Analysis of ATWS for Shin-Kori 1,2 Units according to DPS availability

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1. Introduction

ATWS(Anticipated Transient Without Scram) is an accident that the shutdown control rods cannot be inserted into the core even though the reactor trip signal is occurred in an anticipated operation transient[1]. If the reactor is not tripped during transient conditions, there is a possibility that the pressure at the reactor coolant system pressure boundary rise above its tolerance. The sensitivity analysis using the RELAP5 code for DPS(Diverse Protection System) availability under the ATWS of Shin-Kori 1,2 Units is performed to confirm that the pressure of the reactor coolant system is within the pressure limit.

2. Methods and Results

Typical anticipated operational transients such as a loss of feedwater, a loss of load, turbine trip, an accidental control group withdrawal, a loss of AC power, and a loss of condenser vacuum can have serious consequences when they occur at the same time with the reactor trip. In this study, we evaluated the loss of feedwater without SCRAM because the event is the most limited case in terms of overpressure. For this study, RELAP5 MOD3.3 Patch5 version was used. And the core conditions at BOC(Beginning Of Cycle) and MOC(Middle Of Cycle) are used.

2.1 Acceptance Criteria

ATWS analysis should ensure that the plant condition can be kept within the following limits.

- The PCT(Peak Cladding Temperature) shall not exceed 1204℃(2200℉)
- The maximum pressure at the reactor pressure boundary shall not exceed 3200psig

2.2 Shin-Kori 1,2 units Diverse Protection System

The DPS, which is equipped for the diversity of reactor protection system was installed at the Shin-Kori 1,2 units as a provision for ATWS. The DPS includes the following three functions[2] to prevent the overpressure of reactor coolant system pressure boundary.

- Reactor trip by pressurizer high pressure signal
- Turbine trip due to reactor shutdown
- Auxiliary water injection by SG low level signal

Table 1 shows the DPS set points of Shin-Kori 1,2 units.

Table 1: Shin-Kori 1,2 units DPS set points

Signal	Set points	
	Pre-Trip	Trip
PZR H-pressure	163.5kg/㎠ A	168.5kg/㎠ A
SG L-level	24.0%(WR)	22.1%(WR)
Containment H-pressure	248.6cmH_2O	259.8cmH_2O
Turbine trip	-	Rx. trip
Manual	N/A	N/A

2.3 Sequence of Event

The event of the loss of feedwater without SCRAM starts as interrupts the feedwater during normal operation. The pressurizer safety valve is opened by the pressurization of the reactor coolant system due to the feedwater shutoff. And, as the steam generator level becomes low, auxiliary water is injected. If the main feedwater to the steam generator is lost, the water level of the steam generator decrease and reaches the set point of the auxiliary water operation. After that, the reactor is tripped not by the RPS with the pressurizer high pressure trip set point, but by the DPS with the DPS high pressure set point which is higher than the former. As the main steam isolation valve is closed, the steam generator pressure is increased and the main steam safety valve(BANK1) is opened.

2.4 Primary system TH behavior

Figures 1,2 and 3 show the analysis results for peak cladding temperature. It does not exceed the PCT limit of 1204℃, regardless of whether the DPS is operated or not. However, if there is a difference, PCT is in the 18th thermal structure when DPS is activated, and PCT is in 19th thermal structure when DPS is not activated at MOC core conditions, and PCT is in 20th thermal structure when DPS is not activated at BOC core conditions.

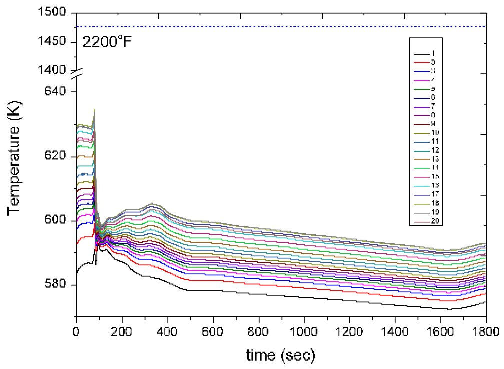

Fig. 1. Peak Cladding Temperature (with DPS).

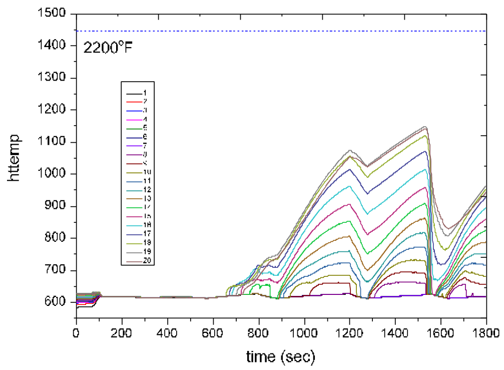

Fig. 2. Peak Cladding Temperature (without DPS - MOC).

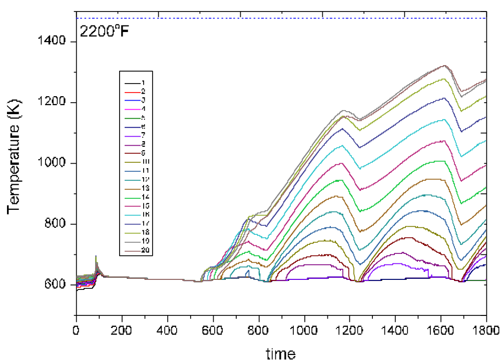

Fig. 3. Peak Cladding Temperature (without DPS - BOC).

Fig. 4,5 show the temperature and pressure of the reactor coolant system. Before the reactor trip, the heat transfer through the steam generator tube is reduced by the total loss of feedwater, increasing the pressure and temperature of the reactor coolant system. The pressure of the reactor coolant system reaches the reactor trip set point, but the reactor does not trip and the temperature and pressure of the reactor coolant system continue to increase. Reactor trip set point due to the high pressure of the pressurizer of the DPS is reached and the reactor is tripped. Thereafter, the pressurizer safety valve set point is reached and the pressurizer safety valve is opened, after that the pressurizer pressure and the reactor coolant system temperature are reduced. If the DPS is not operated, the reactor coolant system temperature is maintained higher than during DPS operation. The pressurizer pressure does not decrease even after the pressurizer safety valve is opened, and the opening and closing of the pressurizer safety valve tends to be repeated. At BOC core conditions, the maximum pressurizer pressure is over the limit value.

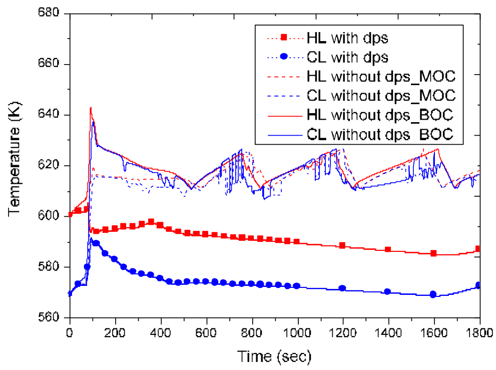

Fig. 4. Reactor Coolant System temperature.

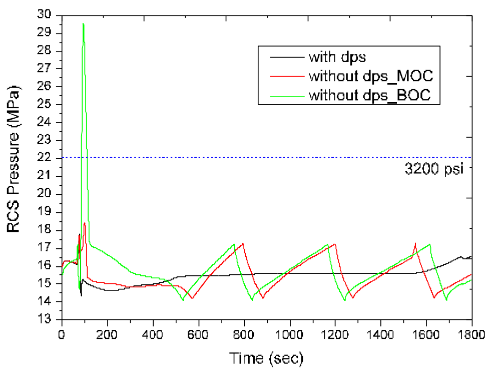

Fig. 5. Pressurizer pressure.

Fig. 6 shows the water level of the pressurizer. The behavior of water level of the pressurizer is similar to the behavior of pressure of the pressurizer. The water level increase before the reactor trip, but the water level decrease after the pressurizer safety valve is opened, and then the water level of 30~35% is maintained by the secondary behavior. If the DPS is not operated, the water level of the pressurizer reaches the full water level due to the opening of pressurizer safety valve, however, after that, it decrease to 0%.

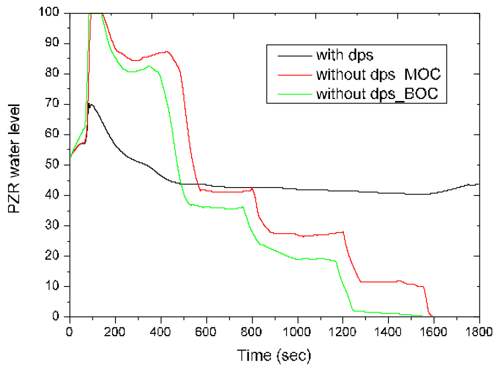

Fig. 6. Pressurizer water level.

2.5 Secondary system TH behavior

Fig. 7 shows the steam generator level. The feedwater is tripped at the same time with the start of the accident, and the water level of the steam generator is exhausted. Subsequently, the water level of the steam generator is recovered as the auxiliary water is injected. After recover of the steam generator level, the behavior is changed as the injection/interruption of the auxiliary water and the main steam safety valve opening/closing is repeated. If the DPS is not operated, the steam generator level remains exhausted because no auxiliary water is injected. Fig. 8 shows the steam generator pressure. In the early time of the loss of feedwater, the pressure slightly increase, but the turbine steam bypass valve opens and the pressure decrease. After the main steam isolation valve is closed, the pressure increases again to reach the main steam safety valve set point. After the main steam safety valve is opened, opening and closing will occur repeatedly. If the DPS is not operated, the steam generator pressure is kept lower than DPS is operated.

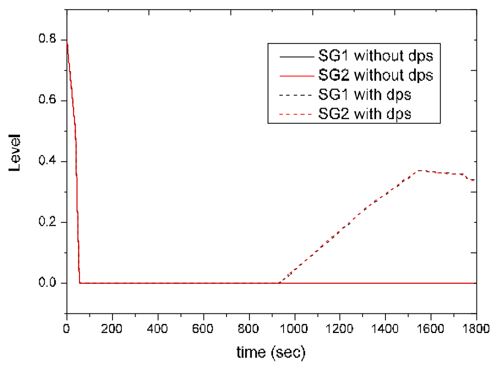

Fig. 7. Steam Generator water level.

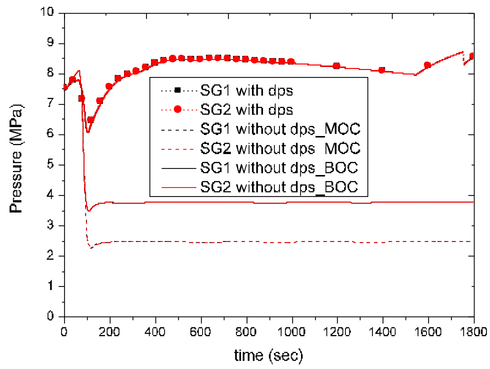

Fig. 8. Steam Generator pressure.

3. Conclusions

In the case of Shin-Kori 1,2 units, the probability of occurrence of ATWS is very low because reactor protection system has very high reliability. Nevertheless, it has been confirmed that the peak cladding temperature remains within the PCT limits and the maximum pressure of reactor coolant system is not exceed the limits, when the DPS is operating, in the event of an ATWS accident due to a loss of feedwater. However, when the DPS is not operated, at the BOC core conditions the maximum pressurizer pressure is over the limit value.

REFERENCES

[1] "KINS/GE-N016(Rev.0) : Safety Review Guideline for Accident Management Program", Korea Institute of Nuclear Safety, 2017.

[2] "Shin-Kori 1,2 units Operator Handbook", Korea Hydro & Nuclear Power Co., Ltd.