Evaluation of the situation of

cores and containment vessels of

Fukushima Daiichi Nuclear Power Station

Units-1 to -3 and examination into unsolved

issues in the accident progression

Progress Report No. 6

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Tokyo Electric Power Company Holdings, Inc.

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１． Introduction

１．１． Response actions taken so far

The Tohoku-Chihou-Taiheiyou-Oki Earthquake and Tsunami (also known as the Off the Pacific Coast of Tohoku Earthquake and Tsunami or the Great East Japan Earthquake and Tsunami), which occurred on March 11th, 2011, led the Fukushima Daiichi Nuclear Power Station (hereinafter referred to as "Fukushima Daiichi NPS") to a situation far beyond design basis accidents, and even further exceeding multiple failures assumed in developing accident management measures. Consequently, Units-1 to -3 finally experienced severe accidents, although they were successfully shut down but lost functions related to cooling.

TEPCO bears a responsibility, as a party who experienced this accident and failed to prevent it, to reveal the complete picture of the accident at Fukushima Daiichi NPS and to contribute to enhancing nuclear power plant safety. In other words, it is critically important to engage as a corporate unit in safety improvement, in order to continue the company's nuclear power business, especially to continue efforts to reveal the process of accident progression and, based upon the findings, to continue implementing further measures for safety enhancement of nuclear power plants.

The accident progression processes to the ultimate severe accidents have been interpreted in the response actions to the accident[†1] taken so far and the knowledge obtained therefrom has been integrated in safety enhancement measures of the Kashiwazaki-Kariwa Nuclear Power Station. (Refer to Figure 1 Examinations of accident progression at Units-1 to 3 by event-tree analysis. [†2] See 8.1 for further details.)

１．２． Continuing improvement of safety measures

The two main pillars of safety enhancement measures currently being taken at Kashiwazaki-Kariwa Nuclear Power Station are: measures for preventing loss of functions due to earthquakes and tsunami; and measures for enhancing safety functions, centering on strengthening redundancy and diversity by additional installation of back-up components and equipment having equivalent safety functions.

Safety measures for safety functions are enhanced mainly by strengthening safety by additional means. Therefore, continuing efforts are needed for assessing the appropriateness of added safety means or their integrities against various causes, being not limited to tsunami. TEPCO is continuing these efforts by, among others, collecting proposals for safety improvement measures broadly from its employees through a company-wide "safety improvement campaign."

１．３． Overall analysis of the accident of the Fukushima Daiichi NPS

On other hand, there are still unclear issues, e.g., the reason why the reactor core isolation cooling (RCIC) system of Unit-2 lost its functions remains unknown, and some observed phenomena cannot be interpreted yet. Also, concerning earthquakes and tsunami, there are some issues for academic researchers to tackle, such as the mechanism of earthquakes of this historically huge scale occurring in the same district and causing massive tsunami.

For instance, discovering the reasons for the safety equipment function loss adds knowledge about ensuring existing system functionality and safety enhancement. Fuel removal and prevention of contaminated water production are crucial for dismantling Fukushima Daiichi NPS. In order to cope with these issues, it is essential to grasp the damage situations as well as the debris in the reactors and containment vessels (hereinafter referred to as "PCVs"), and to tackle the issues related to function loss of safety equipment by the effects of the earthquake and tsunami, and other issues having impacts on accident progression. Even the issues not directly related to accident progression may provide clues to enhancing safety as a result of examining them. Therefore, issues must be extracted from a broad standpoint.

Consequently, it is TEPCO's important responsibility to examine unclear issues of the accident at Fukushima Daiichi NPS. TEPCO has been carrying out examinations into these issues [1]-[6] prior to this report and has expressed its commitment to continue the examinations, together with the outcomes of the examinations done, in its Progress Report of Nuclear Safety Reform Plan.

１．４． Contents of this report

The purpose of this report is to organize and present the examination results into about 50 issues[†3), which are directly and indirectly related to the situations of the cores and PCVs of Units-1 to -3, and identified as requiring examinations as of March 2012, based on the data and investigation outputs compiled in the TEPCO Accident Investigation Report of Fukushima Nuclear Power Station [7].

This particular version presents unclear issues in a list form and identifies the issues to tackle hereafter. The results of examinations into those issues already completed are contained in this report, but examinations will continue on uncompleted issues and the results will be added as soon as they become available. Items to be examined will be added or dropped as necessary.

The Progress Reports 3, 4, 5 and 6 present first the examination results obtained jointly with TEPCO Systems, Ltd., based on its proposal, from among the unsolved issues

identified in Progress Report 1. The results are compiled in Attachments 1-6, 2-9, 2-14, 3-7, 3-9, 3-11 and 3-12. Further, the dose distribution was surveyed in the SGTS room of the Unit-2 turbine building. This is a step toward solving the Unit-2/Issue-9 "The Unit-2 rupture disc actuated?"

It should be noted that this report covers issues in a broad range related to the accident progression until about the end of March 2011, at Fukushima Daiichi NPS, but it is limited[†4] to those issues concerning the release of radioactive materials to off-site, which may contribute to interpreting the accident progression.

[†1] History of TEPCO investigations on the situations of cores and PCVs at Units-1 to -3

TEPCO published its "Fukushima Nuclear Accident Analysis Report" [7] on June 20th, 2012, in which the results of the investigation by the "Fukushima Nuclear Accidents Investigation Committee" (established in June 2011) were presented. In addition, the following investigations and examinations are being continued.

TEPCO examined the plant situations by the accident analysis code (Modular Accident Analysis Program, hereinafter referred to as "MAAP") for the first time and published the results on May 23rd, 2011, by evaluating the relevant information.

On November 30th, 2011, a technical workshop on estimating the damage status of reactor cores of Fukushima Daiichi Units-1 to -3 was held. In its report, TEPCO made open the situations re-estimated by comprehensive considerations of the updated information available, including the temperature changes due to water injection by core spray systems at Units-2 and -3. The report also contained the results of the updated core status from that of May 2011. Site investigation [5], reexamination of records [6], etc. are still continuing.

Further on March 12th, 2012, another report published (separate volume 1) the results of reexamination of plant status using MAAP and the knowledge obtained since the above publication. Also, the evaluation results of actual progression of accident, which were obtained by in-depth analysis of the examination results mentioned above and the gap between the examination results and observed values, have been published [1]-[4].

These investigations and analyses have been conducted aiming at revealing the accident progression and status of the reactor cores and PCVs and this knowledge will be utilized in the dismantling activities. While TEPCO is continuing its efforts to deepen reliabilities of analysis results on the accident progression by examining the information concerning operation and design, government projects have been conducted in parallel to advance the accident analysis codes. Since FY 2016, the government projects have shifted their target from improving the analysis codes to estimating the conditions in the reactor and PCV using

the improved codes for analysis. TEPCO has been engaged in estimating the fuel debris distribution in collaboration with the government projects.

†2) Event-tree analysis

An event-tree analysis is a means to analyze what sequences a system follows starting at an initiating event to the ultimate status via junctions such as a loss of function of safety-related equipment. This approach enables system transition processes to be assessed by knowing simply the loss of function of safety-related equipment without identifying the reasons for the function loss, thus facilitating the arrangement of basic information related to accident progression.

†3) Significance of extracting issues of less importance

When evaluating accident progression, not only items that deteriorate or mitigate the accident status, but also those that accelerate or delay the accident progression need to be included in the conditions. The latter are comparatively less important, but they are being extracted as unclear phenomena because they are needed as input for evaluations.

One example is clarification of the status of the residual heat removal (RHR) system of Unit-2 after arrival of the tsunami. The RHR system started to operate before the tsunami and was cooling the suppression chamber (S/C). This is considered not to have a big influence on the accident progression, but if this system continues cooling (energy removal), the basic energy balance is affected, resulting in a possible delay in the accident progression.

At the very least, its evaluation results may provide meaningful clues to enhance nuclear safety, as hinted at in this text.

†4) Regarding the assessment of radioactive materials discharged outside the nuclear power station site, the report "Estimation of Radioactive Material Released to the Atmosphere during the Fukushima Daiichi NPS Accident" was published (May 2012). As the reliability improvement for evaluating core status is needed for more accurate estimation of released quantities, it will be made using the latest knowledge obtained in this report.

Initiating event	Reactor trip	AC power		DC power	Core cooling	AC power recovery	Ensuring of long term cold shutdown				Core state	Control of PCV	Control of Reactor Build.	Final status
Earthquake (Great East Japan earthquake)	Reactor scram (High earthquake acc.)	Off-site power supply	Emergency D/G	DC power supply	Unit-1：IC Units-2/3：RCIC、HPCI	Off-site power, EDG, power source cross-ties	RPV depressurization (alternative meas. included)	Water injection (alternative meas. included)	RHR	RHR recovery	Cold shutdown, Core damage, PCV damage, etc.	PCV Vent	SGTS, ventilation, vent valve kept open etc.	Cold shutdown, Core damage, PCV damage, Reactor building damage etc.

Cold shutdown

Cold shutdown

PCV damage

Success（with difficulty due to the impacts of X）

Failure（due to X）

Cold shutdown (long term cooling needed)

Core damage

Core damage

FPs release due to overheating, etc. despite PCV integrity

FPs release and hydrogen explosion

PCV damage due to over pressure

Final status of Unit-3

Cold shutdown (long term cooling needed)

Core damage

Core damage

FPs release due to overheating, etc. despite PCV integrity

FPs release and hydrogen explosion

Large FP release

Final status of Unit-2

Cold shutdown (long term cooling needed)

Core damage

Core damage

FPs release due to overheating, etc. despite PCV integrity

FPs release and hydrogen explosion

PCV damage due to over pressure

Final status of Unit-3

(A)

to (A)

Success

Failure

RCIC/HPCI

1F-3

1F-2

RCIC

1F-1,2

1F-1

Legend:

Ⓔ ： **Earthquake inpacts**

Ⓣ ： **Tsunami impacts**

Ⓟ ： **Impacts of AC or DC power loss**

Ⓢ ： **Impacts of the loss of ultimate heat sink**

Ⓗ ： **Impacts of hydrogen explosion**

Figure 1 Event-tree analysis results of Units-1 to -3 of Fukushima Daiichi Nuclear Power Station

２．　The earthquake, tsunami and their impacts

２．１．　Issues concerning the earthquake and its impacts

The Tohoku-Chihou-Taiheiyou-Oki Earthquake, which occurred on March 11th, 2011, was the biggest scale of earthquake ever observed in Japan. Kurihara City in Miyagi Prefecture observed a maximum seismic intensity of 7 on the Japanese (JMA) scale, and high tsunami were observed along the Pacific coast areas in the districts of Hokkaido, Tohoku and Kanto.

It has been reported that the focal area of the earthquake extended from offshore Iwate Prefecture to offshore Ibaraki Prefecture, being about 500 km long, about 200 km wide, and with about 50 m in maximum slip. There was a massive slip observed in the southern trench side off Sanriku coast and part of the trench side off Northern Sanriku coast to far south off the Boso Peninsula. Multiple regions, offshore Central Sanriku, offshore Miyagi Prefecture, offshore Fukushima Prefecture and offshore Ibaraki Prefecture, moved simultaneously and the magnitude was 9.0 on the JMA scale at the hypocenter.

Many unknown matters remain about the causes of such massive synchronized earthquakes. It is necessary, therefore, to monitor the research progress in Japan and overseas on their mechanism and to incorporate the latest knowledge about them in the approach for consideration in design (Common/Issue-12). (The number in the brackets shows the Issue identifying number as described in Attachment 2).

Seismic activities have become active in the southern area of Hama-dori in Fukushima Prefecture after the Tohoku-Chihou-Taiheiyou-Oki Earthquake. A new fault appeared on the occasion of an earthquake on April 11th, 2011 as a normal fault in the Yunodake Fault, which TEPCO had assessed as having had no seismic activity since the Late Pleistocene era.

Investigations in detail thereafter by trench surveys and others in the Yunodake Fault revealed seismic activity marks at several locations, resulting in the judgment that the Yunodake Fault had been a fault which should have been considered in seismic design. Should the investigations by boring or trenching have been done, the evaluation of the activities would have been possible [8]. This knowledge shows that fault activities should be directly confirmed by geographical investigations in detail such as trench surveys, etc. in order to negate possible fault activities. This must be considered in future fault investigations (Common/Issue-13).

Regarding the intensity of ground motions at Fukushima Daiichi NPS, they were about the same level with those assumed in seismic design, when observed values and analysis results were considered. Most of them were below the assumed values for seismic design, although the observed values on the reactor building basemat (the lowest basement floor) had partly exceeded the maximum acceleration corresponding to the design basis earthquake ground motion Ss, which were reported in July 2012 [9]. Concerning the impacts

of the earthquake on reactor systems, TEPCO has evaluated, from the observed plant operation status and the results of seismic assessment using observed ground motions, that the main equipment having important functions for safety was in a situation to maintain its safety functions during the earthquake and right after it [7], [9].

２．２． Issues concerning the tsunami and their impacts

The Tohoku-Chihou-Taiheiyou-Oki Earthquake, which occurred on March 11th, 2011, was followed by tsunami, which caused a large scale disaster in the Pacific Ocean coastal areas. The tsunami was designated as having the tsunami intensity of 9.1 in an index for indicating the scale of tsunami, and was the fourth largest ever observed in the world and the largest ever in Japan.

TEPCO carried out tsunami reproduction calculations in January 2013 by a wave source model (fault lengths, fault widths, locations, depths, slip scales, etc. needed for numerical simulation of tsunami) which could well reproduce tsunami tracks, inundation heights, tsunami bore levels, submerged areas and diastrophism in the area from Hokkaido to Chiba Prefectures. The results indicate that an especially large slip (about 50 m at maximum) occurred near the Japan Trench.

The tsunami heights estimated based on the estimated wave source were about 13 m at Fukushima Daiichi NPS and about 9 m at Fukushima Daini NPS. The main reasons for this difference were considered to be that the peaks of tsunami waves, which were generated in regions with large slips, estimated to be off Miyagi Prefecture and off Fukushima Prefecture, overlapped at Fukushima Daiichi but not so much at Fukushima Daini NPS.

Many unknown matters remain about the causes of such massive tsunami. It is necessary, therefore, to monitor the research progress in Japan and overseas on tsunami generation mechanisms and to incorporate the latest knowledge in the approach for considering massive synchronized earthquakes with accompanying tsunami in design (Common/Issue-12).

Meanwhile, the tsunami waves which hit Fukushima Daiichi NPS flooded not only the 4-m ground level above O.P. (O.P.: Onahama Port construction standard surface) (hereafter described as 4-m ground level), where seawater pumps had been installed, but also the 10-m ground level, where key buildings had been constructed, and also flowed into the buildings through openings and other routes. Consequently, motors and electrical equipment were flooded, and important systems such as emergency diesel generators and power panels were directly or indirectly affected and lost their functions.

It is necessary that investigations should be continued further on the arrival time of the tsunami to Fukushima Daiichi NPS site and the inundation routes in order to clarify their

chronological correlation with the loss of power (Common/Issue-14: Examination completed (Attachments Earthquake-tsunami-1, Earthquake-tsunami-2)).

Concerning the wave force of tsunami, damage was confirmed partially on doors, shutters, etc. of the openings at the ground level, which could be considered as being due to directly tsunami or to floating wreckage. Parts of heavy oil tanks, which had stood on the seaside area, seemed to have been pulled away by wave force and buoyancy. But no significant damage was noticed on the building structures such as walls or pillars of key buildings. Furthermore, most of the breakwater and seawall banks stand as before, with no big impacts having been confirmed although part of northern breakwater with parapet was damaged. Actual wave forces due to tsunami on these building structures or breakwater and seawall banks were not measured, thus the situations at the time of the tsunami are difficult to grasp, but comparative studies referring the actual damage will help to quantify the degree of conservative evaluation by wave force evaluation formulae (Goda Formula, Tanimoto Formula, etc.) (Common/Issue-15).

２．３． Examination results of the earthquake and tsunami

２．３．１． Arrival times of tsunami to the Fukushima Daiichi NPS site

The issue of tsunami arrival times reaching Fukushima Daiichi NPS site (Common/Issue-14) has been evaluated (see Attachment: Earthquake-triggered tsunami-1).

The following findings have been concluded, through analyzing continuous photos, by chronologically arranging the incidents at the time of the arrival at the site of the tsunami that accompanied the Tohoku-Chihou-Taiheiyou-Oki Earthquake.

- Tsunami, which affected various systems and equipment at the power plant, arrived at the Fukushima Daiichi NPS site sometime between 15:35 and 15:36, hereafter described as the 15:36 level.
- The tsunami maximum wave arrived from almost directly in front of the site with no big delay.
- Seawater system pumps located near the sea lost their functions mostly at the 15:36 level.
- Many systems and much equipment lost their functions in a limited time when there were no aftershocks, indicating it was tsunami that caused the losses of power.

２．３．２． Additional examination of station black-out due to tsunami

Based on the investigation results discussed in 2.3.1 on the relationship between arrival times of tsunami and station black-out, it is believed that the station black-out was caused by tsunami. To strengthen the credibility of this estimation, examination was continued on the relationship between the tsunami flooding process and station black-out timings (see Attachment Earthquake-tsunami-2).

Correlation was reviewed between tsunami inundation path lengths to each item of emergency AC power equipment and the timings of emergency AC power supply losses. It was confirmed that the timings of power loss tended to be delayed as the tsunami inundation path lengths became longer. Thus, the grounds for the estimation to date have been strengthened that the emergency AC power supply systems had lost their functions due to tsunami run-up and flooding.

２．３．３． Other examinations

Examination results of other issues extracted in "2.1. Issues concerning the earthquake and its impacts" and "2.2. Issues concerning the tsunami and their impacts" will be appended to this section as soon as they become available.

２．４． Summary of examinations into the earthquake and tsunami

Unclear issues have been extracted concerning the earthquake and tsunami. This report contains facts drawn from organizing the actually observed results. Examinations are due to continue for other issues.

３． Examinations into the accident progression at Unit-1

３．１． Approach for examinations

The analysis results of MAAP (see Attachment 1) have been mainly used to examine the accident progression, excluding the effects of the earthquake and tsunami, of Fukushima Daiichi Unit-1 (hereinafter referred to simply as "Unit-1"). Figure 3.2.1 shows the reactor water level changes, while Figure 3.2.2 shows the reactor pressure changes and Figure 3.2.3 shows the PCV pressure changes. However, the MAAP results cannot perfectly reproduce the actual accident progressions because of the uncertainties in the analysis conditions, analytical models, and consequently the results obtained. In this report, therefore, the following steps were taken to examine unclear issues: first, discrepant points were identified as issues between the MAAP results gotten in the past (see separate Volume 1 for the results by MAAP4, and Attachment 3 for the results by MAAP5) and actually observed measurements; and second, the issues identified were examined one-by-one. Section 3.2 explains in chronological order the issues extracted and Attachment 2 describes each issue individually.

３．２． Issues derived from the comparison between measured information of Unit-1 and analyses

３．２．１． From the earthquake to tsunami arrival

At Unit-1, all control rods were inserted into the reactor core when scram was signaled upon high seismic acceleration (horizontally >135 Gal or vertically >100 Gal on Basement Floor 1 of the reactor building) as caused by the earthquake at 14:46 on March 11th, 2011. Thereafter, two isolation condenser (IC) systems automatically activated at 14:52 due to the reactor pressure increase. The IC system is designed to have capacities to remove decay heat by heat exchange at 5 minutes after the scram by its one sub-system. Therefore, under the then activated IC operation conditions (longer than 5 minutes after the scram and simultaneous operation of two sub-systems), heat removal exceeded the decay heat; resulting in the reactor pressure decrease. Shift operators stopped the two IC systems once, as the reactor pressure was decreasing faster than the rate corresponding to the coolant temperature decrease limit of 55 deg C per hour. After that, in response to the reactor pressure increase (when the IC was in shutdown) and decrease (when the IC was in operation), the shift operators were controlling the reactor pressure by repeating the startup and shutdown of one sub-system (A) of the IC systems. As the IC system automatic startup pressure was set at a value lower than the lowest activation pressure of the main steam safety relief valves (SRVs), no SRVs were activated, no steam was released from the reactor, and the coolant inventory in the reactor did not decrease, while the IC system

operation was on and off. It should be noted that the reactor was in the process of being maneuvered toward cold shutdown by including the startup of the containment cooling system (CCS) in the suppression chamber (S/C) cooling mode. At 15:37 on March 11th, 2011, however, all AC power supplies were lost due to tsunami followed by the loss of the DC power supply. Nothing difficult to explain is seen in the reactor behavior before the tsunami arrival, as confirmed in the recorded results in the charts and transient recorders. According to these records in the charts and transient recorders, the power supplies were lost while the reactor pressure was increasing. It means that all AC and DC power supplies were lost while the IC systems were in shutdown, that is, while the decay heat was not being cooled.

However, the Fukushima Nuclear Accident Independent Investigation Commission of the National Diet of Japan expressed its views, in its report, on the small amount of water observed on the fourth floor of the reactor building immediately after the earthquake (Unit-1/Issue-4: Examination completed (Attachment 1-3)). It pointed out that the possibility of a loss-of-coolant accident (LOCA) on a small scale could not be denied on the following grounds:

- Impacts on plant parameters do not show up when the cross section of the break is small even if the leak occurs from essential pipes; and
- No statements were obtained from shift operators of Unit-1 and Unit-2 that they confirmed the sounds of the SRVs of Unit-1 (the Commission presumed the steam came not from SRVs but from an opening of a broken pipe).

３．２．２． From the tsunami arrival to reactor water level decrease

All cooling capabilities were lost and all displays of monitoring instruments and various display lamps in the main control room went out due to the station black-out caused by tsunami. Approximately from 16:42 to 17:00 on March 11th, 2011, part of the DC power supply was temporarily recovered. The reactor water level measured for a while helped to confirm that it had decreased from the earlier level before the arrival of tsunami. If the IC is not in operation, the reactor pressure increases due to the station black-out: the reactor pressure continues to increase, since the IC does not automatically start up without the DC power source even when the reactor pressure reaches the IC activation pressure. Even when the reactor pressure reaches the SRV activation pressure in the relief mode, the SRVs do not open, either, without the DC power source. The reactor pressure continues to increase, and when it reaches the SRV activation pressure in the safety mode, the SRVs open, and the reactor pressures starts to decrease by releasing steam to the S/C. The loss of water due to steam release to the S/C upon SRV activation in the safety mode caused the

above-mentioned reactor water level decrease. The SRVs are activated in the safety mode by spring forces without power, and therefore, the SRVs are considered to have repeated on and off operations in response to the reactor pressure increases and decreases even during the station black-out. The water level observed (by the wide range water level indicator) at 16:56 on March 11th was at the top of active fuel (TAF) +2,130 mm, and had not decreased yet to the TAF at that time, although it was still decreasing by the above mechanism.

The analysis results suggest that the reactor water level reached the TAF at about 18:10 on March 11th, and the core damage started at about 18:50 (fuel cladding temperatures reached to about 1200 deg C). Although almost no measurements were available during the decrease of the reactor water level, the analysis result of water level at about 17:00 on March 11th was in good agreement with the measured value. Therefore, it can be considered that the timing of the water level reaching TAF is fairly accurate and that of the start of core damage is by and large well predicted.

Even if the fuel starts to be uncovered, steam cooling prevents it from conspicuous temperature rises as long as sufficient steam is supplied from below. Once fuel claddings can no more be cooled by steam cooling and their temperatures reach about 1200 deg C, large amounts of hydrogen are produced by water–zirconium reactions, which are accelerated by positive feedbacks, and the energy released from their oxidation reactions further raises fuel temperatures. It is highly possible the energy released by the water–zirconium reactions in this situation exceeded the decay heat. As the measured information for Unit-1 is significantly less than that of Unit-2 and Unit-3, use of the analysis results is often unavoidable for explaining the phenomena, but they still have big uncertainties as of now.

The situation continued that the IC operation could not be confirmed. When part of DC power supplies was temporarily recovered, it was observed that the isolation valve outside the containment of the IC subsystem-A (see system diagram in Unit-1/Issue-1) was operable (the status display lamp was "closed"). The shift operators took an opening action[*1] of the valve at 18:18 on March 11th. The operators confirmed that the status display lamp changed from "closed" to "open," and they heard the steam generating sounds and saw steam when looking above the reactor building, but the amount of steam was limited, and it stopped a while later. Concerned about the water inventory left in the IC shell side tank, at 18:25 the operators closed the isolation valve outside the containment on the return pipe. Later at 21:30 the operators took action again to open the isolation valve outside the PCV and confirmed the steam generating sounds and saw steam when looking above the

[*1] The operator took action to open the isolation valve (Valve 2A) outside the containment on the incoming pipe as well, not only that on the return pipe (Valve 3A).

reactor building.

It is considered that non-condensable hydrogen gas produced by the water-zirconium reactions deteriorates the heat removal performance when it stays in the IC cooling tubes. But further examinations are needed because it is unknown how much the heat removal capability was deteriorated (Unit-1/Issue-1: Examination completed (Attachment 1-7)).

Post-accident surveys of the water level in the IC shell side tank revealed that the water level indicator of subsystem-A had been 65% (normal level is 80%) and the water in the tank had been sufficient. If the isolation valve had not been closed at 18:25 on March 11th, reactor cooling by the IC might have been continued. It is also important, therefore, to examine the effect on the accident progression, if the isolation valve of IC subsystem-A had been kept open after 18:25 (Unit-1/Issue-2: Examination completed (Attachment 1-7)).

Meanwhile, mechanical seals were mounted on the primary loop recirculation pumps (PLR pumps) as a shaft seal. During normal operations, sealing water for the shaft seals provided from the control rod drive (CRD) pumps prevents reactor water from leaking. When the external power supply was lost, CRD pumps were shut down and sealing water was lost, then the high pressure reactor water was discharged to the drywell (D/W) equipment drain sump via the PLR pump shafts and shaft seals. Examinations are needed to determine how much water actually leaked (Common /Issue-4).

３．２．３． From the reactor water level decrease to PCV pressure increase

The reactor pressure of 7.0MPa[abs] was measured at 20:07 on March 11th, and D/W pressure of 0.6 MPa[abs] at about 23:50; on March 12th, the D/W pressure of 0.84 MPa[abs] was measured at 02:30 and the reactor pressure of 0.9 MPa[abs] at 02:45. In the meantime, although the exact timing is unknown, it was observed that at a certain time after 20:00 on March 11th, the PCV pressure showed a sharp rise and the reactor pressure decreased despite no depressurization actions.

In order to reproduce this pressure changes, a scenario was assumed in the analysis that steam had leaked to the D/W via in-core instrumentation dry tubes or main steam pipe flanges due to temperature rises in the vessel caused by overheating of uncovered fuels and fuel melting. But no direct evidence has been obtained showing any actual leaks at these locations, from either the measured parameters or the observed facts. Including the possibilities of damage to the main steam line due to overheating after the core damage, as hinted at in a study by the US Sandia National Laboratory, or other gaseous phase leaks, further examinations are needed (Unit-1/Issue-5).

There is a record that when operators entered the reactor building at about 21:00 on March 11th, in order to check the water levels of the IC shell tank and the reactor, their alarm

pocket dosimeters (APDs) showed 0.8 mSv shortly thereafter and they reported that upon returning to the main control room at 21:51. The increase of dose levels in the reactor building might have impeded responses to terminate the accident, since it was unknown whether this dose increase was caused by the reactor depressurization, etc. This remains as an issue to examine (Unit-1/Issue-7).

In the investigation thereafter, high dose level contaminations were noticed in the vicinity of the travelling in-core probe (TIP) room in the southeast of the first floor of the reactor building. Examinations are needed about the possibility of a TIP drytube break when the core had been uncovered and overheated (Unit-1/Issue-8: Examination completed (Attachment 1-12).

When operation of the fuel range water level indicators was recovered at 21:19 on March 11th by the temporary power supply, they showed TAF+200 mm but the reactor water level indicators seemed to have been already defective. But still, as the pressure difference between the pipes on the reference water level side (reference leg) and reactor side (variable leg) can be known from the water level measurements, some information might be obtained on the accident progression. Examinations will continue from this standpoint (Common/Issue-3, Unit-1/Issue-3: Examination ongoing (Attachment 1-6)).

Meltdown accidents follow the following progression: When heated up to high temperatures, fuel melts down from the core to the lower plenum, and then further down to the bottom of the PCV by breaking through the reactor vessel.

In the analysis results, the reactor pressure showed a sharp peak at about 22:00 on March 11th. It came from a model used in the MAAP analysis that the molten core collects and stays for a while on the core support plate, and then drops down after the plate is damaged to the lower plenum at one sweep, thus generating a big amount of steam. The relocation mechanism of molten fuel to the lower plenum is based on the knowledge of the TMI accident. As it is difficult to say that the model well simulates the complicated BWR structures, further examinations are needed to evaluate the molten BWR fuel relocation mechanism (Unit-1/Issue-6: Examination ongoing (Common/Isssue-6, Examination ongoing: Attachment 1-8)).

３．２．４． From the containment vessel pressure increase to containment venting operation

At about 23:50 on March 11th, the D/W pressure measured 0.6MPa[abs]. Thereafter the indicator continued displaying high values. At around 04:00 on March 12th, the dose rate near the main gate started to show an upgoing trend, which may show the effect of radioactive materials released from Unit-1. The dose rate increase may have been caused

by the radiation from radionuclides transferred to the reactor building from the PCV, or by the radiation from radionuclides released from the reactor building and transferred to near the measuring instruments (Common/Issue-7 Examination ongoing (Attachment 1-11)).

It is highly possible that the molten fuel dropped to the reactor vessel bottom and further to the bottom of the PCV before 19:04 on March 12th, when fire engines started continuous water injection to the reactor. The relocation of molten fuel to the PCV would raise the PCV pressure and temperature. In addition, most water injected by fire engines from 19:04 on March 12th is highly likely to have failed to reach the reactor according to the investigations to date. It is possible the molten fuel that fell to the PCV bottom remained for some extended time in a situation without being sufficiently cooled.

When the molten fuel cannot be cooled enough, the concrete of the PCV floor is heated up above its melting point and core-concrete reactions start, which dissolve the concrete. The core-concrete reactions produce non-condensable gases such as hydrogen, carbon monoxide, etc., resulting in a big impact on the containment pressure change and radioactive release behavior. But it is unknown to what extent the core-concrete reactions actually occurred. Therefore, examinations are needed to determine the extent of core-concrete reactions as well as their impacts on the accident progression (Common/Issue-5).

The D/W pressure was being maintained at about 0.7MPa[abs] to 0.8 MPa[abs], after reaching 0.84 MPa[abs] at about 02:30 on March 12th, until PCV venting was successful. This fact of constant PCV pressure gives a strong suggestion that the PCV was leaking, despite the constant PCV pressure, because the PCV pressure is expected to rise, when steam is produced due to water injection, PCV temperature rises, and gases are produced by core-concrete reactions, etc.

In the analysis, a gaseous phase leak was assumed from the PCV about 12 hours after the earthquake (at about 03:10 on March 12th) so that the D/W pressure measurements could be well reproduced by the analysis. But no direct evidence was obtained from the parameters measured or facts observed on when and from where the actual leak occurred. Further examinations are needed (Unit-1/Issue-6).

Freshwater was injected by fire engines from about 04:00 to 14:53 on March 12th. But part of the injected water seems to have gone to other systems and equipment, not to the reactor. In the analysis, it was assumed that the injection had not been enough to flood the core region and that only a fairly small amount of water compared to the actual amount of discharged water by the fire engines had been injected to the reactor in view of reproducing containment pressures. The amount of water injected into the reactor represents important information for understanding the accident progression. Further examinations are needed to

know the actual amount of injected water (Common/Issue-2: Examination ongoing (Attachments 1-4, 1-5)). It should be re-emphasized that most of the water injected by fire engines to the reactor during this period was likely to have failed to reach the reactor.

High dose rates were also noticed around the piping and heat exchangers of the reactor building closed cooling water system (RCW) and of the radioactive waste treatment building. There might have been a possibility of FP transfer from the equipment sump in the PCV to the RCW piping, but details of causes are unknown. It is important to clarify the causes of dose rate increase as the unintentional dose rate increase could impede accessibility to the buildings for terminating the accident. Whether part of the water in the RCW flowed into the PCV or gas leaked from the RCW piping also needs to be examined in connection with the accident progression (Unit-1/Issue-9: Examination completed (Attachment 1-9)).

３．２．５． From the containment venting operation to reactor building explosion

Three times at 10:17, 10:23, and 10:24 on March 12th the opening operation of the small S/C vent valve was carried out from the main control room, assuming the availability of residual pneumatic pressures for the valve operation. There was no visible response in the D/W pressure, while the dose rate near the main gate increased temporarily at 10:40. A while later, when a temporary air compressor was connected for opening the large S/C vent valve and it was started up at about 14:00, an up-current of steam above the stack was observed by a live camera (filmed at 15:00 on March 12th) and the D/W pressure decreased from 14:30 until about 14:50. On the other hand, no dose rate increase was observed near the main gate and monitoring post-8 (MP-8).

No details are known concerning the FP release behavior from the PCV before and after the vent valve operation. Examinations are needed concerning what extent the venting operation affected the release (Common/Issue-8).

A high dose rate of 10 Sv/h was noticed around the standby gas treatment system (SGTS) piping connected to the Unit-1 and Unit-2 stack. Also, in the vicinity of the SGTS room a dose rate of several Sv/h was observed. It can be understood that the FPs released during venting stagnated in those areas, but no details are known. It is necessary to examine the emission behavior upon venting as the unintentional dose rate increase could impede accessibility to the buildings for restoration activities (Unit-1/Issue-10).

After the opening operation of the large S/C vent valve, the D/W pressure decreased from 14:30 through about 14:50, as mentioned at the beginning of this section. Later at 15:36, hydrogen in the reactor building exploded and the roof and outer walls of the uppermost floor were damaged. The video film recorded at the time of the explosion confirmed a strong air blast upward with a short time delay after the outer wall damage occurred to the building

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top floor. It is highly possible the camera filmed this strong air blast when it passed through the equipment hatch that ran from the reactor building Floor 1 up to Floor 5 upon the explosion development on Floor 5.

It can be considered that hydrogen gas produced mainly by water–zirconium reactions leaked together with steam and finally reached the reactor building. Although it is difficult to fully identify the hydrogen leak path, volume, explosion aspects and ignition source, the characteristics of the explosion have been found. (Common/Issue-11: Examination completed (Attachment 1-10)).

３．２．６． From the reactor building explosion to March 18th

At 19:04 on March 12th after the reactor building explosion, seawater injection was started by fire engines. But part of the injected water was likely delivered to other systems and equipment, and did not reach the reactor. Actual quantity of water injected into the reactor needs to be examined (Common/Issue-2: Examination ongoing (Attachments 1-4 and 1-5)).

Water injection to Unit-1 and Unit-3 was halted once at 01:10 on March 14th, when the water source used for these two units was depleted. The water injection to Unit-3 was resumed at 03:20 under critical conditions, when the water source was partly recovered by using an additional water supply, but the water injection to Unit-1 was delayed. Water injection to Unit-1 and Unit-3 was again halted with the hydrogen explosion at Unit-3. It is known that water injection to Unit-1 was eventually interrupted from 01:10 to 20:00. Possible impacts of water injection interruption on the accident progression need to be examined (Unit-1/Issue-11: Examination ongoing (Attachment 3)).

Concerning the FP releases after core damage, the analysis showed that almost 100% of the FP noble gases were released to the environment as of 12:00 on March 16th via the PCV leak paths assumed and venting operation. The analysis also showed that about 6% of the total cesium iodide and cesium hydride were released and less than 5% of most other nuclides.

Meanwhile, almost the whole core of Unit-1 dropped down to the lower plenum and of that part most dropped further to the containment pedestal, according to the analysis. There are many unknown matters concerning the location of debris, the final status of accident progression. As data about these matters are important input to future decommissioning steps, further examinations remain based on the outcomes of the investigative research and development projects for the PCV and reactor pressure vessel, and other relevant projects (Common/Issue-10: Examination ongoing (Attachments 4 and 5))

３．２．７．　Other matters

It should be noted that MAAP has uncertainties in its analysis conditions and models, and consequently in its results. In particular, the amount of FP release is strongly affected by these uncertainties. The results should be understood as being simple reference information.

In the main control room for Units-1 and -2, most instrumentation and control power supplies were lost and nothing was available for operators to monitor plant conditions or to take operational actions. There is also the reality, on the other hand, that operators in the main control room were desperately struggling for any actions possible to take at that moment, or later, by referring to system configuration diagrams. An example of such an attempt was that they took an action at the 17:00 level on March 11th to prepare for water injection to the reactor via alternative water injection lines. It is important to verify the psychological conditions that operators and other personnel encountered under such situations in order to extract lessons for implementation to future development of emergency response software (Common/Issue-16).

MAAP has been used in analyzing the accident progression for about a week at the maximum after the earthquake. This is because the uncertainties in analysis results become larger when covering longer time spans, and the result reliabilities decrease accordingly. On the other hand, the FPs released from Fukushima Daiichi NPS around March 20th and 21st might have caused dose rate increases in the Kanto district, as the FPs would be affected by the wind direction; and the authorities recommended the public cut their consumption of tap water, due to concern about increased iodine concentrations and their FP intake. Thus, there is a need to examine the release behavior for a long time after the earthquake, which is difficult to do (Common/Issue-9: Examination completed (Attachment 3-6)).

The issues derived above are shown in Figures 3.2.1 to 3.2.3. They are also described in parallel in Attachment 2. The examination results obtained so far from among the issues extracted as unsolved are compiled in the Attachments and they are summarized briefly in the following section 3.3.
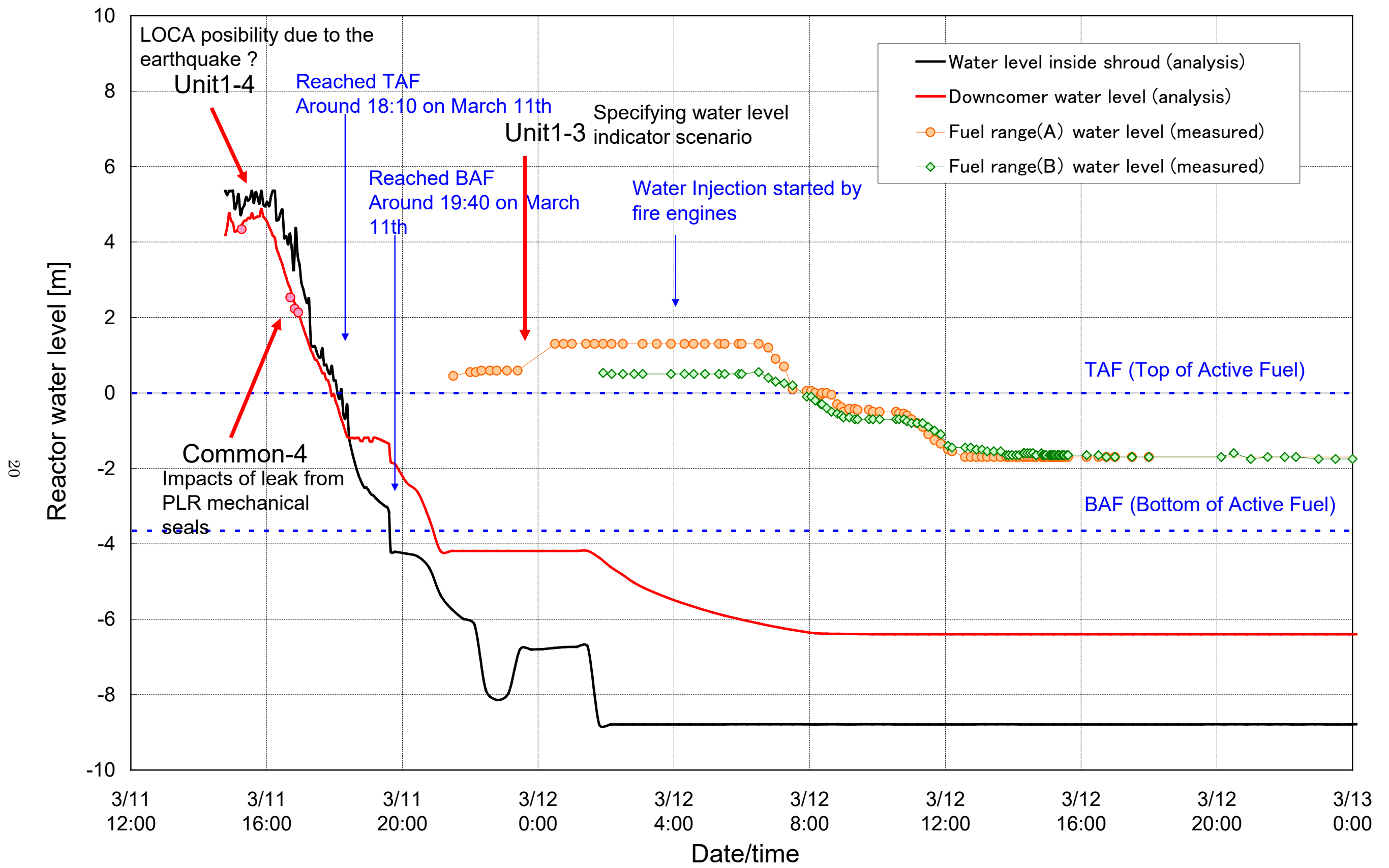
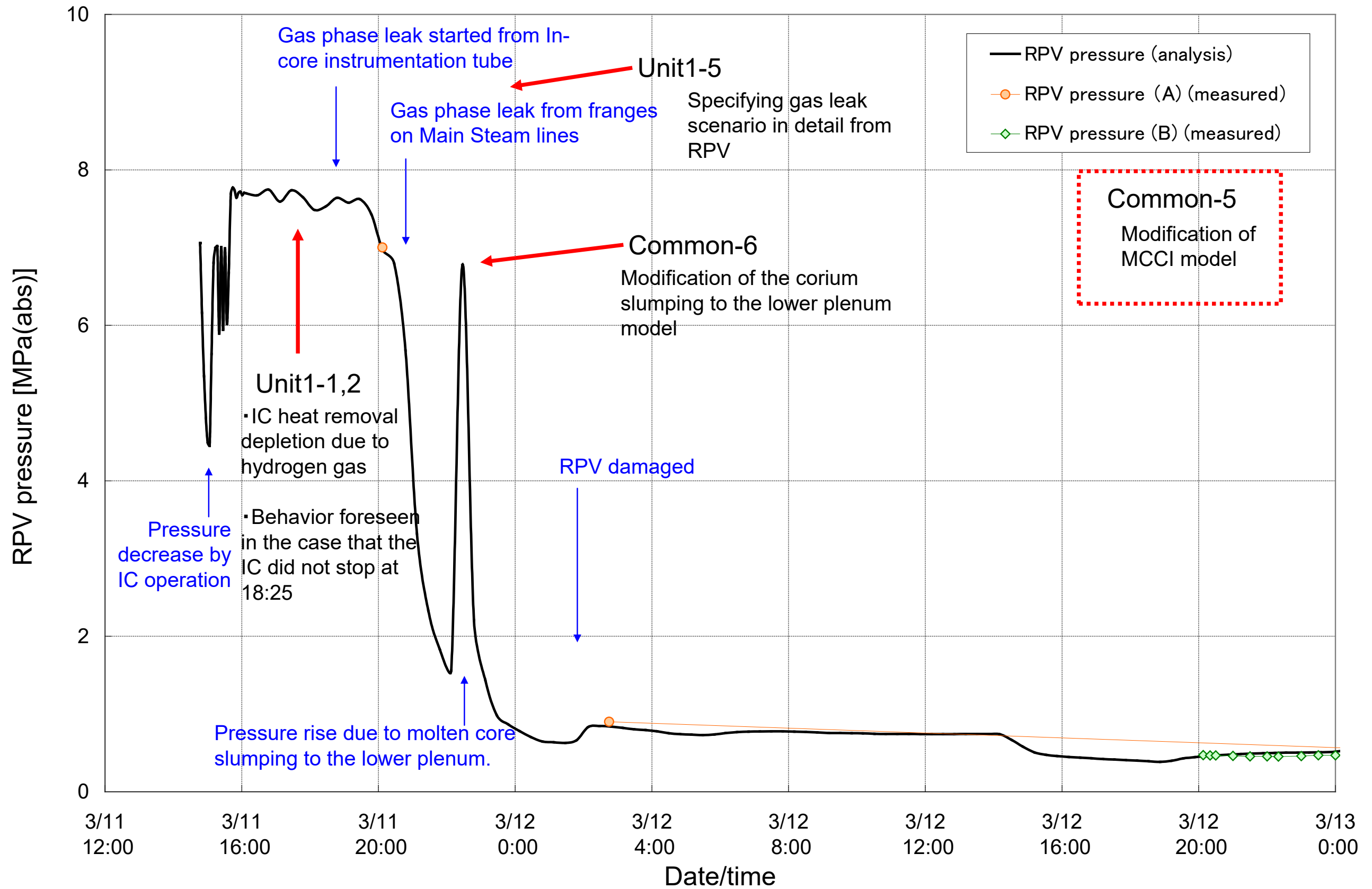Figure 3.2.1 Issues derived from reactor water level change of Unit-1

Figure 3.2.2 Issues derived from reactor pressure changes of Unit-1
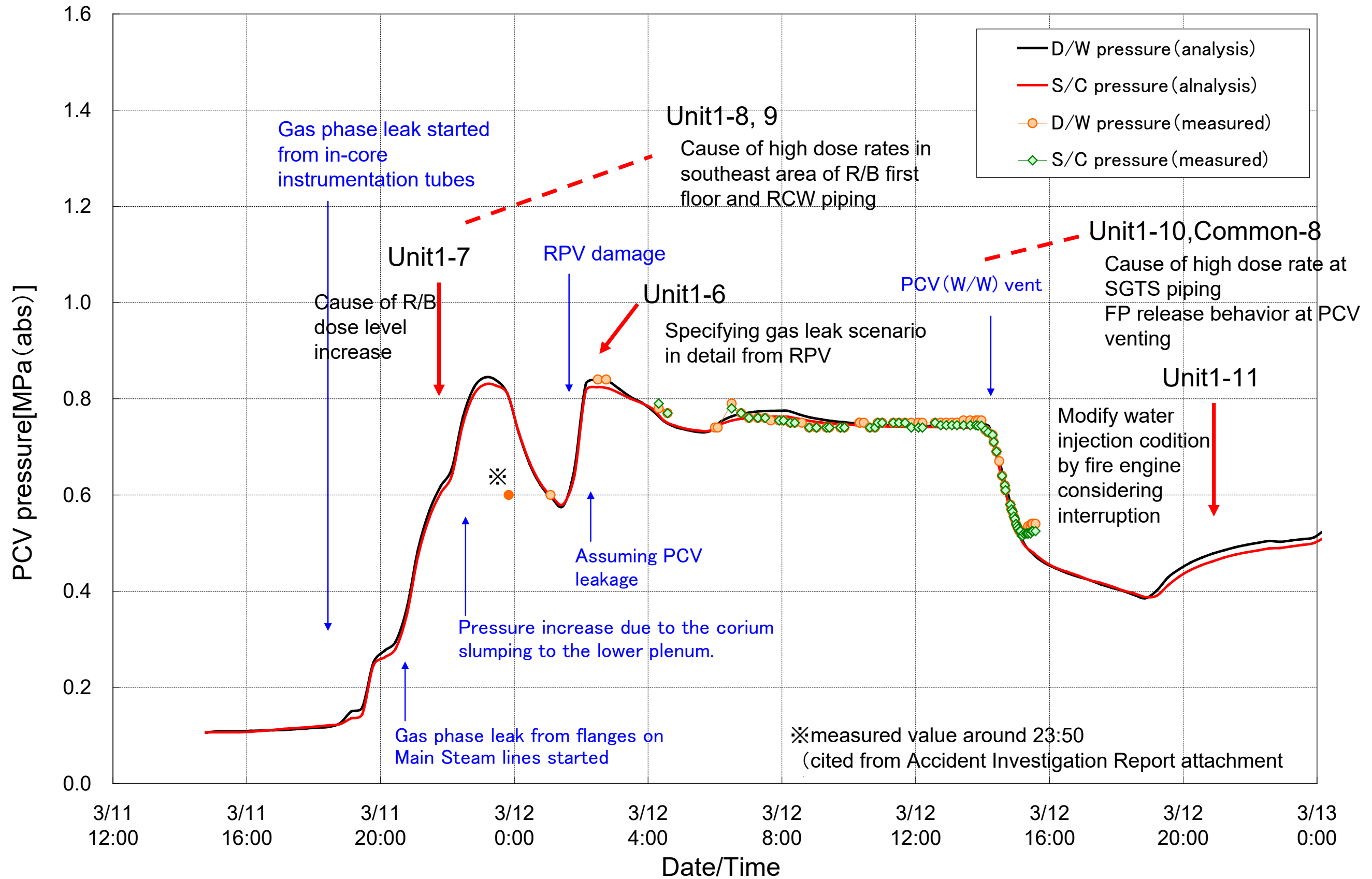
Figure 3.2.3 Issues derived from PCV pressure changes of Unit-1

３．３． Examination results of the issues derived for Unit-1

３．３．１． Impacts of the earthquake

The issue of the possibility of an LOCA caused by the earthquake (Unit-1/Issue-4) was examined (Attachment 1-3).

Unit-1 was assessed to have reached the core damaged situation early in the night of March 11th. Concerning the early development of the accident, a possibility has been noted that the LOCA might have been caused by the earthquake, which advanced the loss of coolant in addition to evaporation, leading to faster accident progression. In the investigation made so far, the LOCA had not been considered as an accident scenario, because the coolant decreasing speed and water level changes measured were consistent. Possible impacts of the earthquake on Unit-1 were examined this time by logical consideration to consistency between LOCA occurrence, measured data and fundamental physical rules such as the law of energy conservation.

As the result, it has been found, by referring to measured data and physical laws, that neither the LOCA caused by piping damage due to the earthquake, or the loss of functions of emergency diesel generators by the earthquake, occurred.

３．３．２． Water injection by fire engines

The issue of the water injection by fire engines (Common/Issue-2) was examined (Attachment 1-4).

The amount of injected water by fire engines has been recorded on-site. But the possibility is well established that part of the injected water flowed not to the reactor but to other systems and equipment, because piping of the make-up water condensate system (MUWC) and the fire protection system is installed at many locations of the plant. MAAP did not assume either that the full amount of injected water had reached the reactor. The amount of water actually injected to the reactor is very important information for examining the accident progression. Therefore, possible leak paths were examined to evaluate their quantities.

It has been found from the piping configuration of the MUWC and the fire protection system that there was more than one path having valves in the "regular open" status or an opening branching from the water injection line from fire engines to the reactor. The leak flow through these paths might have been limited in volume, since the piping was of small diameter or had constant flow valves installed. But the quantitative assessment is important hereafter for reducing uncertainties in injected water quantity to the reactor.

The amount of water injected to the reactor has been made publicly available as a daily average. MAAP has also used it for analysis. In reality, there were water source depletions,

interruptions of injection due to hydrogen explosion, etc. When reanalyzing the accident by MAAP, the interruption time of the injection should be considered, and leaked water volume changes based on reactor pressure changes should also be considered.

３．３．３． Examination of the amounts of water injected by fire engines

The issue of water injection by fire engines (Common/Issue-2) has continued to be examined for evaluating quantitatively the amounts of water injected (Attachment 1-5).

Distribution of water injected by fire engines to the reactor was examined between possible flow paths of the fire protection system lines and the make-up water condensate system (MUWC) lines as well as potential bypass (leak) flow paths. The results showed that about 20 to 50% of the water injected by fire engines reached the reactor. However, this was obtained based on the assumption that fire engines discharged water at about 1 MPa. In reality, operating discharge pressures below this level were occasionally recorded. Therefore, the results still have some uncertainties.

３．３．４． Examination into accident progression from the changes of water level indicator readings

The issue of estimating the accident progression from the readings of defective water level indicators (Unit-1/Issue-3) was examined (Attachment 1-6).

When the fuel range water level indicators were restored at 21:49 on March 11th by the temporary power source, their readings indicated TAF+200 mm. But at that time, the water level indicators are considered to have been already out of order. Still the readings tell the pressure difference between the reference leg and the variable leg. This may give some hints about the accident progression. The energy discharge from the reactor and the containment vessel temperature distribution which could reproduce the fuel range water level indicator readings were calculated by analysis. As a result, it was found that, when gaseous leaks are assumed to occur at an upper position of the RPV, the fuel range water level indicator readings could be well reproduced.

The leak positions assumed in the past analyses could not well reproduce the water level indicator readings. The analysis results in the current examination will be reflected to the MAAP input conditions for reproducing accident progression of higher reliability.

３．３．５． Examination into heat removal by Unit-1 isolation condenser

The Unit-1 IC was started up twice, i.e., before and after the tsunami arrival. The issue of the heat removal by the IC when in operation (Unit-1/Issue-1 and Issue-2) was examined (Attachment 1-7).

After all the AC power supply was lost, the IC was started up at 18:18 on March 11th. The examination was done on the accident progression of a case in which the IC would NOT have been stopped at 18:25 but continued to operate. The examination indicated that, even if the IC had continued its operation, the IC would have lost its cooling capability because the hydrogen gas produced in the core would have stayed behind in the IC line. When compared with the progression in the IC continuous operation after 18:25, the continued IC operation delayed the RPV damage and led to less erosion of the containment vessel concrete. But in the overall progression of the accident it would be quite likely that there was only a minor difference from what actually occurred in Unit-1.

３．３．６． Behavior in molten fuel relocation to below the core

Core support plates and their surroundings in BWRs have complicated structures as compared with those of PWRs. In connection with the issue what paths the molten fuel would have taken when being relocated to below the core (Common/Issue-6), relevant previous studies and the latest research were surveyed (Attachment 1-8).

Concerning the relocation paths from the core to below it, there are five possible paths: (i) the inlet orifice of the fuel support bracket; (ii) the CRDM piping; (iii) the damaged in-core instrumentation tubes; (iv) the broken core support plate; and (v) the broken shroud.

From the examination based on the survey results, it was found that: Paths (i) and (ii) had higher likelihood for molten fuel to have relocated via them; on the contrary, Path (iii) had a low likelihood because molten fuel would have solidified in the small diameter tubes; and Paths (iv) and (v) could not be judged because of dependence on the fuel conditions (sedimenting, solidifying).

The relocation paths will be examined further, as relevant knowledge becomes available and reliable evaluation methods are established by accumulating more information at actual plants relating to the fuel relocation behavior.

３．３．７． Estimation of the cause of high dose contamination of Unit-1 RCW piping

The issue (Unit-1/Issue-9) was examined concerning the high dose rates measured around the reactor building close to the cooling water system (RCW) piping in the reactor building, and the radioactive material treatment building (Attachment 1-9).

As its cause, the possibility has been considered that, since the high dose rates were observed around the RCW heat exchanger, the molten fuel dropped to the pedestal in the PCV and damaged RCW piping housed in the pedestal to cool the equipment drain sump, and thus the radioactive materials were transferred into the RCW piping. In the current examination, the destination of the radioactive materials was investigated, taking all RCW

system components in the buildings into account, under the conditions of high PCV pressures and low PCV pressures. Consequently, the possible destinations and the positions of actually measured high dose rates were found to be roughly consistent.

Thus, the current examination further indicated that the molten fuel had likely dropped from the reactor vessel to the PCV at Unit-1, damaging the RCW piping.

３．３．８． Examination into hydrogen explosion at Unit-1 reactor building

The issue of hydrogen leak paths and the amount of leaked hydrogen to the Unt-1 reactor building (Common/Issue-11) which had caused the hydrogen explosion (Attachment 1-10) were examined.

Among possible hydrogen leak paths from the PCV to the reactor building, it is estimated to be highly possible that hydrogen had leaked from the PCV top flange to Floor 5 of the reactor building via the shield plug and resulted in the hydrogen explosion, since, for instance, the dose rates on Floor 5 had been relatively high. For verifying this estimation, analysis was made on the hydrogen explosion development with the sites of hydrogen leaks and ignition as a parameter. By comparing the results of analysis with the building damage conditions observed, the process of the hydrogen explosion at Unit-1 was estimated. It should be noted that the case of hydrogen leaks from isolation condenser (IC) piping to Floor 4 of the reactor building was also examined, which had been pointed out at the Discussion on Individual Issues of the Fukushima Accident Investigation in Niigata Prefecture Technical Committee "Impacts of Ground Motion on Equipment of Importance." As a result, no inconsistency was noticed with the building damage conditions observed when hydrogen leaks had been assumed only on Floor 5, while it was difficult to make a consistent explanation when hydrogen leaks on Floor 4, in addition to Floor 5, had assumed. Consequently, the scenario to date that hydrogen leaked to Floor 5 of the reactor building was estimated to be more likely.

３．３．９． Estimation of accident progression of Unit-1 based on the air dose rate monitoring data

The initial accident progression behavior at Unit-1 was estimated based on the air dose rate monitoring data (see Attachment 1-11) as part of the examination into the correlation between the timing of a large amount of radioactive materials released to the air and the monitoring data (Common/Isuue-7).

When analyzing the air dose rate change behavior, two patterns were considered: one was the direct and sky shine radiations from radioactive materials that had transferred to the reactor building; and the other was the cloud shine radiation from the plumes of the

radioactive materials that had been released from the reactor building. The transfer and release behaviors of radioactive materials were estimated by examining which radiation source was dominant in each period of interest. The accident progression was also examined from the PCV pressure behavior viewpoint.

Consequently, it was estimated that: radioactive materials were transferred from the PCV to the reactor building at about 04:00 on March 12th in an amount that could be detectable outside the reactor building; and some event occurred at about 06:00 on March 12th to increase PCV pressure, which accelerated the transfer of radioactive materials to the reactor building. This estimation was consistent with the estimations concerning the accident progression in the study to date (Attachment 1-6): "radioactive materials have been transferred to the PCV as of March 11th;" and "the RPV (bottom head) was damaged at about 06:00 on March 12th."

３．３．１０． Identification of the cause of the high radiation dose rate observed in the southeast area of the first floor of the Unit-1 reactor building

Issues related to the cause of the high dose rate observed in the southeast area of the first floor of the Unit-1 reactor building (Unit-1-8) were discussed (Attachment 1-12).

Four possible contamination sources were identified as possible causes of the high dose rate: ① contamination by steam and contamination of the torus room; ② contamination of AC piping; ③ contamination of RCW piping; and ④ contamination of the TIP room. The effects on the southeast area were examined from the following three perspectives: (1) cause of contamination, (2) effect of radiation from the contamination source, and (3) presence of radioactive material migration from the contamination source. Regarding ①, ③ and ④, it was estimated that the radiation from the contamination source was sufficiently shielded by the concrete body and that there was no significant radioactive material migration from the system to the southeast area on the first floor of the reactor building, and therefore, their influence on the southeast area was not dominant. Regarding ②, although it is considered that there was no significant radioactive material migration to the southeast area on the first floor of the reactor building due to damage to the AC piping, it was confirmed that the contamination in the AC piping identified along the piping was consistent with the characteristics of the air dose rate observed in the southeast area.

Therefore, the high dose rate observed in the southeast area was identified as being the dominant factor due to radiation from the AC piping used for containment venting.

３．３．１１． Examinations into other issues

Examination results of other issues derived in "3.2. Issues derived from the comparison

between measured information of Unit-1 and analyses" will be added to this section as soon as they become available.

３．４． Summary of Unit-1 examinations

Some of the issues derived from the comparison between MAAP analysis results and measured information of Unit-1 were examined, and rational interpretations of phenomena have been obtained for some issues as follows.

✓ There were no indications in measured reactor pressures which showed LOCA caused by piping damage due to the earthquake as described in "3.3.1. Impacts of the earthquake."

✓ Part of the injected water by fire engines flowed not to the reactor but to other systems and equipment as described in "3.3.2. Water injection by fire engines."

Hereafter, this latest information will be considered as input to the analysis for increasing reliability.

4． Examinations into the accident progression at Unit-2

4．1． Approach for evaluation

The analysis results of MAAP (Attachment 1) have been mainly used to examine the accident progression process, excluding the effects of the earthquake and tsunami, of Fukushima Daiichi NPS Unit-2 (hereinafter referred to as "Unit-2"). Figure 4.2.1 shows the reactor water level changes, while Figure 4.2.2 shows the reactor pressure changes and Figure 4.2.3 shows the PCV pressure changes. However, the MAAP results cannot perfectly reproduce the actual accident progression because of the uncertainties in its analysis conditions, analytical models, and consequently its results obtained. In this report, therefore, the following steps were taken for examinations: First, discrepant points were identified as issues between the MAAP results done in the past (See separate Volume 1 for the results by MAAP4, and Attachment 3 for the results by MAAP5) and actually observed measurements; and then, the issues identified were examined one-by-one. Section 4.2 explains in chronological order the issues extracted and Attachment 2 describes each issue individually.

4．2． Issues derived from the comparison between measured information of Unit-2 and analyses

4．2．1． From the earthquake to tsunami arrival

At Unit-2, the following operation steps were being taken towards cold shutdown: start up and shutdown of the reactor core isolation cooling (RCIC) system, startup of the residual heat removal (RHR) system in the S/C cooling mode, etc. Unit-2 lost all power supplies due to damage by the tsunami at 15:41 on March 11th. At Unit-2, as the RCIC system had been manually started up at 15:39 just before the DC power for control was lost, water injection to the reactor could continue after the tsunami arrival. This was the big difference between Unit-1 and Unit-2 situations, i.e., at Unit-1 the IC had been shut down before the tsunami arrived, and therefore the IC could not be restarted upon loss of the control power supply.

4．2．2． From the tsunami arrival to reactor water level increase

A possibility was hinted at that the RCIC system was in operation, with no control power supply due to tsunami, being driven by water-steam mixture, i.e., two-phase flow, which had been produced when the reactor water level increased to a level above the main steam line, thus water was flowing into the steam piping, as seen in Attachment 2-1. But no behavior prior to the water level increase up to the main steam line has been confirmed. In the analysis, the water injection rate was adjusted as 30% of the rated value, so that the reactor pressure changes measured could be reproduced during the period while the RCIC was

thought to have been driven by two-phase flow. According to the results under this condition, the reactor levels calculated during the time period prior to the water level increase up to the main steam line increased more slowly than the measured values. This raises the need to investigate the RCIC behavior after loss of power supply due to tsunami (Unit-2/Issue-1).

It should be noted that, from its turbine performance, the RCIC is considered to be operable in two-phase flows containing water in steam. Such operability with no need of control is compatible with the concept of passive safety equipment. Accumulation of knowledge concerning the water injection characteristics when driven in two-phase flows, operable time duration with no auxiliary equipment requiring DC power supplies, and other features will be useful.

In the meantime, during the RHR system operation in the S/C cooling mode, the pumps were considered not to be being operated due to the loss of all AC power supplies. If the RHR system configuration (valve open/closed positions) had been maintained after the power supply loss, plant behavior including the D/W pressures might have been affected due to energy flow to the RHR system. This raises another issue to investigate (Unit-2/Issue-4: Examination completed (Attachment 2-5)).

４．２．３． From the reactor water level increase to loss of RCIC functions

After the reactor water level increased, no accurate water levels could be estimated, because the fuel range reactor water level indicator had reached their upper limit of measurement, as is mentioned in the next paragraph. The reactor pressure, on the other hand, started to decrease after the RCIC started up (it should be noted that MAAP4 gave a later time for the pressure to start decreasing, while MAAP5 overestimated the pressure decrease). When it reached 5.4 MPa[abs] at 01:30 on March 12th, the reactor pressure began to rise again. In the time sequence, this pressure change had no connection with the switchover of water sources from 04:20 through about 05:00 on March 12th. The reactor pressure and temperature changes due to RCIC water injection, and the relationship between lowered saturation temperatures due to pressure decrease, might possibly be able to explain the above pressure change behavior of decrease and increase. Therefore, if the water injection rates from RCIC to the reactor can be determined to reproduce this pressure reversal behavior, although unknown yet, it will help to reveal the accident progression including the RCIC water injection properties.

Incidentally, the reactor water levels measured were higher than the "reactor water level high (L-8)" (Attachment 2-1) after correction of the reactor pressure increase and containment temperature increase, as was shown by the blue points in Figure 2-1 "Reactor Water Level Changes of Unit-2" in Attachment 3. This water level corresponds to the upper

limit of the fuel range reactor water level indicator measurement range.

While the RCIC operation was being continued with no control power supply, the reactor pressure is considered to have stayed, as discussed in Attachment 2-1, at lower levels than the level at normal operation for the following reasons.

- The reactor water level rose above L-8 because of no control of the RCIC valve apertures for adjusting steam flow rates.
- Decay heat energy was removed from the reactor by low quality two phase flows.
- The water was injected by the RCIC at a lower flow rate than the rated value, because the RCIC turbine was operated by low quality two phase flows.
- Thus, the energy in the reactor vessel was kept balanced without the SRV operation.

The reactor pressure varied in a downward trend again from about 06:00 on March 13th. This can be understood as the effect of decreased decay heat with time. Thereafter, the pressure increased again after it was measured as 5.4 MPa[abs] at 09:00 on March 14th and reached 5.6 MPa[abs] at 09:35. MAAP could reproduce, as shown in Attachment 2-1, the gradual reactor pressure increase, assuming interruption of water injection by the RCIC system (but steam supply to its turbine continued) at 09:00 on March 14th. MAAP could also reproduce the sharp pressure increase thereafter, assuming full shutdown of the RCIC system at 12:00 on March 14th. The assumptions made in the analysis could reproduce quite well the reactor pressure changes, but why the RCIC stopped is unknown. It is necessary, therefore, that the RCIC shutdown mechanism consistent with those assumptions in the analysis be investigated (Unit-2/Issue-2).

The containment pressure varied at lower levels than anticipated, despite the fact that all the decay heat was stored in the S/C, because of the loss of the ultimate heat sink (LUHS). In the process of Unit-2 accident progression, it is considered that the SRV did not operate when the RCIC was in operation. This means the RCIC exhausted two-phase steam that flowed into the S/C accompanied by energy equivalent to the decay heat energy. As a result, the energy stored in the S/C raised the containment pressure. It has been noticed, on the other hand, as discussed in Attachment 2-2, that the gradual increase of D/W pressure measurements can be reproduced by assuming the inflow of seawater into the torus room and heat removal from the S/C outer walls by seawater.

４．２．４． From the loss of RCIC functions to forced depressurization by SRV operation

Although it has not been clarified at what time the RCIC system shutdown, the reactor water level started to decrease gradually after RCIC stopped, uncovering the core, and then it rapidly decreased due to flashing (depressurization boiling) by opening an SRV. The core

was completely uncovered and core damage started (Figure 2-1 in Attachment 3). After the reactor pressure increased due to RCIC system shutdown, it was maintained at about 7.5 MPa[abs] due to the SRV relief valve mode; (the SRV(A) had been connected to temporary batteries). Thereafter, the reactor pressure sharply dropped upon opening the SRV and finally approached the ambient pressure.

The reactor pressures and water levels were measured once the water level had gone below the maximum range of the fuel region reactor water level indicator, following the RCIC shutdown. Further, the reactor water levels and pressures could be reproduced with good accuracy (Figure 2-2 in Attachment 3). This was done by appropriate processing of the energy balance and property changes over the time span until the forced depressurization by the SRV, because the water in the reactor decreased monotonously, although it was being accompanied by pressure changes.

The measured values of PCV pressure changed downward from about 13:00 on March 14th after the RCIC system had stopped. It can be considered that this was because heat continued to be removed from the S/C by the seawater which flowed into the torus room, although no more energy was transferred to the S/C through the RCIC turbine. However, the analysis cannot reproduce these transitions. The pressure decrease, judging from the changes in reactor pressure increase, started more than one hour later than the 12:00 time on March 14th assumed in the analysis as the timing of full RCIC shutdown. This coincides roughly with the time period when the energy inflow to the S/C due to SRV operation started but is inconsistent with the scenario of energy inflow termination and continued heat removal from the S/C outer walls. This means the PCV pressure changes need examination, including consideration for the PCV leak scenario (Unit-2/Issue-5: Examination completed (Attachment 2-6)).

Regarding the PCV pressure upon depressurization by the SRV, it remained stable at about 0.4 MPa[abs] from 17:00 through 20:00 on March 14th and the anticipated pressure increase was not seen, despite the big steam (energy) inflow to the S/C upon depressurization by the SRV (MAAP predicted pressure increase upon depressurization by the SRV). This raises another issue to investigate, that is, the pressure behavior upon depressurization by the SRV (Unit-2/Issue-6: Examination completed (Attachment 2-6)). It should be noted that the reactor water level should decrease rapidly due to decompression boiling upon SRV forced depressurization, down to the bottom of active fuel, that is, the fuel becoming fully uncovered. During this period, no core damage and no core melt occurred at Unit-2. This differs from the situations at Unit-1 and Unit-3, in which the reactor water levels decreased gradually only by decay heat while the reactor pressure remained high, and core damage/core melt occurred due to water-zirconium reactions in the uncovered fuel region.

It has been shown, concerning Unit-2/Issue-5 and Unit-2/Issue-6, that the PCV pressure decrease may be explained by stratification of S/C water due to cooling of the lower part of the S/C by the water flowing in the torus room from outside, and by the mixing effect of S/C water with the released water that had been retained in the main steam lines when the SRVs worked at an early stage after the pressure increase. Similarly, concerning Unit-2/Issue-6, that no significant PCV pressure increase was anticipated upon reactor depressurization may be explained by the complete condensation of steam generated upon reactor depressurization by the cooled water at the lower part of the S/C.

４．２．５．　From the forced depressurization by SRV to PCV pressure decrease initiation

About at the same time when depressurization by the SRV was completed, water injection was started by fire engines. But the amount of water set in the analysis was insufficient to flood the core and core damage developed. Sufficient data on reactor water levels were not available, but their increasing trend after 21:00 on March 14th could be confirmed. This reactor water level increase, however, could have been caused by overestimating the real level due to water evaporation inside the reference water level side piping in the accident progression, as in Unit-1. The water level indicator became unable to show accurate values after all, although the timing when this happened is unknown. Therefore, the actual amount of injected water is considered to have been less, too, including its possible leakage from the injection lines of the fire engines (Common/Issue-2, Common/Issue-3: Examination ongoing (Attachment 2-14)).

At Unit-2, the core became fully uncovered while the water level was low, as mentioned in 4.2.4, and fuel cladding temperatures started to rise, a big amount of hydrogen was produced by the water – zirconium reactions (Figure 2-6 in Attachment 3).

The PCV pressure increased to 0.75MPa[abs], thereafter, due to hydrogen formation and SRV opening, etc. The D/W pressure increases were observed at about 20:00, 21:00 and 23:00 on March 14th, probably being effects of hydrogen formation. In the meantime, the S/C pressure measurement started from 04:30 through about 12:30 by the normal pressure indicator, which showed similar values with those of D/W pressures. Thereafter, the measurement was interrupted once due to defective indicators. The measurement resumed at 22:10 using the S/C pressure indicator for accident management. This pressure indicator gave lower values than the D/W pressure from the beginning. As such pressure gaps are unlikely to occur in view of the PCV structure, it is highly possible that these pressure measurements did not show the actual pressures. Eventually, the S/C pressure indicator dropped below the lower end of the scale at 06:00 on March 15th, indicating the instrumentation system malfunction. Since some useful information may be obtained from

the fluctuations of indicated pressure values and the indicator malfunction timing, examinations into the S/C pressure indicator behavior need to be continued (Unit-2/Issue-3: Examination completed (Attachment 2-17)).

SRV opening was repeated after the forced depressurization of the reactor in order to control the reactor pressure increase which had occurred occasionally. But the reactor pressure decrease and SRV manual operation did not necessarily coincide. For example, an SRV opening operation was recorded at 21:00 on March 14th and 01:10 on March 15th, but not at about 23:00 on March 14th, when the reactor pressure increased and decreased. With this background, behavior of the reactor and PCV pressures was examined around this time. As a result, the reactor pressure changes on this occasion are considered due to, not only the SRV operations, but also the progressing core damage/core melt caused by water-zirconium reactions in a large scale. Especially, the reactor pressure increases and decreases at about 23:00 on March 14th are estimated to have reflected the changes of in-core conditions enough to increase the reactor pressures (hydrogen formation, steam generation, temperature increase, etc.) which overcame the effect of SRV openings, because of the accompanying PCV pressure increase (Unit-2/Issues-7, 8: Examination completed (Attachments 2-7, 2-9 and 2-12)).

At Unit-2 preparation was underway for the S/C venting, but no decisive evidence exists as to whether or not the rupture disk was opened. But it was at about 23:00 (measured pressure at 23:00 was 540 kPa[abs]) on March 14th when the D/W pressure exceeded the preset rupture disc operating pressure (528 kPa[abs]), even if the measured S/C pressure was not correct. In the meantime, a radiation monitoring car did record a sharp rise in dose rates at about 21:20 when the SRV opening operation was recorded. It is necessary, therefore, to examine in what state the rupture disc was and why the dose rates rose (Unit-2/Issue-9: Examination ongoing (Attachment 4)). It should be noted that the dose rate around the rupture disk was confirmed to be low and therefore the dose rate increase recorded by the monitoring car could have been caused by radioactive material leaks from other than vent lines, if it originated in Unit-2. The occasional increase in reactor pressure around this time was at most about 1.5 MPa[abs] and non-condensable hydrogen gas is considered to have mixed in the discharged steam upon pressure decrease, because core damage is thought to have developed by this time. Whether or not the S/C integrity was affected by the pressure increase due to the non-condensable gas not being condensed is another issue to examine (Unit-2/Issue-10: Examination ongoing (Attachment 2-8, 2-13)).

4．2．6． From the PCV pressure decrease initiation to March 18th

The measured PCV pressure was 0.73 MPa[abs] at about 07:20 on March 15th, and then

it decreased to 0.155 MPa[abs] at 11:25 on March 15th. It is not clear when the pressure started to decrease, because the measured data are limited around this time period due to the temporary reduction of workforce at Fukushima Daiichi NPS. Still it is highly possible that this pressure decrease occurred during the morning, as suggested by the facts that (1) steam release from the Unit-2 blowout panel was confirmed in the morning on March 15th, and (2) the dose rates measured by monitoring cars increased. The FPs released at this time are believed to have resulted in radioactive contamination in Iitate Village, etc. The mechanism needs to be examined how this pressure decrease of the PCV occurred (Unit-2/Issue-11: Examination ongoing (Attachment 2-16)).

The containment atmospheric monitoring system (CAMS (D/W)) in the meantime, showed a monotonous increase until around 06:00 on March 15th (63 Sv/h at 06:20) and then a lowered value (46 Sv/h at 11:25) after an interruption of data recording for about 6 hours. The PCV pressure decrease would explain the dose rate decrease in the PCV, by the FP release from it. The CAMS (D/W) recorded a sharp rise to 135 Sv/h later at 15:25 on March 15th. This sharp rise suggests some incidents developed abruptly in the reactor and PCV (Unit-2/Issue-12: Examination completed (Attachments 2-10, 2-11, and 2-18)).

The analysis to date predicted that the total amount of hydrogen formed over about a week after the earthquake was about 450 kg (Figure 2-6 in Attachment 3). The reasons for no hydrogen explosion at Unit-2 could possibly be hydrogen leakage from a blowout panel or ceiling holes, or the lower hydrogen formation rate of Unit-2 as compared to Units-1 and -3. It was mentioned earlier in 4.2.5 that a large amount of hydrogen could have been produced by water-zirconium reactions during the reactor pressure increase after forced depressurization. The reason for no hydrogen explosion at Unit-2 is likely due to hydrogen leaks from the reactor building to the outside before an explosion could occur. But the hydrogen leak behavior from the reactor to the reactor building after hydrogen production still remains to be clarified by further examination (Unit-2/Issue-13).

Concerning the FP release, the analysis indicated that the FP noble gases were discharged to the S/C from the reactor vessel after the core damage, and almost all of them were released outside the PCV, based on the leaks from the PCV assumed in the analysis. The release fraction of cesium iodide was about 1%, while most of it remained in the S/C. But there is a possible gap between the analysis results and the reality, since the FP release outside the PCV was based on an assumption of leaks from the PCV.

The analysis also gave the result (both MAAP4 and MAAP5) that the Unit-2 core remained in situ and the reactor vessel was not damaged, although part of the molten fuel remained as a pool. This may be due to such reasons as that water injection by the RCIC system was continued rather well at the beginning, and that the water injection by the fire

engines could be started with a relatively shorter time delay after the RCIC shutdown, in comparison to the situation with Unit-1. But it is known, as described below, that in Unit-3, too, core damage and melting were likely to have started before water injection by fire engines started. This means that only in Unit-2 did the core damage and melting start when the reactor pressure was increasing and decreasing, while water was being injected by fire engines after the reactor depressurization. Therefore, it is considered that the RPV damage is highly influenced by the amount of water injection by fire engines, and uncertainties in analytical conditions have big effects on the analysis results.

4．2．7．　Examinations into other matters

It bears noting again that MAAP has uncertainties in its analysis conditions, models, and consequently, in its results. In particular, the amount of FP release is strongly affected by these uncertainties. The results should be understood as being simple reference information.

If the sharp rise of the (CAMS) (D/W) readings at 15:25 on March 15th was assumed to be due to the debris falling when the reactor vessel was damaged, as elaborated in examining Unit-2/Issue-12 (Attachment 2-10), MAAP would not be able to reproduce the reactor vessel damage around this time period, as far as the current MAAP results indicate. In the meantime, Figure 3-3 in Attachment 4 confirms the control rod drive mechanism (CRD) piping at the RPV bottom remained at least in the peripheral positions. In addition, in Figure 3-22 of Attachment 4, locations where control rod position indicator probe (PIP) cables or local power range monitoring system(LPRM) cables cannot be identified at the RPV bottom and the location where the grating on the platform in the pedestal is missing overlap, both off the center. This indicates that the RPV damage is considered to be present around the RPV bottom center and its peripherals. In order to simulate such phenomena, the effects of debris relocation behavior should be properly analyzed in consideration of the complicated configuration of the BWR vessel lower structures. Improvements in the analytical model will be needed for increasing reliabilities in the analysis results. Currently, the location of debris, the ultimate result of the accident progression, is still unknown. Since this information is important input to decommissioning planning, further examinations are needed, based on the outcomes of the investigative research and development projects of the PCV and reactor pressure vessel, and other relevant projects (Common/Issue-10: Examination ongoing (Main report chapter 7, Attachments 4, 5).

MAAP has been used in analyzing the accident progression for about a week at the maximum after the earthquake. This is because the uncertainties in analysis results become larger when covering longer time spans, and the result reliabilities decrease accordingly. On

the other hand, the FPs released from Fukushima Daiichi NPS around March 20th and 21st might have caused dose rate increases in the Kanto district, as the FPs would be affected by the wind direction; and the authorities recommended the public cut their consumption of tap water, due to concern about increased iodine concentrations and their FP intake. Thus, there is a need to examine the release behavior long time after the earthquake, which is difficult to do (Common/Issue-9: Examination completed (Attachment 3-6)).

The issues derived above are shown in Figures 4.2.1 to 4.2.3. They are also described in parallel in Attachment 2. The examination results obtained so far from among the issues extracted as unsolved are compiled in the Attachments and they are summarized briefly in the following section 4.3.
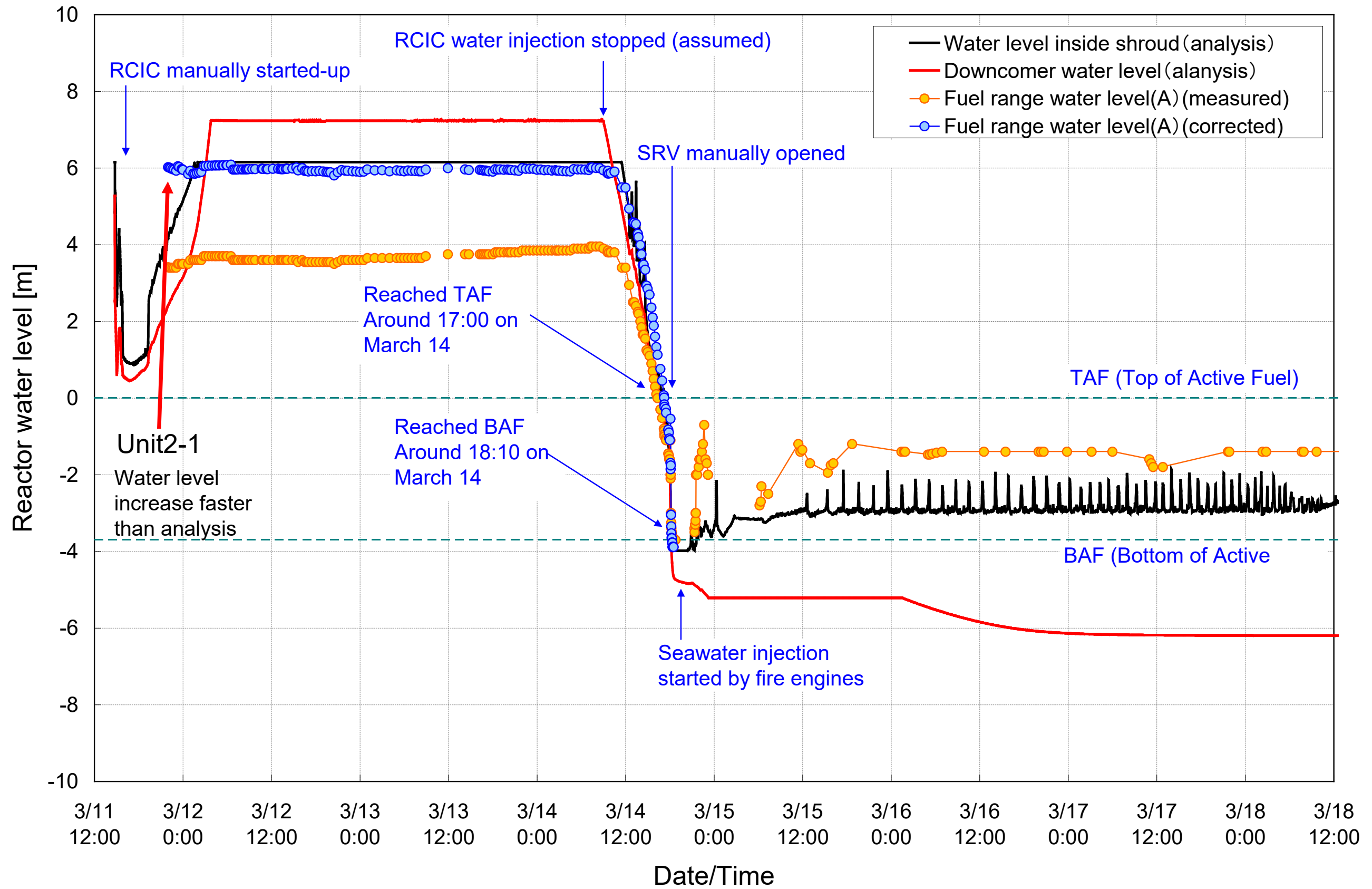
Figure 4.2.1   Issues derived from the reactor water level changes at Unit-2
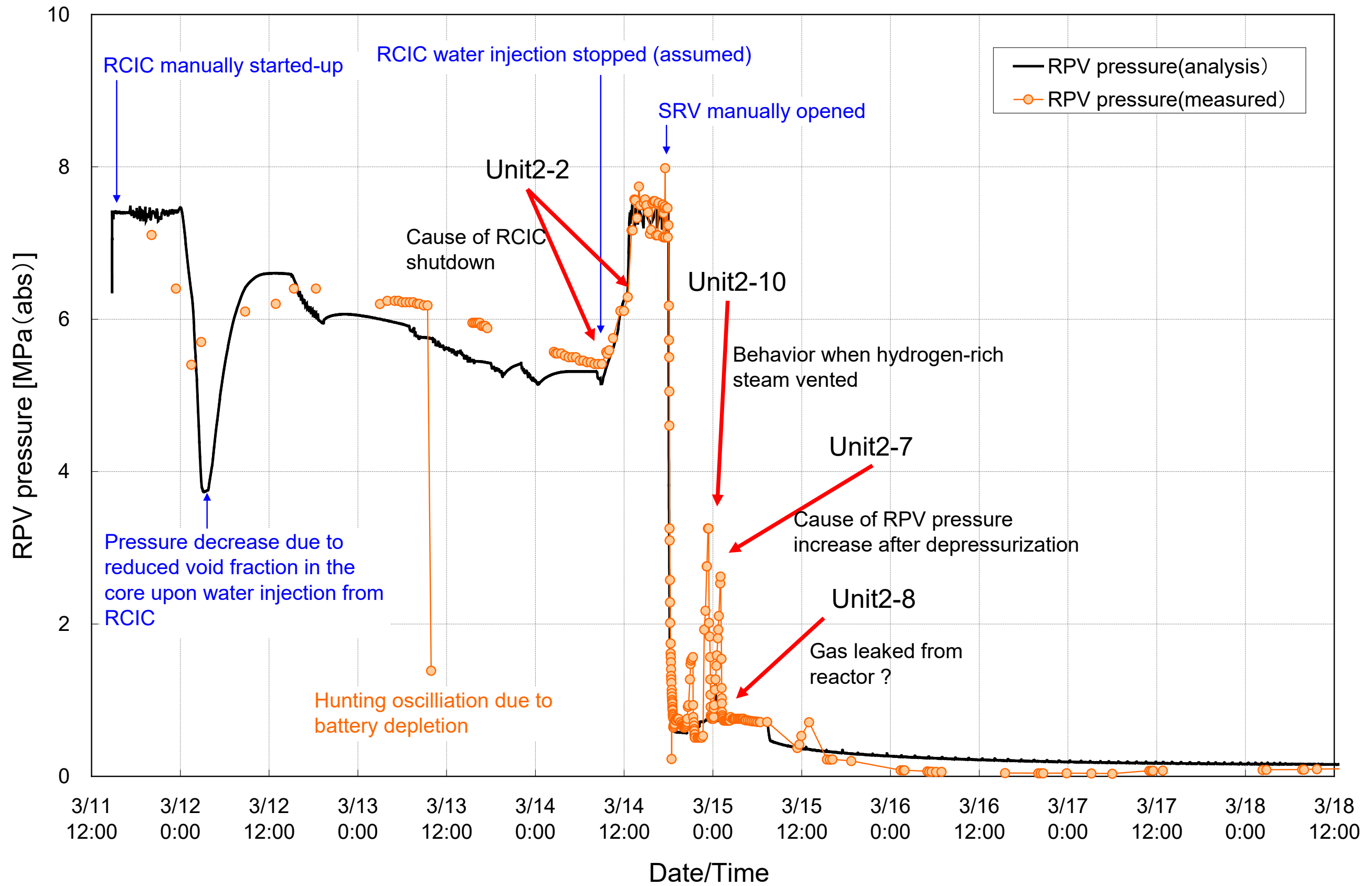
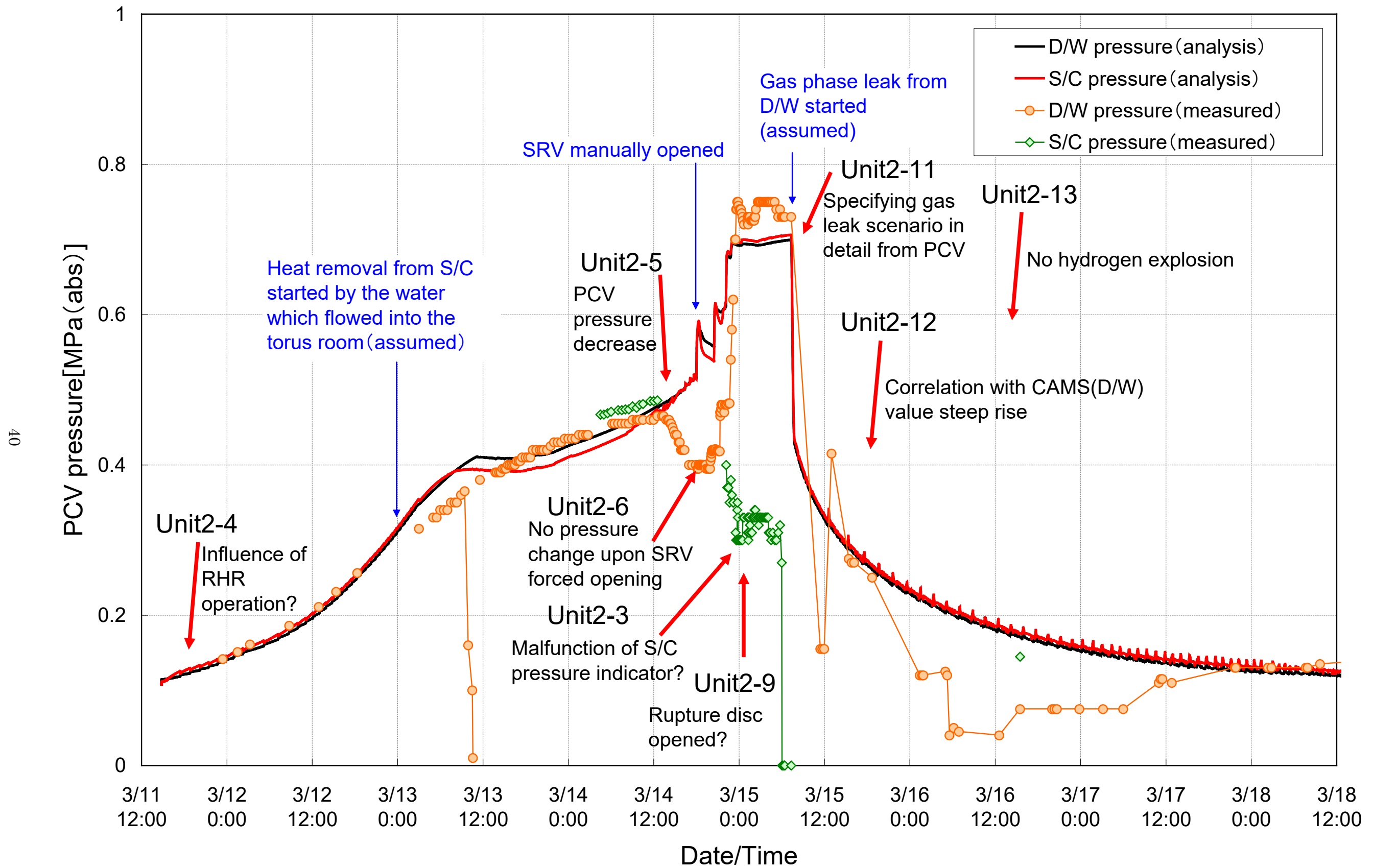Figure 4.2.2   Issues derived from the reactor pressure changes at Unit-2

Figure 4.2.3   Issues derived from the containment pressure changes at Unit-2

４．３． Examination results of the issues derived for Unit-2

４．３．１． RCIC operation behavior without DC power supply

Unit-2/Issue-1 and Issue-2 were examined concerning RCIC operation behavior when the DC power supply for the RCIC control was lost (Attachment 2-4).

It has turned out that the water flow rate increased as the RCIC design provided for fully opening the steam regulator valve once the DC power supply was lost. This flow rate increase is understood to have continued at least until the reactor water level increased to the level of the main steam line.

Further it is known that the RCIC system may trip mechanically by the fully opened steam control valve. But continued examination is still needed, as not everything has been clarified as to why the RCIC system lost its functions.

４．３．２． RHR system configuration after tsunami arrival

Unit-2/Issue-4 was examined concerning the RHR system configuration after the tsunami arrival (Attachment 2-5).

It has turned out through reinvestigation of the actions of operators that an action had been taken to isolate the RHR system before the valves became inoperable due to loss of the power supply by the tsunami. It is found, therefore, that there were no direct correlations between the RHR system configuration and the reactor vessel or PCV behavior, and the RHR system temperature increase measured was caused by some separate reasons such as temperature rise in the reactor building.

４．３．３． Containment vessel pressure decrease after RCIC system shutdown

Unit-2/Issue-5 (related to Unit-2/Issue-6) was examined concerning the PCV pressure decrease after RCIC system shutdown (Attachment 2-6).

It is known that the actual PCV pressure increase was lower than the anticipated value corresponding to the energy inflow by decay heat. It is presumed that seawater, which had flowed into the reactor building basement, removed heat from the PCV from outside it. However, no quantitative explanations have been possible concerning the fact that the PCV pressure reversed downward from past noon on March 14th and that this decrease started at around the time when the reactor pressure control by the SRV was started again.

Plant behavior during the time of pressure decrease has been examined, based on:
- The S/C water temperature measurement chart, which was temporarily resumed to work at that timing; and
- The newly obtained knowledge that the SRV(A) connected to batteries was, in high probability, the only valve working in the relief valve mode.

It has turned out as a result that the PCV pressure decrease would probably be explained by considering the following factors.

- Status of RCIC operation, i.e., energy inflow to the S/C from the RCIC turbine
- Cool water injection to the reactor from the S/C
- Energy balance at the S/C wall outer surface

In addition, a possibility was also examined concerning the timing of the PCV pressure decrease as to whether it had been initiated by discharging the residual water in the main steam line from the SRV.

This work needs to be continued since quantitative reproduction calculations are necessary for verification.

4．3．4． Possible scenario for fuel melting and detection of neutrons

An examination was conducted about the reactor pressure increase "Unit-2/Issue-7" (Figure 4-2-2) among pressure changes after Unit-2 had been depressurized (Attachment 2-7).

The Unit-2 reactor pressure repeated increases and decreases several times after it had been depressurized. This indicated the possibility that steam generated by water injection by fire engines accelerated the water-zirconium reactions, intermittently releasing a large amount of energy, and that caused pressure changes and fuel melting. From the night of March 14th till early in the morning of March 15th, when the reactor pressure increase was observed, neutrons were detected a few times, although the intensity was at the minimum detection level. This indicated a high possibility of detecting neutrons emitted by spontaneous fissions by some actinides of high spontaneous fission probability such as curium, which had been released from the core during fuel melting and leaked from the reactor building　possibly with other actinides such as uranium and plutonium.

Continued examination is needed, since the leak paths of actinides have not been identified.

4．3．5． Impacts of hydrogen-rich steam when released to S/C

Unit-2/Issue-10 was examined concerning the impacts of hydrogen-rich steam released to the S/C, which is considered to have been produced when the reactor pressure increased after depressurization (Attachment 2-8).

The damage location of the Unit-2 S/C has not been confirmed, but by examining the measured temperatures it was estimated to be near the bottom of the S/C or somewhere on the piping connected to the bottom area of the S/C. It was also shown that at least one vacuum breaker was likely to have had problems of closure functionality and that one

reason for the damage could be because the released hydrogen-rich steam had quickly raised the S/C pressure.

4．3．6． Reactor pressure increase after reactor depressurization

The examination of Unit-2/Issue-7 was continued from 4.3.4 above, concerning the reactor pressure increase after depressurization (Attachment 2-9).

The amount of steam and hydrogen in the reactor was estimated which could reproduce the reactor and PCV pressure behaviors after forced depressurization. The results showed that a large amount of hydrogen had been produced by large scale water-zirconium reactions which had occurred while the reactor pressures had repeated increases and decreases. It was also found that, in addition to the scenario presented in Attachment 2-7 (water injection from fire engines repeatedly accelerated hydrogen production in the core region), water-zirconium reactions had been further accelerated by: the increased fuel temperatures due to heat from the water-zirconium reactions; the melting of in-core structures; the falling down of molten objects to the lower plenum; and water evaporation.

4．3．7． Rapid increase of CAMS readings observed on March 15th

Unit-2/Issue-12 was examined concerning the rapid increase of CAMS (D/W) readings to the recorded high of 135 Sv/h observed at 15:25 on March 15th (Attachment 2-10).

Based on water filling of the reference leg side piping of the fuel range water level indicators, the Unit-2 RPV is estimated to have been damaged to the extent that the water level could not be maintained. This means that the RPV damage had occurred at the time of this rapid increase of the CAMS readings. The CAMS readings over a long time span indicate that a drastic change never happened thereafter in the reactor situation and that overheating, melting and relocation of fuel debris are unlikely to have occurred again.

4．3．8． FP transfer behavior estimated from CAMS measurements at Unit-2 obtained on March 14th and 15th

Continuing on from 4.3.7, the Unit-2/Issue-12 was examined based on the CAMS readings (Attachment 2-11).

Attachment 2-10 estimated the timings of core damage and core melt, and the integrity of the reactor vessel and the PCV based on the CAMS data changes in the D/W and S/C. In the current examination, the FP release behavior in the PCV during the core damage and core melt processes was examined by evaluating quantitatively the CAMS readings and the amount of FPs released at each transition event.

Consequently, the CAMS readings in the S/C could be reproduced by considering

releases of iodine and cesium upon core damage, and the CAMS readings in the D/W could be reproduced by assuming the direct release of radioactive materials to the D/W after March 15th onward. Thus, the accident progression scenario of Unit-2 presented in Attachment 2-10 can be concluded to be consistent with the quantitative evaluation results of CAMS data.

４．３．９． SRV operation states after the core damage

The operation states of the main steam safety relief valves (SRVs) after the core damage (Common/Issue-1) were examined (refer to Unit-1, Attachment 1-3; Unit-2, Attachment 2-12; Unit-3, Attachment 3-4).

The SRV operation states of each unit were examined based on the operation records, the records of responses to the accident, the accident progression analysis and the SRV working feasibilities from the design viewpoint. The examination concluded the following three items were of importance: securing a reliable power supply to the solenoid valve of the nitrogen gas feed system for the SRVs; securing the nitrogen gas feed pressure to the SRVs; and implementing measures to lower the possibility of nitrogen gas leaks from the seals of the solenoid and other equipment.

４．３．１０． Water level and temperature changes in the Unit-2 suppression chamber

The examination into the integrity of the S/C of Unit-2 was continued from 4.3.5 (Attachment 2-13).

Attachment 2-8 indicated the possibility that a leakage hole had been present in the lower part of the S/C. This examination was advanced, namely, by taking into account the correlation of the S/C water level changes and the S/C thermometer readings; the S/C water level changes were quantitatively estimated based on the data measured to date, and the scenario was clarified that detailed the impacts of the S/C water level changes on the changes of the thermometer readings. Furthermore, the area of the leakage hole, which had been estimated to be at the S/C bottom area, was determined by reproducing the scenario through a sensitivity analysis. The elevation of the leak hole was also estimated.

The examination indicated it was likely that the leakage hole had been about 9 cm^2 in area and located below O.P.512mm in elevation.

４．３．１１． Estimation of Unit-2 reactor water levels based on water level indicator readings behavior

Water levels and other reactor conditions (Common/Issue-3) and the amount of water injection (Common/Issue-2) were examined for Unit-2, the same as done for Unit-1, based

on the behavior of water level indicator readings, which are considered to have become defective eventually (Attachment 2-14).

Possible scenarios of actual reactor water level changes were estimated based on measured values of plant parameters including water level indicator readings over the night of March 14th, 2011, when the core damage and core melt had developed at Unit-2, the timing having been monitored to date. The reactor water level change scenario estimated was then used to estimate the ranges of reactor water levels by changing, for instance, the amount of water injection depending on reactor pressures.

The results showed that the reactor water level had been below the bottom of active fuel (BAF) before about 22:40 from the reactor depressurization at about 18:00. On the other hand, the reactor pressure increase noticed from about 20:30 to 21:20 could have been caused by molten debris falling to the lower plenum or other reasons. But no clear scenario is yet available to explain this pressure increase, because the pressure increase observed was a slow development in the situation of the reactor water level being below BAF. To sum up, the results of this study are considered to suggest a scenario in which the reactor level changed at a low level and these results will be provided for further examination into the accident progression.

４．３．１２． Estimation of the cause of the lack of high dose rate observed in the reactor auxiliary cooling water system of Unit-2

In relation to Unit-1/Issue-9, the reason for the high dose rate observed around the reactor auxiliary cooling water (RCW) system in Unit-1 is presumed to be that the RCW piping was damaged by fuel debris that fell to the bottom of the containment vessel, causing radioactive materials to diffuse into the system (Attachment 1-9).

On the other hand, the high dose rate around the RCW system of Unit-2 has not been confirmed as in Unit-1. In Unit-2, considering that a part of the fuel assembly has fallen from the reactor pressure vessel to the bottom of the containment vessel, regardless of the similarity with Unit-1 also in which the RCW is not designed to be automatically isolated in the event of power loss, the situation of the RCW systems of Units 1 and 2 is different, and the factors that led to this difference are discussed (Attachment 2-15).

Based on the results of the investigation inside the containment vessel of Unit-2, it is estimated that contamination did not spread into the RCW system of Unit-2 because the RCW system was not damaged by the materials that accumulated at the bottom of the containment vessel.

４．３．１３． Containment pressure drop in Unit-2 in the morning of March 15th

The D/W pressure in Unit-2 remained above 0.7 MPa[abs] from around 23:30 on March 14 to 7:20 on March 15, after which measurements were temporarily interrupted and dropped to 0.155 MPa[abs] when measurements resumed at 11:20 on March 15. The spread of soil contamination in the northwest direction from the power plant was considered to be possibly affected by the release from Unit-2, and the factors of this large pressure drop, which may be related to the gas-phase leak from the containment vessel (Unit-2/Issue-11), were examined (Attachment 2-16).

The possibility of each scenario was examined, assuming the possibility of depressurization due to vapor phase leakage from the containment vessel and the possibility of depressurization due to condensation of water vapor in the containment vessel. Considering the depressurization by the gas phase leakage from the containment vessel, it is necessary to consider that there was a large-scale leakage besides the top head flange due to thermal damage. Regarding the increase and decrease of D/W pressure after around 12:00 on the 15th, the current relatively high tightness of the containment vessel in Unit-2, and the contamination in the building other than for the operation floor being relatively small, it is difficult to explain the observed facts with consistency. On the other hand, it is easier to explain the observed facts with consistency if it is considered that the water level in the torus room rose above the level of the S/C pool due to the inflow of water from other buildings, etc., and the condensation of water vapor in the containment vessel contributed to depressurization in addition to gas phase leakage due to factors such as accelerated cooling of the gas phase part of the S/C, which led to the condensation of water vapor.

４．３．１４． Behavior of S/C pressure gauge at Unit-2 after 21:00 on March 14th

Among the containment pressure gauges used in Unit-2, the S/C pressure gauge for accident management (AM) was connected to the battery and power was restored at 03:00 on March 13th, but it indicated a low value that deviated from the downscale (DS) or D/W pressure and likely did not reflect the actual pressure. S/C pressure gauges are important instruments for understanding plant behavior during accidents, and issues related to their abnormal behavior (Unit-2/Issue-3) were discussed (Attachment 2-17).

Three factors were identified that could possibly cause the abnormal behavior of the S/C pressure gauge for AM: ① mechanical factors, ② factors related to the principle of measurement, and ③ electrical factors; and the possibility of each factor was examined. Regarding ①, the possibility that the pressure gauge itself was mechanically damaged by the earthquake, the explosion of another unit, or the impact of the tsunami was estimated to be low. Regarding ②, it was estimated to be unlikely that the pressure gauges showed

abnormal behavior due to the decrease or loss of water in the condensate tank piping, which is the source of pressure detection. Regarding ③, it was estimated to be quite possible that the submersion of the pressure gauges caused seawater to penetrate into the inside of the gauges, resulting in electrical abnormalities at the terminals, which could cause a drop in the DS or indicated value.

Based on the results of the above study, it was estimated that electrical factors caused by the submergence of the pressure gauges were the main cause of the abnormal behavior.

４．３．１５．Evaluation method for the core damage ratio of the Mark-I containment vessel

In Unit-2, the containment atmosphere monitoring system (CAMS) was able to start measuring dose rates in the containment vessel (D/W and S/C) before the core damage, which enabled the dose rate changes in the containment vessel to be captured before and after the core damage. The relationship between these measurements and the accident progression has been assigned as Unit-2/Issue-12.

Although it was estimated that Unit-2, as well as Units-1 and -3, suffered almost 100% core damage, the assessed core damage ratio published on April 27th, 2011 was 35%, and the degree of core damage estimated from the accident progression and the core damage ratio assessed from the CAMS measurements prepared prior to the accident appear to be significantly different. Also, looking at the D/W and S/C CAMS measurements the tendency for the S/C side to have a value about one order of magnitude lower is also different from the evaluation results prepared prior to the accident.

Since CAMS measurements are important data for understanding the accident progress, these factors were examined (Attachment 2-18).

The results of the study showed that the core damage ratio evaluated from the CAMS measurements prepared before the accident tended to underestimate the core damage ratio due to the fact that the geometry of the Mark-I containment vessel and the location of the CAMS dosimeters did not properly reflect the effects of the geometry.

４．３．１６．Examinations into other matters

Examination results of other issues derived in "4.2. Issues derived from the comparison between measured information of Unit-2 and analyses" will be added to this section as soon as they become available.

４．４．Summary of Unit-2 examinations

Some of the issues derived from the comparison between MAAP analysis results and measured information were examined, and rational interpretations for phenomena have

been obtained for some issues as follows.

✓ Water flow rate of the RCIC system increased as the RCIC design provided for fully opening the steam regulator valve once the DC power supply was lost as described in "4.3.1. RCIC operation behavior without DC power supply."

✓ An action had been taken to isolate the RHR system before the valves became inoperable due to loss of the power supply by the tsunami and, therefore, there were no direct correlations between the RHR system configuration and the reactor vessel or PCV behavior as described in "4.3.2. RHR system configuration after tsunami arrival."

✓ One mechanism which can explain the observed PCV pressure decrease after RCIC shutdown is obtained as described in "4.3.3. Containment vessel pressure decrease after RCIC system shutdown."

✓ A reasonable scenario for the accident progression after the reactor depressurization could be found by "4.3.7 Rapid increase of CAMS readings observed on March 15th" and "4.3.8 FP transfer behavior estimated from CAMS readings at Unit-2 on March 14th and 15th."

Hereafter, this latest information will be considered as inputs to the analysis for increasing reliability.

５．　Examinations into the accident progression at Unit-3

５．１．　Approach for evaluation

The analysis results of MAAP (Attachment 1) have been mainly used to examine the accident progression, excluding the effects of the earthquake and tsunami, of Fukushima Daiichi NPS Unit-3 (hereinafter referred to as "Unit-3"). Figure 5.2.1 shows the reactor water level changes, while Figure 5.2.2 shows the reactor pressure changes and Figure 5.2.3 shows the PCV pressure changes. However, the MAAP results cannot perfectly reproduce the actual accident progression because of the uncertainties in its analysis conditions, analytical models, and consequently the results obtained. In this report, therefore, the following steps were taken for examinations: first, discrepant points were identified as issues between the MAAP results done in the past (see separate Volume 1 for the results by MAAP4, and Attachment 3 for the results by MAAP5) and actually observed measurements; and then, the issues identified were examined one-by-one. Section 5.2 explains in chronological order the issues extracted and Attachment 2 describes each issue individually in parallel.

５．２．　Issues derived from the comparison between measured information of Unit-3 and analyses

５．２．１．　From the earthquake to tsunami arrival

Unit-3 was moving towards cold shutdown after the earthquake by controlling the reactor pressure and water level, etc. through SRV and RCIC operations. But at 15:38 on March 11th all its AC power supplies were lost due to the tsunami. The DC power supply could maintain its function only until the batteries were depleted due to the loss of the AC power supply. This continued availability of DC power source at Unit-3 was a very significant difference from the conditions at Unit-1 and Unit-2.

５．２．２．　From the tsunami arrival to RCIC shutdown

The RCIC had stopped automatically at 15:25 on March 11th due to the high reactor water level before the tsunami arrived. As the DC power supply was available at Unit-3, the RCIC was manually started at 16:03 and water injection was started at 16:16. The reactor pressure and water level were thus controlled by the SRV and RCIC. Operators maintained reactor water levels by adjusting the flow rate set of flow controllers to allow gradual reactor water level changes. This was done using the line configuration where water would pass through both the reactor injection and test lines so that part of the water returned to the condensate storage tank (CST) (water source of the RCIC), which would prevent automatic shutdown due to high reactor water levels and avoid battery depletion due to RCIC

re-activation and valve operations for line switching, and also ensure stable reactor water levels.

During this period the D/W pressure was increasing but the analysis results provided lower values of increase and could not reproduce the pressure behavior observed until about 22:00 on March 12th. (The PCV pressure increased much more than the value predicted from the decay heat until about 12:00 on March 12th when the high pressure coolant injection (HPCI) system started to operate. And thereafter the measured pressure decreases were big while the analysis results showed continued pressure increase.) This discrepancy is being examined in "Examinations into the impacts of thermal stratification in the suppression chamber water on the PCV pressures, etc." [10], in which the possibilities of the following phenomena are being examined.

- The RCIC turbine exhaust steam heated up the S/C pool water near the turbine exhaust pipe exit.
- The high temperature pool water was dispersed horizontally on the pool surface, thus producing thermal stratification in the pool water.
- This stratification caused a bigger PCV pressure increase than the analysis (which assumed a uniform temperature increase of the pool water).

Results of this examination need to be provided to the continuing examination into the PCV pressure behavior until about 22:00 on March 12th (Unit-3/Issue-3: Examination ongoing (Attachment 3-7)).

The reactor recirculation pump (PLR) is equipped with a mechanical seal as its shaft seal. During normal reactor operation, the seal water supplied to this shaft seal by the control rod drive (CRD) pump prevents the reactor water from leaking. Upon loss of off-site power, the CRD pump stops and the seal water is lost. At high pressure, reactor water leaks out to the equipment drain sump in the D/W via the shaft seal. The reactor water leak through the PLR mechanical seal has a significant impact on the reactor because it becomes difficult to maintain the reactor water level; and this leak through the seal also has a significant impact on the PCV because the high temperature leaked water acts as an energy supply causing pressure to increase. As a matter of fact, it is known at Unit-3 that the D/W pressure was higher than the S/C pressure. A possibility is noted that there could have been a PCV pressure increase due to seal water leaks, in addition to the above-mentioned thermal stratification. But it remains unclear how much water did leak through the PLR seal mechanism. Further examination is needed (Common/Issue-4).

The RCIC stopped automatically at 11:36 on March 12th and thereafter its status of shutdown was confirmed on-site as well. There are two steam-driven water injection lines: RCIC and HPCI. The RCIC rated flow is designed to compensate for the water inventory

evaporating due to decay heat, at 15 minutes after the reactor scram, while the designed rated flow of the HPCI is about 10 times larger so that the HPCI could provide cooling water to the reactor at a loss-of-coolant-accident (LOCA) due to, for example, a pipe break. Therefore, when the decay heat decreases about one day after the scram, the RCIC with less rated flow is more appropriate to compensate for the coolant lost by evaporation and to maintain the reactor water level. But the RCIC failed to restart operation. It was found upon an on-site check that the latch for the trip mechanism of the RCIC turbine trip throttle valve had been detached. The high turbine exhaust pressure is considered to have tripped the throttle valve, tripping the turbine (Unit-3/Issue-1: Examination completed (Attachment 3-5)).

５．２．３． From the RCIC shutdown to HPCI shutdown

The RCIC stopped automatically at 11:36 on March 12th and the reactor water level started to decrease. While preparations were being made to restart the RCIC operation, the HPCI started up automatically at 12:35 when the water level was lowered to the low reactor water level (L-2). In addition, the diesel-driven fire pump (DDFP) was started up at 12:06 on March 12th for the S/C spray using the filtrate tank as its water source, since the S/C pressure had risen due to the exhaust steam from the SRV and RCIC.

Operators controlled the HPCI water flows by flow controllers using also, like the RCIC, the line configuration where water would pass through both the reactor injection and test lines so that part of the water was returned to the CST (water source of the HPCI), which would prevent automatic shutdown due to high reactor water levels and avoid battery depletion due to re-activation, and also ensure stable reactor water levels. After the HPCI was started up, the reactor pressure started to decrease because the steam was consumed by the driving turbine. Concerning this pressure decrease behavior, it has been known that the measured behavior of reactor pressure could be well reproduced in the analysis by simulating the flow control.

The HPCI has a bigger flow capacity than that of the RCIC and consumes more reactor steam. As a result, the reactor pressure decreased by operating the HPCI and reached about 1 MPa[abs] at about 19:00 on March 12th. This reduced reactor pressure lowered the HPCI turbine rotation speed and the status continued that it could stop anytime.

In addition, monitoring of the reactor water level became impossible at 20:36 on March 12th due to loss of the power supply for the reactor level indicators.

The reactor pressure started to decrease at about 02:00 on March 13th, which had been stable at about 1 MPa[abs], and it became lower than the allowable HPCI operation limit and reached a situation in which the HPCI could stop anytime. It was manually shut down at 02:42, therefore, in consideration of the preparation underway for reactor water injection

using the DDFP.

5.2.4. From the HPCI shutdown to reactor depressurization

To prepare for water injection to the reactor, the DDFP was switched over from the S/C spray mode to the reactor water injection mode, and the main control room operators were notified of switchover completion at 03:05 on March 13th shortly after the HPCI shutdown. The reactor pressure reversed to an increasing trend after the HPCI had been shut down but the SRV operation attempt failed after all. The reactor pressure further increased and exceeded the DDFP discharge head, thus disabling the alternative water injection. An attempt was made on-site to supply nitrogen gas to drive the SRV via the supply line, but it failed, because the valve on the supply line was an air-driven type and it could not be manually operated due to structural limitations. Operation attempts also failed to start up the HPCI and RCIC: the HPCI failed due to battery depletion, and the RCIC failed because the turbine trip throttle valve was closed again by trip mechanism of the valve.

A nitrogen gas accumulator was in position, which could operate the SRV, even when its nitrogen gas cylinders or nitrogen gas supplied from the atmospheric control (AC) system were not available to drive the SRV to relieve pressure or to open the SRV remotely. The nitrogen gas cylinders and the AC system were designed to be isolated upon loss of AC power supply. Therefore, the SRV could have been operated by the residual pressure of the accumulator or piping. Insufficient driving gas pressure under the high back pressure (PCV pressure) condition or insufficient voltage to energize the solenoid valve could have been the reasons for the inoperable SRV. But details are unknown. Relevance of SRV behavior and its background to the accident progression was examined and the results are presented in Attachments 3-4 and 3-13 (Common/Issue-1: Examination completed (Attachments 3-4, 3-13)).

In the MAAP analysis, which TEPCO published in March 2012, the amount of HPCI water injection had been adjusted so that the measured values of the wide range water level indicators could be simulated, in which the measured values until 20:36 on March 12th had not been corrected for the reactor pressure and PCV pressure. The effect of overestimating the amount of HPCI water injection on the accident progression needs to be examined based on the corrected water levels (Unit-3/Issue-4: Examination completed (Attachments 3, 3-3).

The measurement of reactor water level was interrupted at 20:36 on March 12th due to loss of power supply. When it was resumed upon recovery of power supply at about 04:00 on March 13th, the fuel range water level indicators showed about the top of active fuel (TAF) -2m.

According to the analysis, the reactor water level decreased after the HPCI shutdown and the core was uncovered upon rapid reactor depressurization at about 09:00 on March 13 and core damage started. However, the water level was kept above TAF, overestimating the reactor water levels (Unit-3/Issue-5: Examination completed (Attachments 3, 3-3, 3-4, 3-9)).

As mentioned earlier, a high possibility has been shown that the reactor water level had started to decrease earlier than the HPCI manual shutdown, judging from the overestimated amount of water injection during the HPCI when in operation and the HPCI having been operating under low reactor pressure conditions. Thus, examinations are needed into the reactor level behavior after the HPCI manual shutdown (Attachment 3-3, Attachment 3-4).

In the meantime, the S/C spray was resumed by switching over the DDFP from the reactor water injection mode at 05:08 on March 13th in order to prevent pressure increases of the D/W and S/C. At 07:39 the spray lines were switched over from S/C to D/W and the S/C spray was terminated at 07:43.

At 08:41 on March 13th, the large S/C vent valve (air-operated) was opened and the configuration of the venting line was completed except for the rupture disc.

At about 08:40 through 09:10 on March 13th, the DDFP stopped the D/W spray and switched to water injection to the reactor.

The reactor pressure, in the meantime, reversed to increase by the HPCI manual shutdown at 02:42 on March 13th and reached about 7 MPa[abs] at about 04:30, and stayed thereafter for about 5 hours at about 7.0 to 7.3 MPa[abs]. When battery connection work was ongoing for depressurization, the reactor pressure decreased abruptly at about 09:00 on March 13th down to below 1 MPa[abs].

The mechanism of this rapid reactor depressurization has been found to be not by manual SRV operations, but likely by the automatic depressurization system (ADS) activation (Unit-3/Issue-6: Examination completed (Attachments 3-3, 3-4)). The study has shown that the core damage started before the reactor forced depressurization. This hints that the PCV pressure increase before reactor forced depressurization was due to hydrogen released to the PCV from reactor.

5．2．5．　From the reactor depressurization to reactor building explosion

Following the rapid reactor depressurization, fire engines started freshwater injection at 09:25 through 12:20 on March 13th and later at 13:12 fire engines started seawater injection. The DDFP was also being operated in parallel, but water injection was considered not to be working due to the pressure balance relation between the pump discharge pressure and reactor pressure.

Because of the rapid reactor depressurization, the PCV pressure increased, the S/C

pressure exceeded the rupture disc working pressure and the D/W pressure was confirmed at 09:24 on March 13th to have decreased. This led to the conclusion that the PCV had been vented.

The reactor water level indicators showed hunting oscillatory behavior after the rapid depressurization at about 09:00 on March 13th and a certain constant level after 12:00 regardless of the amount of water injection. It can be understood that the correct water level could not be shown due to water evaporation in the water level instrumentation tube. However, the water level showed at least the pressure difference between the piping on the reference water level side and reactor side. It may provide some meaningful information on the accident progression (Unit-3/Issue-2: Examination ongoing (Attachment 3-9)).

In the analysis, the reactor water level decreased following the HPCI shutdown at 02:42 on March 13th, the core uncovers upon rapid depressurization at about 09:00 and core damage started. A large amount of hydrogen was produced by water-zirconium reactions when the core became uncovered and fuel cladding temperatures started to rise.

The core damage process is greatly influenced by the extent to which the water-zirconium reactions occurred due to the water injected by the fire engines. In reality, part of the injected water is considered to have flowed into systems and equipment other than the reactor itself. In the analysis, the water injected into the reactor was assumed to be considerably lower than the amount of water injected by the fire engines in consideration of reproducibility of the PCV pressure. This is equivalent to assuming that the amount of water injected was not sufficient to cover the core. The amount of water injected to the reactor is important input to examine the accident progression. The actual amount of water injected needs to be examined (Common/Issue-2).

According to the chart records, the reactor pressure after the rapid depressurization at about 09:00 on March 13th showed a sharp rise to several MPa[abs] first at about 10:00 and again at 12:00 and then there was a gradual decrease.

This pressure behavior may have some correlation with the SRV opening/closing operation for connecting batteries to the SRV for opening. But the pressure rise is steeper for the value due to steam generation. The pressure increase can be confirmed to be considerably faster when compared with the pressure increase upon HPCI shutdown. Such pressure behavior may have connections with the core damage process or hydrogen formation, but its details remain unknown and are left for further examination (Unit-3/Issue-7: Examination ongoing (Attachments 3-3, 3-4)).

In the analysis, gas leakage from the reactor vessel was not assumed. But the possibility of gas leakage from the reactor vessel due to reactor temperature rise caused by fuel overheating and melting needs to be examined (Unit-3/Issue-9: Examination ongoing

(Attachments 3-11, 3-12)).

The D/W pressure thereafter repeated up and down swings in response to steam generation by water injection, hydrogen formation, venting operation, etc.

In the analysis, no gas leakage from the PCV was assumed. But the following facts indicate the possibility of gas leakage from the PCV: the hydrogen explosion in the Unit-3 reactor building; steam discharge that was observed above the building even after the temperature of the spent fuel storage pool became sufficiently low; the D/W pressure showed no increase from the atmospheric pressure after March 21st and no response was confirmed when nitrogen gas injection into the PCV was started on July 14th. No direct evidence is known as to when and from where specifically the gas leakage occurred from the PCV, leaving another issue for examination (Unit-3/Issue-10: Examination completed (Attachments 3-8, 3-11, 3-12)).

At 11:01 on March 14th, hydrogen exploded in the reactor building, damaging the whole top floor and the southern and northern walls of the floor next to the top floor.

It is considered that the hydrogen that was produced mainly by water-zirconium reactions leaked, together with steam, to the reactor building eventually and caused the explosion. Its leak paths and amount, mode of explosion, ignition source, etc. are unknown and left for examination (Common/Issue-11).

５．２．６． From the reactor building explosion to late March

Water injection by fire engines was continued after being interrupted at the time of the explosion at 11:01 on March 14th in the Unit-3 reactor building.

Water injection was considered to have been resumed after the explosion at about 16:30 on March 14th. However, it has been concluded to have been about one hour earlier at 15:30 based on the latest investigation of reviewing chronological information including the TV conference records, etc. It was also newly found that water injection to Unit-3 had been interrupted again at 21:14 on March 14th in order to secure water injection to Unit-2 and that it had been resumed again at 02:30 on March 15th. The impacts of water injection by fire engines need examination, including the revised chronological sequence above (Unit-3/Issue-12).

Efforts were continuing to keep the PCV vent valve open since it had been opened at about 09:00 on March 13th when the rupture disc opened upon reactor depressurization. But it was closed thereafter due to failure, etc. of the temporary generator for power supply and the opening operation of PCV vent valve had to be repeated until March 20th to keep it open.

Unclear features remain concerning the D/W pressure: its changes when no PCV venting was recorded; or no pressure decrease when the PCV vent valve was confirmed to have

been opened at 06:10 on March 14th. Details need to be examined (Unit-3/Issue-8: Examination completed (Attachments 3-8, 3-11, 3-12)).

Concerning the FP release behavior from the PCV on the occasion of PCV venting, there are many unknown matters, leaving issues to be examined (Common/Issue-8).

Steam was observed on several occasions, which might have leaked from the PCV: a large amount of steam rising above the building; black smoke rising up at about 16:00 on March 21st; or steam rising up from the west side of the building and above the building on March 29th. They may provide some clues on locating the leak through examination (Unit-3/Issue-11: Examination completed (Attachments 3-8, 3-11, 3-12)).

According to the analysis of FP release, noble gases were released from the reactor vessel to the S/C and almost 100% were released when vented. On the other hand, the fraction of released cesium iodide was about 0.1% and most of it remained in the S/C.

The MAAP4 analysis of core conditions showed that the core remained in situ and the reactor vessel was not damaged, although part of the fuel melted and formed a pool. This may be due to such reasons as that water injection by RCIC and HPCI continued rather stably at the beginning, and that the time delay from HPCI shutdown to water injection commencement was less for Unit-3 than for Unit-1. But the results obtained by MAAP5 analysis were different, in which the possibility of overestimation of water injection by HPCI had been considered, i.e., the reactor water level decreased significantly before the reactor depressurization at 09:00 on March 13th, and this caused the core damage and melting due to water-zirconium reactions, followed by damage of the reactor vessel. Thus, the MAAP5 analysis better reproduced the actual Unit-3 situation. However, it is not necessarily understood that the fraction of fallen fuel and other items were well reproduced, when the results of in-PCV investigations and muon measurements in July 2017 are taken into account.

In any case, there are many unknown matters concerning the location of debris, the final status of accident progression. As these are important input to future decommissioning steps, further examinations remain based on the outcomes of the investigative research and development projects of the PCV and reactor pressure vessel, and other relevant projects (Common/Issue-10: Examination ongoing (Main report chapter 7, Attachments 4, 5)).

5．2．7． Examinations into other matters

As noted many times, MAAP analysis has uncertainties in its analysis conditions, analytical models, and consequently its results for the accident progression. In particular, the amount of FP release is strongly affected by these uncertainties. The results should be understood as being simple reference information.

The D/W pressure of Unit-3 decreased to atmospheric pressure in the early stage, about March 21st, differing in behavior from Unit-1 and Unit-2 where positive D/W pressure was maintained. In addition, no pressure increase was observed at Unit-3 when nitrogen gas injection was started (July 14th), which had been observed at Unit-1 and Unit-2. These observations may indicate that gas leak from the Unit-3 PCV was on a larger scale than that of Unit-1 or Unit-2 (Unit-3/Issue-10).

One of the possible causes of gas leakage from the PCV could be a melt-liner attack, in which molten fuel creates an opening upon its contact with the PCV liner. Therefore, examinations are needed to check for this by on-site observation, etc. (Common/Issue-5). But there was an observation which indicated Unit-3 had a higher water level in the PCV, calculated from the measured S/C pressure, than that of Unit-1 and Unit-2 and that a certain amount of water remained in the D/W. This is not consistent with the theory that a large opening was created by a melt-liner attack.

MAAP has been used in analyzing the accident progression for about a week at the maximum after the earthquake. This is because the uncertainties in analysis results become larger when covering longer time spans, and the result reliabilities decrease accordingly. On the other hand, the FP released from Fukushima Daiichi NPS on around March 20th and 21st caused dose rate increase in Kanto district, as the FPs would be affected by the wind direction, and the authorities recommended the public cut their consumption of tap water due to concern about increased iodine concentrations and their FP intake. Thus, there is a need to examine the plant behavior long time after the earthquake, which is difficult to do (Common/Issue-9: Examination completed (Attachment 3-6)).

The issues derived above are shown in Figures 5.2.1 to 5.2.3. They are also described in parallel in Attachment 2. The examination results obtained so far from among the issues extracted as unsolved are compiled in the Attachments and they are summarized briefly in the following section 5.3.
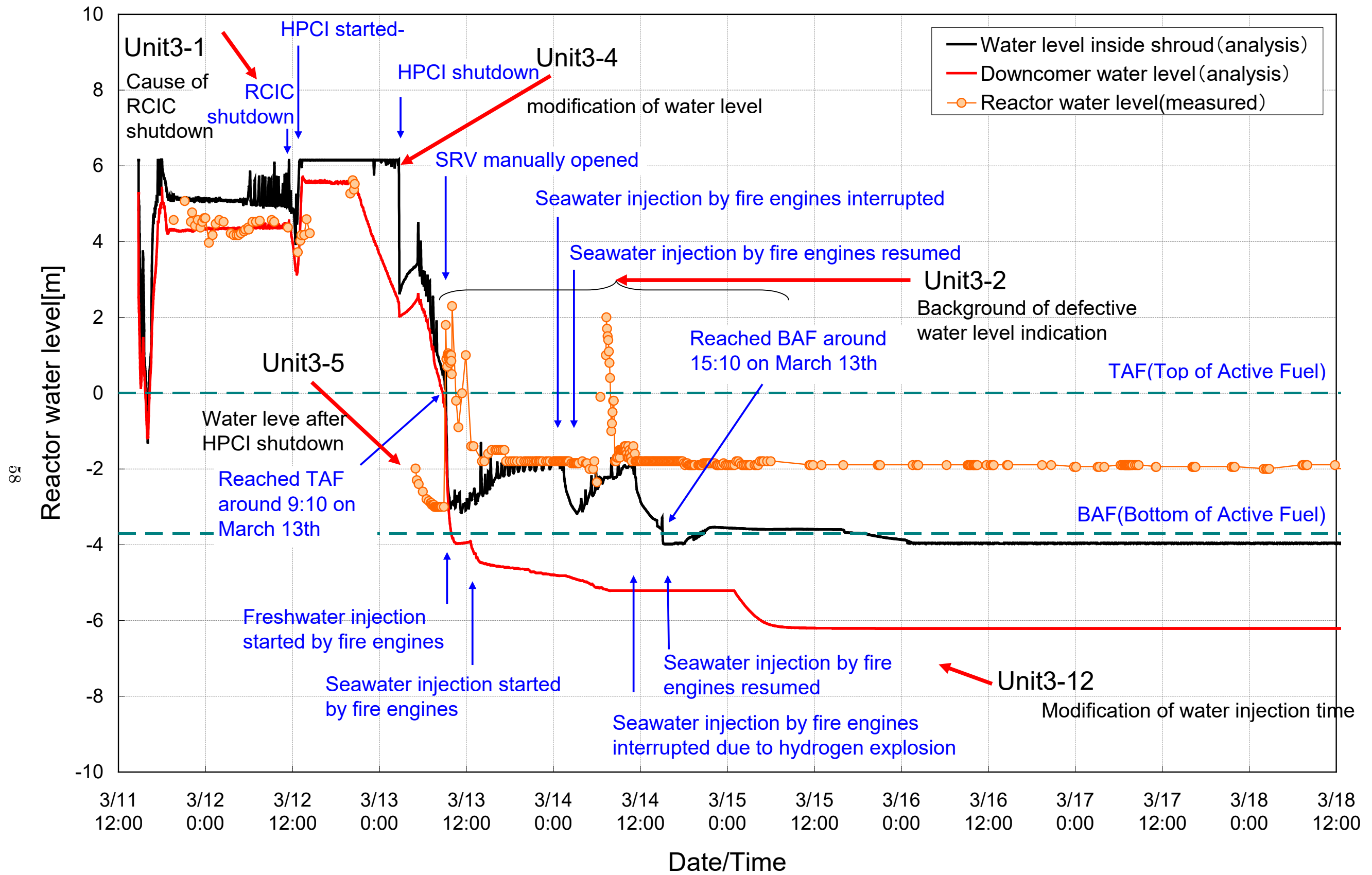
Figure 5.2.1 Issues derived from the reactor water level changes at Unit-3
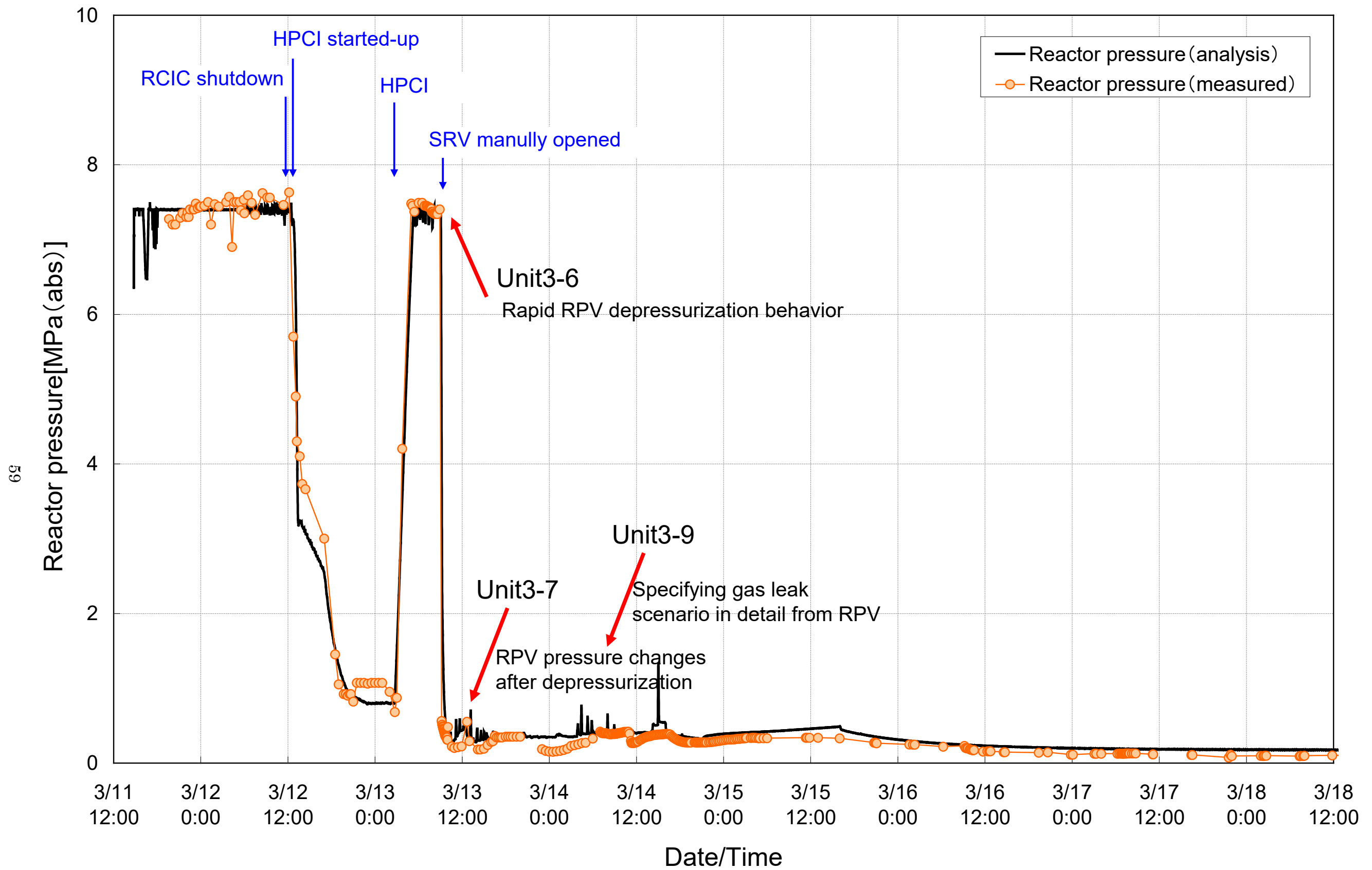
Figure 5.2.2. Issues derived from the reactor pressure changes at Unit-3
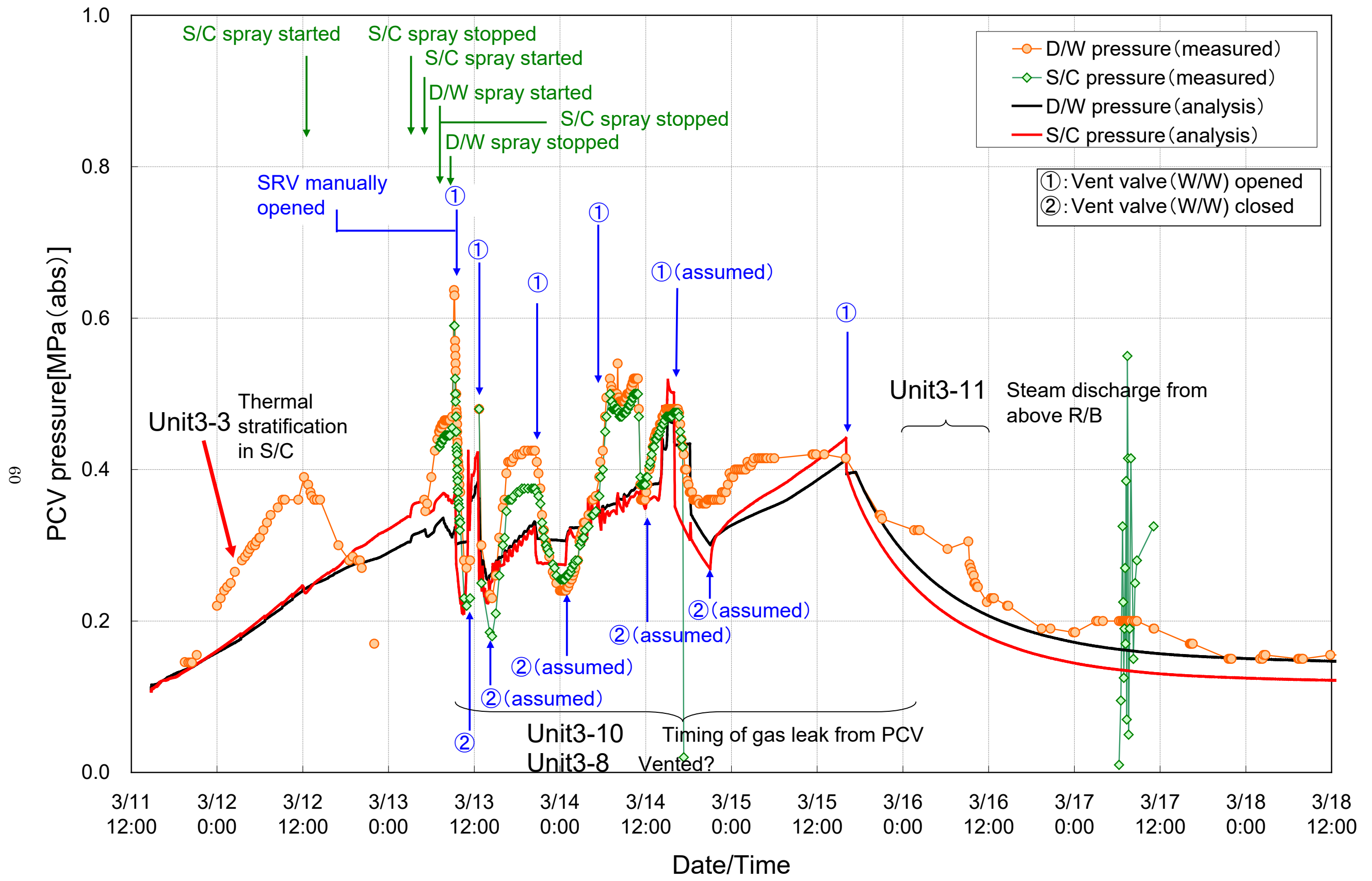
Figure 5.2.3. Issues derived from the PCV pressure changes at Unit-3

５．３． Evaluation results of the issues derived for Unit-3

５．３．１． Depressurization behavior at about 09:00 on March 13th

The behavior of water injection by the HPCI from the night of March 12th and of the rapid pressure decrease of Unit-3 at about 09:00 on March 13th was examined (Unit-3/Issues-6, 4, 5) (Attachment 3-3).

The results confirmed for sure that the rapid pressure decrease of Unit-3, as reported earlier [3], had not been initiated by manual SRV opening. It was also found that the pressure decrease had been too rapid to have been caused by manual opening of one or two SRVs.

There is a theory that this rapid pressure decrease could be due to the reactor vessel damage on the grounds that no SRV operation had been done and the decrease had been very rapid. However, the examination of PCV pressure behavior and automatic startup logic circuits of the automatic depressurization system (ADS) has shown that the probable cause of this rapid pressure decrease was not due to the damaged reactor vessel but because the ADS worked.

In the current analysis, the HPCI was assumed to have continued its water injection to the reactor until it was manually shut down. It has become clear, however, while the pressure decrease starting during the night of March 12th was being examined, that this assumption was inconsistent with the measured reactor water levels. This indicates a high possibility that water injection to the reactor had not been sufficiently done before the operators shut the HPCI down manually. This finding of earlier decrease of reactor water level than having been estimated means faster accident progression, which further indicates larger damage of the reactor vessel. The core conditions need to be examined in consideration of these findings.

The reactor pressure changes (Unit-3/Issue-7) after this period were also examined in parallel (Attachment 3-4).

The latest MAAP5.0.1 analysis shown in Attachment 3 considered there was insufficient water injection before manual HPCI shutdown. Thus, it was shown that fuel melting had started by the time when the reactor water level had dropped to BAF at about 07:30 on March 12th. This finding is consistent with the examination results given in 5.3.2.

５．３．２． Possible scenario of fuel melting and detection of neutrons

The in-reactor behavior of Unit-3 during the time from the night of March 12th was examined in connection with Unit-3/Issue-5 (Figure 5-2-1) concerning water injection by the HPCI (Attachment 2-7).

Attachment 3-3 has indicated water injection by the Unit-3 HPCI became insufficient during its operation. With the reactor water level lowering, heat removal decreased due to reduced steam generation, thus eventually accelerating the water-zirconium reactions and releasing a large amount of energy. It was indicated that this might have caused fuel melting. In the early morning of March 13th, when the reactor water level was decreasing, neutrons were detected a few times, although the intensity was at the minimum detection level. This indicated, as in the case of Unit-2, the high possibility of detecting neutrons emitted by spontaneous fissions by some actinides of high spontaneous fission probability such as curium, which had been released from the core during fuel melting and leaked from the reactor building possibly with other actinides such as uranium and plutonium.

Continued examination is needed, since the leak paths of actinides have not been identified.

５．３．３． Possible cause of RCIC shutdown

An examination was conducted concerning the possible causes of why the RCIC had ceased operation on March 12 (Unit-3/Issue-1). Possible causes were examined by reviewing design conditions which might lead to the RCIC being shut down and by referring to observed measurements and maneuvering of operators after the operation ceased (Attachment 3-5).

It turned out to be highly possible, based on the maneuvering of operators for resetting in the wake of RCIC failure to start up, that the automatic trip logic had ceased RCIC operation; however, it is also evident that no observed measurements were big enough to trigger any trip logics. The observed measurement closest to the automatic trip signal was that of RCIC turbine exhaust pressure. Examination of its pressure changes over a long time span and the RCIC behavior after restart has indicated the possibility that the RCIC turbine exhaust pressure had exceeded the preset automatic trip level.

５．３．４． Examination into the dose increase observed on around March 20th

One of the common issues Common/Issue-9 was examined concerning the dose increase observed on around March 20th (Attachment 3-6).

The FPs released from the Fukushima Daiichi Nuclear Power Station on around March 20th and 21st were carried by winds and caused a dose increase over parts of the Kanto region. The increase of radioactive iodine concentration led to some local government recommendations to refrain from tap water intake. Examination into the reactor behavior over this time period based on the plant data measured showed a better chance that this

dose increase had not been caused by an FP release in a limited time such as from the accident progression or venting operation, but by the continuing release over a longer time span from the PCV which had lost its confinement integrity.

５．３．５． Causes of the PCV pressure increase at Unit-3 from March 11th to 12th, 2011

The issue of the faster than anticipated PCV pressure increase on March 11th to 12th from the decay heat at Unit-3 was examined (Unit-3/Issue-3) (Attachment 3-7).

The faster than anticipated pressure increase would have come from causes in the D/W or S/C. The D/W pressure increase might have arisen from leaks from the reactor vessel to the D/W, but this possibility would be low because, if large enough leaks to cause the measured PCV pressure increase were assumed in the analysis, the pressure decrease after 12:00 on March 12th by the S/C spray could not be reproduced. On the other hand, the exhaust discharge analysis simulating the paths from the RCIC or the SRV outlets could confirm the tendency of thermal stratification in the S/C. The RCIC exhaust outlet was located in the S/C pool water surface layer, which induced promotion of thermal stratification. Moreover, at Unit-3, the RCIC was kept operating in order to control the amount of water injection into the reactor to prevent itself from being stopped by the reactor water level reaching to the high setpoint. This might have accelerated the formation of thermal stratification. It can be concluded that the thermal stratification in the S/C is likely to have caused the faster PCV pressure increase.

５．３．６． Leaks from the Unit-3 PCV and the large steam release

At Unit-3, the PCV venting was attempted several times in order to decrease its pressure, but after the hydrogen explosion of the reactor building, direct releases were also recognized from the building. Release behavior of radioactive materials is considered to be associated with the operation states of vent valves. In this regard, three issues (Unit-3/Issues-8, -10 and -11) were comprehensively examined (Attachment 3-8).

The examination of the PCV pressure changes, photos taken by the live camera installed at the Fukushima Daiichi NPS and other information revealed the following findings.

- Vent operations at about 09:00 and 12:00 seemed to have been successful.
- But vent operations thereafter were likely not to have been successful.
- The PCV lost its integrity by the morning of March 15th at the latest and the radioactive materials could have been released.
- The leak path configured on this occasion caused the continuous releases from the PCV and the PCV pressure decreased eventually to the atmospheric pressure.

- Therefore, the environmental pollution is likely to have occurred by the direct releases from the PCV, not by venting, as far as March 15th and 16th are concerned.

5．3．7． Estimation of Unit-3 reactor water levels based on water level indicator readings behavior

Water levels and other reactor conditions (Common/Issue-3 and Unit-3/Issue-2) were examined at Unit-3, as done for Unit-1 and Unit-2, based on the behavior of water level indicator readings, which are considered to have become defective eventually (Attachment 3-9).

Plant parameters measured between 04:00 and 14:00 on March 13th were chosen for estimating the reactor water level change behavior. This period covers the time of Unit-3 core damage/core melt progression and ends when the level indicator readings became stabilized. It was estimated that the reactor water level had already dropped to near BAF before the reactor depressurization at about 09:00 on March 13th, the water level had further decreased due to decompression boiling by reactor depressurization and had failed to recover to the core level despite water injection by fire engines and other means thereafter. This estimation is consistent with the accident progression scenario estimated to date in other studies (Attachments 3, 3-3). It will be necessary to further estimate the accident progression.

5．3．8． Examination into Unit-3 vent gas reverse flows to Unit-4

The extent of reverse flows of Unit-3 vent gas was examined, in relation to the hydrogen explosion at Unit-4 (Common/Issue-11) (Attachment 3-10).

The whole line configuration including the vent lines was modelled using design information and a thermal-hydraulic analysis code was used to evaluate the fraction of vent gas which had flowed into Unit-4. The approach was different from the earlier analysis [7]. Plant conditions in Unit-3 PCV as the initial conditions of analysis, such as the amount of hydrogen gas at the time of vent, were derived from the plant parameters measured at that time, not from the results of the accident analysis code.

About 35% of the hydrogen-rich vent gas was estimated to have flowed into the Unit-4 reactor building and the high possibility was reaffirmed that this hydrogen inflow had caused the hydrogen explosion.

5．3．9． Examination of the water level in the pressure suppression chamber of Unit-3

The S/C water level data was collected from 17:15 on March 11th to 20:00 on March 12th

for a limited time period because the DC power supply was still active in Unit-3 after the arrival of the tsunami. This information can be used to estimate the accident progress after the reactor depressurization around 09:00 on March 13th (containment vessel venting, gas phase leakage from the reactor pressure vessel/primary containment vessel, hydrogen explosion, etc., related to Units-3/ Issues 8, 9, and 10) and the cooling status of fuel debris (Attachment 3-11).

Based on the actual measurement of the S/C water level and the change in containment pressure, the S/C water level was examined at around 09:00 on the 13th, when no data were available, and it was estimated that the S/C water level was about 7 m above the S/C bottom, which was higher than the vacuum break valve, due to the S/C spraying, etc. that had been conducted until then. This estimated result suggests that water from the S/C pool may have migrated to the D/W side when the D/W pressure dropped later, from around 20:40 on the 13th, and contributed to cooling the fuel debris that fell from the RPV. This could also be related to the current conditions in the D/W of Unit-3 (high water level and high material pile-up in the pedestal).

５．３．１０． Accident progression after reactor depressurization of Unit-3

Based on the results of the S/C water level estimation in 5.3.9, etc., further study was conducted on the accident progression in Unit-3 (related to Units 3/Issues-8, 9, and 10) from 09:00 on March 13th to 0:00 on March 14th (Attachment 3-12).

(1) The possible ranges of important parameters for accident progression scenarios, such as the gas-phase leak area of the pressure vessel and the number of SRV valves open, were evaluated through analysis. (2) There is a high possibility that gas-phase leakage from the pressure vessel to the D/W occurred at about the same time as the ADS activation. (3) There is a high possibility that gas-phase leakage from the D/W occurred at about 16:40 on the 13th. (4) The depletion of the lower plenum water in the RPV may have affected the D/W depressurization at about 20:40 on the 13th.

５．３．１１． Examination of plant conditions during RCIC operation of Unit-3

RCIC operation of Unit-3 after the arrival of the tsunami was based on adjusting the amount of water injected into the reactor, including utilizing the return line to the CST, the water source, in order to reduce battery consumption due to startup and shutdown caused by changes in reactor water level. During this period, the measured reactor pressure behavior was different from that which would be caused by simple SRV operation, specifically, it was a repeating mix of large and small pressure changes. Plant behavior at

that time was examined with a focus on this reactor pressure behavior (related to Common-1) (Attachment 3-13).

Through confirmation of RCIC operation results and analysis to reproduce reactor pressure behavior, it was confirmed that the reactor pressure behavior during the RCIC system operation period is consistent with the conventional understanding that it reflects steam release via the SRV and the depressurization effect due to water injection from the RCIC system to the RPV, although the opening and closing modes cannot be identified.

５．３．１２． Examinations into other matters

Examination results of other issues derived in "5.2. Issues derived from the comparison between measured information of Unit-3 and analyses" will be added to this section as soon as they become available.

５．４． Summary of Unit-3 examinations

Some of the issues derived from the comparison between MAAP analysis results and measured information have been examined, and rational interpretations for phenomena have been obtained for some issues as follows.

✓ There is a possibility that the HPCI system could not inject sufficient water before being manually stopped as described in "5.3.1. Depressurization behavior at about 09:00 on March 13th."

✓ There is a possibility that the reactor pressure vessel depressurization was caused by operation of the ADS function of the SRV as described in "5.3.1. Depressurization behavior at about 09:00 on March 13th."

✓ The latest MAAP5.0.1 analysis shown in Attachment 3 considered there was insufficient water injection before manual HPCI shutdown. It was shown, as the analysis results, that fuel melting had started by the time when the reactor was depressurized, and core melt progression was more severe than that of previous analyses.

Hereafter, this latest information will be considered as input to the analysis for increasing reliability.

6. Sample analysis to determine accident status

Samples of radioactive particulates collected inside and outside the reactor buildings of Fukushima Daiichi Nuclear Power Station Units 1-3 have included particles containing uranium. These particles are considered to have been produced as the accident progressed, with the fuel overheating, fuel and structural materials reacting, and melting. Therefore, the composition and microstructure of the particles may retain information about the environment surrounding the particles at the time of their formation, information that is useful for understanding the accident progression. In addition, the composition and structure of the particles are considered to be different depending on the cooling conditions leading up to particle formation, and may contain information useful for understanding the properties of the fuel debris distributed in the pressure vessel/containment vessel.

Figure 6.1 shows formation and migration mechanisms of radioactive particles and fuel debris as the accident progresses. It is important to obtain information on the conditions at the time of the accident and the current properties of the fuel debris from the analysis of radioactive particulates spread inside and outside the reactor buildings.

The analysis results of the on-site samples are published each time they are analyzed. In this report, information that leads to a better understanding of the accident progression and the properties of the fuel debris is organized as Attachment 5, which is related to Common-10 "Status of Core Damage and Debris Location."
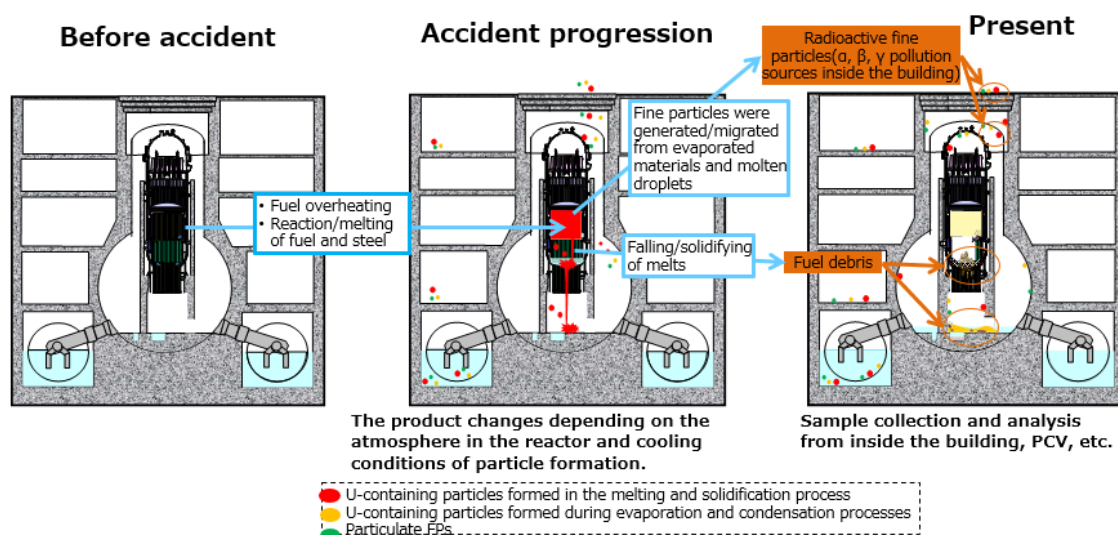
Figure 6.1 Relationship between the properties of radioactive particles and fuel debris and accident progression

7． Estimation of the present situation of core and containment vessel of Unit-1 to Unit-3

Since the accident, TEPCO has continued its efforts to estimate the state of the core and containment vessel of Units-1 to -3, and from FY2016 to FY2017, TEPCO worked with the International Research Institute for Decommissioning of Nuclear Power Plants (IRID) and the Institute of Applied Energy (IAE) in the "Grant-in-Aid for Decommissioning and Contaminated Water Countermeasures Project (Advancement of Comprehensive In-reactor Status Assessment)." Therefore, the figures shown in Figures 7.1.1, 7.2.1, and 7.3.1 include the results of this project. Following the completion of the project, the estimation has been continued by TEPCO.
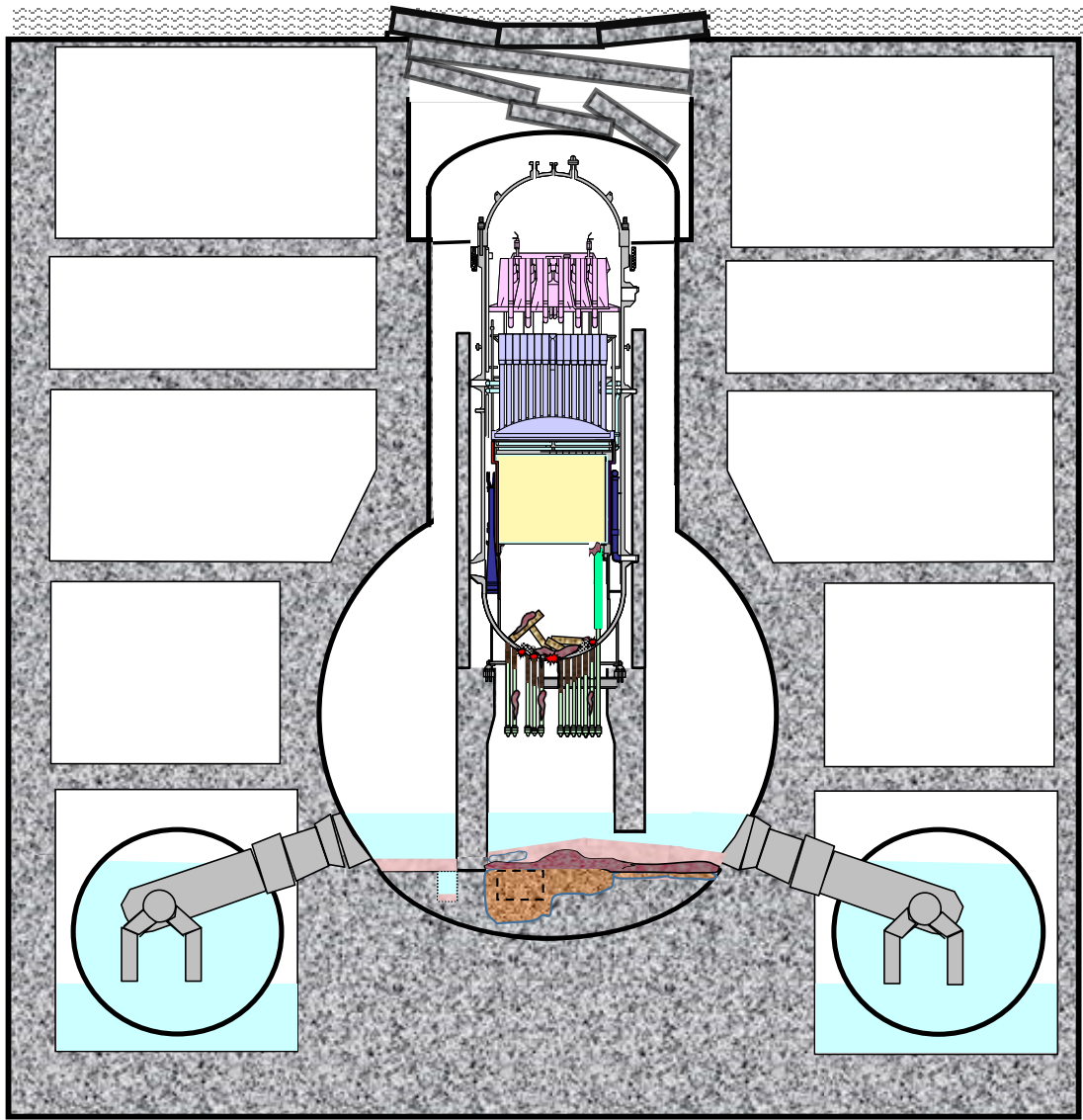
7． 1． The present situation of core and PCV of Unit-1

Water injected into the reactor pressure vessel from the core spray (CS) system is directly sent to the core and water from the feedwater system is sent to the lower plenum via the outer side of the core shroud. The reactor level is confirmed to be below TAF-5m, based on the calibrated results of the water level indicators, that is, no sufficient amount of water exists in the core region. At Unit-1, the temperatures had fallen below 100 deg C as of August 2011 and water injection via the CS system was started in December of the same year. The investigation by muon tomography separately conducted in 2015 found no big fuel inventory in the core region, either (Attachment 4).

The status of Unit-1 core was estimated based on the above facts and aforementioned examination results and is illustrated in Figure. 7.1.1. As can be seen in the figure, most of the molten fuel produced at the accident fell down to the lower plenum below the reactor pressure vessel and only a little fuel remains in the original core location. Most debris, which had fallen to the lower plenum, is believed to have reached the PCV pedestal. It is estimated that, after causing core-concrete interactions, the debris was cooled by injected water, its decay heat decreased terminating the core-concrete interactions and it now remains in the concrete at the bottom of PCV.

On the other hand, regarding the water level in the D/W, the level of residual water in the D/W was checked by cameras at the in-containment investigation in March 2015. It was about 1.9 m above the D/W floor (Attachment 4).

Concerning the status in the S/C, the nitrogen gas injection experiment in September 2012 demonstrated a mechanism in which Kr-85 and hydrogen produced at an early stage of the accident had remained in the upper space of the S/C and they were discharged to the D/W via vacuum breakers when the S/C water level was pushed down. This means that the S/C is currently filled with water (Attachment 4).

Water leaks were witnessed in November 2013 from the sand cushion drain pipe for discharging water accumulated around the D/W. Also, water leaks were witnessed in May 2014 from the vacuum breaker tube. From these findings, the leak of water from the PCV was thought to have occurred near the D/W bottom and vacuum breaker tube (Attachment 4).

７．２． The present situation of core and PCV of Unit-2

Water injected into the reactor pressure vessel from the CS system is directly sent to the core and water from the feedwater system is sent to the lower plenum via the outer side of the core shroud. Based on water filling to the condensing chamber on the reference water level side piping shown by the water level indicators, the reactor water level is estimated to be below TAF-5m, meaning no sufficient amount of water exists in the core region. The muon tomography investigation in FY2016 showed a possibility that fallen fuel on the bottom of RPV had remained there. Another investigation in the PCV conducted in December 2017 found, on the grounds that the grating was missing from the pedestal platform, that the hole at the RPV bottom had not been very big, and therefore the fallen fuel was estimated to have been present at the RPV bottom at its center or in its vicinity. Furthermore, an internal containment survey conducted in January 2018 confirmed an upper tie plate on the inner pedestal wall side of the bottom of the containment vessel, providing information indicating that the bottom of the pressure vessel was damaged. However, since the cable tray and other structures at the bottom of the containment vessel were not noticeably damaged, the deposited materials at the bottom of the containment vessel were considered to contain a large amount of metal. MAAP predicted opposite results, i.e., no damage at the Unit-2 reactor vessel, which is contradictory to the observation, probably due to uncertainties in the analysis.

The situation of Unit-2 core estimated based on the above facts and aforementioned examination results, is illustrated in Figure 6.2.1. As can been seen in the figure, part of the melted fuel produced in the accident fell down to the lower plenum below the reactor pressure vessel or to the PCV pedestal. Some of the fuel may remain in the original core location.

At the monitoring instrument installation work in the PCV in June 2014, the level of residual water in the D/W was confirmed to be about 30cm above the D/W floor.

The nitrogen gas injection experiment to the S/C conducted in May 2013 showed the S/C pressure of 3 kPa[gauge] (as of May 14th, 2013). This meant the S/C water level was at around the nitrogen gas injection inlet (O.P. 3780mm), because a certain water head should

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appear if the S/C was close to being full. When considered together with the low water level in the D/W, the water injected to the reactor is estimated to have flowed into the S/C via the vent lines from the D/W and leaked out to the reactor building from the bottom of the S/C, i.e., the current S/C water level can be estimated to be about the same level as the residual water level in the torus room (Attachment 4).

The water leak paths from the S/C have not been located yet. But at least no leakage was confirmed at the S/C manholes, etc. when, for the internal investigation in the torus room in April 2012, robots accessed the corridor for visual checks; or at the lower ends of the vent tube, when they were checked at the internal investigation of the torus room in December 2012 and March 2013. As the water level in the D/W was low and no damage was identified at the upper part of the S/C, the water leak from the containment vessel is considered to have occurred at the lower part of the S/C. (Attachment 4).

7．3． The present situation of core and PCV of Unit-3

Water injected into the reactor pressure vessel from the CS system and feedwater system is sent to the lower plenum via the outer side of the core shroud. The reactor temperature was lowered to 70 deg C as of November 11th, 2011, which had been achieved by the water injection from the CS system conducted from September 1st, 2011 and the fuel debris could have been cooled, which had remained on the CS water injection path, i.e., in the core position. Measurements by muon tomography in JFY2017 showed a possibility that no big amount of fuel had been left in the core region and part of the fallen fuel had remained at the bottom of RPV (Attachment 4). At the in-containment investigation in July 2017, damage conditions in the pedestal were found to be severer than in the Unit-2 pedestal. The amount of fuel fallen to the Unit-3 PCV is estimated to be bigger than that at Unit-2 (Attachment 4). MAAP4 predicted no damage at the Unit-3 reactor vessel, while MAAP5 predicted its damage. Such a significant difference is considered to come from the uncertainties in the analysis (for example, input conditions of the accident progression, and characteristics of analysis models), but from the observed findings it was likely that the Unit-3 RPV was damaged.

The situation of Unit-3 core estimated based on the above facts and aforementioned examination results, is illustrated in Figure. 6.3.1. As can been seen in the figure, part of the melted fuel produced in the accident fell down to the lower plenum below the reactor pressure vessel or to the PCV pedestal. Some of the fuel may remain in the original core location. It has turned out, however, that there had been a situation of not sufficient water injection possibly due to HPCI manual shutdown by the operators. This indicates the

accident progression was faster than the earlier estimation. In the figure, more fuel than before is assumed to have dropped to the PCV. On this matter, further detailed examinations are needed, including the molten core concrete interaction (MCCI) development behavior.

On the other hand, regarding water level in the D/W, at the in-containment investigation in FY2015, the D/W water level was found to be about 6.3m above the D/W floor. In May 2014, water leaks were witnessed around the expansion joint at the PCV penetration of the main steam piping D. This point is at the same elevation as the estimated water level in the containment vessel. This indicates most water leaks from the containment vessel occurred at this point. Currently, the height is approximately 5 m due to the earthquake that occurred on March 16, 2022, and the implementation of the water injection suspension test.

Legend

Oxide debris (porous)

Particle debris

Concrete-mixed debris

CRGT

Damaged CRGT

CRD

CRD
(containing debris inside)

Shroud

Damaged shroud

RPV damage opening

Deposit
(unidentified material)

Figure 7.1.1 Estimated condition of the reactor core and containment vessel of Unit-1

(Note) The figures shown here are images and do not represent the quantitative reality of the size of the fuel debris.

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Legend

Oxide debris (porous) Damaged CRGT Damaged shroud

Particle debris CRD RPV damage opening

Concrete-mixed debris CRD (containing debris inside) Deposit (unidentified material)

CRGT Shroud

Figure 7.2.1 Estimated condition of the reactor core and containment vessel of Unit-2

(Note) The figures shown here are images and do not represent the quantitative reality of the size of the fuel debris.
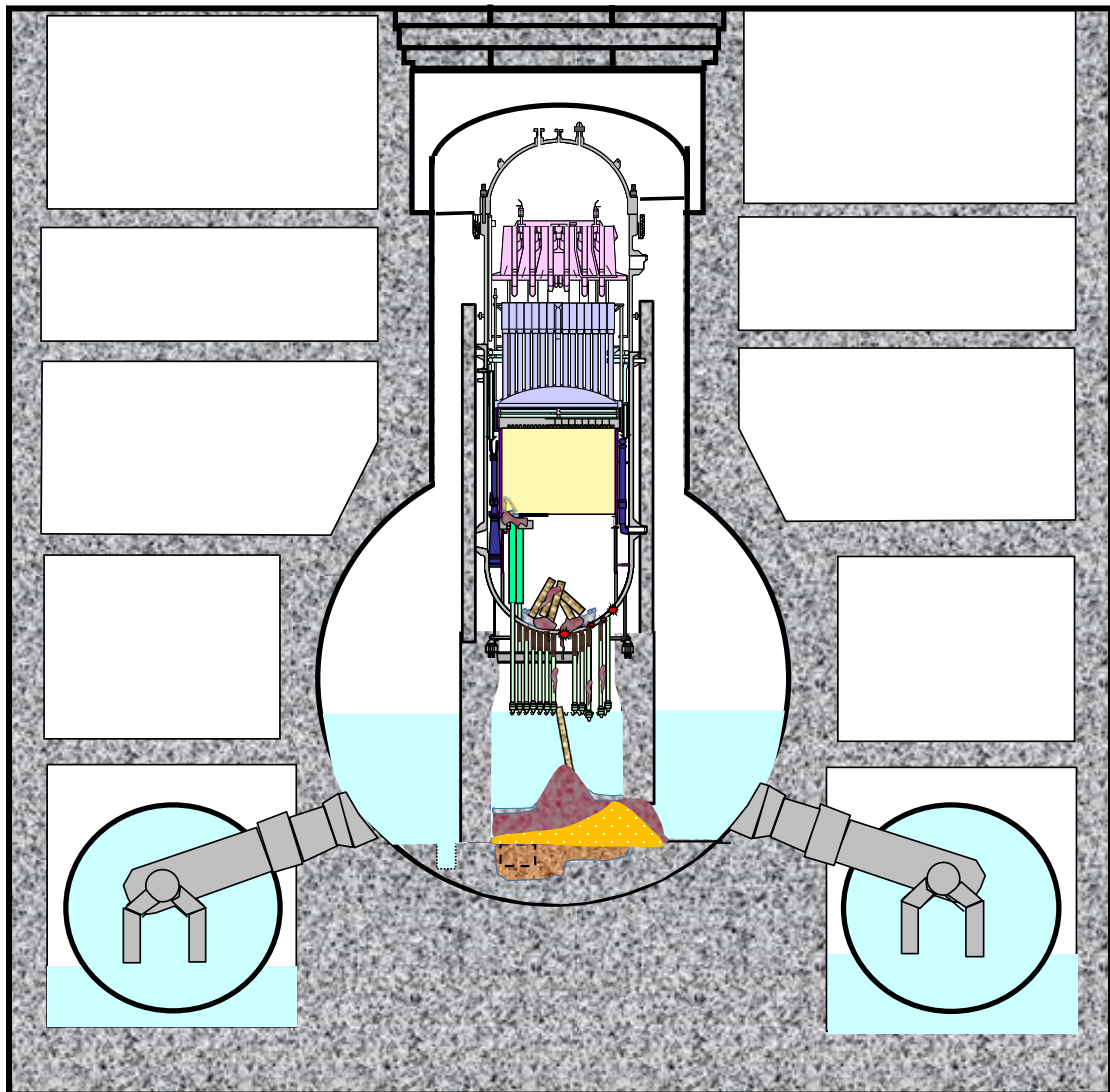
Legend

	Oxide debris (porous)		Damaged CRGT		Damaged shroud
	Particle debris		CRD		RPV damage opening
	Concrete-mixed debris		CRD (containing debris inside)		Deposit (unidentified material)
	CRGT		Shroud		

Figure 7.3.1 Estimated condition of the reactor core and containment vessel of Unit-3

(Note) The figures shown here are images and do not represent the quantitative reality of the size of the fuel debris.

８．　Connection with safety measures

８．１．　Event tree analysis

Below are detailed explanatory remarks about the event-tree analysis illustrated in Figure 1 of the first chapter. An event-tree analysis is a means to analyze what sequences a system follows starting at an initiating event to the ultimate status via junctions such as a loss of function of safety-related equipment. Generally, at a junction, a branch above (success) leads an accident progression in the direction of cold shutdown, while a branch below (failure) leads in the direction to a severe accident. The more outcomes going to branches below, the more dangerous the ultimate plant conditions would be. Junctions are defined as a success or failure of functions of safety-related systems or equipment, etc. in an accident progression. Basically, a junction follows a junction at its left, but the time difference between the two is not fixed, and varies according to the accident progression features at each unit.

Below the results of the accident progressions of each unit as obtained from the event-tree analysis are described.

First, an earthquake (the Tohoku-Chihou-Taiheiyou-Oki Earthquake), the initiating event, led the flow to the first junction of a reactor scram triggered by the earthquake. At all Units-1 to -3, the outcome went to the branch above, because they were successfully scrammed. At the following junctions, the outcomes at all units went to branches below (failure), at the loss of off-site power supplies (E: earthquake) and loss of diesel generator function (T: inundation by tsunami), which resulted in loss of AC power supplies.

At the following junction, the outcomes at Unit-1 and Unit-2 went to branches below (failure) when their DC power supplies were lost simultaneously with the loss of AC power supplies, but at Unit-3 it went to the branch above (success) when its DC power supply survived the tsunami.

At Unit-1, the IC could not be started up due to the loss of DC power supply, because the IC had been shut down immediately before the DC power loss. Consequently, Unit-1 became unable to be cooled by high-pressure means. Unit-2 and Unit-3, however, could continue to be cooled by the RCIC (Unit-2 and Unit-3) and HPCI (Unit-3) systems.

Even at Unit-3, where the DC power supply survived, the high-pressure means for reactor cooling were lost because the off-site power supply and emergency diesel generators could not have been recovered before the DC power supply was depleted. Even if the off-site power supply had been recovered, the high-pressure means for reactor cooling would have been eventually lost as most power panels had lost their function because of the tsunami. Unit-2 continued to be cooled for about 70 hours, much longer than the design value of 8 hours, but the off-site power supply could not be recovered and the RCIC lost its functions

for unknown reasons. Eventually all units lost their cooling capabilities (failure of AC power supply recovery).

Thereafter, Unit-1 could avoid the reactor pressure vessel damage under high-pressure conditions for some unknown reasons[#1], while Unit-2 and Unit-3 were successfully depressurized by the SRV, although much effort had been needed for collecting alternative batteries, etc. But, despite the effort of water injection by fire engines, all units experienced core damage (the RHR could not be used because the power supply could not be recovered and the seawater pumps had lost their functions by tsunami inundation).

After the core damage, the PCV venting succeeded at Unit-1 and Unit-3, but hydrogen gas exploded when fully accumulated in the reactor buildings and that resulted in the release of radioactive materials via unknown paths. Unit-2 could avoid the hydrogen explosion because one of its blowout panels had been opened by the Unit-1 hydrogen explosion impact. Still a large amount of radioactive materials was released due to PCV venting failure.

[#1] In this analysis, "success" includes the case of successful depressurization, not only before the core damage, but even after core damage, if done before the reactor pressure vessel damage. At Unit-3, there is a possibility that the core was damaged before the reactor was depressurized, as discussed in Section 5.3.1. The background to defining the branch (successful or failed) at the junction depending on the reactor damage under high pressure conditions is that, according to the existing knowledge, it is possible for large PCV damage to occur due to direct containment heating (DCH) once the reactor vessel is damaged at pressures of 2 MPa or higher.

8．2． Approach for safety measures

As has been discussed above, it is possible to review the accident progression, by the event-tree analysis, from the viewpoint of whether or not the safety-related functions were lost, although some causes of loss of safety-related functions still remain unknown.

Therefore, there are three ways of approaching safety measures to take based on the accident at the Fukushima Daiichi NPS: (1) to prevent loss of safety-related functions; (2) to mitigate the consequences of the accident; and (3) to strengthen safety measures unrelated to the accident scenario. In approach (1), the mechanism of how the earthquake and tsunami did affect the plant is analyzed and measures are taken for safety-related systems and equipment for excluding those anticipated impacts (examples are construction of breakwaters and installation of watertight doors, etc.); in approach (2), reliabilities of existing systems and equipment are improved, irrespective of the impacts of earthquakes and

tsunami, and their functioning is ensured when in need; and in approach (3), alternative systems are assessed, which are located where no impacts will be received from earthquakes and tsunami (alternative batteries, pumps, etc.).

The examination results of this progress report mention safety measures to take, in which not only those safety measures to directly prevent identified causes, but also further safety measures from the above viewpoints are included.

This progress report series mentions safety measures to take, in which not only those examination results of direct safety measures for preventing the clarified causes, but also those for further safety measures from the above viewpoints are included.

９．Conclusions

The issues, which still remained unclarified at the time of this report concerning the accident at the Fukushima Daiichi NPS, have been identified and their examination results are compiled. These issues remain unclear because they are too difficult to solve in detail in a limited time. Further examinations will continue and their results will be updated.

With the progress of examination of these issues, the estimated situation of the cores and PCVs will also need to be examined for revision.

The examinations will take much time over a long period of time. Their results can be expected to generate the following three pillars of outputs: (i) complete revealing of the whole picture of the accident at the Fukushima Daiichi NPS (estimation of debris location); (ii) upgrading of the analysis code by the newly obtained knowledge; and (iii) contributing to strengthened nuclear plant safety by the obtained knowledge.

Output (i) will immediately contribute to the fuel removal programs by providing debris position information, etc., and to the decommissioning program by providing information of the damage conditions of cores and PCVs. Output (ii) can contribute to the overall improvement activities of nuclear plant safety by applying upgraded analysis codes to the nuclear safety evaluation using the probabilistic risk analysis (PRA) approach, or to the improvement of reliabilities in evaluating effectiveness of accident management measures applied. Output (iii) will help to take measures to prevent unknown occurrence mechanisms which led to the loss of safety functions, to correct the severe accident knowledge that was misunderstood in the past, and to identify items for further improvement in operation procedures and management, etc.

１０． Supplement (Meanings of notation O.P. in this report)

At the Fukushima Daiichi Nuclear Power Station, installation elevations of equipment and systems are currently expressed in terms of T.P. (the mean sea level at Tokyo Bay), not O.P. (the work reference level of Onahama Port) to date, by taking the subsidence due to the earthquake into account.

However, this report continues using expressions in terms of O.P. before the earthquake for the following reasons.

- This report addresses the accident analysis, not the current plant construction work or administration. The O.P. expression causes no problems.
- Throughout this report, installation elevations of equipment and systems are expressed in terms of O.P. But, the extent of subsidence was not quantified at one to two weeks after the earthquake, and the study results in this report do not change with different elevations against O.P. level.

It should be noted that the following conversions from O.P. expressions to T.P. expressions are needed to use the study results of this report in future practical work at the Fukushima Daiichi Nuclear Power Station.

Unit-1 turbine building: O.P. value before the earthquake - 1457mm

Unit-2 turbine building: O.P. value before the earthquake - 1452mm

Unit-3 turbine building: O.P. value before the earthquake - 1437mm

Unit-4 turbine building: O.P. value before the earthquake - 1439mm

Unit-1 to Unit-4 reactor buildings: O.P. value before the earthquake - 1436mm*

(*Currently, the measured elevation of the reference point on the site is being substituted.)

References   (All documents below are in Japanese except 2, 4 and 6)

[1] Detailed analysis of the accident progression of Units 1 to 3 by using MAAP code,
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[7] Fukushima Nuclear Accident Analysis Report, Tokyo Electric Power Company, June 20,
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     Earthquake, Impacts of seismic motions on Unit-1 to Unit-3, Technical Workshop on the
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[10] Examinations into the impacts of thermal stratification in the suppression chamber
     water on the containment vessel pressures, etc., Seventh hearing on technical
     knowledge concerning the accident at the Fukushima Daiichi Nuclear Power Plant of
     Tokyo Electric Power Company, Feb. 1, 2012

List of Separate Documents

[Supporting information 1] Analysis results from MAAP code published on March 12, 2012

[Supporting information 2] Estimation of conditions in the reactor pressure vessels and
            containment vessels after the Fukushima Daiichi Nuclear Power Station accident

List of attachments

[Attachment 1] Overview of MAAP

[Attachment 2] List of issues

[Attachment 3] Findings from the latest analyses using MAAP5

[Attachment 4] Status of investigation on estimating the situation of cores and containment vessels

[Attachment 5] Sample analysis to understand the accident situation

[Attachment Earthquake-tsunami-1] Arrival times of tsunami at the Fukushima Daiichi Nuclear Power Station site

[Attachment Earthquake-tsunami-2] Additional examination of emergency AC power equipment losses due to tsunami

[Attachment 1-1] Amounts of water injection assumed in MAAP analysis for Unit-1

[Attachment 1-2] Evaluation of plant status by the fuel range water level indicators of Unit-1

[Attachment 1-3] Impacts of the earthquake on Unit-1

[Attachment 1-4] Examination into water injection by fire engines

[Attachment 1-5] Evaluation into the amounts of water injected to Unit-1 by fire engines

[Attachment 1-6] Estimation of Unit-1 accident progression based on the measured data and results of analysis to date

[Attachment 1-7] Examination into heat removal by the Unit-1 isolation condensers

[Attachment 1-8] Relocation behavior of molten fuel to below the core

[Attachment 1-9] Estimation of causes of high contamination of RCW piping at Unit-1

[Attachment 1-10] Analysis of the hydrogen explosion at Unit-1 reactor building

[Attachment 1-11] Estimation of accident progression at Unit-1 based on the air dose rate monitoring data

[Attachment 1-12] Identification of the cause of the high radiation dose rate observed in the southeast area of the first floor of the Unit-1 reactor building

[Attachment 2-1] Reactor pressure behaviors at Unit-2

[Attachment 2-2] Containment vessel pressure behaviors at Unit-2

[Attachment 2-3] Amounts of water injection assumed in MAAP analysis for Unit-2

[Attachment 2-4] RCIC flow rates of Unit-2 after the loss of power supply

[Attachment 2-5] RHR system situations after tsunami arrival at Unit-2

[Attachment 2-6] Behavior of primary containment vessel pressure starting about 12 o'clock on March 14th in Unit-2

[Attachment 2-7] Correlation between neutrons detected outside the reactor building and fuel melting

[Attachment 2-8] Evaluation of integrity of suppression chamber (S/C) at Unit-2