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THERMAL-HYDRAULIC SYSTEM ANALYSIS USING THE MARS CODE FOR THE TRANSIENT STEAM GENERATOR TUBE RUPTURE ACCIDENT

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Abstract

A postulated SGTR accident of the APR1400 was analysed using the best estimate safety analysis code, MARS. 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 a HSGL and a LPP on the thermal-hydraulic system response. As for the tube rupture modelling method, double tube modelling was adopted. Broken U-tubes were modelled as a separate assembly of a single volume. 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.

Introduction

The SGTR (Steam Generator Tube Rupture) 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. Break flow from ruptured U-tubes can increase a water level and a pressure of the affected steam generator. Prior to turbine trip, the radioactivity is transported through the turbine to the condenser where the noncondensible radioactive material would be released via the main condenser evacuation system (MCES). 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.

In this study, a postulated SGTR accident of the APR1400 (Advanced Power Reactor 1400 MWe) was analysed using the best estimate safety analysis code, MARS (Multi-dimensional Analysis of Reactor Safety) [1]. As one of the most limiting SGTR accidents, a leak flow equivalent to a double-ended rupture of five U-tubes was analyzed. A multiple steam generator tube rupture (MSGTR) event became a safety issue in the early 1990s, even though there is not any report of a MSGTR event, because of two safety concerns; i.e. the containment bypass of radioactive inventory and the increase in reactivity of the reactor core [2]. The US NRC (Nuclear Regulatory Committee) suggested the improvement of design and operational procedures for mitigation of the MSGTR consequences up to five U-tubes rupture in the SECY-93-087. 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 high steam generator level (HSGL) trip and the low pressurizer pressure (LPP) trip on the thermal-hydraulic system response.

1. 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. The backbones of the MARS are the RELAP5/MOD3.2.1.2 [3] and the COBRA-TF codes [4] of the US NRC. Figure 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 modelled by using a time dependent volume (TDV). The APR1400 has two identical steam generators and the generated steam from each steam generator is collected in the common header and delivered to the turbine. In the MARS input model, the steam generator was divided into a downcomer, an economizer, an evaporator and a riser, a separator, a steam dryer, and a steam dome as shown in Figure 1.

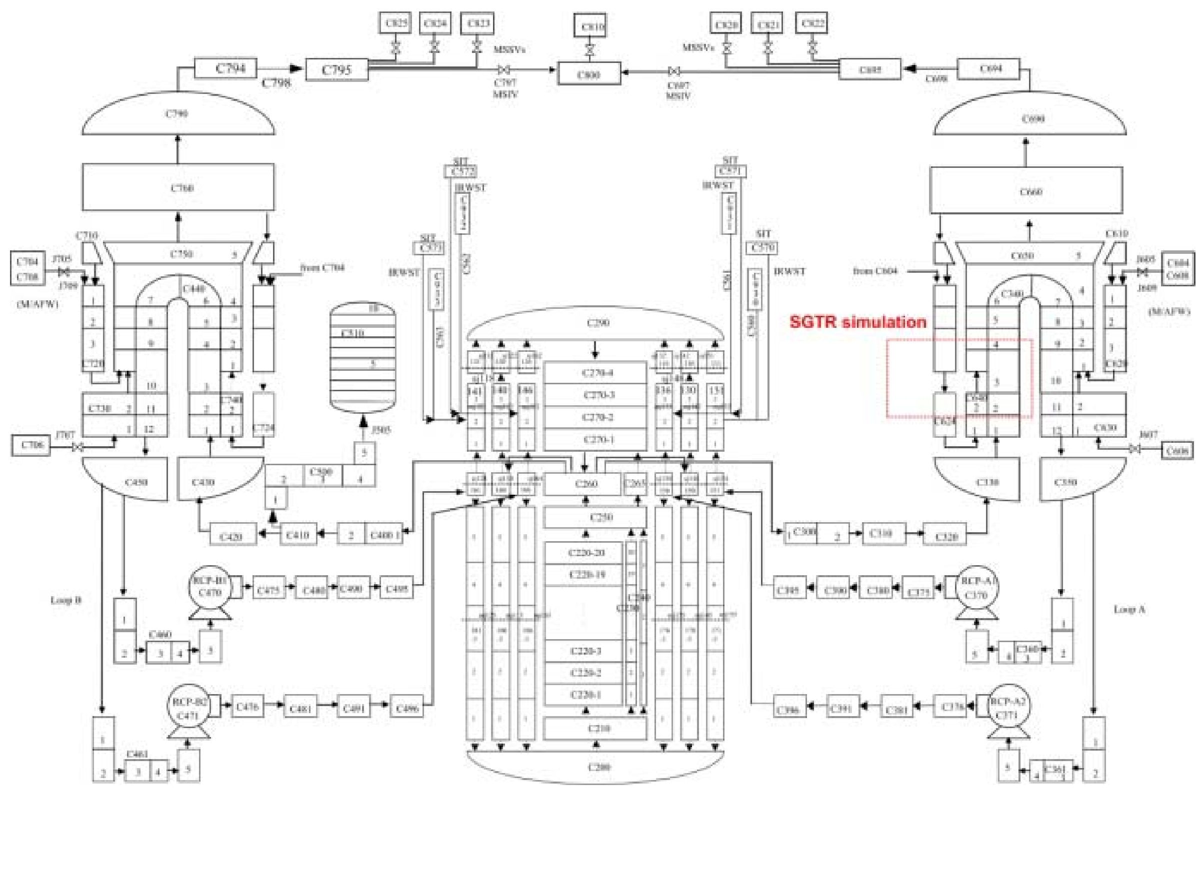

Figure 1 MARS nodalization scheme for the SGTR analysis of the APR1400.

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 modeling method, double tube modeling (DTM) was adopted as shown in Figure 2. Broken U-tubes were modeled as an assembly of a single volume (C341 and C342). And intact U-tubes were modeled as a separate assembly of a single volume (C340). The break location was 4.03 m above inlet of the U-tube at the hot side of steam generator. 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.

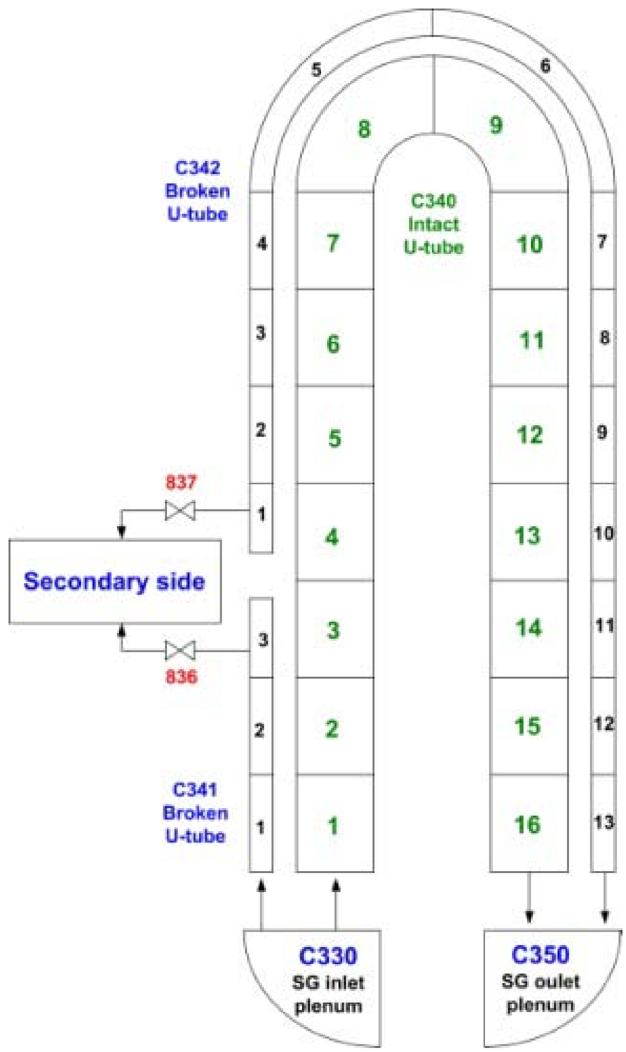

Figure 2 Schematic diagram of the double tube modelling (DTM) for the SGTR simulation.

2. 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. 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 Figure 3. As the SGTR accident progressed, the primary system pressure decreased below 10.7244 MPa and the SIP was actuated with a pre-specified delay time of 39.99 seconds. The injection of safety injection (SI) water resulted in an increase in the RCS pressure because the break 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.

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

Table 2 Summary of sequence of the events.

Figure 4 shows the variation of the collapsed water level in the pressurizer and the secondary side of the steam generators. Even though there was some difference in the timing of the initial reduction, the variation of the collapsed water level in the steam generators secondary side were similar between two cases of reactor trip. The collapsed water level of the affected steam generator continuously increased and the affected steam generator became filled with water which could be attributed to the relatively large break flow. The collapsed water level of the intact steam generator repeated peak shapes due to the on-off of the auxiliary feedwater actuation signal. The collapsed water level in the pressurizer slowly increased due to the supply of the SI water. Figure 5 shows the flow rate variation of the auxiliary feedwater in both calculations. Supply of auxiliary feedwater was actuated when the collapsed water level of the steam generator reached the 25% of the wide range level and was terminated when the collapsed water level reached the 40% of the wide range level.

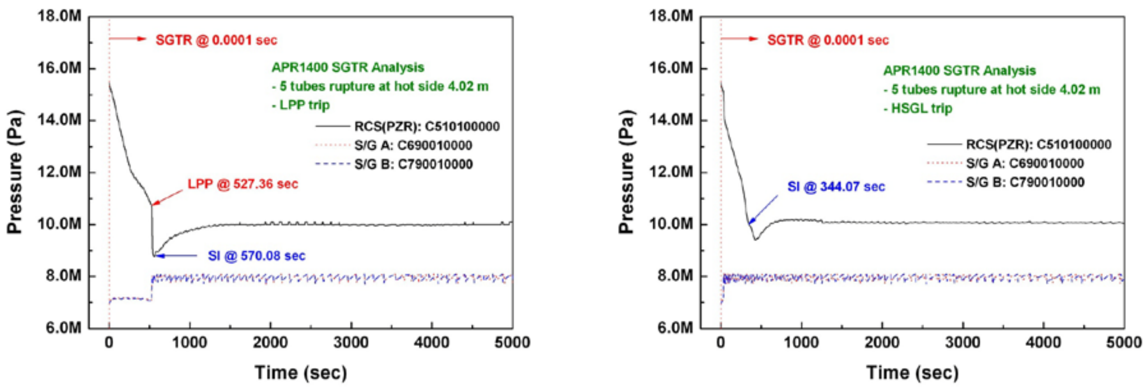

Figure 3 Variation of the pressure of the primary and the secondary systems.

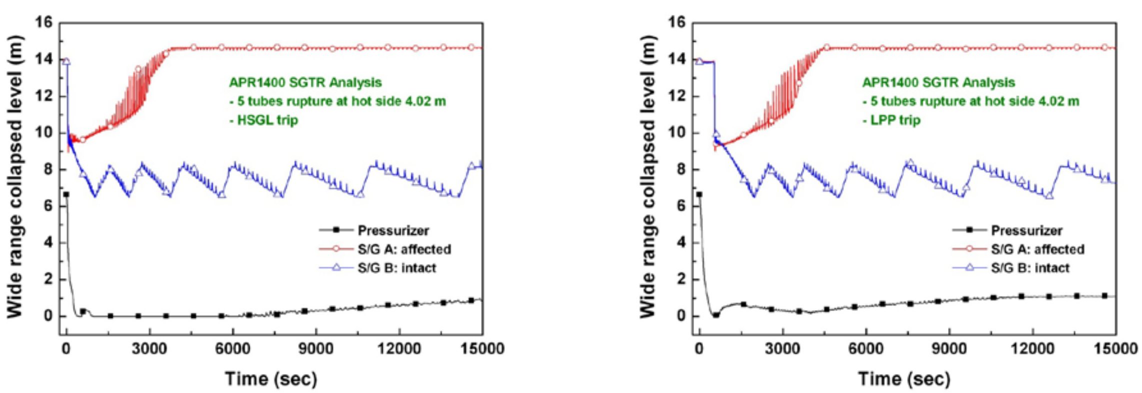

Figure 4 Variation of the collapsed water level of the primary and the secondary systems.

Following the reactor trip, the secondary pressure increased until the main steam safety valves (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 first opening times of the MSSV were 31.48 seconds and 530.52 seconds in the HSGL and the LPP trips, respectively. Even though the first opening times were different, the accumulated mass of discharged flow through the MSSVs was similar in both reactor trip cases applying the synchronized time frame as shown in Figure 6.

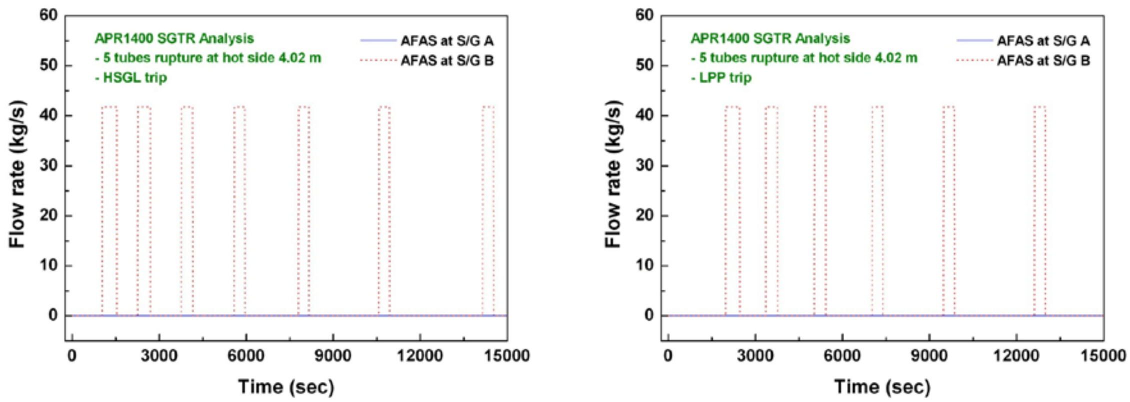

Figure 5 Variation of the auxiliary feedwater flow rate.

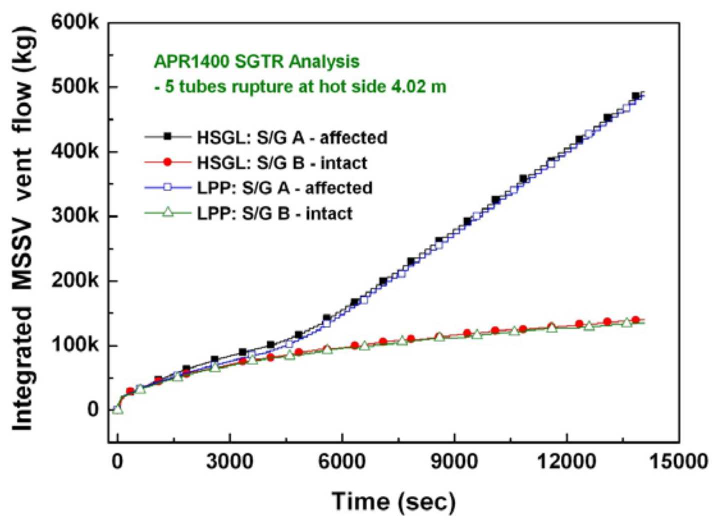

Figure 6 Variation of the accumulated mass of discharged flow through the MSSVs.

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. Figure 7 shows the break flow rates and the accumulated break flow for both reactor trip cases. The initial time was synchronized in Figure 7. Break flow rates show similar trends of the variation of the differential pressure between the primary and the secondary systems. The reactor trip type did not affect the break flow from the primary to the secondary side as shown in Figure 7.

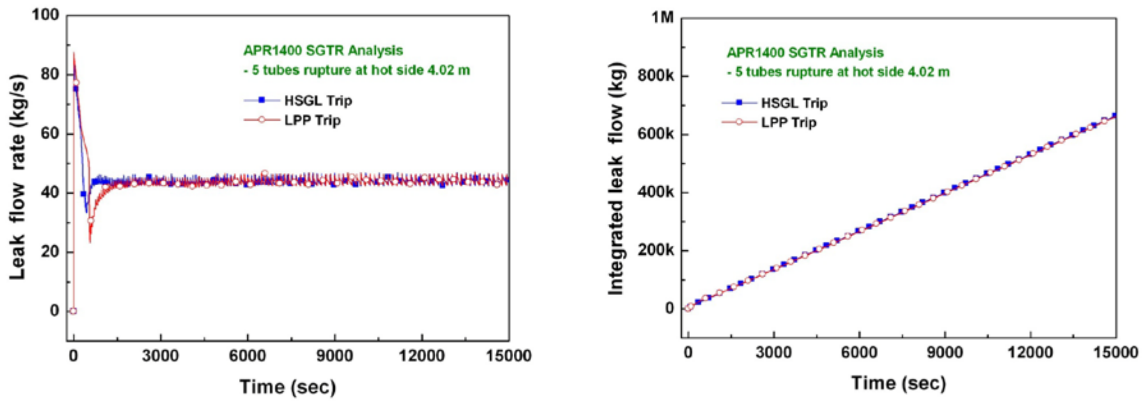

Figure 7 Variation of the break flow rate and the accumulated mass of break flow.

Mild change of the water levels in the core and the down-comer was observed, which was attributed to the relatively small break size of the present SGTR simulation compared with the conventional LOCA (Loss of Coolant Accident) simulations. Since the water level in the core was maintained the initial value, no excursion in the cladding temperature was observed.

3. Conclusion

A postulated steam generator tube rupture (SGTR) accident of the APR1400 was analyzed using the best estimate safety analysis code, MARS. The SGTR accident is one of the design basis accidents having a significant impact on safety in a viewpoint of radiological release. As one of the most limiting SGTR accidents, a break flow equivalent to a double-ended rupture of five U-tubes was analyzed 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 a high steam generator level (HSGL) trip and a low pressurizer pressure (LPP) trip on the thermal-hydraulic system response. As for the tube rupture modeling method, double tube modeling (DTM) was adopted. Broken U-tubes were modeled as an assembly of a single volume. And intact U-tubes were modeled as a separate assembly of a single volume. 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. 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. 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. In the case of the LPP trip, depressurization of the reactor coolant system (RCS) was remarkable simultaneously with the reactor trip which was 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. The reactor trip type did not affect a break flow from the primary to the secondary systems.

4. 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. Vol. 26, 1999, pp. 1611-1642.

[2] J. H. Jeong and K. Y. Choi, “Effects of Tube Rupture Modeling and the Parameters on the Analysis of Multiple Steam Generator Tube Rupture Event Progression in APR1400,” Nuclear Engineering and Design, Vol. 224, 2003, pp. 313 – 336

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