Preliminary Analysis of Steam Generator Tube Rupture accident Induced by Severe Accident for OPR1000 Plant

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1. Introduction

After Fukushima accident, a lot of countries make an effort to ensure the ability to cope with severe accident. In Korea, regulations on the severe accident were legislated at June 2016. The regulation requires that the cumulative frequency of the accident with Cs-137 emissions greater than 100 TBq should be less than 10^{-6} per reactor year. It means that minimizing the release of fission product outside plant is essential to protect public health and minimize environmental damage.

There are many tries to prevent the fission product release with ensuring the containment integrity, and it increases the safety of nuclear power plant. However, in case of containment bypass accident such as steam generator tube rupture (SGTR) and interface system loss of coolant accident (ISLOCA), the damage would be severe because the radioactive nuclides are released to environment directly. In this reason, USNRC has studied on the containment bypass accident, especially SGTR, and it is indicated that the probability of the SGTR is not small [1-3].

In this study, SGTR accident induced by severe accident for OPR1000 plant was calculated with MELCOR and the result analysis was conducted. In addition, KAERI will perform the experiment to evaluate the risk of fission products during SGTR. The products of this study will provide a guideline for the experiment conditions.

2. Calculation conditions

It was assumed that the reactor trip occurred with SBO scenario in this calculation, and SGTR was found with creep rupture as the accident progression. MELCOR 2.2_9607 was used as a calculation tool and MELCOR input to make natural circulation from core to steam generator was described in this section [4, 5].

2.1. Plant nodalization

It was defined by U.S.NRC that one of the uncertain factor on severe accident phenomenon is natural circulation in reactor coolant system (RCS) [6]. In the severe accident condition, decay heat in core is transferred to other part with natural circulation, and it could increase the possibility of SGTR before failure of the lower head vessel.

In this calculation, hot leg and steam generator tube were divided into two parts to make natural circulation, as shown in Fig. 1. Inlet plenum of steam generator was divided into three parts to simulate 1) flow rate from hotleg to steam generator, 2) flow rate after mixing, and 3) flow rate from steam generator to hotleg without mixing.

To simulate temperature induced SGTR, a flow path was modeled from SG tube to SG secondary system, and it was assumed that the path should be fully open if the tube creep rupture condition is satisfied. Main steam safety valves (MSSVs) and atmospheric dump valves (ADVs) were also modeled with considering the operating conditions.

2.2. Accident scenario

Accident scenario considered in the calculation is summarized in Table 1. At time of 0 s, reactor trip occurred with station black out (SBO) accident. The core cooling was conducted until the coolant was remained in the SG secondary vessel. After dryout of the SG secondary volume, water level in the reactor pressure vessel decreased with operation of SRV at 12,465 s. Fuel temperature start to increase and core support plate failure was observed at 22,513 s. Finally, SG tube rupture occurred and after that lower head vessel failure was found. As the reactor vessel pressure decreased, safety injection tank (SIT) injection was initiated, however, fuel debris cooling was not proper because of vessel failure. The calculation was finished at 30,000 s.

Table 1 Summary of accident scenario

Event	Time (s)
Reactor trip (SBO)	0.0
MSIV close	0.0
MSSV stuck open	9000.0
SG secondary dryout	9044.3
First operation of SRV	12465.8
Fuel gap release	18587.2
Core support plate failure	22513.5
SG tube rupture (1 tube)	24937.4
Lower head failure Debris ejection to cavity	25786.5
SIT injection start	25806.7
End of calculation	30000.0

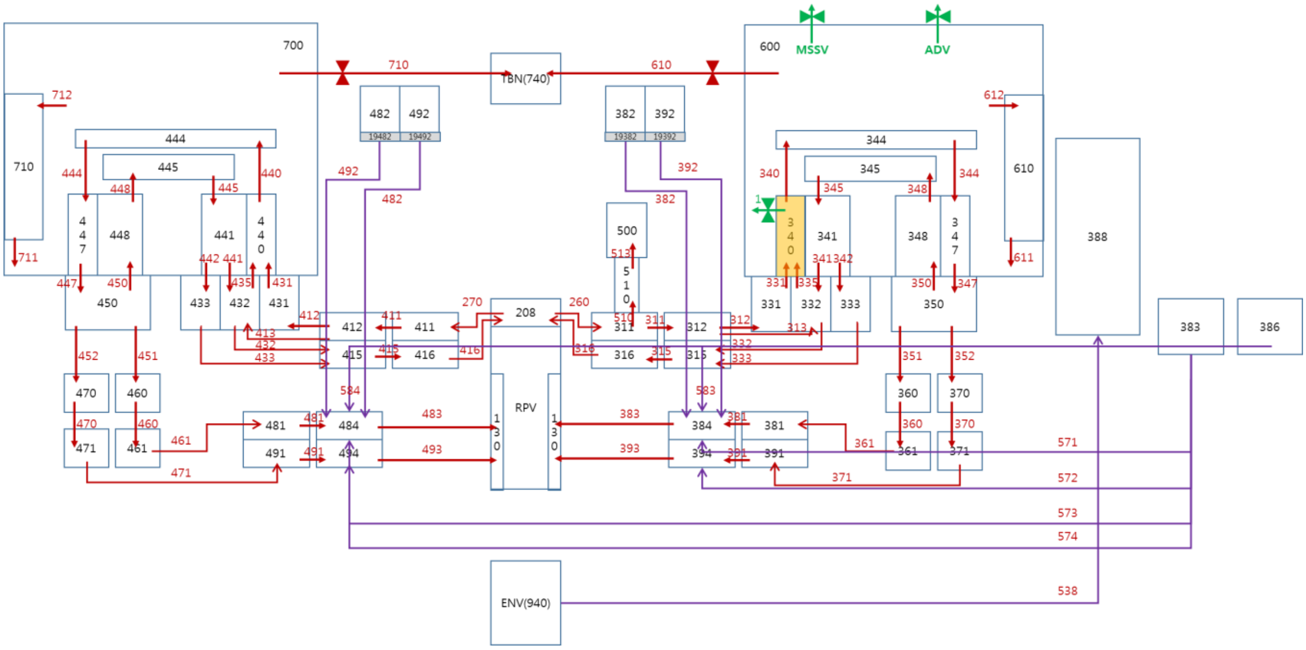

Figure 1 OPR1000 nodalization for MELCOR calculation

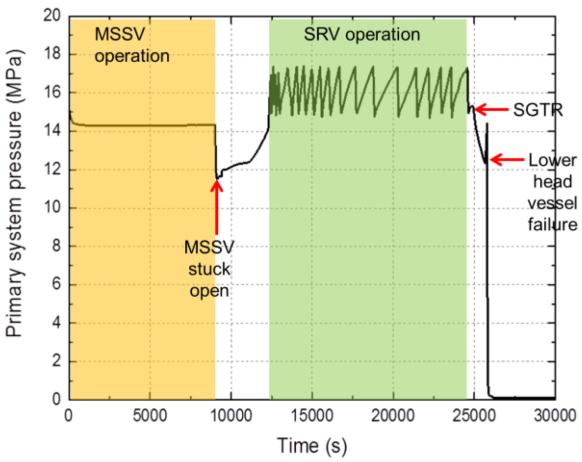

Figure 2 Primary system pressure

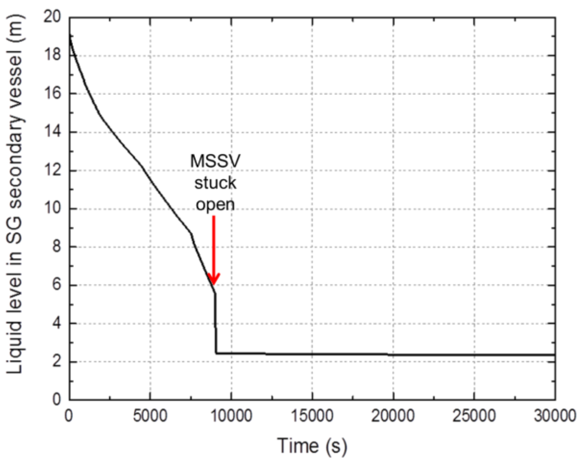

Figure 3 Water level in SG secondary side

3. Results and analysis of accident

Primary system pressure during accident was indicated in Fig. 2. Before 9,000 s, core decay heat was removed with MSSV operation and the water level in the SG secondary side decreased as shown in Fig. 3. At 9,000, RCS pressure was decreased sharply as the MSSV was stuck open. This is because that the gas temper ature in the SG secondary side was decreased with depressurization. As the fuel temperature increased, the RCS pressure increased again and the water level in the reactor vessel decreased with SRV operation. As shown in Fig. 4, the water level reached to bottom of active fuel at about 18,000 s. Decay heat in the core was transferred to the steam generator and the SG tube temperature increased and SGTR with creep rupture was observed at about 25,000 s, as indicated in Fig. 5.

The thermo-hydraulic conditions during the SGTR period are important to plan the experimental conditions to be conducted in KAERI. The SG pressures of primary and secondary are indicated in Fig. 6 and the flow rate through the ruptu re area was shown in Fig. 7. In this calculation, only one tube rupture was assumed.

4. Conclusion

Experiment for evaluating the risk of SGTR accident induced by severe accident will be performed in KAERI. To design the test condition, analytical study was preceded with MELCOR. MELCOR input was developed to simulate natural circulation that could simulate the temperature induced SGTR. Some works are still remained with changing conditions, such as

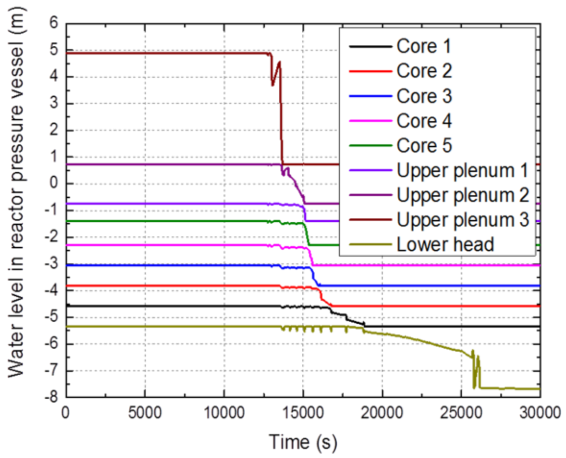

Figure 4 Water level in reactor pressure vessel

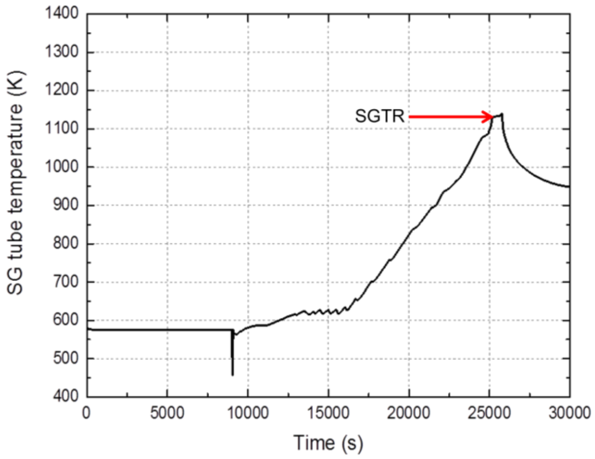

Figure 5 SG tube temperature

different accident scenario, the number of ruptured tube, and the amount of released fission products.

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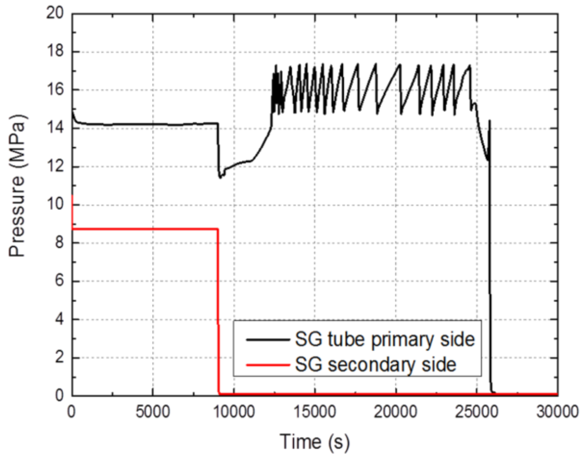

Figure 6 SG pressure of primary and secondary side

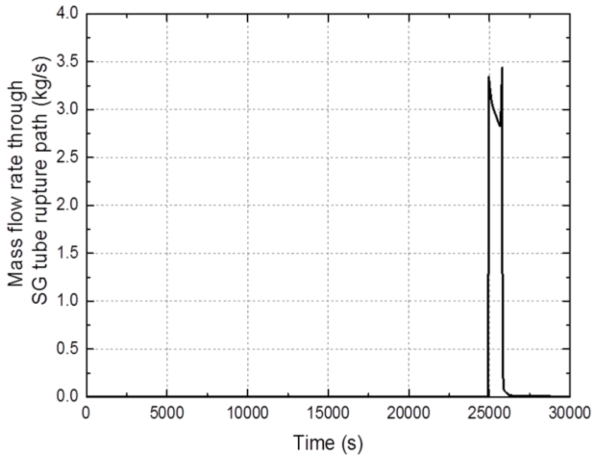

Figure 7 Mass flow rate through rupture area

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