Thermal-Hydraulic Integral Effect Test Result on the Steam Generator Tube Rupture Accident in the APR1400

Kyoung-Ho Kang*, Seok-Cho, Hyun-Sik Park, Nam-Hyun Choi, In-Cheol Chu, Byong-Jo Yun, Kyung-Doo Kim, Yeon-Sik Kim, Ki-Yong Choi, Won-Pil Baek
Korea Atomic Energy Research Institute, 150 Dukjin-dong, Yusong-gu, Daejeon 305-353, Korea
**Corresponding author: khkang@kaeri.re.kr*

1. Introduction

A postulated SGTR (Steam Generator Tube Rupture) event of the APR1400 was experimentally investigated with the thermal-hydraulic integral effect test facility, ATLAS (Advanced Thermal-hydraulic Test Loop for Accident Simulation) [1]. It is generally known that the leak flow rate from the primary to the secondary side is the most important factor affecting the overall thermal-hydraulic behaviors such as the depressurization rate of the RCS system, the water level increase and pressurization rate of the secondary system, and the consequent MSSV opening time, etc. As one of the most limiting SGTR accidents, a leak flow equivalent to a double-ended rupture of single and five U-tubes was simulated in this study. The main objectives of these tests were not only to provide physical insight into the system response of the APR1400 reactor during a transient situation of the SGTR but also to present integral effect test data for the validation of the SPACE (Safety and Performance Analysis Computer Code), which is now under development by the Korean nuclear industry.

2. Description of the ATLAS

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.

3. Experimental Conditions and Procedures

In the present SGTR-HL-04 and SGTR-HL-05 tests, considering the safety analysis result for the SGTR accident of the APR1400 [2], a reactor trip was assumed primarily to occur by an increase of the steam generator level as a High Steam Generator Level (HSGL) trip signal. In addition, a single-failure of a loss of a diesel generator, resulting in the minimum safety injection flow to the RPV (reactor pressure vessel), was assumed to occur in concurrence with the reactor trip. Therefore, the safety injection water from the SIP (Safety Injection Pump) was only available through the DVI-1 and -3 nozzles, and the safety injection water from the SIT (Safety Injection Tank) was available through all of the DVI (Direct Vessel Injection) nozzles.

In order to simulate the SGTR accident of the APR1400 as realistically as possible, a pertinent scaling approach was considered from a leak flow rate point of view. And the leak flow rate was directly measured with a combination of an orifice flow meter and a CVM (Capacitance Void Meter). The leak flow can be choked or non-choked depending on the differential pressure between the primary and the secondary systems. For both the cases of discharged flow, the leak flow rate should be scaled down appropriately in the ATLAS test. Considering the velocity scaling factor of the ATLAS, the break areas for a single and a five tubes rupture case were obtained. A geometry of the break nozzles used in the present tests is shown in Fig. 1.

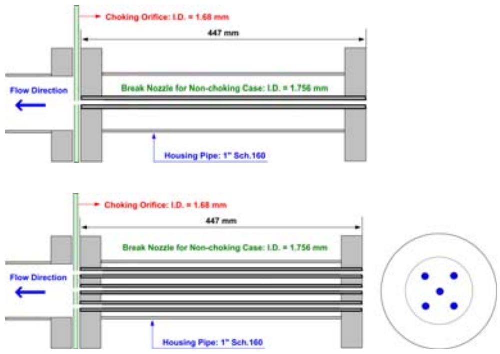

Fig. 1. Break nozzles used in the present tests for simulating a single and a five tubes rupture.

Contrary to the real accident situation of the SGTR in nuclear power plants, the primary inventory was discharged from the hot side of the lower plenum to the upper location of steam generator secondary hot side. The discharging location was 2015.0 mm above the inlet of the U-tube. The present test conditions were determined by a pre-test calculation with a best-estimate thermal hydraulic code, MARS-KS.

4. Experimental Results and Discussions

When the SGTR event was started by opening the SGTR simulation valve, the water level of the affected steam generator (SG-1) increased rapidly and reached the set-point of the HSGL reactor trip. When the HSGL signal occurred, the RCP and the pressurizer heater were stopped, and the main feedwater isolation valves (MFIVs) were closed with pre-specified delay times. As the SGTR accident progressed, the primary system pressure decreased below 10.7214 MPa and the SIP was actuated with a pre-specified delay time of 28.28 seconds as shown in Figs. 2 and 3. The injection of SI water resulted in an increase in the RCS pressure because the leak flow rate of the primary coolant through the break was less than the safety injection flow rate by the SIPs. Depressurization rate of the RCS was smaller in the SGTR-HL-04 test in which a single U-tube rupture was simulated.

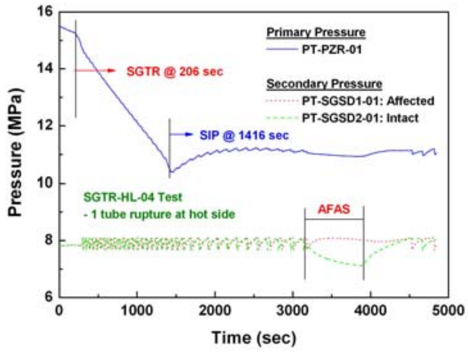

Fig. 2. Pressure trend measured in the SGTR-HL-04 test.

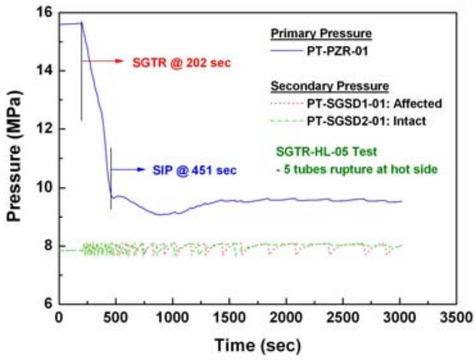

Fig. 3. Pressure trend measured in the SGTR-HL-05 test.

Fig. 4 shows the variation of the collapsed water level in the secondary side of the steam generators. Due to the leak flow, the collapsed water level of the affected steam generator was maintained a quasi-steady state even though there were level fluctuations resulting from the discharged flow through the MSSVs in the SGTR-HL-04 test. The secondary side of the affected steam generator in the SGTR-HL-05 test was filled with water due to the relatively large amount of leak flow. The collapsed water level of the intact steam generator, however, gradually decreased due to the cyclic discharge of inventory through the MSSVs as shown in Fig. 4. In the SGTR-HL-04 test, the injection of the auxiliary feedwater recovered the water level of the intact steam generator after about 3160 seconds. During the whole test period, the secondary side water level of the intact steam generator was maintained above the set-point for actuation of the AFAS in the SGTR-HL-05 test.

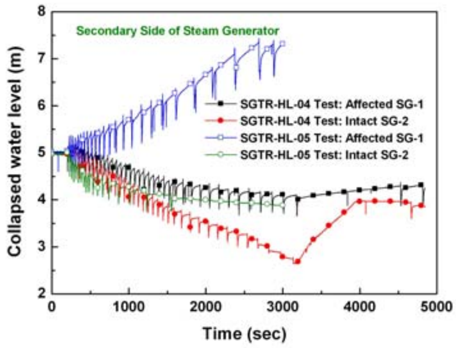

Fig. 4. Variation of the collapsed water level in the secondary side of the steam generator.

As the SI water was injected, the water level in the down-comer of the RPV fluctuated. Even though the water level in the core decreased slightly with the time, the change of the level could be considered to be trivial. This mild change of the water level in the core could be attributed to the small break size of the present tests compared with the LOCA (Loss of Coolant Accident) test. No excursion in the cladding temperature was observed in both the tests.

5. Conclusions

In order to simulate the SGTR accident of the APR1400, the SGTR-HL-04 and the SGTR-HL-05 tests were performed by simulating a double-ended rupture of single and five U-tubes at the hot side of the ATLAS steam generator. Following the reactor trip induced by HSGL, the primary system pressure decreased and the secondary system pressure increased until the MSSVs was opened to reduce the secondary system pressure. The MSSVs repeated on and off status depending on the secondary pressure during the whole test period. Mild change of the water level in the core was observed, which could be attributed to the small break size of the present tests. No excursion in the cladding temperature was observed in the present test. This integral effect data will be used to evaluate the prediction capability of existing safety analysis codes as well as the SPACE code. Furthermore, this data can be utilized to identify any code deficiency for a SGTR simulation, especially for DVI-adapted plants.

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.
[2] Korea Electric Power Corporation, "KNGR Standard Safety Analysis Report," Chapter 15, 2001.