A realistic assessment of ISLOCA in ABWR

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Interface-System Loss-of-Coolant Accident (ISLOCA) assumes failure of isolation valves in piping systems. Probabilistic safety assessments for advanced boiling water reactor (ABWR) have been shown a risk associated with ISLOCA has been sufficiently low on the ground of exceedingly small values of frequency of ISLOCA. On the other hand, understanding of plant response after ISLOCA is important from the viewpoint of defense in depth for the plant safety. Therefore a realistic assessment of plant response after ISLOCA in Hitachi-GE standard ABWR plant has been conducted by using best estimate codes in this study.

At first, transient behavior of pressure propagation inside piping systems after ISLOCA has been simulated. Complicated network of piping systems has been modeled in detail by combining 1D thermal-hydraulic components. Simulation has been shown that a pressure peak transiently occurred by water hammer effect inside piping systems after ISLOCA has been at most 8.2MPa. Then, transient behavior of environmental temperature in the reactor building on the assumption of leak from piping systems has been simulated. A realistic operation after ISLOCA in Hitachi-GE standard ABWR plant has been modeled. Simulation has been shown that an environmental temperature in the reactor buildings has been decreased by realistic operations such as depressurization systems of reactor pressure vessel(RPV) and control of RPV water level. Capability for mitigation and isolation of ISLOCA in Hitachi-GE standard ABWR plant has been demonstrated.

I. INTRODUCTION

Interface-System Loss-of-Coolant Accident (ISLOCA) has been assumed to be caused by failure of isolation valves such as check valves and motor-operated valves (MOVs). An overpressure of low pressure designed piping systems after ISLOCA might result in a leak from the reactor pressure vessel (RPV) to the reactor building.

Probabilistic risk assessments have been evaluated for ISLOCA in Hitachi-GE standard advanced boiling water reactor (ABWR). It has been understood that a risk associated with ISLOCA is sufficiently low on the ground of exceedingly small values of occurrence frequency of ISLOCA. Therefore ISLOCA has been identified as one of beyond design basis accidents. On the other hand, understanding of plant response after ISLOCA is important from the aspect of defense in depth for the plant safety.

Transient behavior of pressure propagation inside piping systems after ISLOCA is one of the important phenomena to understand the plant safety. Especially water hammer effect is well known risk for the pipe design which transiently causes large dynamic loads to piping systems. Kellly et al. simulated pressure distribution for the failure of cold leg in residual heat removal (RHR) system of pressurized water reactor (PWR) by using RELAP code[1]. Despite a steady analysis, they pointed the valve opening period has a large effect on the transient behavior of pressure propagation inside piping systems from the behavior of pressure approaching to steady state. In our previous work, transient analysis of pressure propagation were conducted for simple pipe line and Hitachi-GE standard ABWR plant by using TRACG code[2]. Applicability of TRACG model for ISLOCA was demonstrated by comparison with linear theory. Besides the effects of valve opening area and valve opening period were shown a larger effect on transient behavior of pressure propagation.

In this study, a realistic assessment of ISLOCA in Hitachi-GE standard ABWR plant has been demonstrated by using best estimate codes. At first, water hammer effect in piping systems after ISLOCA has been investigated by using TRACG code. Then, an environmental simulation in the reactor building on the assumption of leak from piping systems has been demonstrated by using MAAP code.

II. SCENARIO OF ISLOCA FOR HITACHI-GE STANDARD ABWR PLANT

High Pressure Core Flooder (HPCF) system is selected as a representative piping systems for Hitachi-GE standard ABWR plant. Because HPCF system has a large flange which has a possibility for a large leak. And the water hammer effect is comparatively larger than those of other piping systems.

Figure 1 shows a HPCF system for Hitachi-GE standard ABWR plant. Low pressure region at 1MPa is isolated from higher pressure region connected to RPV which is about 7.5MPa at normal operation. HPCF system has a check valve and a motor operated valve (MOV) as isolation valves. A pump is under suspension. Water filled in two regions is considered as quiescence state.

ISLOCA is postulated the failure of a check valve at first. Then, ISLOCA initiates from wrong opening of an MOV. After the opening of an MOV, low pressure region of piping systems will be received the overpressure which is possibly not only a static pressure of 7.5MPa but also dynamic loads by the water hammer effect. An overpressure of low pressure region might result in a leak to the reactor building.

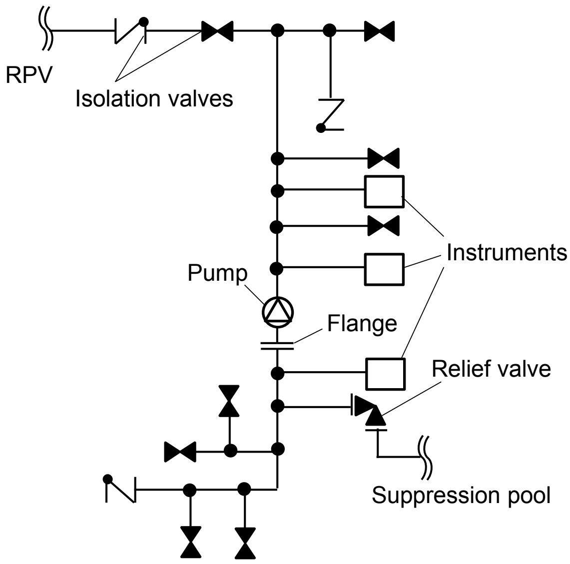

Fig. 1. HPCF system for Hitachi-GE standard ABWR plant

III. SIMULATION OF PRESSURE PROPAGATION INSIDE PIPING SYSTEMS

III.A. Analysis Model and Conditions

TRACG is one of the best estimate codes available for BWR transient analysis [3] [4]. The two-fluid model and conservation equations for mass, momentum and energy are used for the thermal hydraulics in TRACG. TRACG possesses basic 1D component models, such as pipe, tee and valve. By piecing these 1D components together, TRACG can model a complicated network of piping systems.

Figure 2 shows TRACG nodalization of HPCF system for Hitachi-GE standard ABWR plant. Length and inner diameter of pipe line are modeled in detail. Form losses at elbows and bifurcations in pipe line are also modeled. Elevation changes are neglected since the maximum value of head in HPCF system is 0.3MPa at most, which is insignificant as a driving force of transient behavior after ISLOCA compared to 6.5MPa of pressure difference between high pressure region and low pressure region. Relation between opening area and form loss of the MOV are modeled based on designed values. Behavior of the relief valve is also modeled based on designed values. Pressure boundary condition at the RPV side is set as 7.5MPa of saturated water. And pressure boundary condition at the outlet of relief valve to the suppression pool is set as 0.1MPa. 1D components are divided into some meshes at the size of about 0.05m. Initial pressure is 7.5MPa at high pressure region and 1MPa at low pressure region. Initial temperature is set as between 7.5MPa of saturation temperature and 311.15K at high pressure region and set as 311.15K at low pressure region. Initial velocity is 0m/s at all meshes. MOV is set to initiate opening at the start of calculation. Sensitivity analysis is conducted for the opening period of MOV.

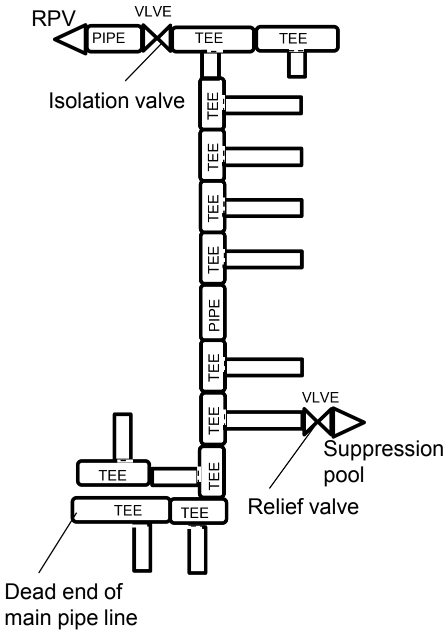

Fig. 2. TRACG Nodalization

III.B. Analysis Results

Figure 3 shows time history of pressure at a nearest mesh of dead end of main pipe line at various opening period of MOV. Maximum dynamic loads are seen at this position. Where α denotes a non-dimensional parameter defined as the ratio of opening period to the actual opening period of MOV, which means that α=1 corresponds to the actual case. After the initiation to open the MOV, water transiently flows from high pressure region to low pressure region in every case. Therefore, pressure in the low pressure region increases. On the other hand, the transient flow is interfered by a form loss of the MOV. As the value of α becomes smaller, a form loss of the MOV comparatively becomes smaller at the same time because the fraction of opening area of the MOV is larger when it is compared at the same time. As a form loss of the MOV becomes smaller, the steeper pressure increases, and it results in the larger dynamic loads at the dead end of main pipe line. Dynamic load oscillates with gradual damping after that, pressure in low pressure region finally settles into 7.5MPa.

Figure 4 shows maximum pressure of first peak oscillation at a nearest mesh of dead end of main pipe line. Maximum pressure of actual HPCF system for Hitachi-GE standard ABWR plant is at most about 8.2MPa.

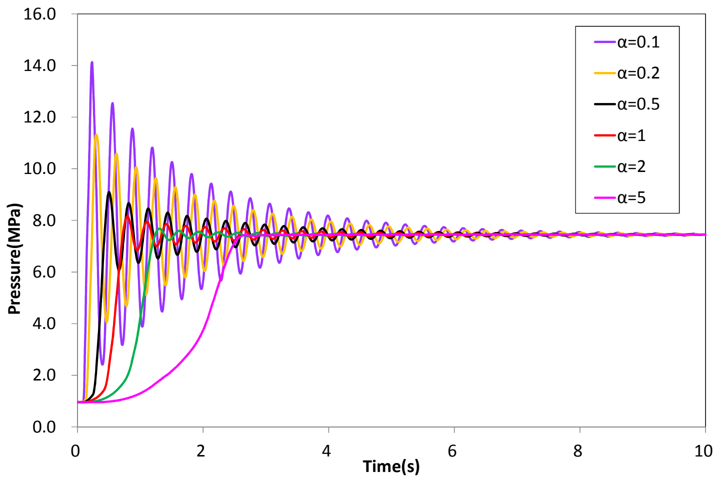

Fig. 3. Time history of pressure at a nearest mesh of dead end of main pipe line

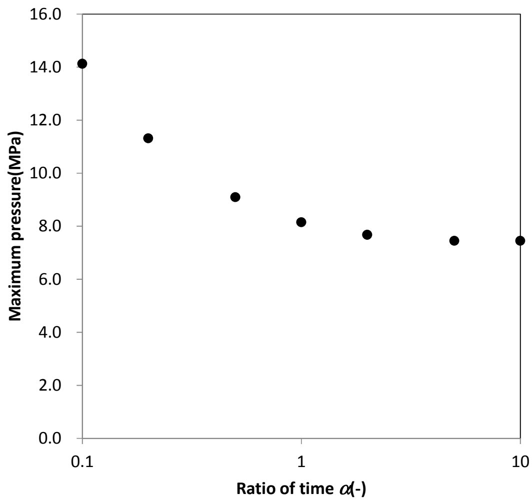

Fig. 4. Maximum pressure at a nearest mesh of dead end of main pipe line versus ratio of time α

IV. SIMULATION OF ENVIRONMENTAL TEMPERATURE IN THE REACTOR BUILDING

IV.A. Analysis Model and Conditions

MAAP is one of the best estimate codes available for BWR severe accident analysis. Transient behavior in RPV, pressure containment vessel (PCV) and Reactor building by realistic operations can be simulated.

Figure 5 shows MAAP nodalization for the simulation after ISLOCA in Hitachi-GE standard ABWR plant. RPV, PCV and reactor building are divided into some region and a thermal hydraulic behavior such as pressure, temperature and also water level are simulated in each region. Leak from RPV to reactor building is modeled by using junction model shown in red dotted line in Figure 5. Blowout panel is modeled to open at a designed pressure difference between other reactor building and environment. Major analysis conditions are shown in Table1. Leak room and leak area are assumed based on the result of realistic fragility evaluations. Flange and instrument in the HPCF pump room are pick out as possible components for leak. Although at most 1cm^2 of leak area in total by realistic fragility evaluations, 10 cm^2 of leak area is assumed for MAAP analysis as a severe condition. Some operations after ISLOCA are assumed in a realistic manner shown in Table1.

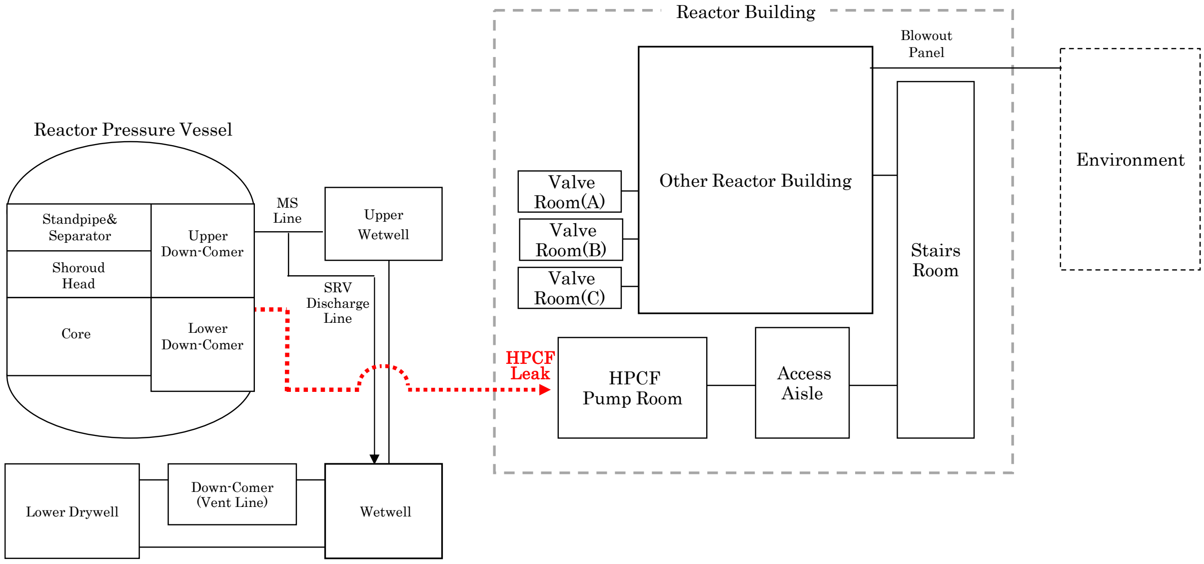

Fig. 5 MAAP Nodalization

TABLE 1. Major Analysis Conditions

Items	Conditions
Leak Room	HPCF(B) Pump Room
Leak Area	$10cm^2$
Operation after ISLOCA	・Scrum after 10 minutes ・MSIV closed after 25 minutes ・RPV depressurization after 30 minutes ・Start-up of S/P cooling system after 60 minutes ・Control of RPV water level under the leak height

IV.B. Analysis Results

Figure 6 shows time history of leak rate to reactor building. Leak rate shows steep decrease by the depressurization of RPV after 30minutes. Furthermore leak rate is decreased after 2.5hours. It is because the RPV water level becomes lower than the leak height. It means that the water leak is substituted for the steam leak at this point of time. After that leak rate is almost maintained lower value due to the control of RPV water level under the leak height.

Figure 7 shows time history of temperature in the reactor building. Provided that a remote control of MOV is failed to close, operator needs to access in the valve room (B) to close the MOV manually for the isolation of ISLOCA. In that case, other reactor building is also expected as an access route. Temperatures in both regions are decreased by the depressurization of RPV after 30minutes. It is because that an air in the environment starts to flow into the reactor building through the blowout panel by depressurization of RPV. Temperatures in both regions can be mitigated below 40degrees C after 2 hours.

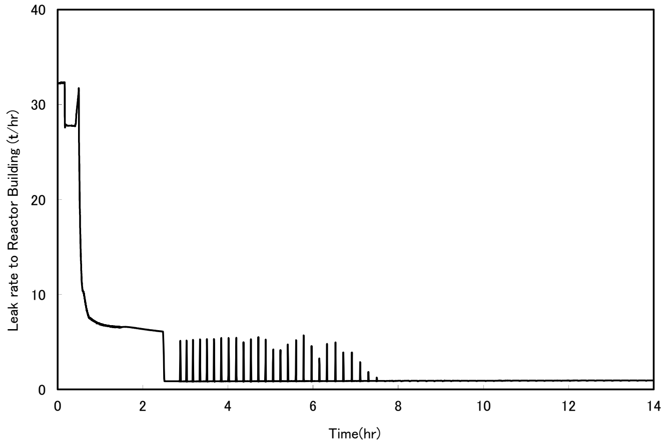

Fig. 6 Time history of leak rate to reactor building

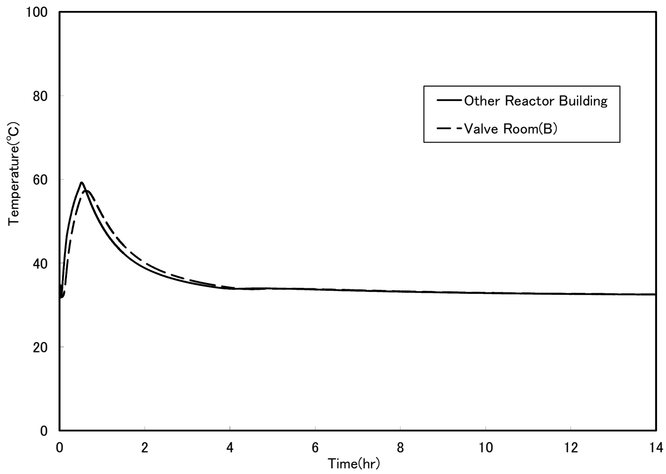

Fig. 7 Time history of temperature in the reactor building

V. CONCLUSIONS

A realistic assessment of ISLOCA in Hitachi-GE standard ABWR plant has been demonstrated by using best estimate codes. Effect of the opening period of MOV is quantitatively demonstrated. Maximum pressure of actual HPCF system for Hitachi-GE standard ABWR plant is at most about 8.2MPa. An environmental simulation in the reactor building after the leak of piping systems has been demonstrated. RPV depressurization has been considered as an effective operation to decrease the leak rate to reactor building and temperature in reactor building. Even though we have assumed leak area more than ten times larger than that of realistic fragility evaluation, Hitachi-GE standard ABWR plant has been capable of mitigation and isolation of ISLOCA by realistic operations.

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