Success Criteria Analysis for PSA of Hanul Units 3&4: Results and Insights

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1. Introduction

Recently, the probabilistic safety assessment (PSA) model for the Hanul units 3&4 is being re-developed in KAERI as the one of the national project aiming at increasing and assuring the safety of the nuclear power plant (NPP). The success criteria (SC) analysis of the at-power Level 1 internal accident of the PSA model was also conducted in this project.

This paper deals with the SC analysis of the Level 1 internal accident of PSA model for Hanul units 3&4. From the SC analysis, the valuable findings and insights were obtained.

2. Methods

A TH calculation for SC analysis has been performed with the MARS (Multi-Dimensional Analysis of Reactor Safety) code as a best-estimate TH analysis computer code. The MARS code has been developed for a realistic analysis of two-phase thermal-hydraulic transients for pressurized water reactor (PWR) plants.

The Hanul NPP units 3 and 4 is an OPR-1000 (Optimized Power Reactor) type. The OPR-1000 is a two-loop 1000MWe PWR generation-II nuclear reactor [3]. The MARS model for the Hanul NPP units 3 and 4 is illustrated in Fig. 1 [4]. It consists of two SGs (steam generators), a pressurizer, four RCPs (Reactor Coolant Pump), HPSI pumps, LPSI (Low Pressure Safety Injection) pumps, AFW (Auxiliary Feed Water) pumps, four MSSVs (Main Steam Safety Valves), four MSIVs (Main Steam Isolation Valves), four ADVs (Atmospheric Dump Valves), a PSV (Pressurizer Safety Valve), and an SDS (Safety Depressurized System) valve. With regard to the major contributor to the reactor transients, the following have been modeled.

- One HPSI pump (1 out of 2 trains) is available and SIAS (Safety Injection Actuation Signal) is generated at 124kg/cm^2 of the RCS pressure, and the delay time of the injection is 30 seconds.
- One LPSI pump (1 out of 2 trains) is available.
- One AFW pump (1 out of 4 pumps) is available and AFAS (Auxiliary Feed Actuation Signal) is generated at below 23.5% of the SG wide-range level, and the delay time of the injection is 45 seconds.
- The temperature of the injection water by the HPSI and AFW pumps is 30^{o}C.
- Four RCPs are automatically shut down at below 15^{o}C of the sub-cooled margin.
- MSIV is closed at below 62kg/cm^2 of SG pressure.
- PSV is gradually opened from 86.2e^{5}Pa to 88.8e^5 Pa of RCS pressure.
- While the RCS pressure increases, the MSSV is suddenly opened at 1.75e^7 Pa, and while the RCS pressure decreases, MSSV is gradually closed at up to 80% from 1.75e^7 Pa to 1.43e^7 Pa and suddenly closed at 1.43e^7 Pa.

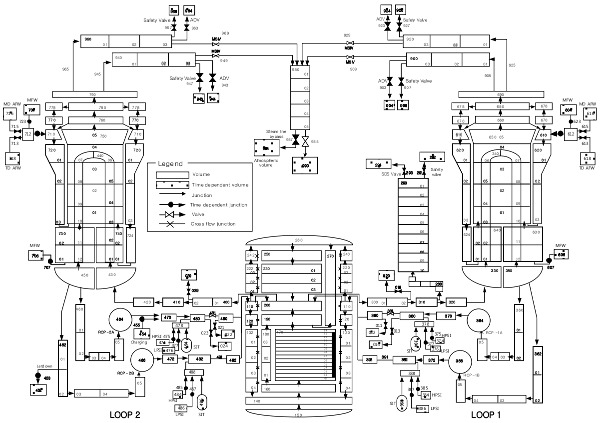


Fig. 1. Nodalization of Hanul units 3 and 4 MARS model

The core damage is defined as a peak cladding temperature (PCT) of 2200^{o}F (1477K) [5].

3. Results and Discussions

In TH analysis for the selected scenarios, there were no core damage because the scenarios were already developed by iterating between thermal-hydraulics analysis and accident sequence analysis.

Below sub-chapters are the major findings and insights obtained in SC analysis.

3.1 A role of safety functions in LOCA

The figure 2 shows the success criteria of safety functions with break size in LOCA. There are three categories along the safety functions: safety injection only, secondary cooling only, and combination.

In "safety injection only" category, only safety injection features such as HPSI pump, LPSI pump, safety injection tank (SIT) are available. For 0.8 inch to 9.4 inch of break size, one HPSI pump was enough for success criteria because the break flow is well balanced with injection flow rate. Break flow is the means to remove the core residual heat. For below than 0.8 inch, the HPSI pump was not available due to high pressure of RCS. RCS pressure is increased to PSV open pressure, after that, SDS valve is manually opened by operator. Based on the TH results, safe operator's action time for SDS valve is within 35 minutes. In break size above than 9.4 inch, one HPSI pump is not sufficient for supplementation of break flow. HPSI pump with two SIT is success criteria for up to 17.8 inch. For 60 inch break size (double-side break of cold-leg pipe), LPSI pump should be operate to assure the no core damage.

For "secondary cooling only" category, AFW pump with ADV open was only available. For the range of 1.4inch to 60.0 inch of break size, PCT reaches the limit because accumulated loss of coolant is so large that secondary cooling is meaningless.

For "combination" category, safety injection and secondary cooling together are available. For below than 10.2 inch, HPSI pump with secondary cooling was also success criteria.

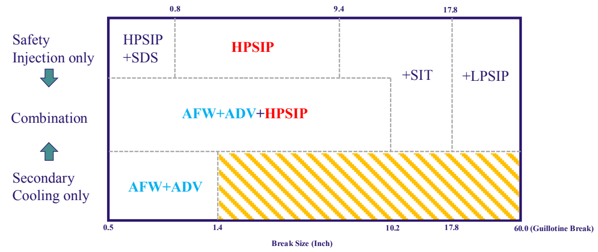
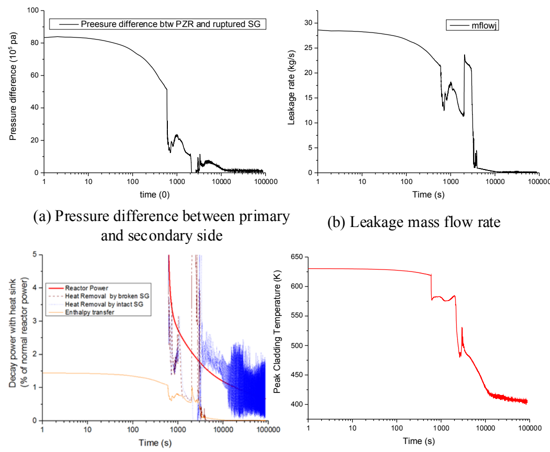


Fig.2. Success Criteria of safety functions in LOCA

3.2 Coolant inventory in leakage accident

In case of leakage accident such as cold-leg pipe break, steam generator tube break, and RCP seal LOCA, PSA traditional manner qualitatively evaluates that the scenarios, without HPSI or LPSI pump, directly go to the core damage because primary coolant inventory is not sufficient to remove the residual heat. However, several TH results show that the accumulated leakage is not significantly large. If there is secondary cooling, then residual heat is properly removed.

In LOCA results as shown in figure 2, only secondary cooling could be success criteria for below than 1.4 inch. This is because that accumulated leakage weight is not critical to RCS inventory. Figure 3 shows that the TH results of the scenario that SGTR with secondary cooling available. Although the HPSI and LPSI pump are not available, there is no core damage by only secondary cooling within 24 hours. Because of pressure balance between primary side and affected SG and drying the primary coolant of affected SG up, leakage flow rate rapidly decreased. Then, accumulated RCS leakage mass is not significantly large and unaffected SG secondary cooling removed the residual heat.



(a) Pressure difference between primary and secondary side
(b) Leakage mass flow rate
(c) power balance
(d) Peak cladding temperature

Fig. 3. Transient of main variables in SGTR (secondary cooling only available)

4. Conclusions

In this paper, based on best-estimate TH results for exhaustive case of event tree of the Hanul units 3&4 PSA model, several insights were identified. It is not recommended that these issues directly apply the present PSA model because of several remain problems such as uncertainty analysis, and its impact on the core damage frequency. Major contribution of this work is to identify the problems in present PSA event tree model.

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