Transactions of the Korean Nuclear Society Autumn Meeting
Changwon, Korea, October 20-21, 2022

Evaluation for MSGTR Accident Mitigation following Operator Action using RELAP5/MOD3.3

Seong-Su Jeon [a*], Jae-Ho Bae [a], Jungjin Bang [a], Do Hyun Hwang [b]
[a]*FNC Technology Co., Ltd., 32F, 13 Heungdeok 1-ro, Giheung-gu, Yongin-si, Gyeonggi-do, 16954*
[b]*KHNP Central Research Institute, 70, 1312-gil, Youseong-daero, Yuseong-gu, Daejeon, 34101*
[*]Corresponding author: ssjeon@fnctech.com

1. Introduction

After the Fukushima Daiichi plant accident in 2011, concerning the safety of Nuclear Power Plant (NPP) in extreme external events has increased. Also, the concept of Design Extension Condition (DEC) have been introduced, and there is a demand for accident prevention and mitigation under DEC from the step of design [1].

Among DEC-A accidents, Multiple Steam Generator Tube Rupture (MSGTR) is an accident when 2 or more U-tubes are broken simultaneously in a single SG. The 5 tubes rupture is considered as MSGTR accident by regulatory guideline in Korea [2]. When MSGTR occurs, the discharge flow and the released radioactive materials from the Reactor Coolant System (RCS) are relatively larger. Thus, the accident proceeds more rapidly compared to SGTR. If the initial actions to mitigate the accident are not properly taken, the large amount of radioactive materials could be discharged into the atmosphere through the Main Steam Safety Valve (MSSV) opening. However, MSGTR has not been taken into account in depth in the NPP design and only few researches have been conducted as part of SGTR because of its low frequency of occurrence.

This study developed MSGTR analysis model of 1000 MWe Pressurized Water Reactor (PWR). And we conducted MSGTR simulation following operator action in order to effectively mitigate the accident using RELAP5/MOD3.3 [3].

2. Modeling for MSGTR Analysis

For MSGTR analysis using RELAP5 code, this study used a model of a 2-loop 1000MWe PWR as shown in Fig. 1. The Reactor Pressure Vessel (RPV), Pressurizer (PZR), RCP, hot and cold-leg, and SG U-tube were modeled as the primary side. The SG, main feedwater system, main steam line, turbine, etc., are contained as modeling on secondary side. In addition, safety systems (e.g. High-Pressure Safety Injection (HPSI), Low-Pressure Safety Injection (LPSI), Safety Injection Tank (SIT), and Aux-Feedwater System (AFWS)) were included for mitigating accidents. The analysis for MSGTR among DEC-A was performed using Best Estimate (BE) analysis methodology with realistic assumptions and conditions [1, 4]. Also, the operator actions for mitigation of accident were considered on this analysis. Thus, the analysis model contained PZR Pressure Control System (PPCS), PZR Level Control System (PLCS), Feedwater Control System (FWCS), and SBCS. The components and systems, which include Main Steam Isolation Bypass Valve (MSIBV), PZR aux-spray, Atmospheric Dump Valve (ADV), and SG Blowdown (SGBD) considered as operator action, were also added. To follow realistic assumption for MSGTR analysis, the nominal values at 100 % core power condition were assumed to be the initial and boundary conditions.

The simulation of MSGTR was initiated by assuming that the rupture occurs at 0 sec, and 5 U-tubes are instantaneously broken in the hot-leg side of SG-2. The rupture was modeled as double-ended guillotine break. It was assumed that the first operator action, the RCP trip, is performed 10 min after the reactor trip. Afterward, it is assumed that the operator conducts a procedure of actions to mitigate the MSGTR 15 min after the reactor trip [5]. The operator action time for each procedure was considered to be 2 min.

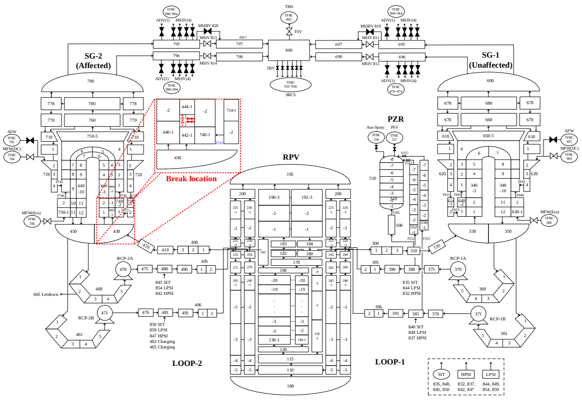

Fig. 1. RELAP5 nodalization for MSGTR analysis

3. Simulation Results of MSGTR Analysis

The simulation results of MSGTR in this study are shown in Figs. 2 and 3. MSGTR accident progresses as follows: after 5 tubes rupture happen at 0 sec. The RCS coolant discharges to SG. The reactor trip occurs by hot-leg saturation temperature set-point signal at 71 sec. Following the loss of RCS inventory, PZR pressure and level decrease. Therefore, the charging flow increase, and PZR heater turns on to compensate for this. But, PZR level decreases below the set-point on PZR heater off, and PZR pressure rapidly drops. The turbine trip occurs due to the reactor trip, and steam of SG is automatically controlled and released to condenser by

SBCS. After the turbine trip, the SG pressure is maintained at about 8 MPa by SBCS.

The RCS pressure continuously decreases and HPSI starts at 127 sec. As HPSI starts, the PZR pressure maintains at about 9 MPa, and the depressurization does not proceed anymore. After 10 min of reactor trip, the operator stops the all four RCP in consideration of RCS pressure and sub-cooling. Due to the tubes rupture, the inventory of affected SG continues to rise. The level of the affected SG rises faster than the unaffected SG, and Main Steam Isolation Signal (MSIS) is generated by the SG high-level at 732 sec. The SG isolation by MSIS causes the stop of RCS heat removal. So the pressure of RCS and affected SG increases.

At 15 min after the reactor trip (i.e. 971 sec), since Main Steam Isolation Valve (MSIV) is closed by MSIS, MSIBV is opened to release steam from SG to condenser using SBCS. This operation conducts for RCS temporary-cooldown. Therefore, SG is depressurized to rapidly decrease RCS temperature to MSSV opening prevention temperature with maximum RCS cooldown rate (55.6 K/hr). Two minutes later, PZR aux-spray is operated to make pressure balance on the PZR and the affected SG and to depressurize the RCS. At 2,378 sec, the hot-leg temperature reaches 558.15 K, and the operator closes MSIBV. After that, the operator identifies and isolates the affected SG. After MSIBV is closed, the SG pressure starts to rise again.

The affected SG level reaches 100 % or higher than the Wide Range (WR) level. So, SGBD is operated at 2,618 sec. Two minutes after the SGBD operation, at 2,738 sec, ADV of unaffected SG is opened to perform RCS controlled-cooldown. The affected SG pressure increases by isolation and becomes the same as the PZR pressure. The pressure of PZR and affected SG is balanced at approximately 3,500 sec. And depressurization is occurred through controlled-cooldown using ADV and PZR aux-spray. The RCS pressure is hardly decreased because of natural circulation cooling by all RCPs stop. After one hour of ADV opening, because the operator restarts one RCP per loop, the RCS pressure rapidly decreases. In about 11,000 sec, the RCS pressure and temperature reach the Shutdown Cooling System (SCS) entry condition. It can be evaluated that the MSGTR accident is well mitigated.

4. Conclusions

Since MSGTR progresses rapidly due to relatively higher break flow rate than in SGTR, the appropriate operator action is crucial to prevent the release of radioactive materials into the environment. In order to effectively mitigate the accident, it is necessary for the operator to properly perform each operator action within the appropriate time. In this study, MSGTR analysis model was developed using the RELAP5/MOD3.3 code. And MSGTR analysis with considering operator action was conducted. As a result, it was evaluated that MSGTR is well mitigated by proper operator action and appropriate time.

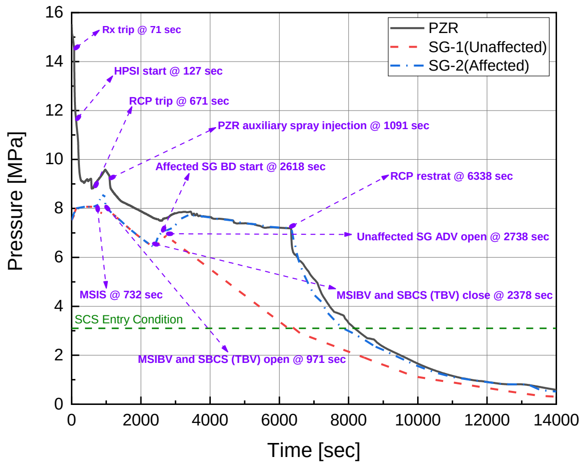

Fig. 2. MSGTR Simulation results: PZR and SG pressure

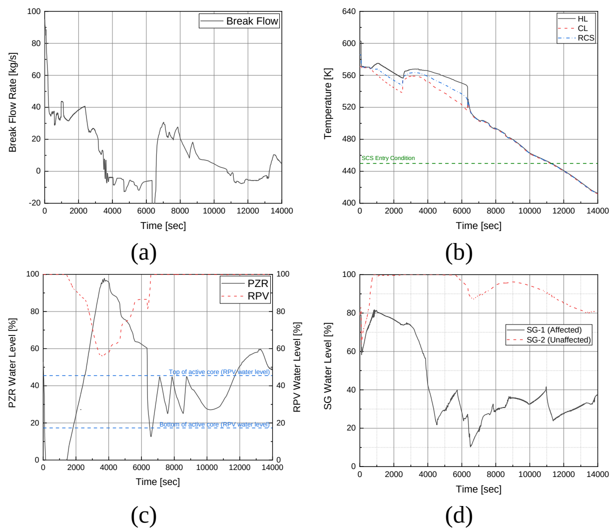

Fig. 3. MSGTR Simulation results: (a) break flow, (b) RCS temperature, (c) PZR and RPV level, (d) SG level

ACKNOWLEDGEMENT

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) and the Ministry of Trade, Industry & Energy (MOTIE) of the Republic of Korea (No. 20208540000020).

REFERENCES

[1] IAEA, Safety of Nuclear Power Plants: Design, SSR-2/1, Rev.1, 2016.
[2] KINS, Regulatory Guideline 16.1, "Assessment of accidents due to multiple failures", KINS/RG-N16.01, Rev.0, 2016.
[3] USNRC, RELAP5/MOD3.3 Code Manual, Information Systems Laboratories, Inc., USA, 2016.

[4] USNRC, Policy, technical, and licensing issues pertaining to evolutionary and advanced light water reactor (ALWR) designs, SECY-93-087, 1993.
[5] Yoon, B.K. et al., Analysis of Multiple Steam Generator Tube Rupture with Emergency Operating Guidelines for APR1400, Transaction of the Korean Nuclear Society Virtual Autumn Meeting, October 21-22, 2021.