Best Estimate Thermal-Hydraulic System Analysis using the MARS Code for the Steam Generator Tube Rupture Accident in the APR1400

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1. Introduction

A postulated SGTR (Steam Generator Tube Rupture) accident of the APR1400 was analysed using the best estimate safety analysis code, MARS (Multi-dimensional Analysis of Reactor Safety) [1]. 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 the failure of a steam generator U-tube. The SGTR has an importance in safety due to a concern of a containment bypass of radioactive inventory. In the course of the SGTR, the radioactivity leaking from a broken steam generator U-tube mixes with the shell-side water in an affected steam generator. Leak flow from ruptured U-tubes can increase a water level and a pressure of the affected steam generator. Following a reactor trip and a turbine trip, 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 system resulting in the release of fission products into the atmosphere.

As one of the most limiting SGTR accidents, a leak flow equivalent to a double-ended rupture of five U-tubes was analysed in this study. The main objective of this study is not only to provide physical insight into the system response of the APR1400 reactor during a SGTR but also to investigate the effect of reactor trip type of the HSGL (High Steam Generator Level) trip and the LPP (Low Pressurizer Pressure) trip on the thermal-hydraulic system response.

2. Methodology of MARS Analysis

The MARS code has been developed at the KAERI for the realistic multi-dimensional thermal-hydraulic system analysis of light water reactor. Fig. 1 shows the MARS nodalization scheme used for the present analysis. The nodalization scheme includes all the reactor coolant systems of the APR1400 such as the reactor pressure vessel, primary piping, steam generators, a pressurizer, steam lines, and a safety injection system. The feedwater and the turbine systems were treated as boundary conditions and were modeled by using a time dependent volume.

The total number of U-tubes in each steam generator of the APR1400 is 12559. In this study, a double-ended guillotine break of five U-tubes of the steam generator was assumed from a conservative point of view. As for the tube rupture modelling method, double tube modelling (DTM) was adopted as shown in Fig. 2. Broken U-tubes were modelled as an assembly of a single volume (C341 and C342). And intact U-tubes were modelled as a separate assembly of a single volume (C340). The break location was 4.03 m above inlet of the U-tube at hot side. For the simulation of the critical flow discharge at the break location, the Henry-Fauske critical flow model was used and the discharge coefficient and the thermal non-equilibrium constant were assumed to be 1.0 and 0.14, respectively.

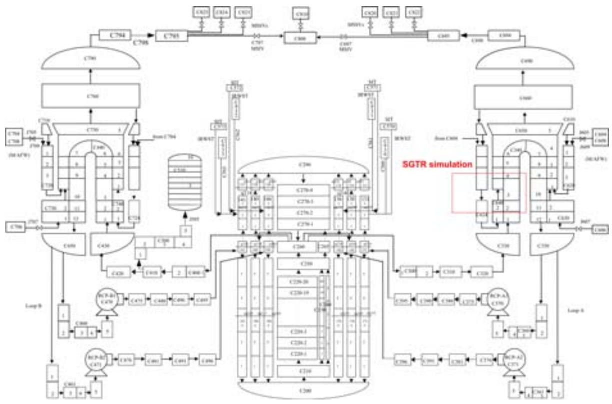

Fig. 1. MARS nodalization scheme for the SGTR analysis of the APR1400.

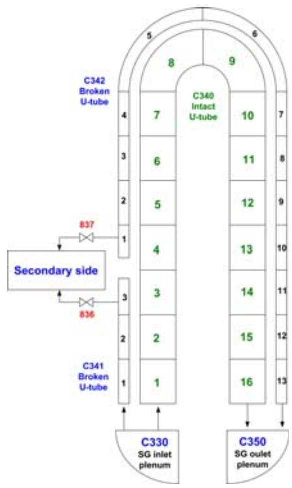

Fig. 2. Schematic diagram of the double tube modeling (DTM) for the steam generator tube rupture simulation.

3. Analysis Results

In this study, the effect of the reactor trip type of the HSGL and the LPP trips on the thermal-hydraulic system response was investigated. Table 1 shows the sequence of the events for the two reactor trip cases. The main steam isolation signal (MSIS) was not actuated in the case of the LPP trip.

Table 1: Sequence of the Events

Events	Time (sec)	
	HSGL	LPP
Break starts	0.0001	0.0001
Reactor trip	28.96	527.36
RCP trip	29.46	527.86
MSIS	29.96	-
Turbine trip	30.02	528.43
1st MSSV opening	31.48	530.52
MFIS	39.46	537.88
Decay power reach at 8 %	46.08	543.80
SIP	344.07	570.08
AFAS	1025.67	1975.00

As shown in Table 1, overall progress of the events was calculated to be rapid in the case of the HSGL trip. When the SGTR event was started, the RCS depressurized until the safety injection pumps (SIPs) were actuated as shown in Figs. 3 and 4. The injection of safety injection (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. In the case of the LPP trip, depressurization of the RCS was remarkable simultaneously with the reactor trip which could be attributed to that the time of the reactor trip was coincident with the time when the pressure started to rapidly decrease due to the steam condensation resulting from the pressurizer emptiness.

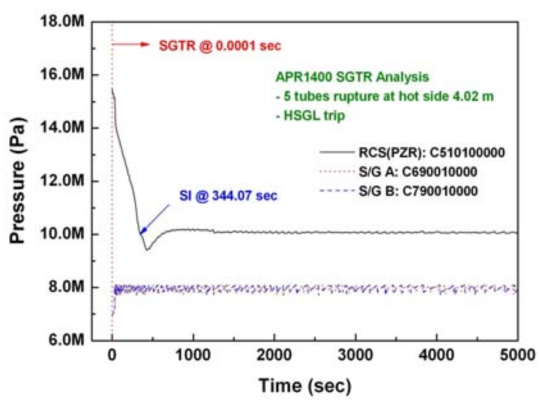

Fig. 3. Variations of the primary and the secondary pressures in the case of the HSGL trip.

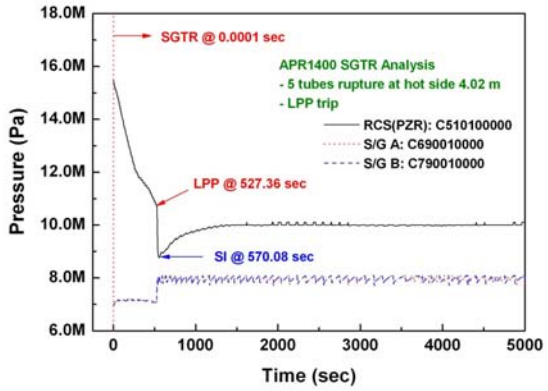

Fig. 4. Variations of the primary and the secondary pressures in the case of the LPP trip.

The SGTR accident can provide a direct flow path from the primary to the secondary systems due to a failure of a steam generator U-tube. Fig. 5 shows the accumulated leak flow for both the reactor trip cases. The initial time was synchronized in Fig. 5. The reactor trip type did not affect the leak flow from the primary to the secondary side as shown in Fig. 5.

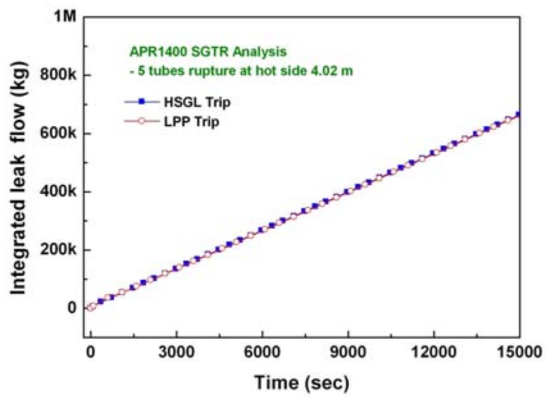

Fig. 5. Comparison of the accumulated leak flow.

4. Conclusions

As one of the most limiting SGTR accidents, a leak flow equivalent to a double-ended rupture of five U-tubes was analysed using the best estimate thermal-hydraulic system code, MARS. In this study, the effect of the reactor trip type of the HSGL and the LPP trips on the thermal-hydraulic system response was investigated. The reactor trip type affected the overall progress of the major events. However, the effect on the thermal-hydraulic response of the plant was trivial.

REFERENCES

[1] Jeong, J.J., Ha, K.S., Chung, B.D., Lee, W.J., Development of a Multi-Dimensional Thermal-Hydraulic System Code, MARS1.3.1, Ann. Nucl. Eng. 26, 1611-1642, 1999.