Thermal-Hydraulic Integral Effect Test with the ATLS for Investigation on CEDM Penetration Nozzle Integrity

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1. Introduction

Primary water stress corrosion cracking (PWSCC) was found at the control element driving mechanism (CEDM) penetration nozzle of the Younggwang nuclear power plant unit 3 (YGN-3) in 2012. As for the integrity of the CEDM penetration nozzle, a safety issue was already raised by the Davis Bessel nuclear power plant accident where a significant wall thinning was found at the CEDM penetration nozzle in 2002.

In this study, thermal-hydraulic integral effect test with the ATLAS (Advanced Thermal-Hydraulic Test Loop for Accident Simulation) [1] was performed for simulating a failure of CEDM penetration nozzle. The main objectives of the present test were not only to provide physical insight into the system response during a failure of CEDM penetration nozzle but also to establish an integral effect test database for the validation of the safety analysis codes. Furthermore, present experimental data were utilized to resolve the safety issue raised by the PWSCC at the CEDM penetration nozzle of the YGN-3.

2. Experimental Methodology

2.1 Description of the ATLAS Facility

A thermal-hydraulic integral effect test facility, ATLAS, has been operated in order to investigate major design basis accidents and operational transients for a 1400 MWe-class advanced pressurized water reactor, APR1400 (Advanced Power Reactor 1400). The ATLAS has the same two-loop features as the reference plant of the APR1400 and is designed according to the well-known scaling method suggested by Ishii and Kataoka to simulate the various accident scenarios as realistically as possible. The ATLAS is a 1/2 reduced height and a 1/288 volume scaled integral effect test facility with respect to the APR1400. It has a maximum power capacity of 10% of the scaled nominal core power, and it can simulate full pressure and temperature conditions of the APR1400.

2.2 Experimental Conditions and Procedures

In the present test, named SB-CEDM-01, failure of two penetration nozzles of the CEDM in the APR1400 (Advanced Power Reactor 1400 MWe) was simulated. Initial and boundary conditions were determined with respect to the reference conditions of the APR1400. However, with an aim of corresponding to the YGN-3 situation, the safety injection water was supplied via cold leg injection (CLI) mode which is adopted in the OPR1000 (Optimized Power Reactor 1000 MWe). A single-failure of a loss of a diesel generator, resulting in the minimum safety injection flow to the reactor pressure vessel (RPV), was assumed to occur in concurrence with the reactor trip. Therefore, the safety injection water from the safety injection pump (SIP) was only available through the CLI-1 and -3 nozzles, and the safety injection water from the safety injection tank (SIT) was available through all of the CLI nozzles.

Break flow from the failure of the CEDM penetration nozzle was discharged to the containment simulation system, which consists of separating vessels and measuring vessels. The break flow was collected into the separating vessel. The steam, which was separated in the separating vessel, was discharged through a silence to the atmosphere. The steam flow rate was measured by a vortex-type flow meter at the discharge line. The separated water was drained to the measuring vessel and the accumulated mass of water was measured by a lead cell. Fig. 1 shows a schematic diagram of the simulation of the CEDM penetration nozzle failure and the break nozzle in the present SB-CEDM-01 test, respectively.

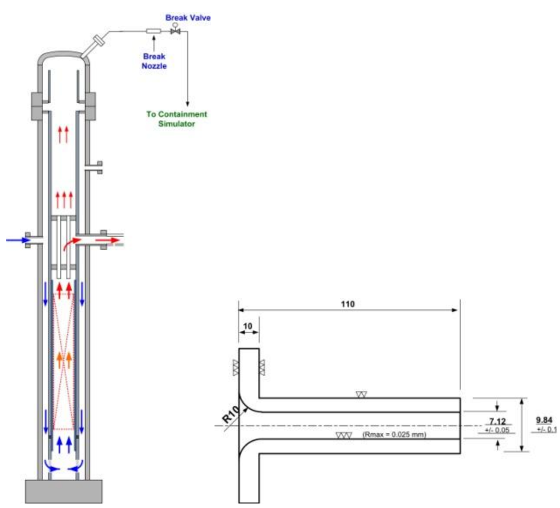

Fig. 1. A schematic diagram of the simulation of the CEDM penetration nozzle failure and the break nozzle

3. Experimental Results and Discussions

The test was started by opening the break valve shown in Fig. 1 after steady state data were measured for 305 seconds. The major sequence of events observed during the whole test period is summarized in Table 1.

Table I: Sequence of Events

Event	Time (sec)	Remark
Break open	305	
LPP	419	Pressurize pressure @ 10.7124 MPa
MSIS	420	LPP + 1.0 s delay
MFIS	427	LPP + 7.07 s delay
MSSV opening	432	Steam generator pressure @ 8.1 MPa
Decay power start	432	LPP + 12.07 s delay
SIP	448	LPP + 28.28 s delay

On break, the primary system pressure decreased abruptly and then the safety water from the SIP was supplied via CLI-1, and -3 nozzles. Supply of the safety injection water mitigated the depressurization of the primary system pressure as shown in Fig. 2. Following the reactor trip, the secondary system pressure increased until the main steam safety valves (MSSVs) were opened to reduce the secondary system pressure. The MSSVs were opened three times in both steam generators.

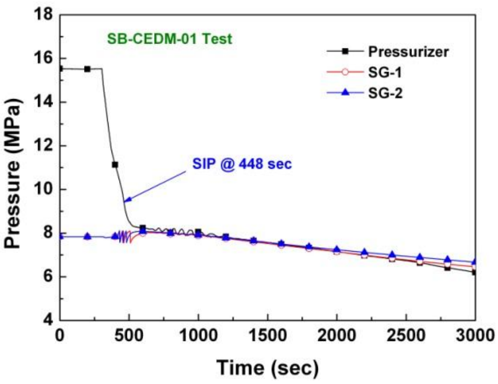

Fig. 2. Variation of the system pressures

The collapsed water level in the core decreased abruptly after the break. However, supply of the safety injection water mitigated the level decrease and the collapsed water level in the core increased again as shown in Fig. 3. Since the core was not uncovered, peak cladding temperature (PCT) was not observed in the SB-CEDM-01 test.

In the present SB-CEDM-01 test, a loop seal clearing was not occurred in any loop. The fluid temperatures in each cold leg showed different trends depending on supply of the safety injection water. The fluid temperatures in the cold leg-1A and -2A were remarkably lower than those in the cold leg-1B and -2B which could be attributed to that the safety water was injected into the cold leg-1A and -2A in the SB-CEDM-01 test. Compared to the break flow measured in the previous small break loss of coolant accident (SBLOCA) test simulated at the cold leg whose break area was same as the present SB-CEDM-01 test, the break flow rate was lower by 1/4 times. In case of the failure in the RPV upper head, the time period for discharging of single-phase water is shorter than that of the cold leg break SBLOCA. This induces the present test result of small break flow rate compared to that of the cold leg break SBLOCA test.

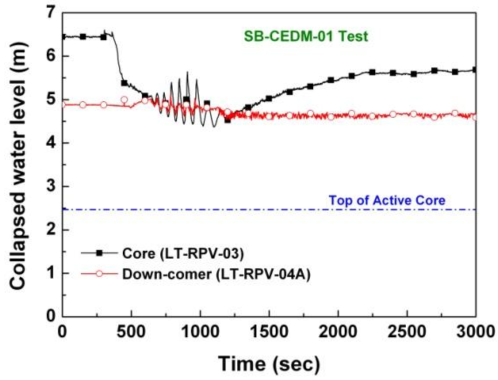

Fig. 3. Variation of the collapsed water levels

4. Conclusions

Thermal-hydraulic integral effect test with the ATLAS was performed for simulating a failure of CEDM penetration nozzle. Failure of two penetration nozzles of the CEDM in the APR1400 was simulated. Initial and boundary conditions were determined with respect to the reference conditions of the APR1400. However, with an aim of corresponding to the YGN-3 situation, the safety injection water was supplied via CLI mode. Compared to the cold leg break SBLOCA, the consequences of the event were milder in terms of a loop seal clearance, break flow rate, collapsed water level, and PCT. This could be mainly attributed to the small break flow rate in case of the failure in the RPV upper head. Present experimental data were utilized to resolve the safety issue raised by the PWSCC at the CEDM penetration nozzle of the YGN-3.

Acknowledgements

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REFERENCES

[1] W.P. Baek, C.H. Song, B.J. Yun, et al. “KAERI Integral Effect Test Program and the ATLAS Design,” *Nuclear Technology*, Vol. 152, p. 183 ~ 195, 2005.