A Comparison of Nuclear Power Plant Simulator with RELAP5/MOD3 code about Steam Generator Tube Rupture

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1. Introduction

A nuclear power plant simulator is a training device for operator license examination, requalification, and verification of emergency operating procedures under the circumstance similar to practical plant [1]. The United States began to consider how to regulate the nuclear power plant simulator after the TMI accident. Licensees in the U.S. are required to submit the results of the simulator performance test on the basis of the NRC (Nuclear Regulatory Commission) form 474 once every four years in accordance with 10CFR55.45 (b). In case of the Korea, licensee makes its own simulator performance tests on each training center, but independent verification of performance test results is not performed by any organization [2, 3].

The RELAP5/MOD3 code introduced in cooperation with U.S NRC has been utilized mainly for validation calculation of accident analysis submitted by licensee in Korea. The Korea Institute of Nuclear Safety has built a verification system of LWR accident analysis with RELAP5/MOD3 code engine [4].

Therefore, the simulator replicates the design basis accident and its results are compared with RELAP5/MOD3 code results that will have important implications in the verification of the simulator in the future. The SGTR simulations were performed by the simulator and its results were compared with ones by RELAP5/MOD3 code in this study. Thus, the results of this study can be used as materials to build the verification system of the nuclear power plant simulator.

2. Methods and Results

2.1 Initial conditions

The initial conditions that were used for this analysis were set equal to the initial state of the SGTR of the OPR-1000 FSAR (Ulchin units 3 and 4) that is assumed by loss of offsite power (LOOP) and single failure. At the beginning of the accident the thermal power was 2871.3 MW (102% of the rated full power). Table 1 shows detailed initial conditions and major parameters in the OPR-1000, RELAP5/MOD3 and simulator in Yonggwang training center well agreed with the measured values. Assumptions used to make the analysis are as follow [4, 5].

1) It is assumed the LOOP due to the unstable power grid after 3 seconds following turbine trip.
2) The turbine operator opens main steam atmosphere dump valve (MSADV) to decrease RCS temperature below 550˚F.
3) The MSADV is stuck open due to the single failure.

Table 1.Initial Conditions for OPR-1000, RELAP5 and Simulator

Major Parameters	OPR-1000	RELAP5	Simulator
Core thermal power [MWt]	2871.3	2871.3	2815
Core inlet temperature [˚F]	570	570.9	564.5
Pressurizer pressure [psia]	2325	2325	2250
Pressurizer water volume/level [ft^3, %]	1038	1041.9	52.6 (%)
SG pressure [psia]	1148	1148	1070
SG water volume/level [lbm, %]	184000	182190	79 (%, WR)

2.2 Comparisons of simulation results of the simulator with RELAP5/MOD3 CODE

We have replicated the pressurizer pressure/level and steam generator pressure/level that are the main parameters associated with the SGTR.

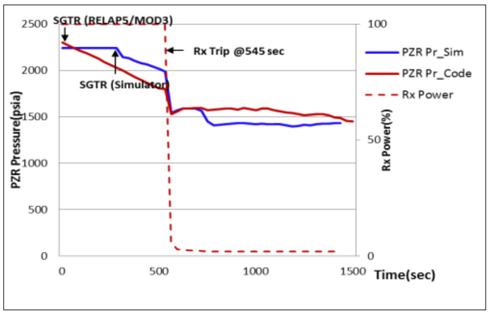

Fig. 1.Pressurizer (PZR) pressure during the SGTR

The reactor coolant system (RCS) leakage to steam generator through ruptured tube is much greater than the amount of coolant water supplied from charging pump.

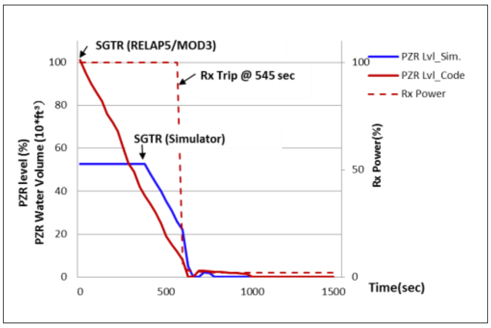

Fig. 2.Pressurizer (PZR) level during the SGTR

Therefore the Pressurizer (PZR) pressure and level gradually decreases as shown in figures 1 and 2. Following a reactor trip at 545 seconds, the PZR pressure rapidly drops until safety injection initiates as shown in figure 1.

The PZR pressure obtained with the simulator in the initial accident decreased relatively a little. The reason is that the PZR back-up heaters were turned on to maintain the pressure of the PZR by the pressurizer pressure control system (PPCS).

In particular, the pressure in the broken SG first increased in early phase of the transient in the simulator. We estimated that the increasing SG level from RCS leakage could increase SG pressure with slow feedwater control system (FWCS) and steam bypass control system (SBCS) response. But the broken SG pressure by RELAP5/MOD3 code decreases until reactor and turbine trip as the RELAP5/MOD3 code could not adopt the FWCS and SBCS in detail. In addition, as the RCS temperature is going down, the broken SG pressure by RELAP5/MOD3 also decreases. The broken SG pressure continues to increase due to the closing of the turbine throttle valves after reactor and turbine trip as shown in figure 3.

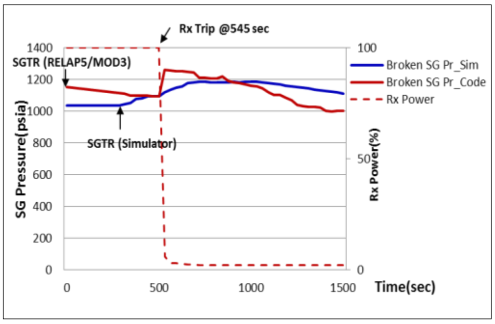

Fig. 3.Broken SG pressure during the SGTR

For that reason, the SG pressures between the simulator and RELAP5/MOD3 code show different direction until reactor trip. However the SG pressure from the simulator and RELAP5/MOD3 after reactor trip generally agreed. As a result of the loss of normal AC power, electrical power would be unavailable for the station auxiliaries such as the reactor coolant pumps, the main feedwater pumps. Under such circumstances the plant would experience a loss of normal feedwater flow, condenser vacuum, and ability to control of SG pressure [5]. That's why the broken SG pressure of two methods increases until main steam safety valves open.

After the reactor trip, the SG level of two methods is in direct opposition to each other as shown in figure 4. The opening period of the main steam safety valve (MSSV) in the simulator was relatively late, compared with that of RELAP5/MOD3 code. Because of this, the simulator SG inventory released to the atmosphere is smaller than the SG level increases by ruptured tube.

However, the SG inventory with RELAP5/MOD3 code affected by loss of steam, feedwater flow and relatively long time to the MSADV stuck open could cause SG water level decreases.

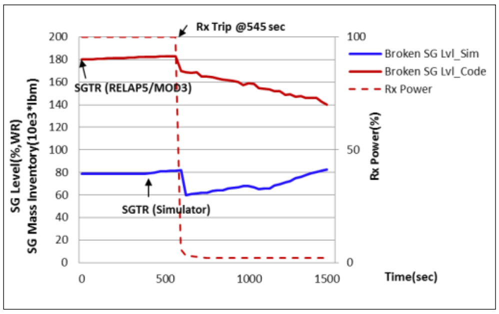

Fig. 4.Broken SG level during the SGTR

3. Conclusions

We tried to compare with RELAP5/MOD3 verification code by replicating major parameters of steam generator tube rupture using the simulator for OPR-1000 in Yonggwang training center. By comparing the changes in temperature, pressure and inventory of the reactor coolant system and main steam system during the SGTR, it was confirmed that the main behaviors of SGTR which the simulator and RELAP5/MOD3 code showed are similar. However, the behavior of SG pressure and level that are important parameters to diagnose the accident were a little different. We estimated that RELAP5/MOD3 code was not reflected the major control systems in detail, such as FWCS, SBCS and PPCS. The different behaviors of SG level and pressure in this study should be needed an additional review.

As a result of the comparison, the major simulation parameters behavior by RELAP5/MOD3 code agreed well with the one by the simulator. Therefore, it is thought that RELAP5/MOD3 code is used as a tool for validation of NPP simulator in the near future through this study.

ACKNOWLEDGMENTS

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