Feasibility Study on the Application of the High Containment Pressure Trip Function during Feedwater Line Break Accident

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1. Introduction

The feedwater line break (FLB) is an accident in which heat removal by the secondary system decreases as the steam generator inventory decreases due to the break in the main feedwater system (MFS) piping. In particular, if the main feedwater line break occurs at the downstream of the check valves, the reduction of primary-to-secondary heat transfer rate get worse as a decrease of the affected steam generator inventory due to the reverse flow from break side.

Overall sequence of accident is that, when the FLB accident is initiated, the affected steam generator inventory begins to decrease due to a break discharge flow. It causes to increase in the steam generator temperature and decreasing of the liquid inventory. And the reactor coolant system (RCS) is continuously pressurized and the heatup becomes more severe as the affected steam generator experiences a further reduction in its heat transfer capability due to a depletion of liquid inventory. The reactor trip can occur by the high pressurizer pressure (HPP), low steam generator water level (LSGL) or high containment pressure (HCP) trip signal. And the RCS pressure rapidly increases as a result of the turbine trip coincident with the reactor trip, and it reaches the pilot operated safety relief valve (POSRV) opening setpoint. The POSRV opening results in abrupt depressurization and the maximum RCS pressure reaches at this point. Thereafter, the primary and secondary system enters into a quasi-steady state due to decreasing of core decay heat, cycling of the main steam safety valves and auxiliary feedwater injection. In short, the RCS pressure and temperature are tended to increase and the departure from nucleate boiling ratio (DNBR) decreases during a transient. Thus, the FLB analysis is performed with respect to the primary peak pressure, secondary peak pressure and the minimum DNBR. For each viewpoint, the analysis methodology and initial conditions are established to derive the most conservative result, respectively.

With respect to primary peak pressure, the existing analysis methodology is to derive the maximum RCS peak pressure after the reactor trip by the HPP or LSGL trip that may occur during a transient by delaying of the trip as much as possible. It gives sufficient time to rise a RCS pressurization. Meanwhile, during a transient, the containment pressure and temperature are also tended to increase as the steam generator water inventory with high enthalpy flashes into the containment atmosphere through the break, and there is some possibility of the reactor trip signal by HCP. Therefore, if reactor trip by HCP trip signal occurs earlier than other trip signals such as a HPP or LSGL trip, the RCS pressurization before the reactor trip can be mitigated, and it is expected that the maximum RCS pressure would be derived as a lower value. Therefore, this feasibility study is to consider the reactor trip signal by HCP during the FLB transient and to verify that the existing methodology and result for FLB analysis are conservative.

2. Analysis Method

2.1 Analysis Software

The simulation of FLB accident is performed by the CESEC-III [1] computer program. It computes key system parameters such as core heat flux, pressures, and temperatures. And also the break flow rate and the fluid enthalpy through the break area during FLB accident are calculated. The CESEC-III simulated the nuclear steam supply system but it does not have the containment model. And, it means that HCP trip signal is excluded in existing analysis methodology. Therefore, to predict the pressure and temperature transients in the containment, the CONTEMPT4/MOD5 [2] computer code is used. It is developed to calculate the containment pressure and temperature behavior in case of the postulated loss-of-coolant accident (LOCA) and main steam line break (MSLB) accident. The program is capable of demonstrating the containment responses on the effects of leakage at RCS and secondary-side break accidents. Thus, it can be used to predict the containment pressure behavior during FLB accident.

2.2 Discharge Flow Rate and Energy Release

Major parameters can have an effect on the containment pressure during FLB transient based on a break flow rate and an enthalpy. In the CESEC-III computer code, the break flow is modeled assuming frictionless critical flow as calculated by the Henry-Fauske/Moody correlation [3]. In existing analysis methodology, the limiting break size with respect to the primary peak pressure is 0.4 ft^2 and its critical flow rate is about 4,000 lbm/sec. In addition, the limiting break size of 0.1 ft^2 with respect to minimum DNBR is considered to evaluate the delayed trip signal of HCP with a smaller discharge flow rate.

Break flow enthalpy is also calculated by the CESEC-III computer code. For FLB evaluation, it is especially assumed that saturated liquid is discharged until the steam generator is depleted. This is to minimize the effect of heat removal by the break flow although the break flow enthalpy physically depends upon the location of the break relative to fluid condition within the affected steam generator. It is also conservative in terms of minimizing pressure rise in containment because it has lower enthalpy than saturated vapor. In addition, the feedwater to both steam generators is assumed to be terminated so steam generator inventory has a high enthalpy due to loss of feedwater flow and continuous heat exchange from the primary side. So, it is expected that the containment pressure will rise much faster when this inventory begins to discharge through a break area. It is reasonable in case that the break area is sufficiently large to discharge the entire feedwater flow capacity. On the other hand, the feedwater from the opposite side through the main feedwater pumps has relatively lower enthalpy. With considering a feedwater from both side, there are so many uncertainties to simulate an exact enthalpy of the discharged flow. So minimum feedwater enthalpy during the normal operation is considered for comparison.

Table I shows the cases with the break size, the discharge flow enthalpy and the energy flow rate at 0 second after the accident initiation.

Table I: Initial Conditions of Sensitivity Study

Case	Break size, ft^2	Enthalpy, BTU/lbm	Energy flow rate, BTU/sec
1	0.4	557.0	2.26E+6
2	0.4	419.5	1.70E+6
3	0.1	557.0	1.13E+6
4	0.1	419.5	8.51E+5

2.3 Containment Model

Table II shows initial conditions for containment pressure analysis. These initial conditions are assumed to minimize the containment pressurization with maximum containment cooling capacity and to delay the reactor trip by HCP conservatively. The heat removal capacity of the containment active cooling systems such as the containment sprays and fan coolers are assumed to be actuated after the break with their maximum cooling capacity.

The setpoint of HCP is considered as the maximum value of 4.0 psig. And total response time including sensor response time and signal delay time is considered as 1.15 seconds.

Table II: Initial Conditions for Containment Pressure Analysis

Physical Parameter	Initial Condition
Maximum Free Volume	96,277 m^3 (3.4x10^6 ft^3)
Temperature	10°C (50°F)
Pressure	1.024 kg/cm^2A (14.56 psia)
Relative Humidity	90%

3. Analysis Results

As shown in Table III, when applying the limiting break size 0.4ft^2 with respect to the RCS peak pressure, the time to reach the HCP analysis setpoint varies depending on the enthalpy of the discharge flow, but the reactor trip by HCP occurs within 20 seconds in both cases. In case of 0.1 ft^2 break size, the containment pressure reaches the HCP analysis setpoint at about 24.3 seconds with the enthalpy of discharge flow but it did not reach the HCP analysis setpoint considering minimum feedwater enthalpy. Figure 1 and Figure 2 show the containment pressure behaviors. After an initiation of the accident, the containment pressure rapidly increases due to a mass release and reaches the containment peak pressure. An increasing ratio depends on the initial condition. After the depletion of the steam generator inventory, the pressure smoothly decreases by the containment cooling system.

Table III: Results of the Sensitivity Study

Case	Break size, ft^2	Enthalpy, BTU/lbm	Time to reach the HCP analysis setpoint, sec
1	0.4	557.0	11.5
2	0.4	419.5	16.6
3	0.1	557.0	24.3
4	0.1	419.5	Not occur

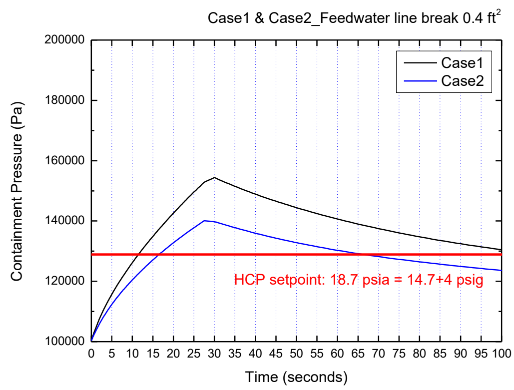

Fig. 1. Containment Pressure during the FLB Transient (Case1 & Case2)

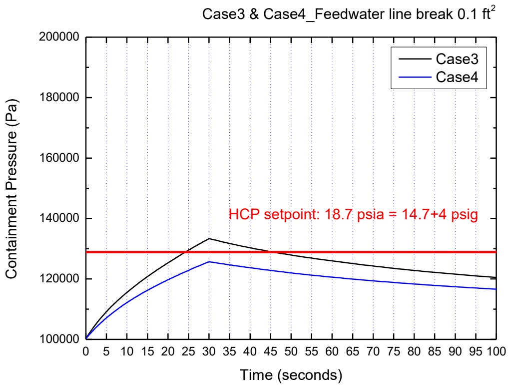

Fig. 2. Containment Pressure during the FLB Transient (Case3 & Case4)

Figure 3 and Figure 4 show the RCS pressures and the steam generator mass inventories with different initial conditions. According to the existing analysis results, the reactor trip time by the HPP is 27.23 seconds. The latest reactor trip by the HCP occurs at 25.45 seconds in Case3. As shown in Figure 3, the earliest trip by the HCP can mitigate a RCS peak pressure from 2,795.88 psia to 2,756.47 psia.

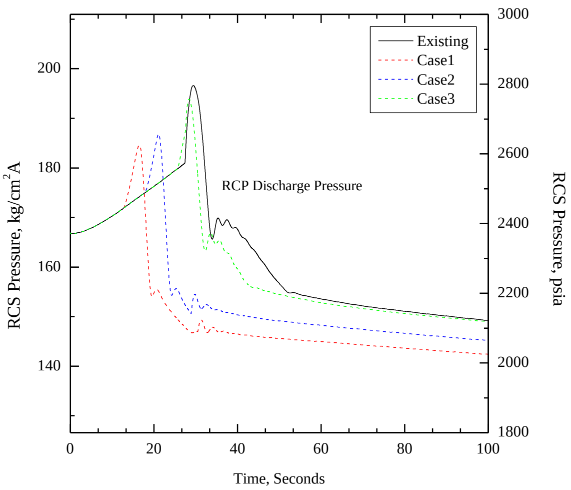

Fig. 3. RCS Pressure vs. Time

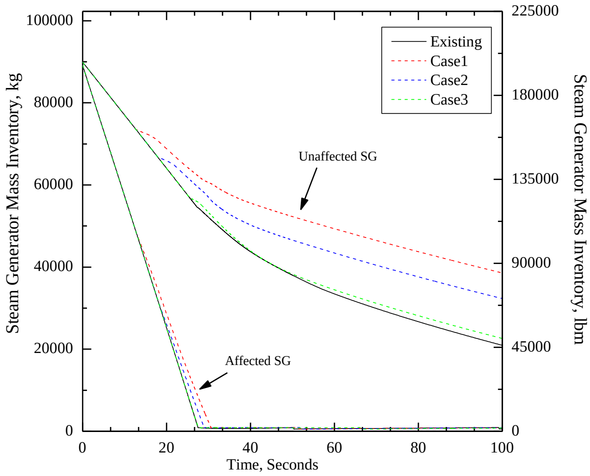

Fig. 4. Steam Generator Mass Inventory vs. Time

4. Conclusions

A feasibility study was performed on the trip time of HCP due to reverse flow from the steam generator during FLB accident. As a result of the study, it was derived that the trip time of HCP was shortened as the break size and discharge enthalpy are increased, so that the maximum RCS peak pressure of the existing result could be mitigated.

The result was evaluated with discharge flow and enthalpy as a simple assumption. Additional studies such as the sensitivity study on the initial condition and an actual experiment to verify the energy flow rate to the containment are necessary for safety analysis methodology in the future.

REFERENCES

[1] CENPD-107, “CESEC Digital Simulation of a Combustion Engineering Nuclear Steam Supply System”, April 1974

[2] CONTEMPT4/MOD5, "An Improvement to COTEMPT4/MOD4 Multicompartment Containment System Analysis Program for Ice Containment," Idaho National Engineering Laboratory, September 1984.

[3] R. E. Henry, H. K. Fauske, “The Two Phase Critical Flow of One-Component Mixtures in Nozzles, Orifices and Short Tubes,” Journal of Heat Transfer, Transactions of the ASME, May 1971.