Results of Our Investigation of the Accidents and Subsequent Development at Fukushima Daiichi Nuclear Power Station

Described below are the results of our latest investigation of pressure behavior in the Reactor Pressure Vessels and the Primary Containment Vessel (PCV) drywells (D/W) of Units 2 and 3 of Fukushima Daiichi Nuclear Power Station and the venting operation of Units 1, 2 and 3 of Fukushima Daiichi Nuclear Power Station in and after the accidents.

This report describes the latest information obtained from interviews, etc. Conditions required for the analysis cover presumptions and assumptions based on equipment design data and plant data. Thus, as the investigation progresses, new findings and analytical results may emerge, which differ from the information in this report.

1. Our Opinion on the Reactor Pressure and Primary Containment Vessel Pressure of Unit 2

(1) Introduction

The followings are our presumptions of the reasons for the inconsistency between the measured values and analytic values of changes in the reactor pressures in Figs. 8-13-2 and 8-13-12, and the reasons for the inconsistency between the measured values and analytic values of the Primary Containment Vessel (D/W) in Figs. 8-13-3 and 8-13-13 in the report titled "Impacts on the Nuclear Facilities of Fukushima Daiichi Nuclear Power Station in the Tohoku-Pacific Ocean Earthquake" submitted to the Nuclear and Industrial Safety Agency on September 9, 2011 (hereafter referred to as "the September 9 Report"). Figures 8-13-2, 8-13-12, 8-13-3, and 8-13-13 shown in the September 9 Report respectively correspond to Figs. 1-1 through 1-4 in this report. Boxes surrounded by a red boarder in the figures are areas where analytic values are inconsistent with measured values.

(2) Regarding changes in the reactor pressure

(2)-1 Analysis in the September 9 Report

Although the control power supplies of Unit 2 were lost due to the effect of the tsunami, the Reactor Core Isolation Cooling System (RCIC) remained in service. It was at 13:25 on March 14, 2011 that the RCIC was judged to have been shut down by a drop in the reactor water level. Over the two days following the earthquake until the halt, the RCIC continued to inject water into the reactor core. In the interim, the fuel region water level indicator had measurement values of the reactor water level between approx. 3,400 mm to 3,950 mm above the top of active fuel (TAF). The measured value of the reactor pressure fell from approx. 7.1 MPa[abs], measured at 20:07 on March 11, and remained stable in a range of approx. 5.4 to 6.4 MPa[abs], which are pressures slightly lower than in ordinary operation.

Considering that the RCIC's service was not confirmed after the loss of the power supply and that the reactor water level was maintained while the RCIC was still in service, the September 9 Report in its analysis assumed that the RCIC continued operating at the rated flow (95m^3/h) and repeated automatic stopping and automatic startups in the reactor water levels ranging between L-2 and L-8. The analysis accordingly assumed that the reactor pressure was maintained due to closings and openings of the Safety Relief Valve. This resulted in the inconsistency with the measured values. Moreover the operating state of RCIC until discontinuance of water injection to the reactor has barely any effect on the state of the reactor core after the discontinuance of water injection, as long as the reactor water level is maintained.

Since changes in the behavior of the reactor pressure were presumably closely related to the RCIC's state of operation, we examined the RCIC's state of operation as outlined below.

(2)-2 Presumed state of the RCIC operation

In light of items 1) and 2) below, we consider that the RCIC that lost its control power supply was neither in service in design operation mode (rated flow) nor in repetition of halts and startups due to the reactor water level (L-2 and L-8).

1) Compensation of the reactor water level indicator

The fuel region water level indicator of Unit 2 continued to measure the reactor water level after the accident on March 11. Since the fuel region water level indicator primarily aims to monitor the water level in the loss-of-coolant accident of the reactor, the indicator is compensated for the atmospheric pressure and saturated temperature. Thus, when the reactor is in high pressure or when the D/W is in high temperature, the fuel region water level indicator requires compensation because it does not indicate the actual water level.

When we compensated(*) the measured reactor water levels with the reactor pressures and the D/W temperatures, the values indicated around the surface of the reference leg of the water level indicator (TAF + approx. 5,916 mm) (Fig. 1-5). Since the RCIC is supposed to trip when the reactor water level reaches L-8 (TAF + 5,653 mm), the water level will not rise to L-8 or higher. However, since the control power supply was lost, we presume that the RCIC continued operating without control.

Taking into account a decrease in the decay heat, there was a strong possibility that the reactor water level rose to L-8 or higher. Due to the mechanism of the water level indicator, if the reactor water level rises above the surface of the reference leg, the differential pressure between the pipe on the reference leg side and the pipe on the reactor side (Hs-Hr indicated in Fig. 1-6) no longer indicates changes. As a result, the reactor water level apparently remains at the height of the surface of the reference leg.

Thus, based on our explanation above, we consider that the reactor water level while the RCIC was in service exceeded L-8 and further rose above the surface of the reference leg.

*: Compensating the reactor water level requires the value of the reactor pressure and of the D/W temperature measured at the time the reactor water level was measured. Thus, in compensating the water level at that time without having the actual measurement of the reactor pressure, we applied linear interpolation based on the reactor pressure measured at another time to obtain a rough estimation at the said time. In addition, since no actual measurement value of the D/W temperature at the said time was available, we used a value from analytic results indicated in the report titled "Analysis of Operation Records and Accident Records of Fukushima Daiichi Nuclear Power Station and Assessment of Impacts at the Time of the Tohoku-Pacific Ocean Earthquake" reported to the Nuclear and Industrial Safety Agency on May 23, 2011.
Accordingly, the compensation value of the reactor water level in Fig. 1-5 includes errors of assumption of the reactor pressure and of the D/W temperature, in addition to measurement errors.
In connection with this, the compensation curve of the reactor water level based on the reactor pressure and the D/W temperature is described in the Accident Operation Procedure.

2) Motive steam of the RCIC

As mentioned earlier, there was a possibility that the reactor water level was above the surface of the reference leg of the water level indicator. In addition, if the water level rose above the height of the Main Steam line (TAF + approx. 7,301 mm) or higher, it is considered that the carryover of water droplets on the Main Steam line would be significant. Thus, there was a possibility that the motive steam of the RCIC was in a two-phase flow. Although quantitative assessment of the RCIC's water injection capability in a two-phase flow and in a low-quality state was difficult, it was possible that the RCIC was injecting water at a flow rate lower than the rated flow rate due to a decline in the number of turbine revolutions.

(2)-3 Results of MAAP analysis

Figure 1-7 shows behavior of the reactor pressure obtained from the MAAP analysis performed based on the presumptions in item (2)-2. On the presumption that the RCIC's flow rate was 30m^3/h, or approx. 1/3 of the 95m^3/h rated flow, we obtained the behavior of the reactor pressure that can roughly reproduce the actual measurement.

The reactor pressure remained in stable pressure (approx. 5.4 to 6.4 MPa[abs]) during the RCIC operation, which is lower than the pressure in normal operation. This is attributable to the presumption that the RCIC was actuated in a two-phase flow where saturated energy becomes larger than steam. In such case, the heat transfer quantity from the Reactor Pressure Vessel to the suppression chamber (S/C) becomes greater than that in ordinary operation. As a result, there was a possibility that heat carried out of the Reactor Pressure Vessel was balanced with the decay heat.

(2)-4 RCIC operation in light of design

It is generally assumed that even when the quality of steam flowing into the RCIC turbine becomes somewhat inferior to the design conditions, it will neither damage the wings nor immediately decelerate the turbine. Moreover, it is assumed that drain water discharged into the S/C direction will not be immediately accumulated in the turbine. Thus, the RCIC operation in a two-phase flow could persist.

Furthermore, if the water level rises and the Main Steam line (RCIC steam supply line) submerges or nearly submerges, steam will not be supplied adequately to the RCIC turbine. As a result, the turbine could be decelerated and then stopped. However, it is possible that the reactor water level is maintained at a height proximate to the height of the Main Steam line when the turbine does not cease immediately and when decrease in injection water resulting from deceleration leads to a drop in the reactor water level, allowing steam inflow.

In connection with this, if the RCIC's control power supply is lost, under the current design, the regulator valve will be opened fully and unable to adjust the flow rate (Figure 1-8 shows a schematic diagram of the RCIC system).

(2)-5 Presumptions of the RCIC's functional decline

It is judged at 13:25 on March 14 that RCIC had stopped by a drop in the reactor water level. As mentioned earlier, however, we consider that the value of the water level indicator after pressure compensation indicated constant values around the surface of the reference leg, and falls in the measured values of the water level observed from around 12:00 indicated that the water level at a higher level had dropped to that level. Accordingly, we consider that the RCIC's function declined earlier than around 12:00 when the drop in the water level was observed. According to changes in plant data, the reactor pressure rose from around 9:00 on March 14. We presume the rise was attributable to a decrease in the injection water from the RCIC due to a decline in the RCIC's function and to a decrease in the steam supply to the RCIC turbine.

Moreover, the rise in the pressure is more moderate than the rise in the pressure anticipated in ordinary halt of the RCIC (steam supply stopped by closing the turbine valve). This is presumably because the valve on the steam supply side did not close due to the loss of the control power supply.

(2)-6 Conclusions

Based on our description above, while uncertainty still remains, we presume that the reactor pressure remained stable in pressures lower than that in ordinary operation. This is attributed to a number of assumptions that the RCIC continued operating without being controlled due to the loss of its control power supply; a resulting rise in the reactor water level to L-8 or higher led to the transfer of energy equivalent to decay heat from the reactor due to the RCIC's operation at a low-quality two-phase flow; which also led to a smaller amount of water injection than that of the rated flow rate; and a series of these factors established a balance of energy in the Reactor Pressure Vessel without the actuation of the Safety Relief Valve.

(3) Changes of pressure in the Primary Containment Vessel

(3)-1 Regarding analysis in the September 9 Report

If removal of heat from the Primary Containment Vessel is inadequate, the D/W pressure and the S/C pressure will rise because steam generated in the reactor core is exhausted into the S/C through the RCIC and the SRV. The measured values of the D/W pressure and the S/C pressure of Unit 2 indicated a slower rise than the anticipated behavior during the period from around 0:00 on March 12 to 12:00 on March 14, 2011.

Analysis in the September 9 Report (Figs. 1-3 and 1-4) assumed in its simulation that, with limited information available, the slow rise in the pressure of the Primary

Containment Vessel was attributable to leakage from the D/W, which in reality is an unlikely event in our opinion. Based on the analytic value, the analysis determined that the leakage occurred at the time when the temperature of the Primary Containment Vessel exceeded the design temperature (138°C).

According to findings in past researches(*), however, leakage from the Primary Containment Vessel due to hyperthermia is likely to occur on a gasket, etc., and we know that the temperature when such leakage may occur is approx. 300°C. Thus, leakage from the Primary Containment Vessel is unlikely to occur when the temperature reaches the design temperature (138°C). In addition, since the analysis in the September 9 Report assumes leakage from the Primary Containment Vessel, it failed to reproduce states of the rapid rise in the pressure of the Primary Containment Vessel from around 22:40 on March 14 and a continuous high pressure state.

Based on our description above, therefore, we consider a scenario, besides the leakage, whereby a rise in the pressure of the Primary Containment Vessel was suppressed. We investigated this scenario as follows.

*: Hirao, T. Zama, M. Goto et al., "High-temperature leak characteristics of PCV hatch flange gasket," Nucl. Eng. Des.,145, 375-386 (1993).

(3)-2 Regarding possibility other than leakage

The September 9 Report adopted the assumption of a leak and conducted analysis on that assumption. However, the mechanism of heat removal from the Primary Containment Vessel must be considered in order to reproduce a state where a rise in the D/W pressure and the S/C pressure is suppressed while heat transfers into the S/C due to exhaust steam, etc. from the RCIC. In particular, there could exist a state where spraying, etc. from an external water source continued to cool down the inside of the Primary Containment Vessel or where surface heat on the wall of the Primary Containment Vessel transferred adequate heat outside. During the period from around 0:00 on March 12 through around 12:00 on March 14, no operation was conducted to cool down the Primary Containment Vessel. Thus, there is a possibility that surface heat of the wall of the Primary Containment Vessel transferred outside.

The S/C is doughnut-shaped with a very large surface area. However, since heat transfer by air is limited, ample heat transfer is unlikely to occur. Meanwhile, on the assumption that the basement of the building was submerged due to the effects of the tsunami, one scenario for the heat transfer route could be flooding of the torus room containing the S/C; the heat transferred to the S/C wall conveyed the heat to the water flooding the torus room. Since surface heat transfer by water is efficient, it is possible that adequate heat removal was performed to suppress a rise in the pressure of the Primary Containment Vessel.

Thus, we performed MAAP analysis on the presumption that the torus room was gradually submerged by flooded seawater (approx. 10°C) and that ultimately the lower half of the S/C was submerged. As a result, we were roughly able to reproduce a slow rise in the pressure for the period from around 0:00 on March 12 through 12:00 on March 14 (Fig. 1-9). We also computed and obtained an analytic value that is lower than the measured value for a rapid rise in the pressure from around 22:40 on March 14. The lower analytic value could be attributable to a number of factors, including the factor that the amount of hydrogen generation in our analysis was smaller than the actual value amount. Because we presume no leakage from the Primary Containment Vessel, we were roughly able to reproduce the behavior, indicated in the measured values, of the pressure of the Primary Containment Vessel (Fig. 1-9), including the behavior whereby the Primary Containment Vessel maintained the raised pressure.

(3)-3 Regarding possibility that the torus room is submerged

We did not inspect whether the torus room was actually submerged or not. However, it is confirmed that the RCIC room and the basement of the turbine building, etc. were submerged in an early stage after the tsunami. Water levels of present stagnant water in buildings indicate that flooded seawater has been moving through cable penetrations, etc. among these buildings. These observations indicate that there is a possibility that the torus room, which is located on the lowest floor of the reactor building, was submerged due to the effects of the tsunami.

In connection with this, it is known that the lower half of the S/C in the torus room of Unit 4, whose structure is roughly the same as that of Unit 2, was submerged (Fig. 1-10). Although there is a difference in that Unit 4 was under regular inspection on March 11 while Unit 2 was in service, it is possible that the torus room of Unit 2 was submerged, as that of Unit 4 was underwater.

(3)-4 Conclusions

The assumption in the September 9 Report that leakage occurred when the temperature of the Primary Containment Vessel reached the design temperature is presumably unlikely to occur in view of the design.

Our analysis on the presumption that the removal of heat from the Primary

Containment Vessel was attributable to the stagnant water in the torus room was able to reproduce, more precisely, a state of slow rise in the pressure of the Primary Containment Vessel from around 0:00 on March 12 through around 12:00 on March 14 and a state of rapid rise in the pressure of the vessel from around 22:40 on March 14. Therefore, we consider that this mechanism suppressed a rise in the pressure of the D/W.

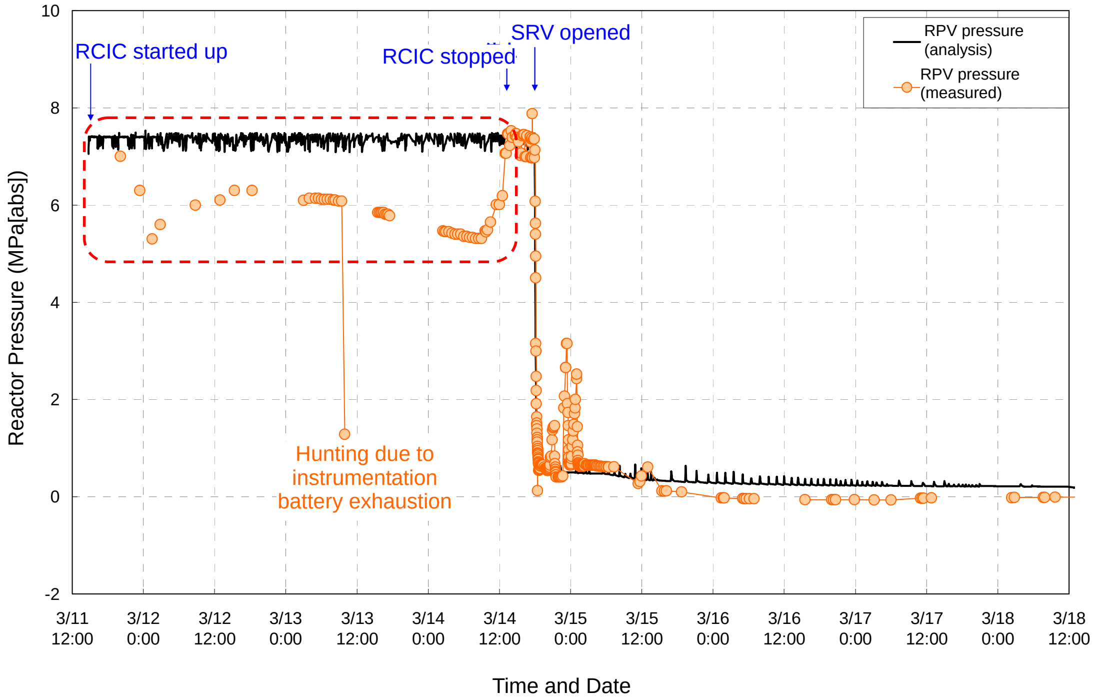

Figure 1-1. Behavior of Reactor Pressure of Unit 2 (Fig. 8-13-2 in the September 9 Report)

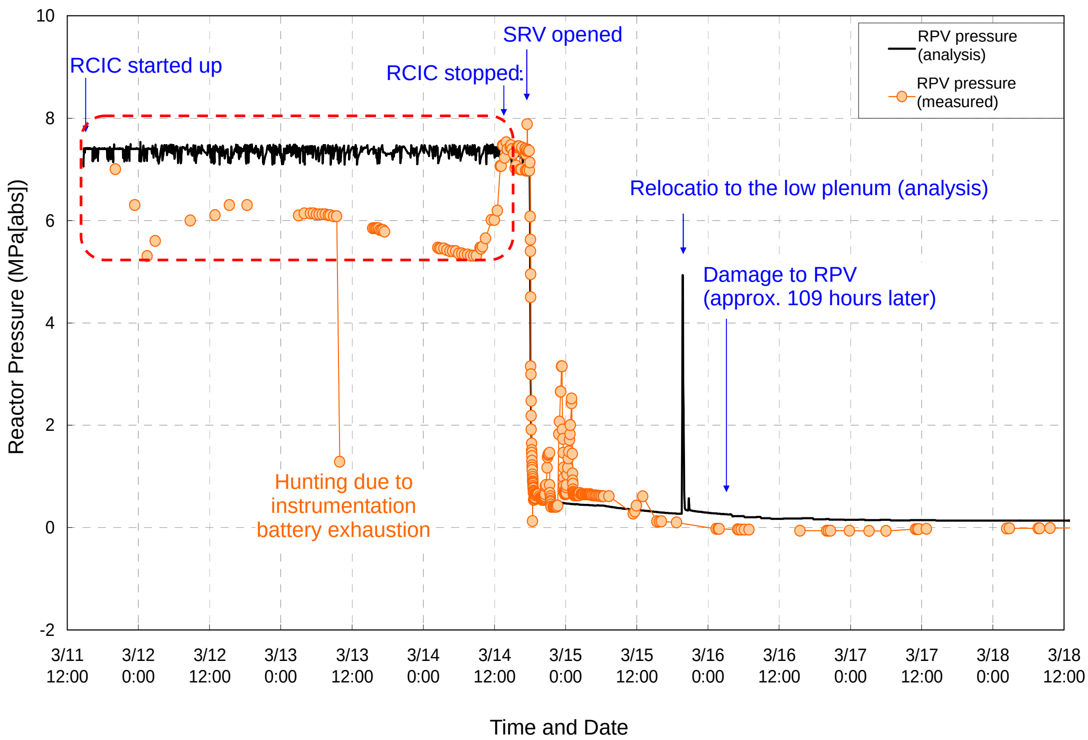

Figure 1-2. Behavior of Reactor Pressure of Unit 2 (Fig. 8-13-12 in the September 9 Report)

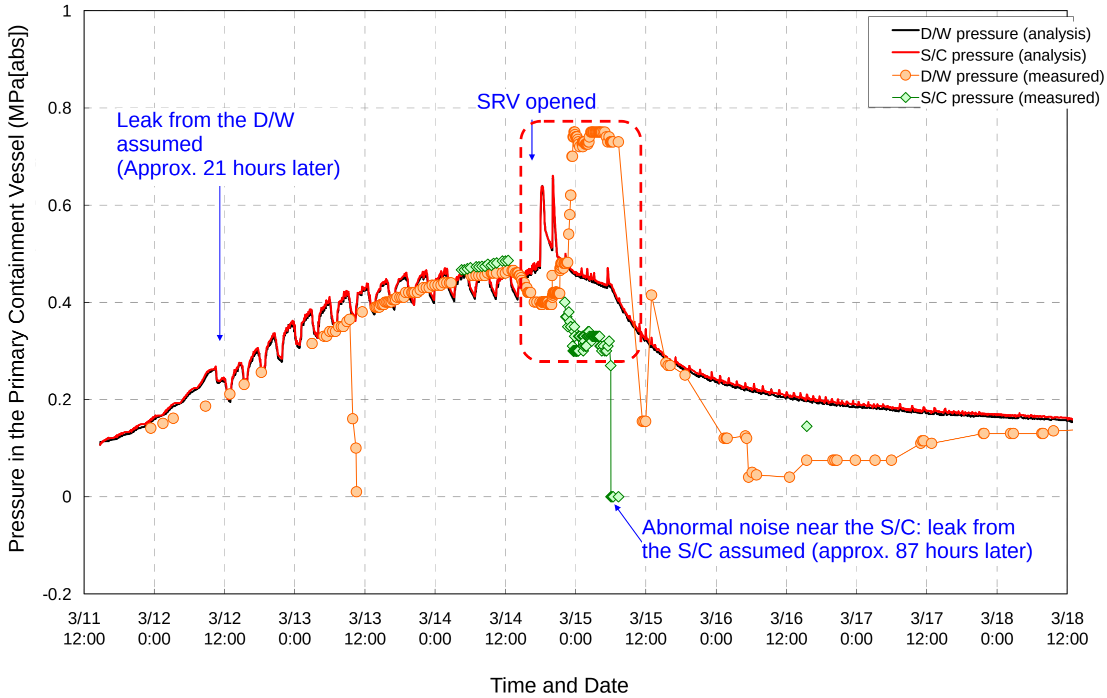

Figure 1-3. Behavior of Pressure of Unit 2 Primary Containment Vessel
(Fig. 8-13-3 in the September 9 Report)

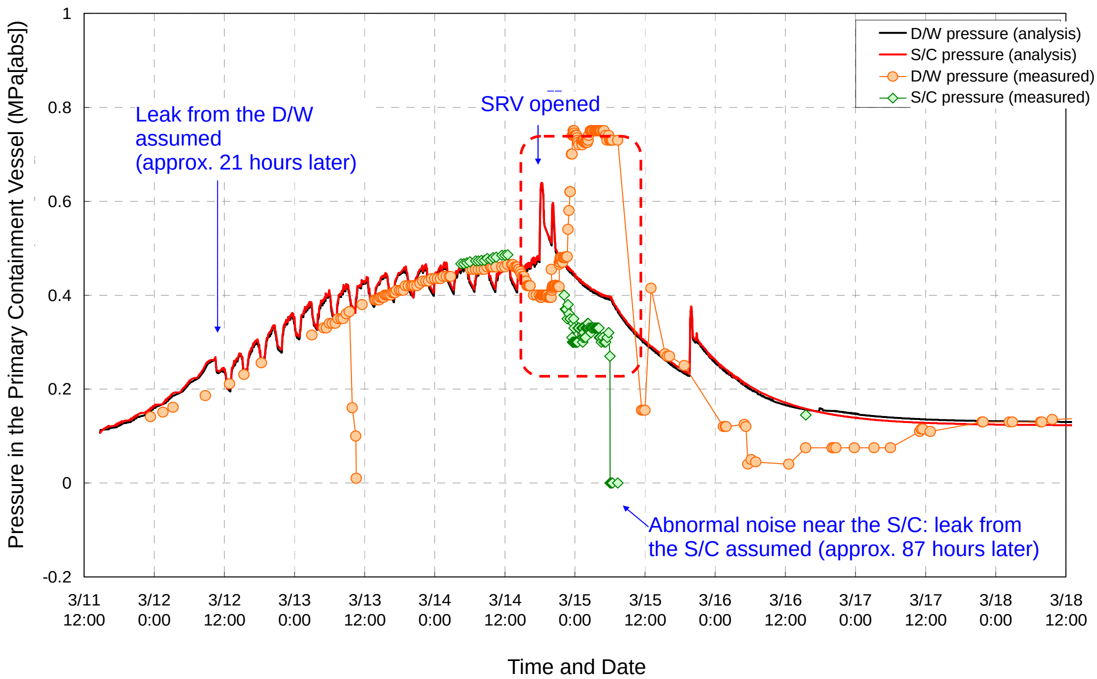

Figure 1-4. Behavior of Pressure of Unit 2 Primary Containment Vessel
(Fig. 8-13-13 in the September 9 Report)

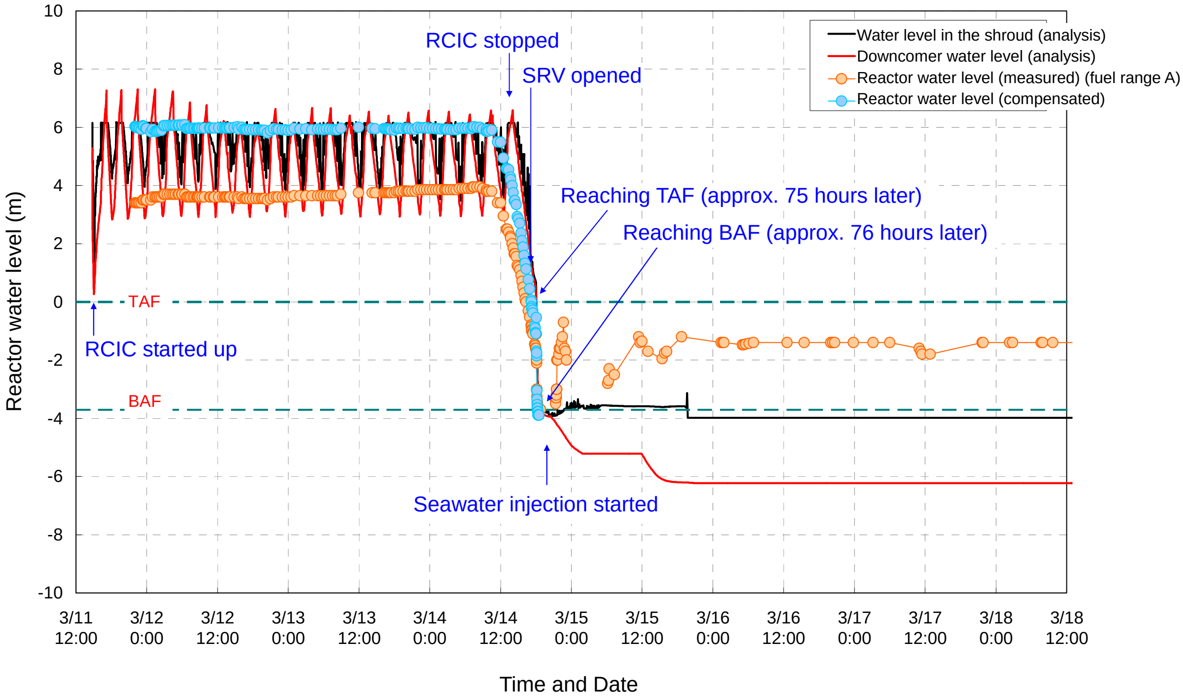

Figure 1-5. Changes in Reactor Water Level of Unit 2

(Water level after compensation was added to the result of MAAP analysis of May 23, 2011 [2/n])

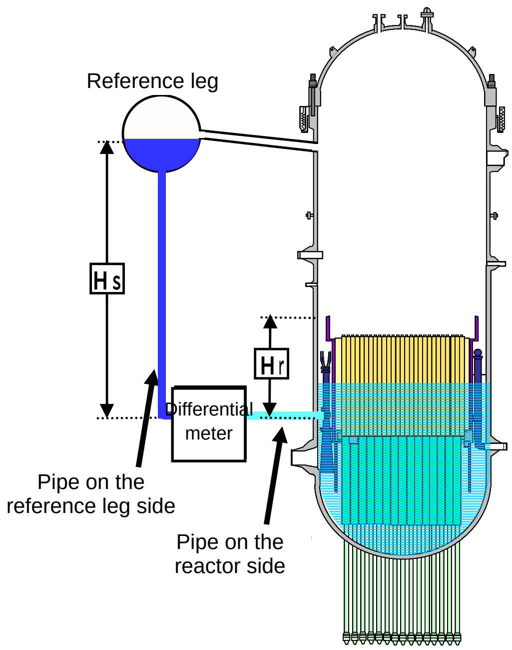

Figure 1-6. Structure of Reactor Water Level Indicator

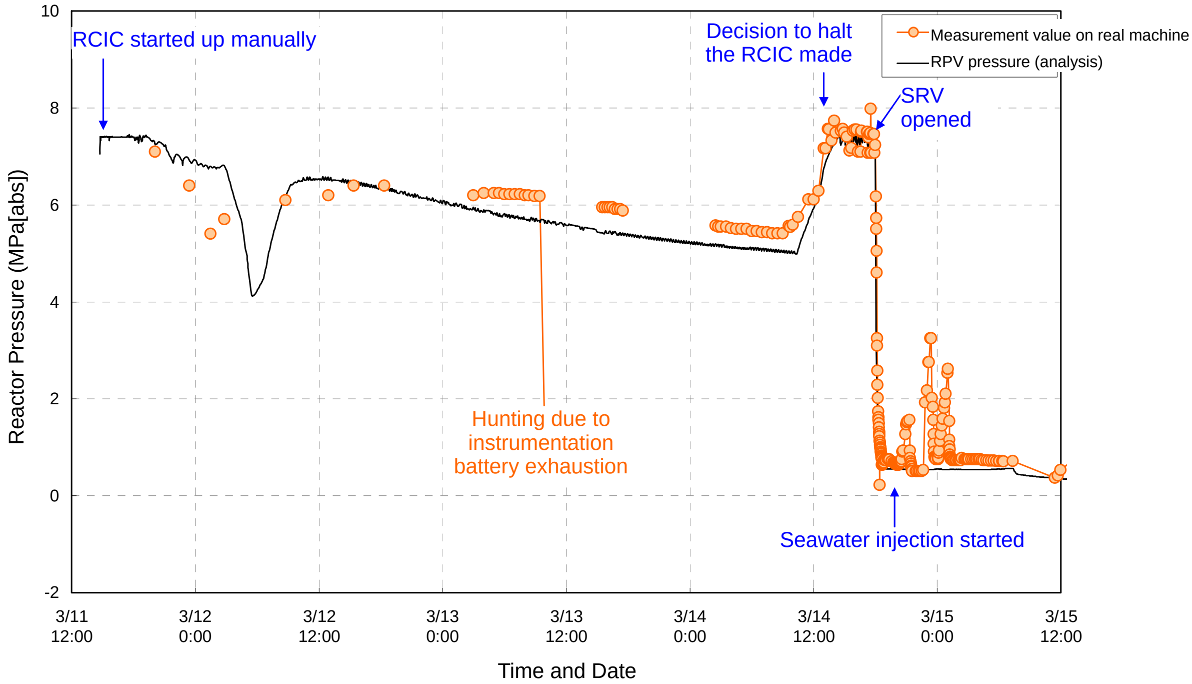

Figure 1-7. Changes in Reactor Pressure of Unit 2

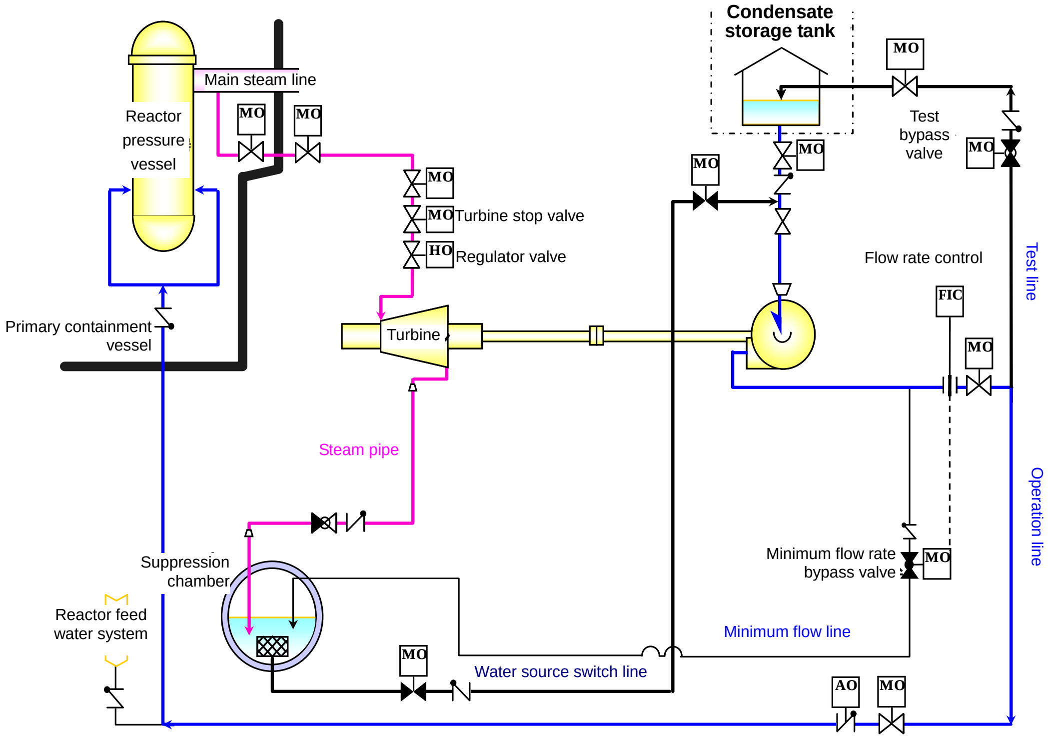

Figure 1-8. Schematic Diagram of the RCIC System

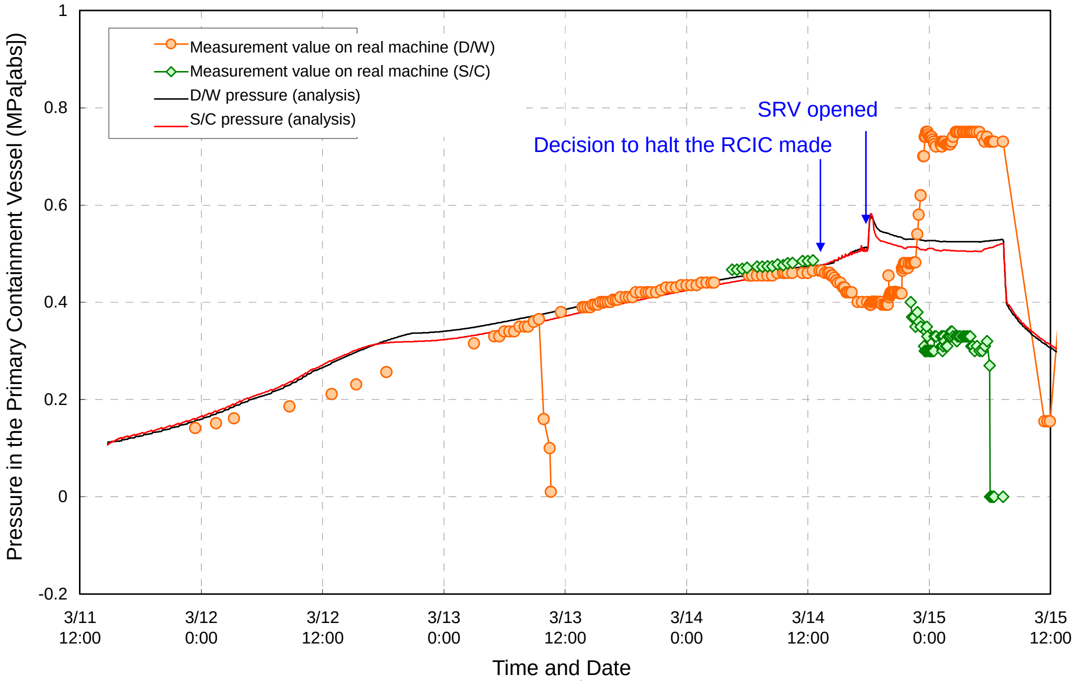

Figure 1-9. Changes in Pressure of Unit 2 Primary Containment Vessel

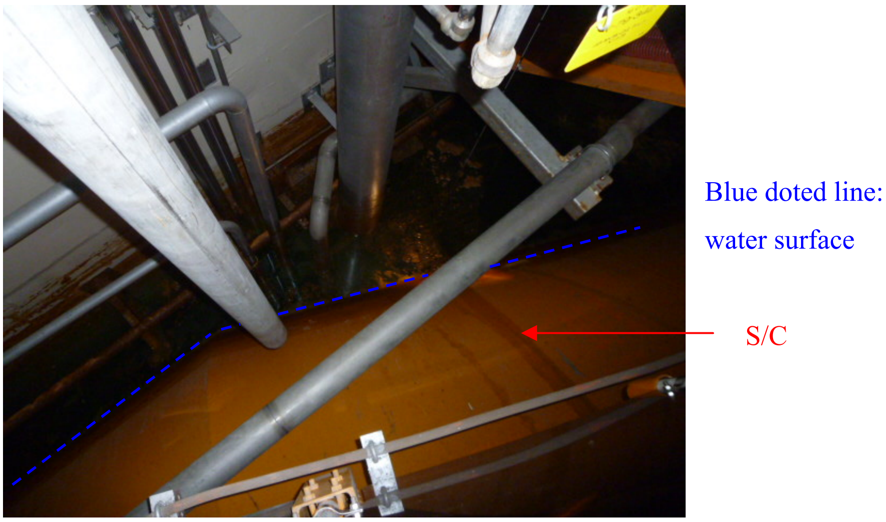

Figure 1-10. Photo Beneath the Catwalk in the Torus Room of Unit 4

2. Our Opinion on Unit 3 Operation State and Reactor Pressure while the High Pressure Coolant Injection System (HPCI) was in Service

2.1 Operation state of Unit 3 while the HPCI was in service

Indicated below are the state of operations, including flow rate adjustment and other manual operations, and reactor pressure conditions while the HPCI system was in service. Figure 2-1 shows a schematic diagram of the HPCI system.

- It is confirmed that the RICI system had stopped automatically at 11:36 on March 12. Later at 12:35, the HPCI system started up automatically due to a low level of reactor water and resumed water injection into the reactor. Decline in the pressure of the reactor began when the turbine to actuate the HPCI system exhausted the steam of the reactor.
- Operators operated the HPCI control panel in the main control room to supply the reactor with water through the water injection line and the test line of the HPCI line formation. Moreover, the minimum flow line was set to shut-off in fear of a rise in the S/C water level.
- The HPCI system has a large flow rate capacity. Thus, the adjustment range of reactor water levels was set wider so as to prevent the HPCI's automatic shut down due to a high reactor water level.
- Concerning the flow rate adjustment, operators set the flow rate by adjusting the aperture of the valve for the test line and with the flow controller (FIC) to save battery power so that the reactor water level has a slow change; operators adopted a method of repeating changes in the flow rate setting (from 100% rated flow to approx. 75%) whenever the reactor water level approached the upper or lower end of the adjustment range of water levels.
- The power supply of the reactor water level indicator was lost at 20:36 on March 12. This prevented monitoring of the reactor water level. Thus, operators raised the HPCI flow rate setting slightly and continued monitoring the operation state by using the reactor pressure and HPCI discharge pressure and others.
- The reactor pressure that had remained stable at approx. 1 MPa began to decline at 2:00 on March 13. It was feared that damage would occur in some equipment items due to stronger vibrations of the turbine that would occur when the number of HPCI turbine revolutions decreased further due to the decline in the reactor pressure. In addition, it was presumed that the HPCI system failed to inject water

into the reactor because the reactor pressure was roughly the same as the HPCI discharge pressure. Based on these conditions, a decision was made to use a Diesel-Driven Fire Pump (DDFP) as a substitute for the HPCI to inject water into the reactor and to shut down the HPCI promptly.

- The HPCI system was manually halted at 2:42 on March 12 via the HPCI control panel in the main control room to start water injection in the reactor via the DDFP, a substitute for the HPCI to inject water.

2.2 Our opinion on Unit 3 reactor pressure while HPCI was in service

(1) Background

Concerning the HPCI operation of Unit 3, Appendix-1, States of Cores of Units 1 to 3 of Fukushima Daiichi Nuclear Power Station, attached to the report titled "Analysis of Operation Records and Accident Records of Fukushima Daiichi Nuclear Power Station and Assessment of Impacts at the Time of the Tohoku-Pacific Ocean Earthquake" reported to the Nuclear and Industrial Safety Agency on May 23, 2011, contains the following description in the assessment on the state of the reactor core of Unit 3, referring to an example of conditions that meet measured behavior: “Pressure was on the decline in areas where the HPCI was in service. Analysis, for example, on the assumption that steam leaked through the HPCI steam pipe outside the Primary Containment Vessel obtained the result of behavior roughly corresponding to the behavior of the reactor pressure and the Primary Containment Vessel pressure.”

In the subsequent investigation and assessment, the HPCI system, with adjustment of the flow rate, was found to be in continuous service. The analysis on the assumption that water from the condensate storage tank, as the water source, flowed through the minimum flow line into the S/C to maintain a constant flow rate of the HPCI system obtained results that can explain the behavior of the reactor pressure and the Primary Containment Vessel pressure. Thus, the abovementioned description was announced on July 28, 2011.

Recently, we obtained the following new findings: the minimum flow line had been shut off in fear of a rise in the S/C water level; the HPCI flow rate was adjusted by appropriating a portion of the injection water for the reactor to the test line; and spraying had been conducted to cool down the Primary Containment Vessel. Based on these new findings, HPCI operation was organized as follows.

In addition, although the assumption of the HPCI operation state of July 28 was different from the actual state of operation, it had no significant modification in its major parts that steam from the reactor continued flowing out through the HPCI steam pipe and that low-temperature water brought in from outside into the S/C suppressed a rise in the pressure of the Primary Containment Vessel. As a result, our analysis resulted in a similar trend to those of the analysis results dated July 28. In addition, analysis of the reactor core state of Unit 3 indicated no particular condition because the reactor water level was maintained while the HPCI system was in service.

(2) Regarding field conditions and operation

○ Regarding the field conditions

· In the event that a large quantity of steam leaked out from the Primary Containment Vessel through the HPCI steam pipe, the reactor building including the HPCI room would be in a high temperature or high steam atmosphere, preventing workers from entering the building. On the contrary to this assumption, however, after the HPCI system was halted on March 13, operators did enter the RCIC room through the HPCI room to try to restart the RCIC system.

○ Regarding operation of the High Pressure Coolant Injection System (HPCI)

· After the HPCI system started up, operators adjusted the HPCI flow rate by checking the reactor water level to prevent halts and startups of the HPCI system due to a high reactor water level (*).

*: The flow rate adjustment was made through the test line. In addition, the minimum flow line was shut off in fear of a rise in the S/C water level (Fig. 2-1).

· Meanwhile, the HPCI system continued injecting water into the reactor from the condensate storage tank as a water source.

○ Regarding spraying the Primary Containment Vessel with water

· Workers sprayed the Primary Containment Vessel with water to lower the pressure and temperature of the Primary Containment Vessel under the time series indicated in Table 2-1 (Fig. 2-2) below.

Table 2-1. Time Series of Spraying Unit 3 Primary Containment Vessel

Date	Time	Event
Mar. 12	12:06	Started spraying the S/C with water from the DDFP
Mar. 13	3:05	Stopped spraying the S/C with water from the DDFP
	5:08	Started spraying the S/C with water from the DDFP
	7:39	Started spraying the D/W with water from the DDFP
	7:43	Stopped spraying the S/C with water from the DDFP
	8:40 to 9:10	Stopped spraying the D/W with water from the DDFP

*: The operation procedure stipulates that repeated operation of startups and halts due to a low level of reactor water (L-2)/high level of reactor water (L-8) shall not lead to disruption in the continuous operation of the HPCI system.

(3) Regarding possibility of HPCI pipe rupture in light of equipment design

The analysis released on May 23, 2011 introduces the assumption that steam leaks through the HPCI steam pipe out of the Primary Containment Vessel, as an example of a condition that meets the measured behavior of the reactor pressure and the Primary Containment Vessel pressure. As we released on July 28, however, it is unlikely in light of equipment design that the HPCI pipe ruptured and led to a large quantity of steam leakage.

- While the HPCI system was in service, the reactor water level was maintained. Thus, steam generating from the reactor was supplied for the HPCI system that injects water into the reactor.
- If it is assumed that the HPCI steam pipe ruptured and steam leaked out of the Primary Containment Vessel, an isolation signal would have been issued upon the HPCI steam rupture (steam flow maximum); temperature sensors installed in the HPCI turbine/pump room, steam supply line penetration room, etc. around the HPCI steam pipe will issue an isolation signal in high atmospheric temperatures; as a result, the HPCI system will not start up or halt operation. This state does not coincide with the state whereby the reactor water level was maintained.

(4) Regarding factors of plant behavior including decline in the reactor pressure

Figures 2-3 and 2-4 show the results of MAAP analysis (changes in the reactor water level and changes in the reactor pressure) after the HPCI flow rate was adjusted and the operation in Table 2-1 was taken into account. Moreover, in our analysis, we changed injecting quantities of the water from the RCIC and HPCI systems to simulate the measured water levels.

- In general, when the HPCI system starts injecting water, the reactor pressure declines as the HPCI system continues to inject water. Meanwhile, the temperature of the water in the S/C increases because steam exhausted by the HPCI system for water injection is compressed by the S/C, leading to a rise in the pressure of the Primary Containment Vessel.
- In our analysis of the behavior, the quantity of water injected by the HPCI system was adjusted through the test line to prevent unnecessary halt of the HPCI system due to the reactor water level (L-8) although the reactor pressure declines due to the HPCI system's continuous operation. In addition, we consider that spraying the Primary Containment Vessel with water suppressed a rise in the pressure and temperature.

(5) Conclusions

As described in the report released on July 28, 2011, we consider that the decline in the reactor pressure was not attributable to a rupture of the HPCI pipe but to continuous operation of the HPCI system after considering the fact that workers entered the building after the accidents and in light of equipment design. Moreover, since the minimum flow line was shut off, we believe that spraying the Primary Containment Vessel with water suppressed a rise in the pressure.

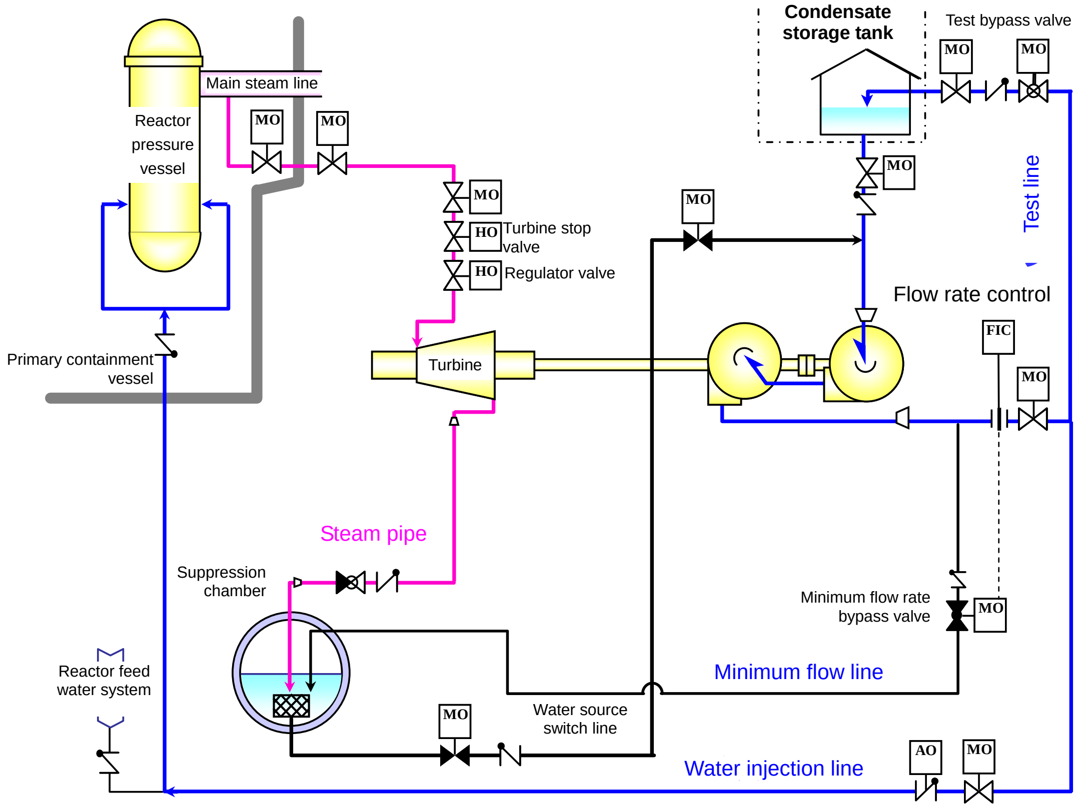

Figure 2-1. Schematic Diagram of the HPCI System

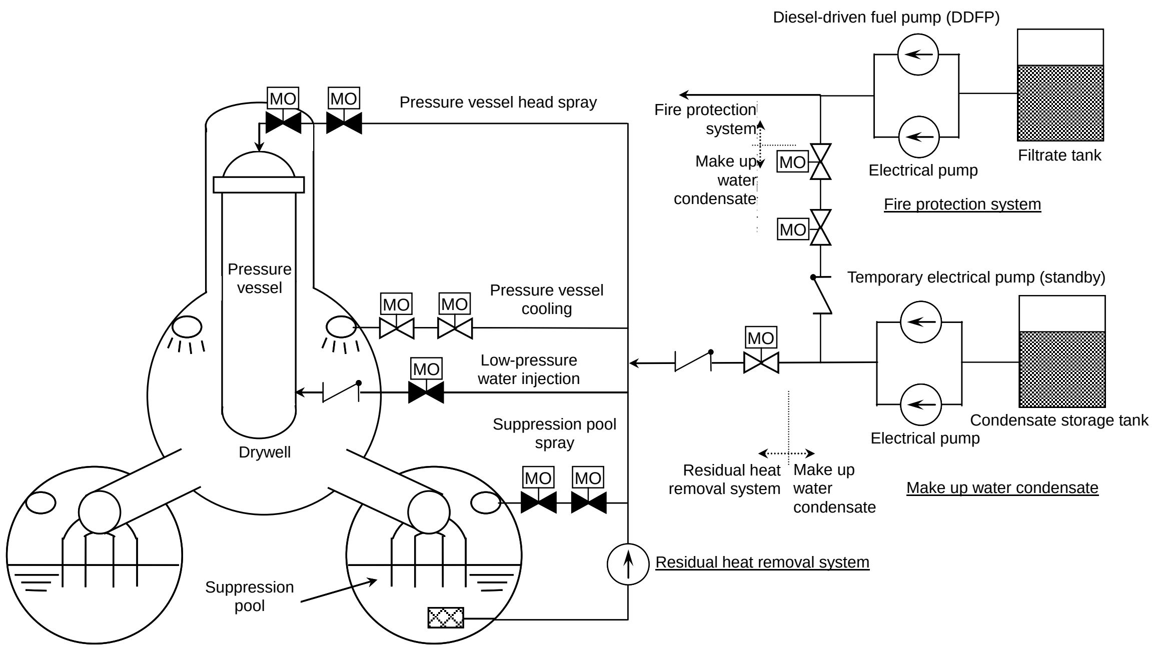

Figure 2-2. Schematic Diagram of Alternative Spray System for Unit 3 Primary Containment Vessel

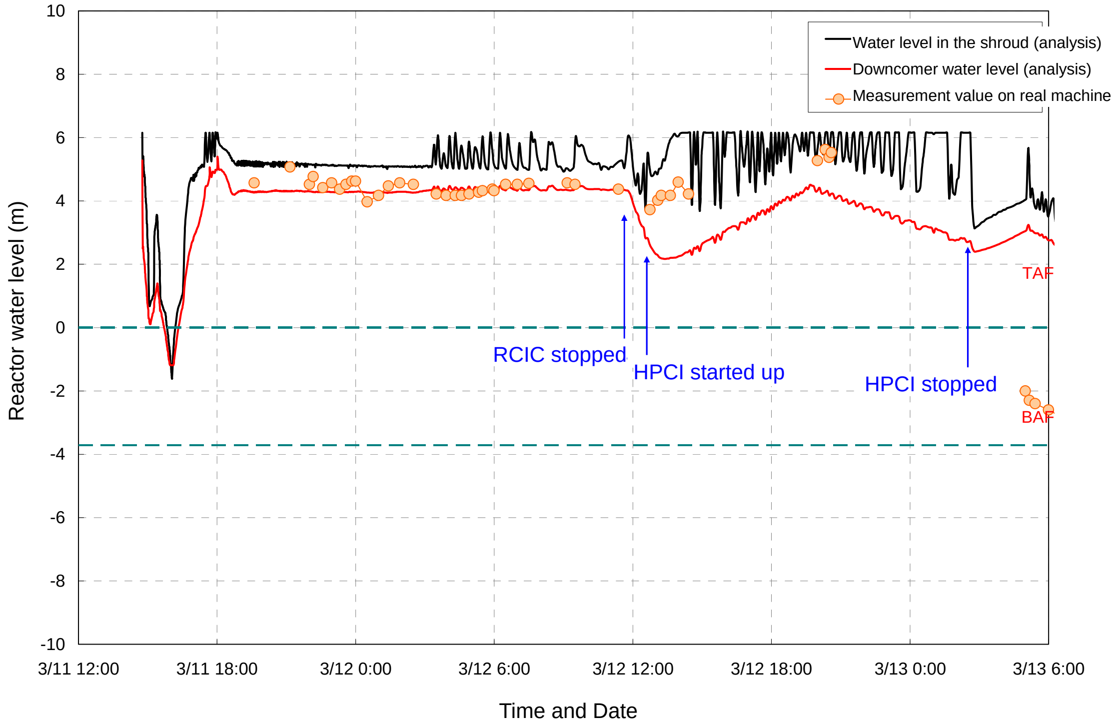

Figure 2-3. Changes in Reactor Water Levels of Unit 3

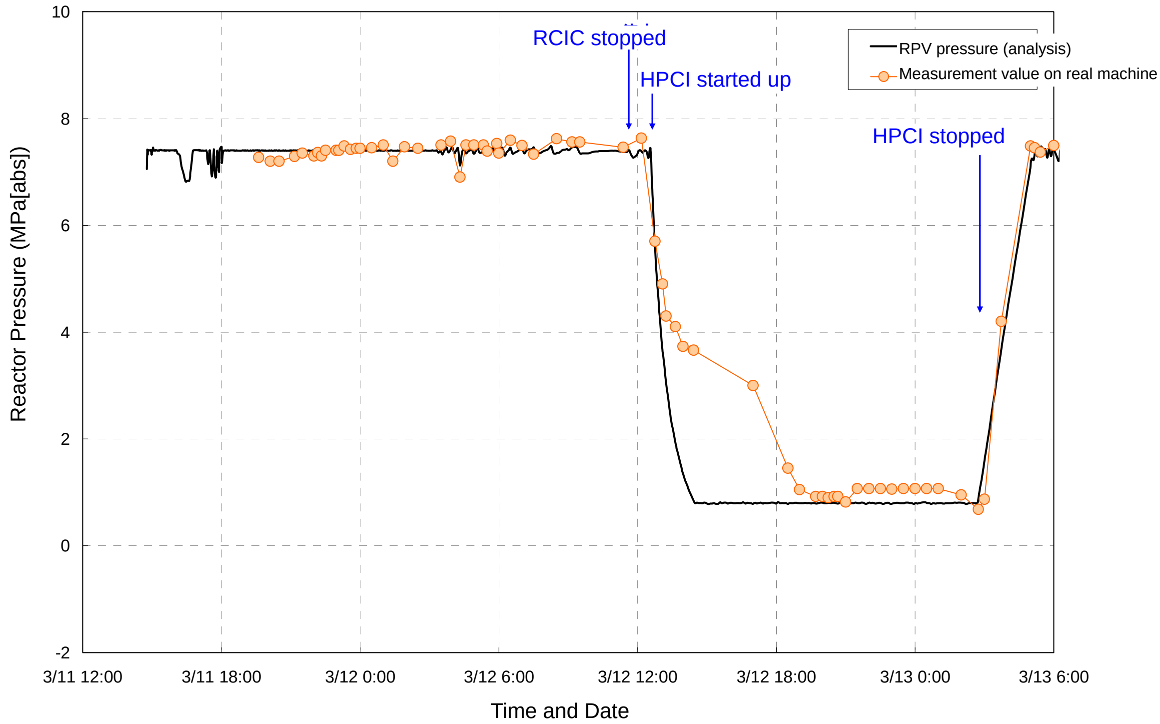

Figure 2-4. Changes in Reactor Pressure of Unit 3

3. Our Opinion on the Inconsistency between Analytic Values Relating to Performances of Manual Operations Including Spraying Unit 3 Primary Containment Vessel or the S/C and Changes in the D/W Pressure, and Values Measured with Actual Devices.

3-1 Performances of manual operations including spraying the Primary Containment Vessel or the S/C

[March 12]

- Operators manually opened valves, including the valve of the residual heat removal system (RHR), on the morning of March 12 and formed an S/C spray line.
  Operators actuated the DDFP at 12:06 on March 12 and started alternative spraying of the S/C with water.

[March 13]

- Operators reported to the main control room at 3:05 on March 13 that the formation of an alternative line of water injection in the reactor from the alternative spaying of the S/C was completed. (Alternative spraying of the S/C stopped)
- At 5:08 on March 13, operators started alternative spraying of the S/C with water to suppress the pressure of the Primary Containment Vessel.
- At 7:39 on March 13, operators shifted the alternative S/C spray line to the D/W spray line and started spraying the D/W. At 7:43, the S/C spray valve was shut off manually.
- During the time from 8:40 through 9:10 on March 13, operators shifted the D/W spray to the alternative water injection line for the reactor (the D/W spray stopped).

3-2. Our opinion on the inconsistency between analytic values relating to changes in the pressure of Unit 3 D/W and values measured with actual devices

(1) Difference between analytic results and values measured with actual devices in the September 9 Report

Figures 9-13-3 and 9-13-14 in the September 9 Report have some inconsistency between the measured values of changes in the pressure of the Primary Containment Vessel and analytic values. Specifically, measured values of the pressure of Unit 3

Primary Containment Vessel continued to rise until around 12:00 on March 12, 2011 and then began to decline around 22:00 on March 12. When compared with the analytic results, the analysis failed to reproduce measured values that indicated, at maximum, approx. 100 kPa higher than analytic values for the period until 12:00 on March 12 and then indicated a decline until around 22:00 on March 12.

Figures 9-13-3 and 9-13-14 in the September 9 Report correspond to Figs. 3-1 and 3-2. The periods of (i) and (ii) in Figs. 3-1 and 3-2 indicate that the analytic values are inconsistent with the measured values.

(2) Investigating changes in the pressure of the Primary Containment Vessel

Our investigations focus on two time periods, namely (i) the period until 12:10 on March 12 from the earthquake (period where measured values of the pressure in the Primary Containment Vessel were increasing) and (ii) the period from 12:10 on March 12 until 22:00 on March 12 (period where measured values of the pressure in the Primary Containment Vessel were declining).

Moreover, we organized operating states of the cooling systems that contribute to the behavior of the pressure in the Primary Containment Vessel, in Table 3-1 as indicated earlier. Figures 3-3 and 3-4 show the results of our analysis with the MAAP code based on the time series in Table 3-1. (Analysis results are identical to 2.2, Our opinion on Unit 3 reactor pressure while HPCI was in service)

Table 3-1. Time Series of Operations of Cooling Systems

Date	Time	Event
Mar. 11	16:03	RCIC started up
Mar. 12	11:36	RCIC stopped
	12:06	S/C sprayed with water from DDFP started
	12:35	HPCI started up
Mar. 13	2:42	HPCI stopped
	3:05	S/C sprayed with water from DDFP stopped
	5:08	S/C sprayed with water from DDFP started
	7:39	D/W sprayed with water from DDFP started
	7:43	S/C sprayed with water from DDFP stopped
	8:40 to 9:10	D/W sprayed with water from DDFP stopped

(2)-1 Regarding the period of (i)

During this period, measured values were, at maximum, approx. 100 kPa higher than analytic values. It is presumed that the rise in the pressure in the Primary Containment Vessel during this period was attributed mainly to the operation of the Safety Relief Valve (SRV) and exhaust steam from the RCIC system. Since both systems condense steam with water from the S/C pool, a rise in the pressure of the Primary Containment Vessel is suppressed. Thus, instead of the S/C, we presume a route where energy is directly transferred to the D/W, which would be able to reproduce the state of a rise in the pressure of the Primary Containment Vessel. Moreover, based on plant parameters after the earthquake, we presume that the Reactor Pressure Vessel boundary is sound. Thus, we investigated mechanisms other than damage to the boundary.

One of such mechanisms could be leakage of the reactor water from the pump mechanical seal of the Primary Loop Recirculation System (PLR) that is not taken into account in the MAAP analysis shown in Fig. 3-3. The PLR pump mechanical seal is usually structured so that it seals the reactor water with the seal water supplied from the Control Rod Drive (CRD) pump and that a portion of the seal water from the PLR pump main shaft drips down into the D/W system drain sump (this quantity of drips is called the "control bleed-off flow rate"). However, if the external power supply is lost, the seal water being supplied from the CRD pump is also lost. Thus, we consider that reactor water at a high temperature from the PLR pump main shaft dripped down into the D/W system drain sump.

Although the analysis on the assumption that the leak quantity from the PLR pump mechanical seal is set to approx. 3L/min, which is the same as the control bleed-off flow rate, was able to reproduce the behavior in the initial stage, the analysis failed to reproduce a rise in the pressure in the subsequent stage (Fig. 3-4).

At present, we are unable to reproduce the state of the measured pressure in the Primary Containment Vessel. However, we will continue investigating other factors that may lead to a rise in the pressure of the Primary Containment Vessel.

(2)-2 Regarding the period of (ii)

Since the S/C was sprayed with water from 12:06 on March 12, the spray presumably had an effect on the behavior of the pressure in the Primary Containment Vessel during the period of (ii). However, although the result of the MAAP analysis (Fig. 3-3) indicates an effect that suppresses a rise in the pressure of the Primary Containment Vessel, it does not indicate that the pressure of the Primary Containment Vessel is lowered. While RCIC and HPCI are in service, the water level is maintained and heat from fuel is removed. Thus, the pressure in the Reactor Pressure Vessel and the pressure in the Primary Containment Vessel are determined depending on how the integral quantity of the heat decay after the loss, due to the tsunami, of the heat sink of the seawater system was distributed to the gas phase and water phase of the reactor water, structure, D/W and S/C. Since present analysis of the water levels while HPCI is in service has different values from measured values, the assumed distribution may differ from the actual distribution state. As a result, the pressure in the Primary Containment Vessel may have been overestimated. Investigations into other factors for the decline in the Primary Containment Vessel must be continued.

Moreover, it can be considered that leakage from the PLR pump mechanical seal described in item (2)-1 above occurred during the period of (ii) where the reactor water level was maintained. It is also considered that since the reactor pressure declined substantially due to the effect from HPCI in service since 12:35 on March 12, the quantity of leaking water declined and the enthalpy of the leaking water also decreased. Accordingly, we believe that the leaking water from the PLR pump mechanical seal had made a smaller contribution than in the period of (i) to the rise in the pressure of the Primary Containment Vessel.

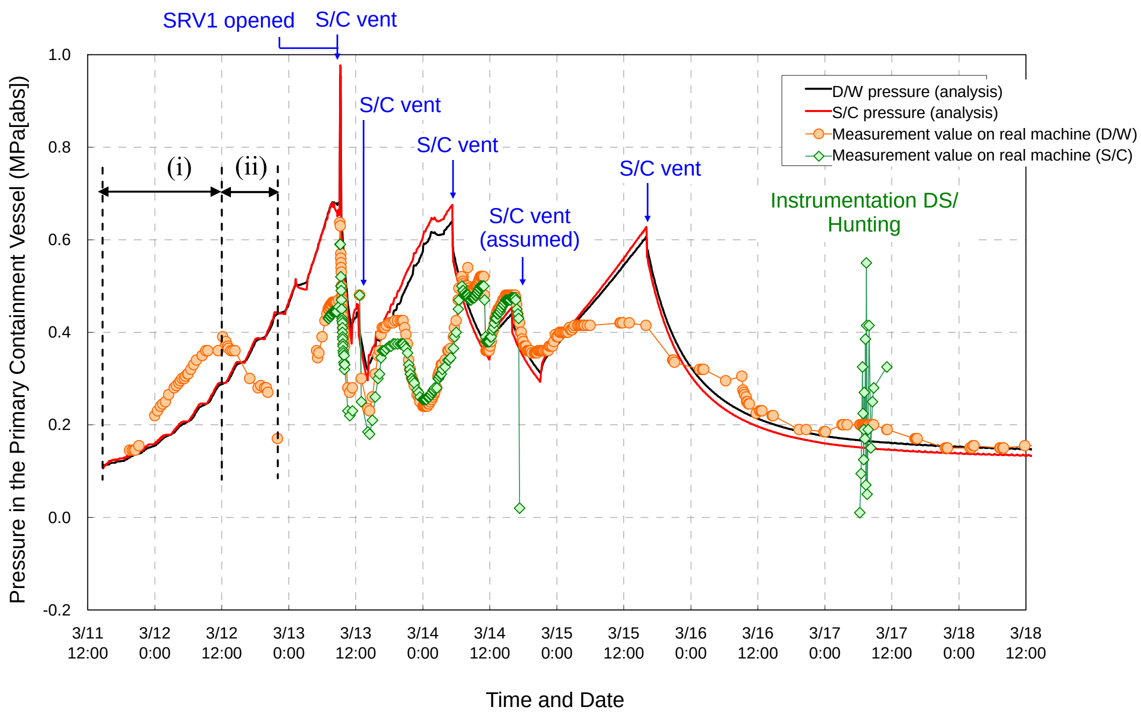

Figure 3-1. Changes in Pressure of Unit 3 Primary Containment Vessel (Fig. 9-13-3 in September 9 Report)

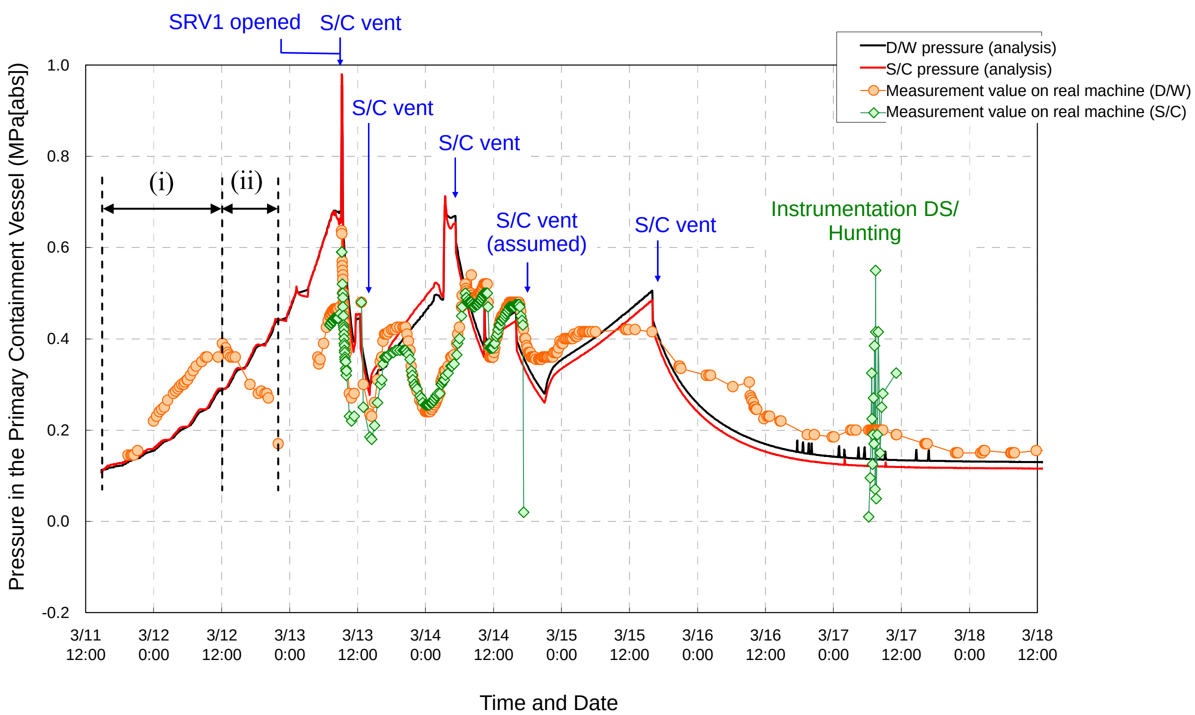

Figure 3-2. Changes of Pressure in Unit 3 Primary Containment Vessel (Fig. 9-13-14 in September 9 Report)

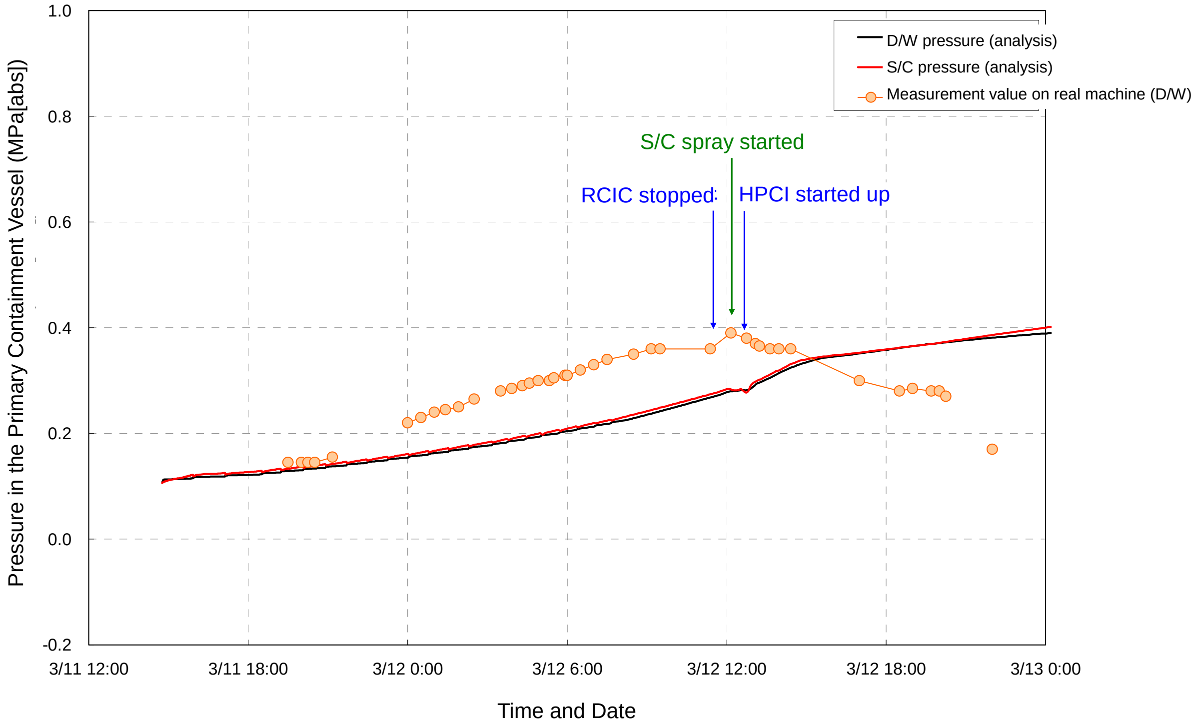

Figure 3-3. Changes of Pressure in Unit 3 Primary Containment Vessel

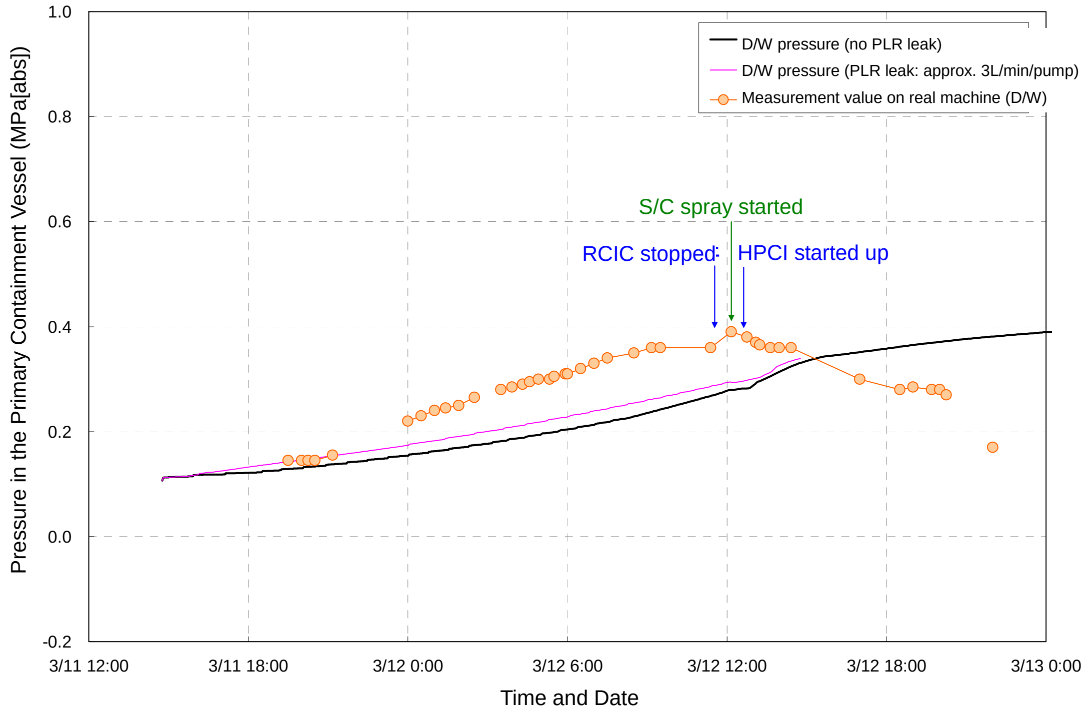

Figure 3-4. Changes of Pressure in Unit 3 Primary Containment Vessel (leakage from PLR pump mechanical seal)

4. Performances of Necessary Isolation Operations, Including Shut-off of Outlet Valve of the Standby Gas Treatment System, and Facts relating to Isolation State Checks, in Venting Operations of Units 1, 2 and 3.

- Since this Fukushima disaster far exceeded anything we could have previously anticipated, we investigated procedures based on our existing accident management operation procedure. However, such procedures have yet to be identified. Thus, we are unable to check for compliance with your inquiry.

- The compressed air system for instrumentation stopped after all the alternative current power supplies were lost following the tsunami. Since the source to actuate the air-operated valves (AO valves) was lost, the standby gas treatment system was presumably in a state as shown in Figs. 4-1 and 4-2. When we checked the valves, such as the outlet valve of the SGTS exhaust fan, for opening and shutting in the Unit 3 field investigation conducted on December 22, 2011, the valves we were able to check (circled valves in Fig. 4-2) were in the same state as that when the power supply was lost. We believe that checking valves in the field, including the valves for Units 1 and 2, is necessary. Thus, we will continue to investigate such valves within the range possible after considering radiation doses in the field.

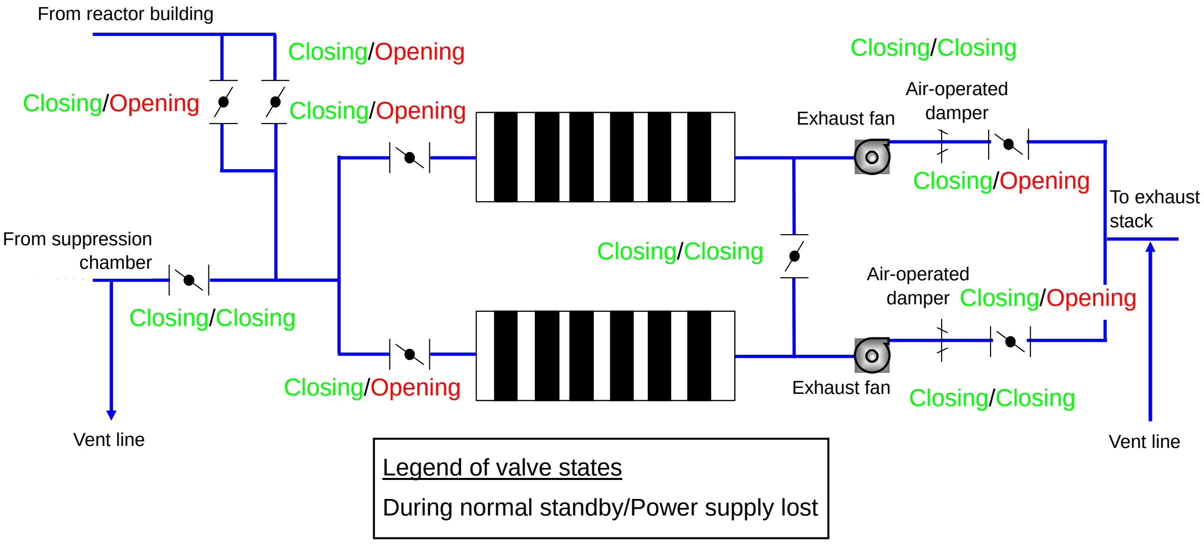

Figure 4-1. Ordinary State of Unit 1 SGTS Valves and State When the Power Supply Was Lost*

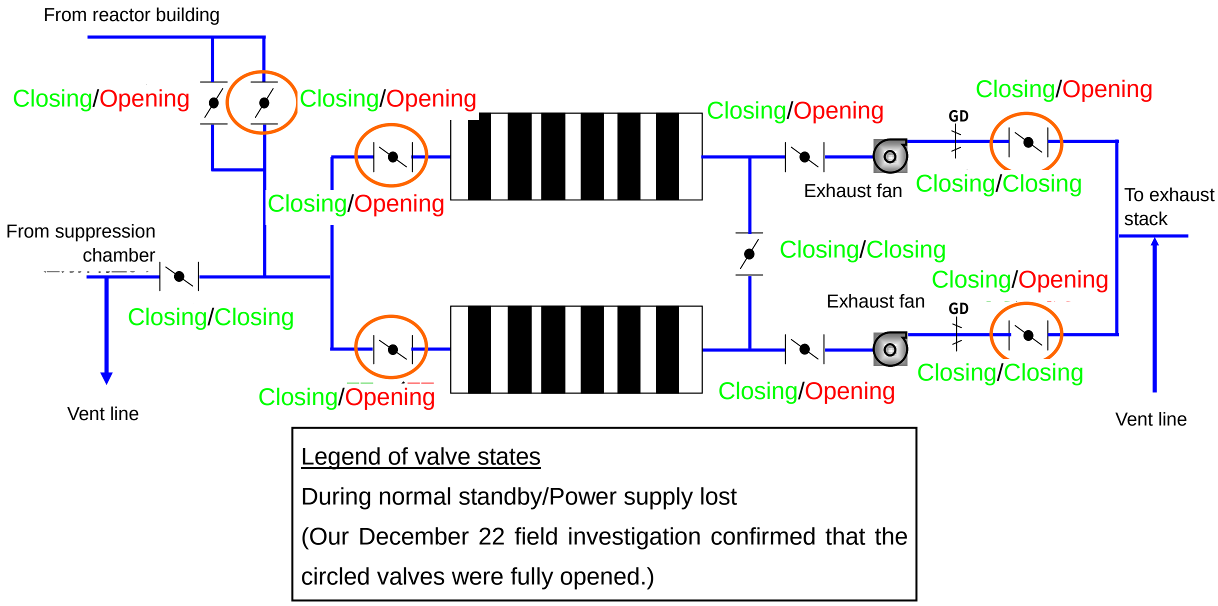

Figure 4-2. Ordinary State of Unit 3 SGTS Valves and State When the Power Supply Was Lost*

*: The valves in the above figures are all air-operated valves. Furthermore, the configuration of the Unit 2 system is roughly the same as that of the Unit 3 system, except for the exhaust fan outlet GD and AO valves that are in opposite locations to each other.