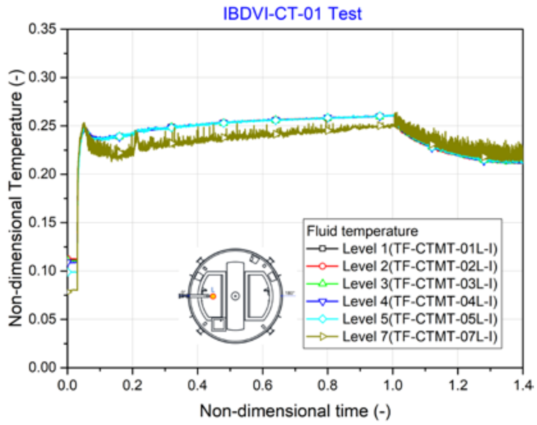

Fig. 5. Steam-gas mixture temperature of the containment (Point L)

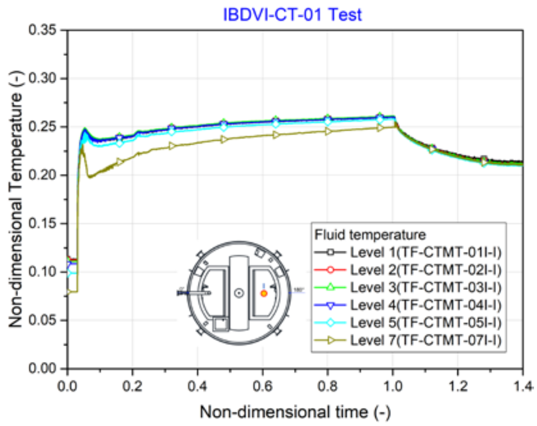

Fig. 6. Steam-gas mixture temperature of the containment (Point I)

5. Conclusions

The IBDVI-CT-01 test was performed by utilizing the ATLAS-CUBE facility to investigate an interactive thermal hydraulic transient of a RCS and a containment during a DVI line IBLOCA scenario. The test condition was similar to that of the OECD-ATLAS2 B3.2 test, which simulated an 8.5-inch break of a DVI line with an assumption of a single failure of SIPs. The initial and boundary conditions for the test were determined referring APR1400 as a prototype with the three-level scaling methodology.

As the experimental result, the DVI line break induced a rapid depressurization of the primary system by a blowdown of the coolant of the reactor pressure vessel, which showed the quantitatively similar behavior to the OECD-ATLAS2 B3.2 test result. Both of the RCS and the containment were effectively cooled down during the transient, by activation of the safety injection system and the containment spray system, respectively. The M/E supply from the RCS increased a pressure and a steam-gas mixture temperature of the containment, while a wall condensation on the passive heat sink and a spray injection contributed to cool down the steam-gas mixture. A containment dome region showed a well-mixed behavior of the steam-gas mixture with a nearly homogeneous temperature distribution.

The present integral effect test data can be used to evaluate the capability of thermal hydraulic safety analysis codes for the RCS and the containment, with considering the multi-dimensional or the multi-compartment models. Also, the test data will be able to contribute to validating the evaluation methodology for M/E and P/T transient of a reactor containment.

ACKNOWLEDGMENTS

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