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Evaluation of the Operator Actions to Maintain the Fuel Integrity during the Small Break LOCA with Safety Injection Failure for Westinghouse type 2-loop Plant

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1. Introduction

Accident Management Plan (AMP) has been requested to get capability of accident mitigation strategies for Nuclear Power Plant (NPP) during multiple failure accident[1]. Small Break Loss Of Coolant Accident (SBLOCA) with Safety Injection (SI) failure is one of multiple failure accidents to be evaluated to mitigate the NPP transient condition keeping not to grow up into severe accident.

Westinghouse type (W type) 2-loop plant such as Kori 2 NPP has two steam generators and Power Operated Relief Valve (PORV) connected to the pressurizer (PZR) and steam generators to maintain the system pressure by discharging coolant during abnormal condition. To maintain the reactor system safe during this event, appropriate operator actions on right time should be performed, therefore, some trials to find the right operator actions using PORV and RCP to prevent the NPP not accessing the initiating condition of severe accident had been considered.

Fuel cladding temperature is referred to demonstrate the result of operator actions during SBLOCA with SI failure. It is confirmed that the decay heat removed stably and continuously preventing the fuel failure by this evaluation.

2. Analysis Methodology

2.1 Plant Modeling and Initial Conditions

Pressurized water reactor W type plant has two loops of reactor coolant system which circulates the primary side coolant in closed loop and transfer heat from the core and internal structure to secondary side systems. The main components of the reactor coolant system are reactor vessel, two heat transfer loops, PZR and piping systems to connect component, respectively. Feedwater systems, steam generators, main steam piping systems and turbine compose the secondary side system.

Best estimated analysis is performed with control systems model including Pressurizer Pressure Control System (PPCS), Pressurizer Level Control System (PLCS), Steam Generator Level Control System (SGLCS) and Steam Dump Control System (SDCS) composed with PORVs. Multiple failure accidents such as SBLOCA with SI failure are analyzed through realistic simulation using the RELAP5/Mod3.3 code, therefore the nominal design value of 100% reactor power is assumed as an initial condition as shown in Table I. RELAP5 is a useful code to predict various physical behavior and thermal hydraulic phenomena in the reactor coolant systems during transient state.

Table I. Initial Conditions

Parameter	Design Value	Analysis Value
Core Power, MWt	1,876	1,876
PZR pressure, MPa(a)	15.51	15.51
RCS flow rate, kg/s	8,832.5	8,836.9
Core inlet temperature, K	560.59	560.59
Secondary pressure, MPa(a)	6.3	6.33
Secondary steam flow rate, kg/s	1028.2	1041.8
PZR level, %	60	60
Steam Generator level, % NR	50	50

2.2 Assumptions

The 2-inch break area in the cold leg is considered to analyze the operator actions during SBLOCA with SI failure, because the limiting size of the small break in the Probability Safety Assessment (PSA) is 2-inches in view point of the core damage frequency[2]. Break flow of the cold leg is used with Henry-Fauske critical flow model.

High Head Safety Injection (HHSI) and Low Head Safety Injection (LHSI) are operated by the PZR low pressure signal cooperating with isolation valve connected from the safety injection pump to the safety head pipe. HHSI as means of the coping facility for SBLOCA is assumed not working during this event but LHSI working if the Reactor Coolant System (RCS) pressure is less than LHSI pump shutoff head.

2.3 Operator Actions

Operator actions should be performed to depressurize RCS and bleed the core with makeup water from Safety Injection System (SIS) or Residual Heat Removal System (RHRS). Through the PORV of the Steam Generator (SG) steam is discharging, then the RCS pressure is started to decrease if the operator can open the PORV effectively. Some of PORV open times from the beginning of the accident and open methods by operator are applied to depressurize the RCS during SBLOCA with SI failure as shown in Table II. Moreover, Reactor Coolant Pump (RCP) trip by operator is also compared. Heat removal by RCP from the core after the accident is not conservative assumption for the traditional accident analysis. However, the appropriate operator action by means of maintaining core integrity should be evaluated. Operator would stop RCP at the 10 minutes and open SG PORV at 30 minutes from the initiating accident in Case 2. For Case 1, operator would be expected to open SG PORV at the same time in Case 2 without stopping RCP. In Case 3 and Case 4, the operator also stops RCP at the same time, but the PORV is opened earlier and the opening rate is determined by cooling down rate of 55.55 °C/hour.

Table II. Operator Action

	SG PORV Open (minutes)	Open Method (°C/hour)	RCP Trip (minutes)
Case 1 (Reference)	30	Full Open	X
Case 2	30	Full Open	10
Case 3	20	55.55	10
Case 4	15	55.55	10

3. Analysis Results

Various reactor system behaviors during the SBLOCA with SI failure are shown in Table III for Case 1 as a reference case. In sequences of event, detailed time is described with related event.

Table III. Sequences of Event of the Reference Case

Sequences	Time (second)
Event Start	0.0
Reactor Trip	22.0
Safety Injection Actuation Signal	27.0
HHSI injection fail	39.0
PZR depletion	41.0
Auxiliary feedwater injection	87.0
SG PORV open	1800.0
MSSV Isolation Signal	1805.0
SIT injection	1831.0
Peak Cladding Temperature	1924.0
LHSI injection stat	2719.0
RCS condition reaches the RHRS operating condition	2820.0

The RCS pressure decreases rapidly discharging the flow from the break location in cold leg. Reactor trip occurs after the pressure reaching the low PZR pressure at 22.0 seconds, and the void fraction start to increase maintaining RCS pressure in quasi equilibrium state. Operator action to open a PORV connected with one of SG is effective to depressurize the RCS as shown in Figure 1. The beginning time of the depressurization of the RCS in Case 4 which opens PORV with the cooling rate of 55.55 °C/hour is earlier than other cases. However, the RCS pressure does not reach Low Head Safety Injection (LHSI) injection pressure. For Case 1 and Case 2, the RCS pressure is depressurized to enough to reach LHSI actuation condition.

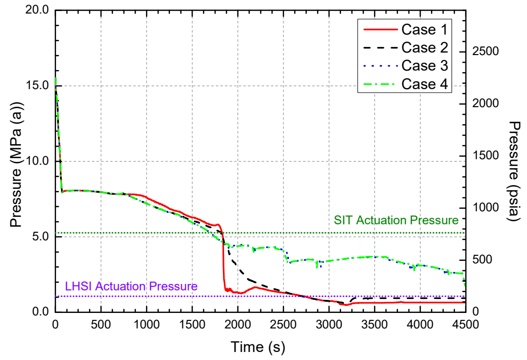

Figure 1. RCS Pressure

Discharging coolant through break the PZR pressure level decreases rapidly and is depleted. When the RCS pressure is low enough, the makeup water from Safety Injection Tank (SIT) is started to be injected and LHSI is actuated subsequently. At the beginning of the LHSI injection with cooling water from SIT, RCS pressure and temperature decrease effectively. As shown in Figure 2, SIT water is injected inconsistently for a short time for all cases, however, only for Case 1 and 2 the LHSI injects large amount of makeup water at about 3,000 seconds. Thereafter, a small amount flow form LHSI injected continuously.

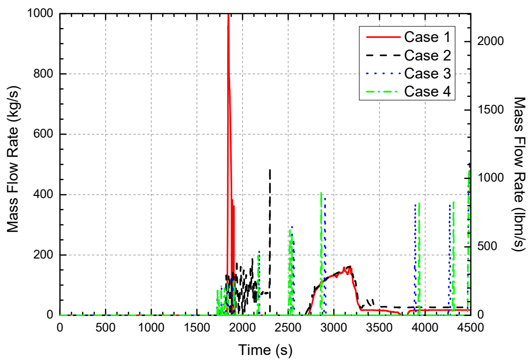

Figure 2. SIT and LHSI injection

RCS inventory decreases because the void occurs in the core. RCS inventory is started to be recovered when the makeup water from SIT and LHSI is larger than break flow. For Case 1 in Figure 3, not tripped RCP flows reactor coolant continuously then core collapsed level decreases rapidly more than other cases as shown in Figure 3. However, the core collapsed level is recovered to top of the active core at about 2,000 seconds by the large amount of the safety injection from the SIT. Since the SIT finished to inject makeup water, core collapsed level decreases again until LHSI is started to cover the core level. As the end of the large amount of the LHSI flow, a small amount of LHSI water flow is balanced with break flow. In this reason, the core collapsed level could be maintained on stable. For Case 3 and Case 4, core collapsed is varied with water injection from SIT without LHSI injection. In these cases, the RCS cooling down is not stable because break flow is larger than injected water.

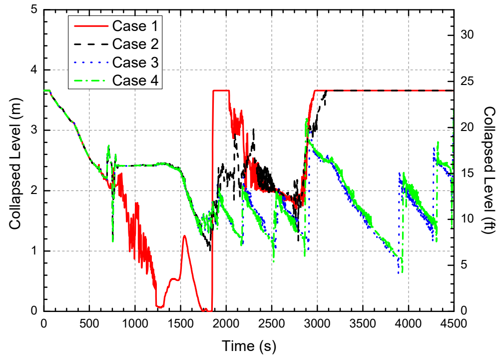

Figure 3. Core Collapsed Level

In Case 1, core collapsed level decreases under the bottom of the active core from 500 to 3,000 seconds. Though it is predicted to fuel dry out in this interval, the continuous core flow by RCP prevents rapid heat up as shown in Figures 4 and 5. Nevertheless, the core collapsed level of the Case 2 does not decrease rapidly, the fuel is heated up obviously owing to the RCS thermal hydraulic conditions such as pressure, temperature and core flow. For Case 3 and Case 4, the core collapsed level remains over the bottom of the active core, however, the RCS temperature and pressure are not decreased enough to prevent fuel heat up as shown in Figure 4 and 5. The time that peak temperature occurs in Case 3 and 4 are related to water injection from SIT and unstable to maintain core cooling ability during SBLOCA with SI failure.

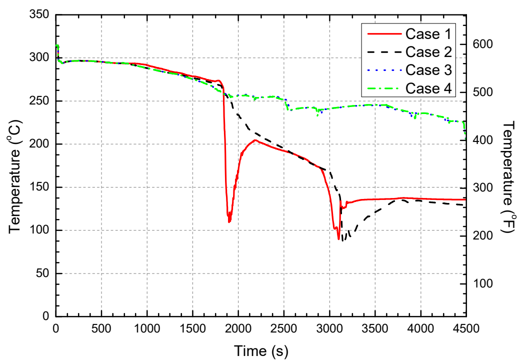

Figure 4. RCS Temperature

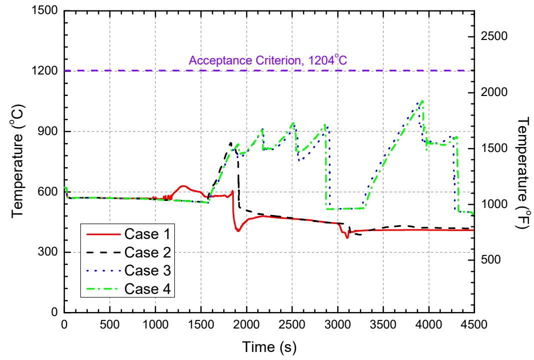

Figure 5. Fuel Cladding Temperature

As a result, for Case 1 the RCS pressure and temperature reach RHRS condition at 2,820 second, then reactor system goes into the stable shutdown state, finally. As similar with Case 1, the operator actions in Case 2 are effective to maintain reactor system on stable during this event. Although the fuel cladding temperatures for Case 3 and Case 4 do not exceed the acceptance criteria, Case 1 is most effective operator actions for SBLOCA with SI failure.

4. Conclusions

SBLOCA with SI failure as one of the multiple failure events had been evaluated with various operator actions using the best estimated methodology. Analyses for the W type 2-loop plant has shown that if the operator fully open SG PORV without any other actions at 30 minutes after the initiation of this event, the SIT, LHSI and RHRS could successively provide stable core cooling ability.

REFERENCES

[1] KINS/RG-N16.01, Rev.0, Regulatory Guideline 16.1, "Assessment of accidents due to multiple failures", 2016.
[2] S.H. Jee, K.M. Park, 11E54-PSA-TH-AR002 : Thermal-hydraulic Analyses for MLOCA, SLOCA and SGTR, 2015.