Benchmarking Simulation of Long Term Station Blackout Events

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1. Introduction

Station blackout (SBO) events occurred due to a historic tsunami at the Fukushima Daiichi nuclear complex. The residual heat removal (RHR) systems functioned as designed for some periods following the shutdown of the reactors. However, some systems degraded over some periods after the accident. Therefore, the assessment for degradations of the systems related to the SBO events should be considered.

The importance of passive cooling systems has emerged since the SBO events. Turbine-driven auxiliary feedwater (TD-AFW) system is the only passive cooling system for steam generators (SGs) in current PWRs. During SBO events, all alternating current (AC) and direct current (DC) are interrupted and then the water levels of steam generators become high. In this case, turbine blades could be degraded and cannot cool down the SGs anymore. To prevent this kind of degradations, improved TD-AFW system should be installed for current PWRs, especially OPR 1000 plants. A long-term station blackout (LTSBO) scenario based on the improved TD-AFW system [1] has been benchmarked as a reference input file.

The following task is a safety analysis in order to find some important parameters causing the peak cladding temperature (PCT) to vary. This task has been initiated with the benchmarked input deck applying to the State-of-the-Art Reactor Consequence Analyses (SOARCA) Report [4].

2. Methods

2.1 Modification of input deck to design SBO events

In this section, the benchmarking of SBO transient calculations has been performed with the MARS code. The initial MARS input deck simulated a large break loss-of-coolant accident (LBLOCA) [2].

The followings show the modification of the input deck of LBLOCA to model SBO events.

- Reactor trip due to a pressure drop of the pressurizer (PRZ) below 12.1 MPa
  → Reactor trip at the beginning
- Cold leg or hot leg break at 0.01 second
  → No leg break
- Motor-driven auxiliary feedwater (MD-AFW) actuates at 2 seconds after the main feedwater isolation
  → No actuation
- High pressure safety injection (HPSI) and low pressure safety injection (LPSI) actuate at 30 seconds after the pressure of the PRZ becomes lower than 12.1 MPa
  → No actuation
- Safety valve in PRZ is always closed
  → open when the pressure of PRZ becomes higher than 2500 psia and closed when the pressure of PRZ becomes lower than 2375 psia
- TD-AFW actuates 2000 seconds after the main feedwater isolation
  → actuates when the wide range water level of the SG is lower than 80% until the total used coolant inventory in the condensate storage tank (CST) reaches 10^6 kg
- The flowrate of the TD-AFW is 45.48 kg/s after 5 seconds from actuation
  → dependent on the pressure of the SG (linearly increasing from 30 kg/s at 1.5 MPa to 40 kg/s at 8.0 MPa)

2.2 Modeling for uncertainty quantification

To initiate studies for uncertainty quantification, the SOARCA has been reviewed. Based on the SOARCA, the TD-AFW is available until the battery depletes or CST empties. Batteries typically last for 2 to 8 hours as indicated in the SOARCA. In this paper, batteries are assumed to last for 2 to 8 hours.

Another case is concerned about a loss of pump seal cooling causing a reactor coolant pump seal to leak. In the previous cases, The RCPs have been unavailable. However, in this case, the RCPs are assumed to be available for 8 hours by DC station batteries. The RCP seals initially leak at 21 gpm/pump (= 1.3249 kg/s) assumed in the SOARCA. Finally, the RCP seals fail at 14 hours and 46 minutes and leak at nominal rate of 182 gpm (11.48 kg/s) per pump after failure. Actually, in this paper, some relevant results for the leakage of the RCP seals are not included because the actual modeling of the leakage of the RCP seals is somewhat troublesome. The results could be included in the next paper.

3. Results

3.1 Benchmarking results

The results of the benchmarking show the same trend as performed in a recent paper [1]. Figure 1 is the most important results to show the water levels of the SG and PRZ during an SBO accident.

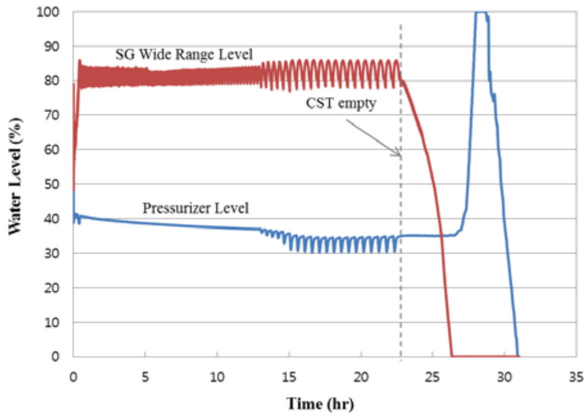

Figure 1. Water levels of the SG and PRZ

The water level of the SG is controlled to be around 80% of the normal operation level. The design criteria of the improved TD-AFW system result in the oscillation of the water level of the SG. The water level of the pressurizer increases after the SG inventory is depleted. Eventually, the flow rate of the reactor coolant system (RCS) in cold legs decreases due to a loss of coolant through the PRZ safety valve (PSV). Consequently, the fuel cladding temperature increases rapidly resulting in the damage of fuel.

3.2 Time dependent failures of the TD-AFW

In this section, the simulation has been performed with the cases that the failure time of the TD-AFW varies depending on the battery depletion.

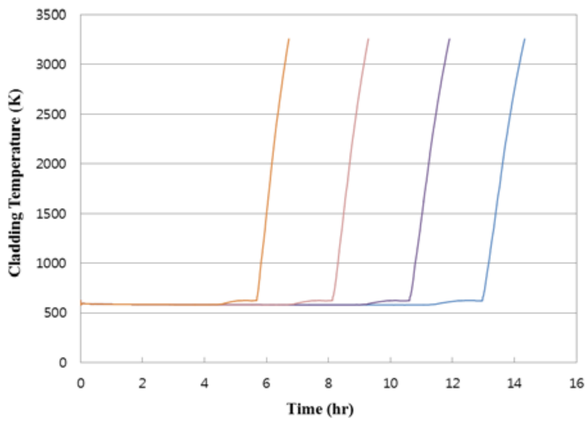

Figure 2. PCT depending on the battery depletion

Figure 2 shows the peak clad temperature (PCT) depending on the battery depletion from 2 to 8 hours with 2 hour intervals. The PCTs exceed 1200 °C of the safety criteria earlier than the case of the TD-AFW supplied by the CST. Another important thing is that the PCT increases in 3 – 4 hours after the battery depletion. This means that the operator has some time to control during that period even though the battery is already depleted.

4. Conclusions

The point of the improved TD-AFW is to control the water level of the SG by using the auxiliary battery charged by a generator connected with the auxiliary turbine. However, this battery also could be disconnected from the generator. To analyze the uncertainties of the failure of the auxiliary battery, the simulation for the time-dependent failure of the TD-AFW has been performed.

In addition to the cases simulated in the paper, some valves (e.g., pressurizer safety valve), available during SBO events in the paper [1], could be important parameters to assess uncertainties in PCTs estimated. The results for these parameters will be included in a future study in addition to the results for the leakage of the RCP seals.

After the simulation of several transient cases, alternating conditional expectation (ACE) algorithm [5] will be used to derive functional relationships between the PCT and several system parameters.

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REFERENCES

[1] S.W. Lee, K.D. Kim, S.W. Bae, and B.D.Chung, "Study on the PWR Steam Generator Behavior with Improved Steam-Driven Auxfeedwater System under Prolonged SBO Accident", 18th Pacific Basin Nuclear Conference, Busan, Korea, 2012.

[2] D. A. Fynan, K.-I. Ahn, H.-G. Lim, and J. C. Lee, "Uncertainty Quantification of LBLOCA PCT for a Pressurized Water Reactor by ACE Algorithm and Gaussian Process Model", 18th Pacific Basin Nuclear Conference, Busan, Korea, 2012.

[3] KAERI, "MARS Code Manual Volume I: Code structure, System Models, and Solution Methods," KAERI/TR-2812/2004, Korea Atomic Energy Research Institute, 2009.

[4] USNRC, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report", NUREG-1935, U.S. Nuclear Regulatory Commission 2012.

[5] H. G. Kim and J. C. Lee, "Development of a Generalized Critical Heat Flux Correlation through the Alternating Conditional Expectation Algorithm", Nuclear Science and Engineering, **127**, 300, 1997.