Transactions of the Korean Nuclear Society Autumn Meeting
Gyeongju, Korea, October 27-28, 2011

ATLAS Test Result on the Five Tubes Rupture at Cold Side of Steam Generator

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

The SGTR accident is one of the design basis accidents, which has a unique feature of the penetration of the barrier between the reactor coolant system (RCS) and the secondary system resulting from failure of steam generator U-tubes. The SGTR has an importance in safety due to a concern of a containment bypass of radioactive inventory. The break flow from ruptured U-tubes can increase a water level and a pressure of the affected steam generator. Following a reactor and a turbine trips, the main steam safety valves (MSSVs) can be open to mitigate an increase in the secondary system pressure. Meanwhile, the SGTR can provide a direct flow path from the primary to the secondary systems resulting in the release of fission products into the atmosphere.

In this study, the SGTR-CL-02 test was performed to simulate a double-ended rupture of five U-tubes at the cold side of the ATLAS steam generator. With an aim of simulating the SGTR accident of the APR1400 as realistically as possible, a pertinent scaling approach was taken, especially focussing on a break flow. Besides, a direct break flow measurement technique was applied to the present SGTR-CL-02 test in order to quantify the break flow rate. The main objectives of this test were not only to provide physical insight into the system response of the APR1400 during the SGTR accident but also to produce integral effect test data to validate the thermal-hydraulic safety analysis code.

2. Description of the ATLAS

A thermal-hydraulic integral effect test facility, ATLAS [1], 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-CL-02 test, considering the safety analysis result for the SGTR accident of the APR1400, 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 by an orifice flow 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 five tubes rupture case were obtained.

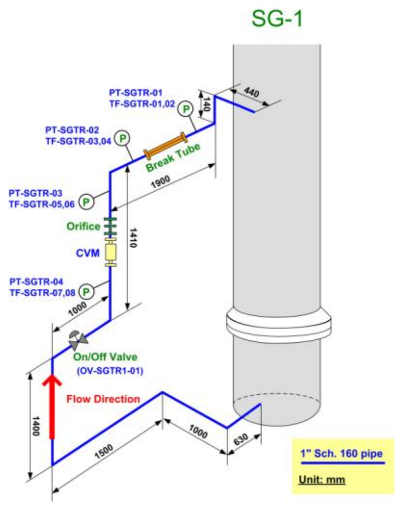

Fig. 1. Piping arrangement of break simulation system

Contrary to the real accident situation of the SGTR in nuclear power plants, the primary inventory was discharged from the cold side of the lower plenum to the upper location of steam generator secondary cold side as shown in Fig. 1. The discharging location was 2015.0 mm above the inlet of the U-tube.

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 Fig. 2. Depressurization rate of the RCS was estimated to be 21.56 kPa/sec.

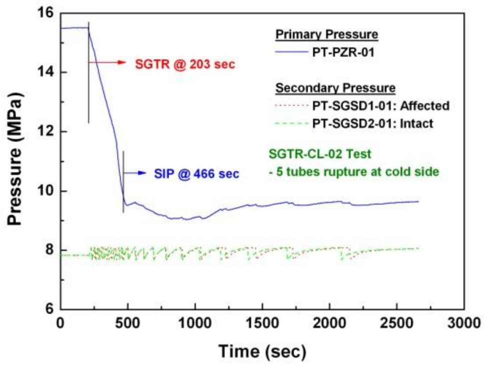

Fig. 2. Pressure trend measured in the SGTR-CL-02 test.

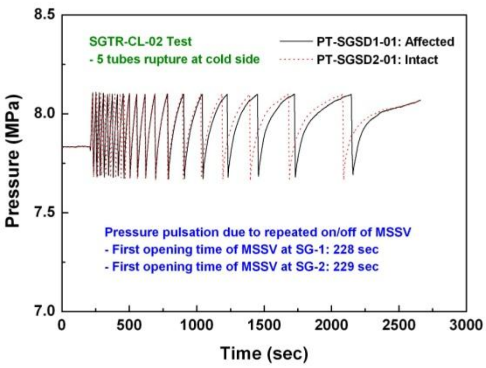

Fig. 3. Secondary system pressure trend

Fig. 3 shows the variation of the secondary system pressure. Following the reactor trip, the secondary pressure increased until the MSSVs were opened to reduce the secondary system pressure. Subsequent to the peak in the secondary system pressure of the steam generators, the secondary system pressure decreased, resulting in closure of the MSSVs. Then, the secondary system pressure started to increase again until it reached the MSSV set-point because the steam generators were isolated due to the previous MSIS and the MFIS actuations. The MSSVs in two steam generators showed almost simultaneous opening and closing behaviors. However, the time difference of the actuation of the MSSVs in two steam generators became larger as time went by.

Compared to the SGTR-CL-01 test, opening frequency of the MSSVs in the intact steam generator (SG-2) was highly reduced after 500 seconds in the present SGTR-CL-02 test. Large discharge of the primary inventory resulted in rapid depressurization of the primary system and consequently early injection of the SIP in the SGTR-CL-02 test. Supply of cold ECC water by the SIPs reduced the energy transfer to the secondary side compared with the single U-tube rupture case. Meanwhile, the secondary pressure of the affected steam generator (SG-1) is more likely to increase due to higher break flow than the single U-tube rupture case. However, less heat transfer to the secondary side caused by earlier actuation of the SIPs had more influence on the secondary pressure of the affected steam generator than the break flow.

In order to directly measure the break flow rate, an orifice flow meter was installed at the upstream of the break nozzle for the break flow to be single-phase water flow at the measurement location. In this study, as a complementary method to the direct measurement of the break flow, a RCS inventory-based break flow estimation method was applied. The accumulated break flow measured by assuming a discharge of pure water shows similar trends with that estimated by the RCS inventory change. And the calculated break flow rate using the Henry-Fauske critical flow model was 0.589 kg/s which was larger than the maximum break flow rate measured in the test. It could be confirmed that the break flow was discharged as single-phase water at the location of the break flow measurement and the measurement accuracy was acceptable in the present SGTR-CL-02 test.

5. Conclusions

In order to simulate the SGTR accident of the APR1400, the SGTR-CL-02 test was performed by simulating a double-ended rupture of five U-tubes at the cold 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. This integral effect test data will be used to evaluate the prediction capability of existing safety analysis codes of the MARS and the RELAP5 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.