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A Study on Securing Core Cooling Ability in Case of Total Loss of Feedwater for Westinghouse Type NPP

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1. Introduction

Total Loss of Feed Water (TLOFW) belongs to the Beyond Desgin Basis Accident (BDBA) of which the results should meet the requirements of Appendix 19-1 of the LWR Safety Review Guidelines. In case of the accident, the rapid depressurization system should function properly to enable the ECCS water to remove the core residual heat by Feed and Bleed (F&B) operation.

As this is an optimal evaluation method can be applied and a single failure is assumed for the analysis. The rapid decompression system should have sufficient capability to take into account the uncertainties related with the operator's intervention, and the performance of the safety injection system. In this paper, a F&B operation strategy for a Westinghouse (WH) two-loops power plant[1] which can cope with a TLOFW accident is proposed. The proposed strategy is introduced as the another process except for current procedures. MARS-KS1.4 computer code was used for analysis.

2. Analysis Methodology

In TLOFW, both the main feedwater system and the auxiliary system are lost and the core heat cannot be removed by the secondary system. Therefore a rapid depressurization should be available until the start of the DHRS (Decay Heat Removal System) which removes the core decay heat in the primary system. Since the chemical and volumetric control system is not a safety class, the operation of the charging pump and the pressurizer auxiliary sprinkler system connected to this system is not considered[2,3].

2.1 Plant Modeling and Initial Conditions

The analyzed plant is a two-loop 1876 MWt pressurized water reactor. Figure 1 shows the MARS nodalization of the plant. There are 2 PORVs(ACT/PID) and 1 safety valve at the top of pressurizer, and these PORVs are used for F&B operation.

Table 1 shows the setpoints of the parameters used in the analysis including the PZR pressure, PORV open pressure, SG water level and RCP trip conditions.

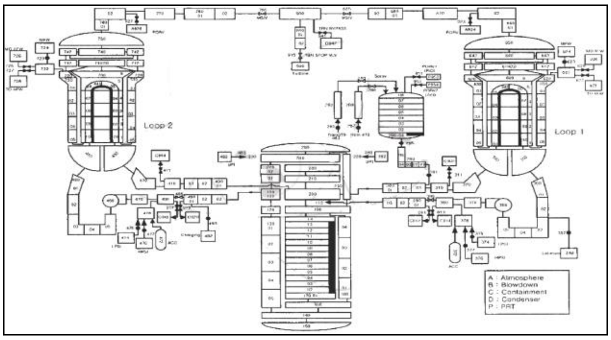

Figure 1. Nodalization of WH(two-loop) Plant

The initial conditions established for the analysis are are listed Table 2, which are reasonably identical with the nominal operation conditions.

Table 1. Setpoints of parameters used in the analysis

	Parameter		Set Point	Comparison
1	PZR, kg/cm^2 (psia)	High Pressure	164.44 (2,395)	Rx Trip Signal
		Low Pressure	129.9(1,885)	SI Signal
2	SG WR Lo-Lo Level, (%)		7	-
3	PZR PORV Open pressure, g/cm^2(psia)		162.0(2,350)	-
4	RCP Trip		164.44 (2,395	Operator Action at SG WR 6%

2.2 Assumptions

TLOFW accident is initiated by the loss of all main and auxiliary feedwater supply to steam generator. The Reactor Coolant Pump (RCP) is assumed to stop when the wide range water level of all SGs decrease to 6% after the reactor trip. Based on the realistic analysis method, nominal design values are used for all reactor trip setpoints.

Control systems such as reactor power control system, pressurizer level control system, pressurizer pressure control system are available. ANS79-2σ decay heat model is assumed. Henry-Fauske/Moody critical flow model is used for PORV discharge rate. The valve area of a PORV, 0.00932m^2, is determined based on the PORV design steam flow capacity. The PORV area was adjusted to fit the flow resistance so that the saturated steam discharge flow rate correspond~~ed~~s to the design value.

In this paper, the analysis was performed using the available number of PORV and HPSI, and the F&B strategy is proposed based on the analysis.

Table 2. Initial Conditions for TLOFW Analysis

Parameter		Nominal Operation Value	Simulation Value
Core Power (MWth)		1876	1875
RCS Flow Rate(kg/sec)		9184.64	9677.20
Pressurizer	Pressure(MPa)	15.52	15.51
Hot Leg Temperature(K)		612.0	612.1
Cold Leg Temperature (K)		550.0	550.5
RCS Average Temperature(K)		581.0	581.3
Steam Generator	Tube Plugging Rate(%)	5	5
Main Feed Water	Flow Rate(kg/sec)	511.25	513.80
	Temperature(K)	494.26	505.4
Steam	Flow Rate(kg/sec)	511.25	513.84
	Pressure(MPa)	63.40	60.98

3. Analysis Results

3.1 Analysis based on the number of PORV and HPSI available

As indicated in Table 3, when the accident occurs the sequences including the loss of main feedwater and aux feedwater, SG Lo-Lo level Reactor trip, manual opening of PORV(s) and automatic SI injection follow.

Table 3. Sequences of Event for TLOFW for WH(2-LOOP)

Sequence	Operator Action			
	Case 1	Case 2	Case 3	Case 4
	1PORV + 1HPSI	1PORV + 2HPSI	2PORV + 1HPSI	2PORV + 2HPSI
Accident occurred (Loss of MFW & AFW supply)	0.0	0.0	0.0	0.0
SG Lo-Lo Level (SG NR 18.2%)	45.9	45.9	45.9	45.9
Rx trip(Delay 2sec) by SG Lo-Lo Level	47.9	47.9	47.9	47.9
SG WR 7%	505.5	505.5	505.5	505.5
RCP 1,2 manual trip(Delay 60sec) by SG WR 7%	565.5	565.5	565.5	565.5
SG inventory dry out	642.5	642.5	642.5	642.5
PZR PORV manual open	642.5	642.5	642.5	642.5
SI Injection (Delay 2sec)	644.5	644.5	644.5	644.5
SL Lo pres trip	821.5	802.2	821.5	808.2
PZR Lo pres trip	849.1	836.8	832.8	819.8

But after SI initiation the depressurization and heat removal depend on the numbers of the PORVs and SI pumps.

Four cases were compared according to the available number of the PORVs and high pressure SI pumps.

The case 1, the scenario of 1 PORV + 1 HSPI, is as follows(Table 3). The loss of all main and auxiliary feedwater is initiated at 0.0 sec. The reactor trip occurs at 45.9 seconds due to steam generator low-low level. After reactor trip, pressurizer pressure decreases by the RCS shrinkage due to sudden decrease of core power. However, the pressurizer pressure starts to increase again soon since the heat removal capacity of the SG is decreasing due to loss of SG inventory. SG inventory is depleted at 642.5 seconds. PORV is opened by the operator at 642.5 seconds and SI injection starts at 644.5 seconds taking into account the 2 seconds delay time. The low pressure trip of main steam line occurs at 821.5 seconds. The pressurizer low pressure trip signal is generated at 849.1 seconds.

For the other 3 cases, 1 PORV+ 2HPSIs, 2PORVs + 1 HPSI, 2 PORVs + 2HPSIs follow the same sequences but at different time steps. Figures 2 and 3 show the pressurizer pressure and water pressurizer level respectively. In the case of 2 PORV + 1 HSPI, it can be seen that the pressure level of the pressurizer is temporarily lowered due to the opening of two PORVs, and SI is returned after the injection. Figures 4 and 5 show PORV discharge flow rate HPSI injection flow rate respectively. Figure 6 shows average temperature of reactor coolant system. Figure 7 shows collapsed core level. Through F&B operation by the operator, the water level revcovery in the core region was achieved. The collapsed core level is maintained at higher than 6.55 m (The location of core top is 4.08m). However, when 2 PORVs are opened, it is confirmed that core exposure is more than other cases. Therefore, as can be seen from the Figure 2 and 6, it can be seen that one(Case 4) of the four conditions satisfies the entry condition (400psia and 350 ^{o}F) of the DHRS. But the safety margin of the case is not sufficient.

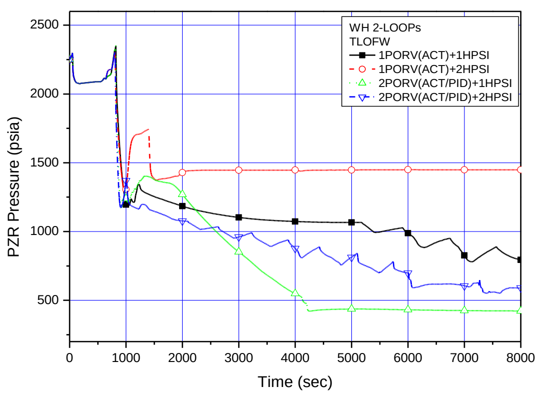

Figure 2. Pressurizer Pressure (Case 1,2,3,4)

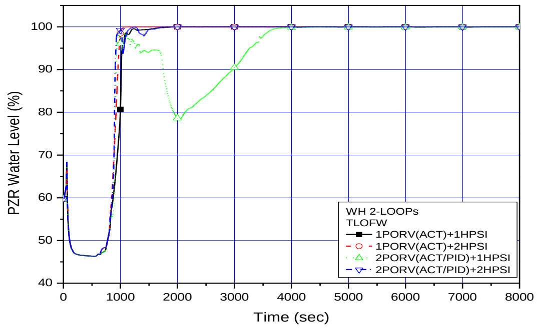

Figure 3. PZR Water Level (Case 1,2,3,4)

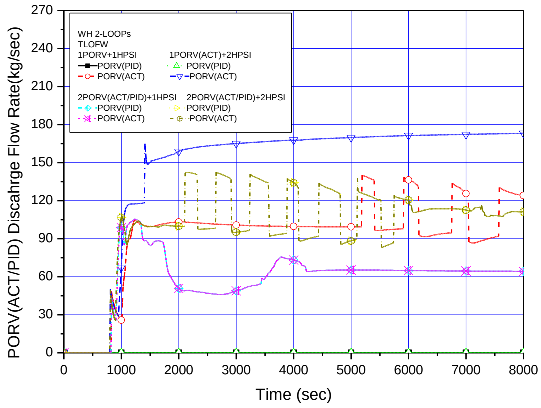

Figure 4. PORV Discharge Flow Rate (Case 1,2,3,4)

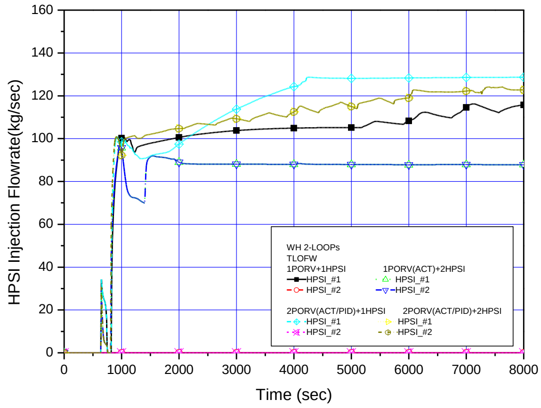

Figure 5. HPSI Injection Flowrate(Case 1,2,3,4)

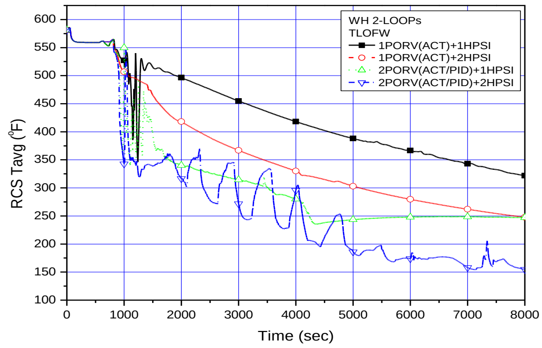

Figure 6. Average Temperatue of RCS (Case 1,2,3,4)

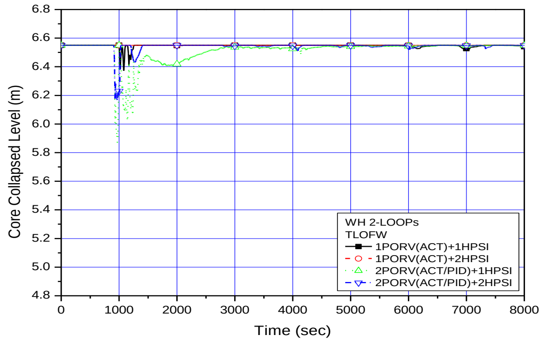

Figure 7. Core Collapsed Level (Case 1,2,3,4)

3.2 F&B Proposal Strategy

It is difficult to reach the start condifition of DHRS using the F&B operation mentioned above. It implies that the simple mass and energy transport of the F&B operation does not bring the reactor condition as desired and the operator's intervention is needed to adjust the feed and/or bleed flow rate at a certain moment. Therefore a new F&B operation strategy consisting of three stages is proposed. First, 1 PORV and 2 HPSI are operated for decompression and cooling of the primary coolant system. As a second step, 1 PORV and 1 HPSI are operated to perform the depressurization of the primary coolant system. As a final step, the DHRS operating condition is reached through decompression of the coolant system by 1 PORV and 60% of the design flow rate of a HPSI.

Figure 8 shows the pressure trend of the pressurizer. It uses 1 PORV + 2 HPSI for at 5,000 seconds to lower the temperature of the coolant while lowering the pressure of the RCS system. The operator action of 1 PORV + 1 HPSI then reduces the pressure in the coolant system at t=8,000 seconds. In the final step, 60% of the design flow of 1 HPSI is injected into the coolant system, so that the temperature and pressure of the coolant reach the start condifition of DHRS (400psia and 350 ^{o}F).

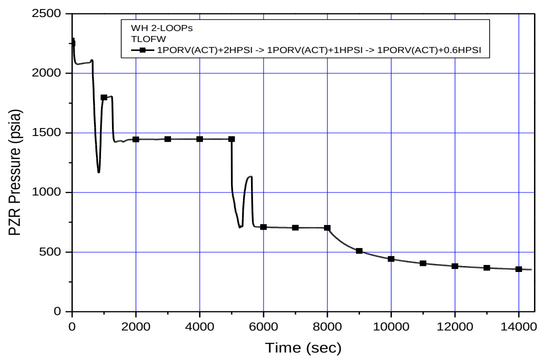

Figure 8. Pressurizer Pressure(Proposal)

Figure 9 shows the pressurizer water level. After the HPSI system is activated, the water level of the pressurizer shows a quick recovery.

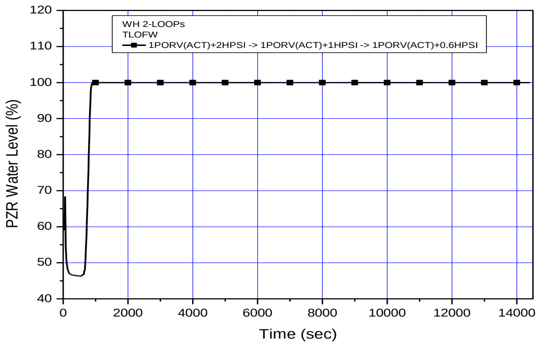

Figure 9. PZR Water Level(Proposal)

Figure 10 shows the PORV discharging flow rate of the plant. As shown in the figure 10, it can be seen that the discharge flow rate of PORV is changed according to the number of HPSI drives for one PORV.

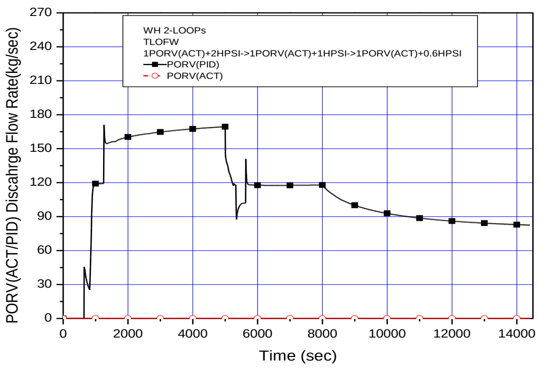

Figure 10. PORV Discharge Flow Rate(Proposal)

Figure 11 shows the HPSI injection flow rate. According to the HPSI injection operation strategy, 2 HPSIs inject cooling water by the reator protection singal, and 1 HPSI is terminated at t=5,000 seconds by the operator. And 60% of the design flow rate of a HPSI is injected at t=8,000 seconds by the operator action.

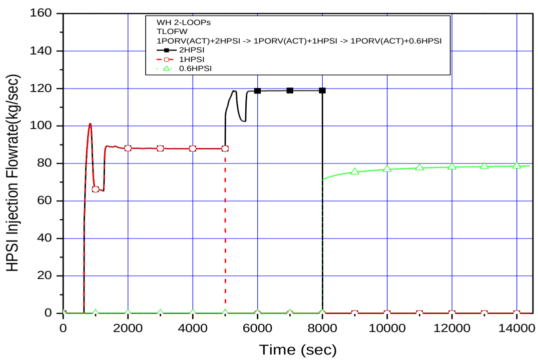

Figure 11. HPSI Injection Flowrate (Proposal)

Figure 12 shows the average coolant temperature As shown in the figure 8, it is seen that the residual heat removal system operating condition of 350°F can be entered.

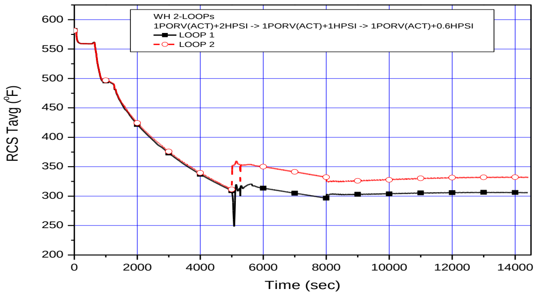

Figure 12. Average Temperatue of RCS(Proposal)

Figure 13 shows the core collapsed level. It is confirmed that the water height is more than 2ft higher than the core upper part.

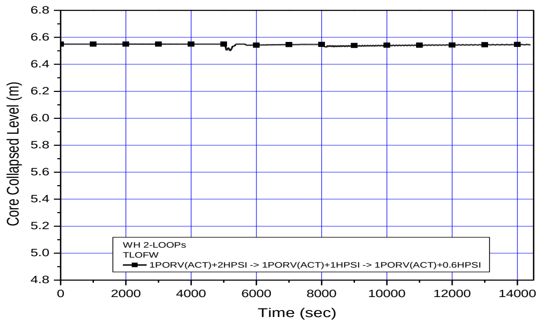

Figure 13. Core Collapsed Level (Proposal)

4. Conclusions

This paper may be to propose new simple routine work for operator without current procedures. The TLOFW analysis was carried out to evaluate the capability of decay heat removal for WH(2-LOOP) using MARS-KS1.4 code. As shown in Figures 2 and 6, it is not sufficient and difficult, in the view of the safety margin, that the 2 PORV plus HPSIs strategies bring the primary coolant system pressure condition for the residual heat removal system and meet the core collapsed level requirement of 2ft. As shown in the results of the MARS analysis, the newly proposed strategy has been proposed. It is a combination of variation of pumps and operator intervention at the appropriate moments. The strategy can reduce the pressure to the start condition of the residual heat removal system and the coolant level is easily maintained at least 2ft from the upper part of the core.

REFERENCES

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