Analysis of a MSLB-induced Steam Generator Tube Rupture Accident for APR1400

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1. Introduction

The main steam line break (MSLB) induced steam generator tube rupture (SGTR) accident, as part of the design extension conditions (DECs) for APR1400 is analyzed in this work.

Based on the probabilistic safety assessment (PSA), total 8 accident scenarios were chosen for the DEC-A conditions. Design extension conditions are accidents that are very unlikely to occur, excess the design basis accident (DBA) limits and in case of the DEC-A scenarios, they do not lead to severe core melting. Those include following scenarios:

1. SBO – Station blackout
2. ELAP – Extended loss of all AC power
3. SBLOCA-LOSIP – Small break LOCA with loss of safety injection pump
4. MSGTR – Multiple steam generator tube rupture
5. MSLB-SGTR – Main steam line break induced steam generator tube rupture
6. SGTR-LOSIP – Steam generator tube rupture with loss of safety injection pump
7. SGTR-FSGI – SGTR with failure of steam generator isolation
8. ELAP-LUHS – ELAP with loss of ultimate heat sink

The MSLB-SGTR accident, as one of accidents included in the DEC-A scenarios, is initiated by a double-ended rupture of the main steam line (MSL) in one of the steam generators (SGs). The water level and pressure of the affected SG decrease quickly due to the rapid release of steam through the rupture in the MSL, which is enhanced by the pressure difference between the secondary side and the containment. This is followed by a single tube rupture in the affected SG. Additional flow from primary to secondary side, which are directly connected during the accident due to the SG tube rupture, leads into decrease of the RCS water inventory and pressure that is directed to the secondary side through the break in the affected SG.

Also, several operator actions are involved in the mitigation and the following actions are assumed:

- Stop two reactor coolant pumps (RCPs)
- Decrease of the safety injection pump (SIP) flow rate to minimize leak from the primary to the secondary circuit

2. Literature review

The final safety analysis report (FSAR), as described in the design control document (DCD) for APR1400 reactor includes anticipated operational occurrences (AOOs) as well as postulated accidents (PAs). However, the accident scenarios belonging under DEC category are not included. Therefore, MSLB and SGTR accidents are simulated there separately and their concurrent occurrence is not simulated. [1]

Although Park et al. [2], [3] investigated the steam line break accident with the steam generator tube rupture, the main focus was to conduct and analyze an experiment in the ATLAS facility, which is a small-scale model of APR1400 used to simulate various scenarios under real conditions.

No further analyses focusing on the analysis of MSLB-SGTR accident using system codes were found and this paper therefore presents the accident analysis using RELAP5 system code. The goal of this research is to find out the plant response and verify if SCS entry conditions are satisfied for successful plant cooldown.

3. APR1400 model

A thermal-hydraulics model of APR1400 plant has been developed for several years and is validated against the final safety analysis report (FSAR), which is also reported in the design control documents (DCD) for US NRC. Major steady-state parameters are listed in Table 1 and are compared to DCD values, showing a reasonable agreement. The model consists of primary and secondary side, with key systems and components that are relevant for the accident simulation. The plant model nodalization is shown in the Figure 1.

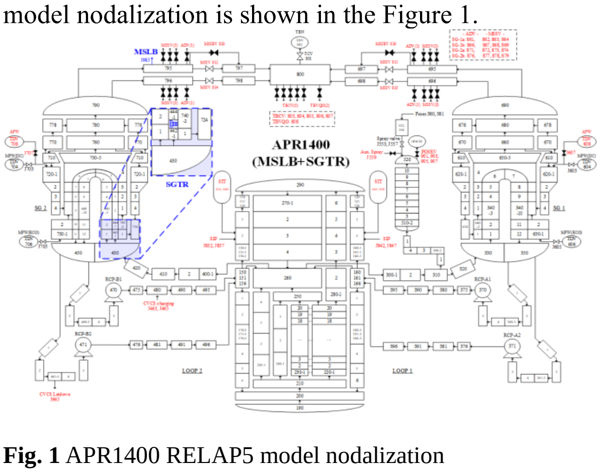

Fig. 1 APR1400 RELAP5 model nodalization

3.1 Primary circuit

The primary circuit consists of the main components, namely reactor pressure vessel (RPV), two vertical steam generators (SGs), each connected by a hot leg (HL) and two cold legs (CLs) to the RPV. Detailed RPV model reflects its real geometry and structure, including the downcomer, lower head, reactor core divided into an average and hot channel, core bypass, upper head and other supportive structures and parts, which together with heat structures model the RPV precisely. On each cold leg, a reactor coolant pump (RCP) is attached, forcing the flow in the reactor coolant system (RCS), allowing the heat generated by fission in the reactor core to be transferred to the secondary side in each SG. Also, a pressurizer (PZR), connected via a surge line to one hot leg, accommodates pressure changes in the primary system and maintains the RCS design pressure. The heat generated in the reactor core is exchanged from the primary to the secondary side by SG U-tubes, which are modeled by a heat structure.

3.2 Secondary circuit

On the secondary side, the two steam generators, as the main part of the nuclear steam supply system (NSSS) are modeled, with their appropriate structures. The main feed water system (MFWS), represented as a time-dependent volume is modeled and serves as the flow boundary, delivering constant feed water flow to each SG. Also, two main steam lines are connected from the upper part of each SG and deliver steam to turbine, which is modeled by another time-dependent volume, representing the second boundary of the secondary side, imposing a constant pressure.

3.3 Safety systems

Several additional systems are modeled to mitigate the accident and safely cooldown the plant. The system diversity and redundancy is therefore granted in case of additional failure.

On the primary side, safety systems include pilot-operated safety relief valve (POSRV), attached to the pressurizer head, protecting the RCS from over pressurization, together with auxiliary spray, which delivers water from the cold leg to decrease the RCS pressure. Another safety systems of the primary side are safety injection tanks (SITs), delivering borated water to maintain the RCS inventory and RPV water level, to protect the core from uncovery, potentially leading into fuel damage and radioactive material release. Those are supported by safety injection pumps (SIPs), forcibly supplying water from the in-containment refueling water storage tank (IRWST). [4]

Passive safety systems of the secondary side consist of main steam safety valves (MSSVs) on each of the main steam lines (MSLs) to maintain the secondary side pressure. Each MSSV operates according to the pressure set point and they are modeled with the conservative minimum mass flow rates, according to the parameters reported in the APR1400 DCD Chapter 10. [5] Main steam isolation valve (MSIVs) and main steam isolation bypass valves (MSIBVs) are included on each MSL to isolate the affected SG and prevent from radioactive material release to the environment. Another valves, atmospheric dump valves (ADVs), which are similar to MSSVs, but their operation is controlled by the operator, are implemented and allow depressurization of the plant via the unaffected SG, as one of the accident mitigation and plant cool down strategies. Lastly, the auxiliary feed water system (AFWS) is modeled by a time-dependent volume on each SG to deliver feed water in case of low SG water level, preventing mainly the unaffected SG from dryout.

Table 1 Steady-state parameters of the plant model

Parameter	DCD	Model
Core power level, MWt	3983.0	3983.0
Pressurizer pressure, MPa	15.51	15.51
Pressurizer lever, %	52.8	50.01
Hot leg temperature, °C	323.9	324.6
Cold leg temperature, °C	290.6	291.7
Total RCS mass flow rate, kg/s	21000.0	20994.7
Steam generator pressure, MPa	6.89	6.57
Feed water flow rate per SG, kg/s	1130.57	1130.13
Steam flow rate per SG, kg/s	1130.56	1130.26
Steam generator water level, %	77.0	77.0

4. Accident description

The MSLB induced SGTR accident combines two DBAs scenarios happening concurrently. Firstly, the break on the main steam line occurs, leading to a steam discharge and SG dryout on the affected steam generator. This is followed by a break in the SG U-tube allowing the primary coolant to leak into the secondary circuit.

4.1 Sequence of events

The sequence of events for MSLB-SGTR accident scenario is following. When the MSLB occurs at 0 seconds, it is assumed that both ends of one MSL on the affected SG break (28 inches) and discharge steam. As a result, the SG pressure decrease in the affected steam generator and reactor and turbine are tripped. When the water level of affected SG reaches 0 %, the SGTR occurs and a break in one of the SG U-tubes is assumed, leading into discharge of the RCS inventory to the secondary side through this break, thanks to a pressure difference. As the RCS pressure decreases rapidly and is followed by decrease in the RCS inventory, the safety injection pumps start operation and deliver coolant to the RCS. Several operator actions are involved in the accident mitigation and plant cooldown, as described in the following chapter.

4.2 Operator actions

Following operator actions and safety systems operation are involved in the plant cooldown and successful accident mitigation:

- Operator action #1: SIP control (Assumption: 30 minutes after accident initiation, set to 10 % of nominal flow rate)
- Operator action #2: Stop RCP (Assumption: stop two RCPs 30 minutes after the accident initiation)
- Operation of SG auxiliary feed water system on the affected SG
- As the pressure of affected SG approaches the atmospheric pressure due to the break on MSL, the RCS also continuously depressurizes by flow of the coolant from the primary to the secondary side through the break in SG tube.
- Supply of cold water through SIPs to maintain RCS inventory and sufficient core cooling
- Reaching the SCS operation entry condition (RCS pressure of 3.099 MPa)

5. Analysis results

The accident analysis was conducted using the RELAP5 system code and results of the major parameters, such as pressure, temperature and mass flow rate are presented in this part.

As the simulation starts and break on MSL occurs, reactor is tripped by the reactor protection system (RPS) and core power decreases, as shown in the Figure 2. Decay heat generated by the fission products is present and continuous cooling of the RCS to dissipate the heat from the reactor core at a sufficient rate is required. This is provided by the RCP and SIP operation on the primary side and by steam discharge in the affected SG on the secondary side. Moreover, after dry-out, break on the affected SG occurs in a single tube and primary coolant leaks to the secondary side due to the pressure difference. As a result, the primary system pressure slowly decreases.

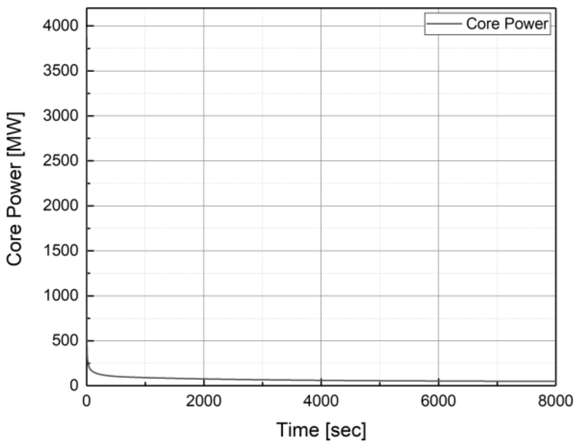

Fig. 2 Core Power

At the first stage of the accident, RCPs are fully operational, RCS pressure decreases due to the reactor trip but then rises again, as the affected SG dries out. Moreover, decay heat is accumulated and SIPs start operation. The pressure then remains around 11 MPa until the operator stops two of the RCPs and decreases SIP flow rate from nominal to 10 % flow after 30 minutes from the accident initiation. The pressure then decreases rapidly, due to the tube break and decreased SIP flow rate. The RCS temperature constantly decreases as the core cooling is provided in a sufficient rate.

Pressure in the affected SG drops immediately after the break on MSL occurs and steam is released. Pressure of the unaffected SG decreases at a slower rate, as the plant is being cooled down mainly by the affected SG. The primary and secondary pressures are shown in Figure 3 and RCS temperatures then in Figure 4.

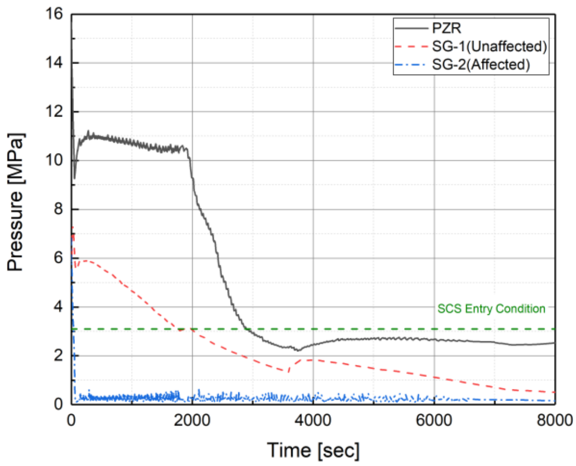

Fig. 3 PZR and SG Pressure

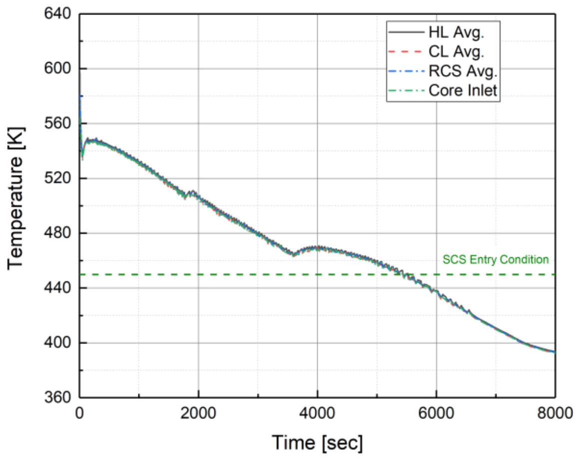

Fig. 4 RCS Temperature

The pressurizer water level follows the RCS pressure and after initial decrease rises again. As the SIP flow rate is decreased by the operator after 30 minutes, the level decreases again. Due to the accumulated decay heat and slower cooldown rate, the PZR level then rises and is maintained around 20 %, as shown in Figure 5.

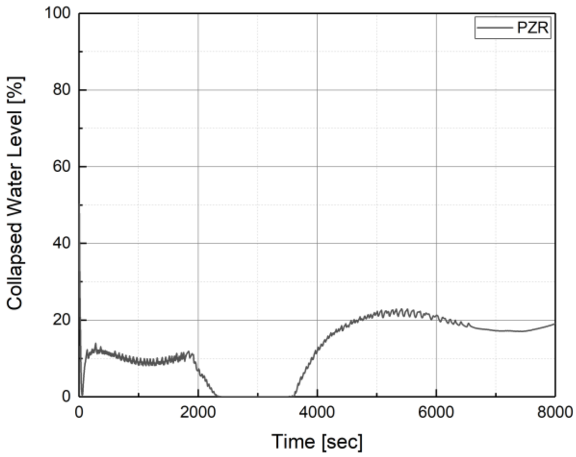

Fig. 5 PZR Collapsed Water Level

The water level of affected SG drops immediately after the break on MSL occurs. Due to the passive auxiliary water system, which delivers water to the affected SG, cooling of the primary side is provided via the affected SG and unaffected SG water level remains constant. The collapsed water level of both SGs for first 30 minutes of simulation is shown in Figure 6.

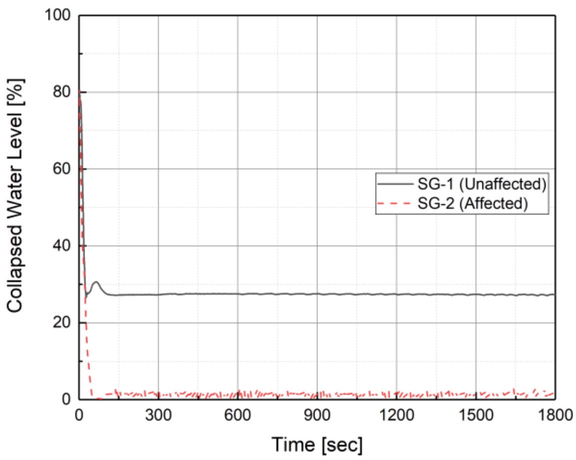

Fig. 6 SG Collapsed Water Levels (first 30 minutes)

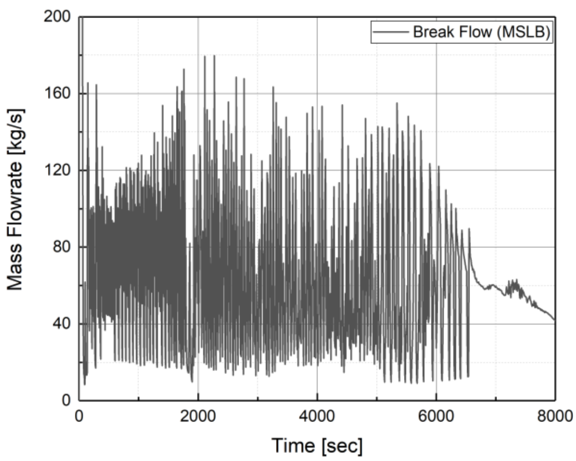

Fig. 7 Mass Flow Rate through MSL Break

The mass flow rate through MSL break is shown in the Figure 7. Break mass flow rate through a ruptured tube in the affected SG is shown in Figure 8. The flow follows the RCS pressure trend, and firstly increases as RCPs and SIPs operate within nominal values and then drops after conducting operator actions to decrease the SIP flow rate and stop two RCPs. Then the break flow slightly increases with time and stabilizes.

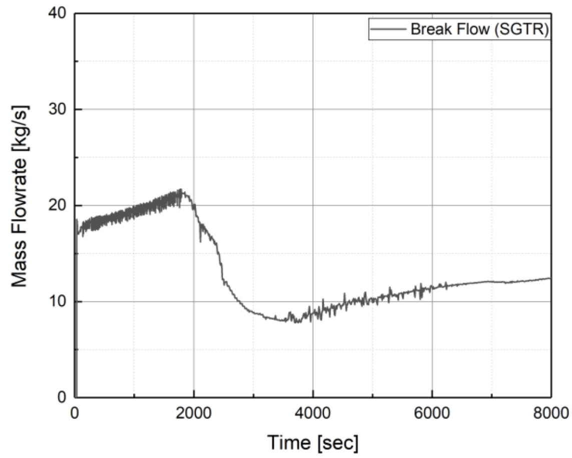

Fig. 8 Mass Flow Rate through SG Tube Break

As suggested during the review process, the DNBR calculation was included in the simulation using control variables and W3 CHF correlation. [6] Due to the early reactor trip and remaining core cooling with full RCP operation, minimum DNBR does not approach safety limits. The DNBR for first 10 seconds of the simulation is shown in Figure 9.

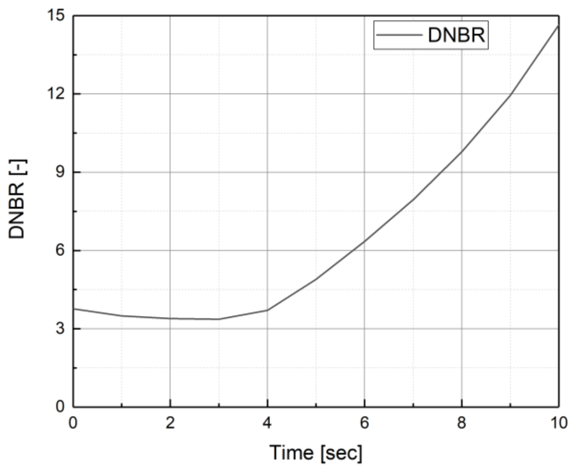

Fig. 9 DNBR for first 10 seconds of simulation

6. Conclusions

The main steam line break induced steam generator tube rupture (MSLB-SGTR) accident, as part of the DEC-A scenarios for APR1400, is analyzed in this work, using RELAP5 system code.

The results of the analysis confirm that the APR1400 plant can withstand MSLB-SGTR accident, when proper operator actions are involved in the mitigation process. The safety functions are preserved and cooling of the reactor core by sufficient supply of the coolant by safety systems ensure the water level in the RPV, RCS inventory and pressure, together with cooling of the secondary side for reaching the SCS operation entry conditions and successful plant cooldown.

Further accident investigation including sensitivity analysis on accident and plant cooldown strategy is considered as part of the future research.

Acknowledgement

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