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. 1

December 13, 2013

Tokyo Electric Power Company, Inc.

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1. Introduction

1. 1. Response actions taken so far

The Tohoku–Chihou-Taiheiyou-Oki Earthquake and ensuing 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 Kashaiwazaki-Kariwa Nuclear Power Plant. (Refer to Fig. 1 Examinations of accident progression at Units-1 to 3 by event-tree analysis. [†2)] See 7.1 for further details.)

1. 2. Continuing improvement of safety measures

The two main pillars of safety enhancement measures currently being taken at Kashaiwazaki-Kariwa Nuclear Power Plant 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 still 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 generating contaminated water 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 (containment vessel hereinafter referred to as “PCV”), 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.

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.

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†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 are also ongoing in parallel to advance the accident analysis codes.

†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 plant, 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.

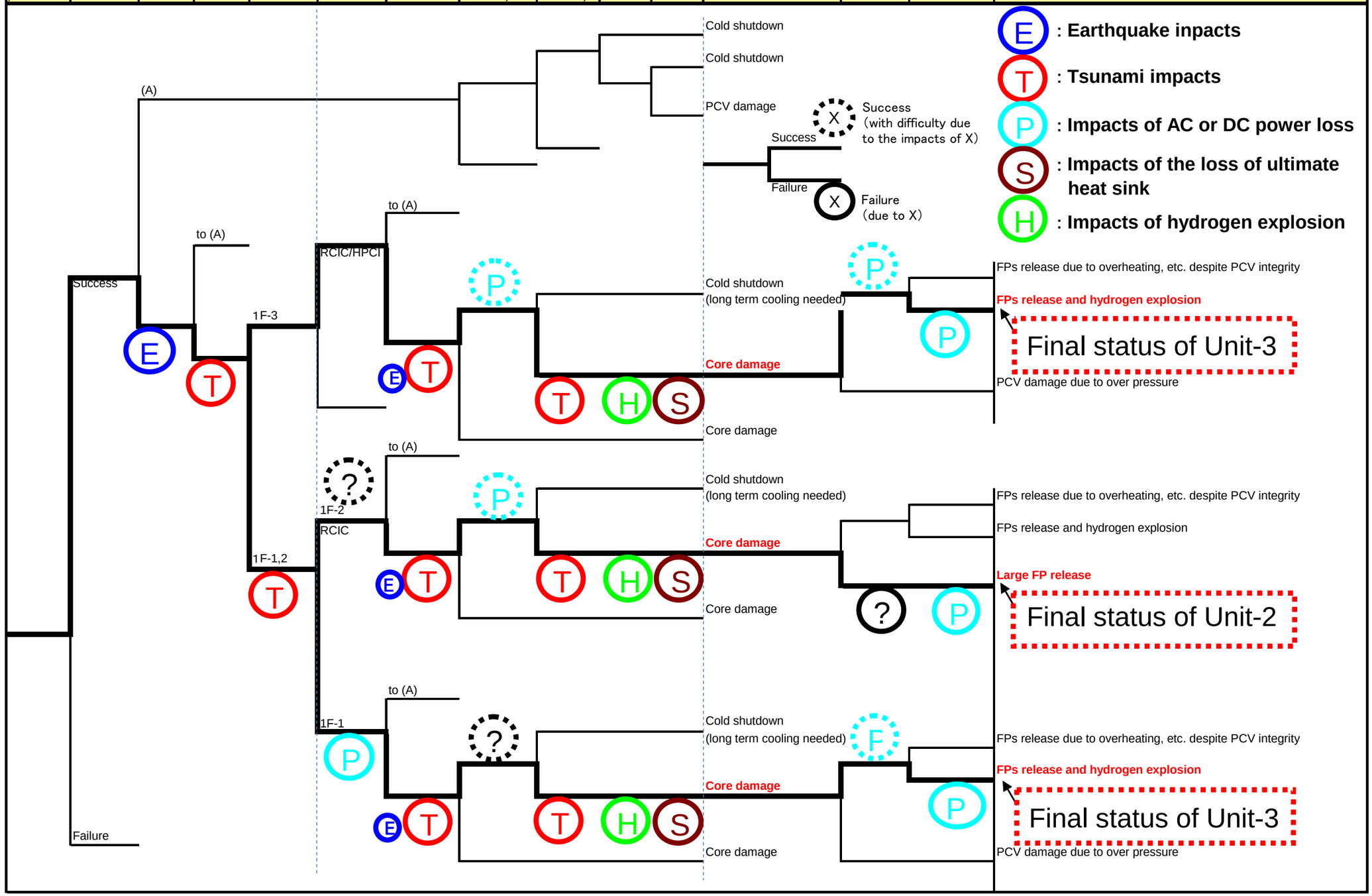

Fig. 1 Event-tree analysis results of Units-1 to 3 of Fukushima Daiichi Nuclear Power Station

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 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 duringthe 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 NPSand 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.

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).

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 15:36 and 15:37, 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．　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 actual 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”). Fig. 3.2.1 shows the reactor water level changes, while Fig. 3.2.2 shows the reactor pressure changes and Fig. 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 (separate Volume 1) 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, two isolation condenser (IC) systems automatically activated due to the reactor pressure increase following the scram caused by the earthquake. After that, the two IC systems were manually stopped and then IC subsystem-A was started up. The reactor pressure was controlled by manually repeating the start-up and shutdown of the IC subsystem-A. Maneuvering actions such as the starting up of the suppression chamber (S/C) in the cooling mode of the containment cooling system (CCS) were also being taken in parallel for cold shutdown of the reactor. At 15:37 on March 11th, 2011, however, all AC power supplies were lost due to tsunami followed by the loss of 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.

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). 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 main steam safety relief valve (SRV) 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 supplies 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. The level observed (by the wide range water level indicator) at 16:56 on March 11th was at the top of active fuel (TAF) +2,130mm and had not decreased yet to the TAF at that time, although it was still decreasing.

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 showing the decreasing trend of the reactor water level, the analysis 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 generated by water–zirconium reactions and the energy released from their oxidation reactions further raises fuel temperatures. 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

*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).

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 reactor building.

It is considered that noncondensable hydrogen gas generated 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).

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).

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 0.9MPa[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. 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).

When operation of the fuel range water level indicators were recovered at 21:19 on March 11th by the temporary power supply, they showed TAF+200mm 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 shall continue from this standpoint (Unit-1/Issue-3).

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).

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 0.6 MPa[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.

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.

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 generate noncondensable 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 0.7 MPa[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, because the PCV pressure is expected to rise, when steam is produced due to water injection, PCV temperature rises, and gases are generated 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. 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).

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).

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 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 10Sv/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 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. Later at 15:36, hydrogen in the reactor building exploded and the roof and outer walls of the uppermost floor were damaged.

It can be considered that hydrogen gas generated mainly by water–zirconium reactions leaked together with steam and finally reached the reactor building, resulting in the hydrogen explosion. But its leak path, volume, explosion aspects and ignition source are unknown. Examinations of these items remain (Common/Issue-11).

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).

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).

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)

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 12th 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 long time after the earthquake, which is difficult to do (Common/Issue-9).

The issues derived above are shown in Figures 3.2.1 to 3.2.3. They are also described in parallel in Attachment 2.

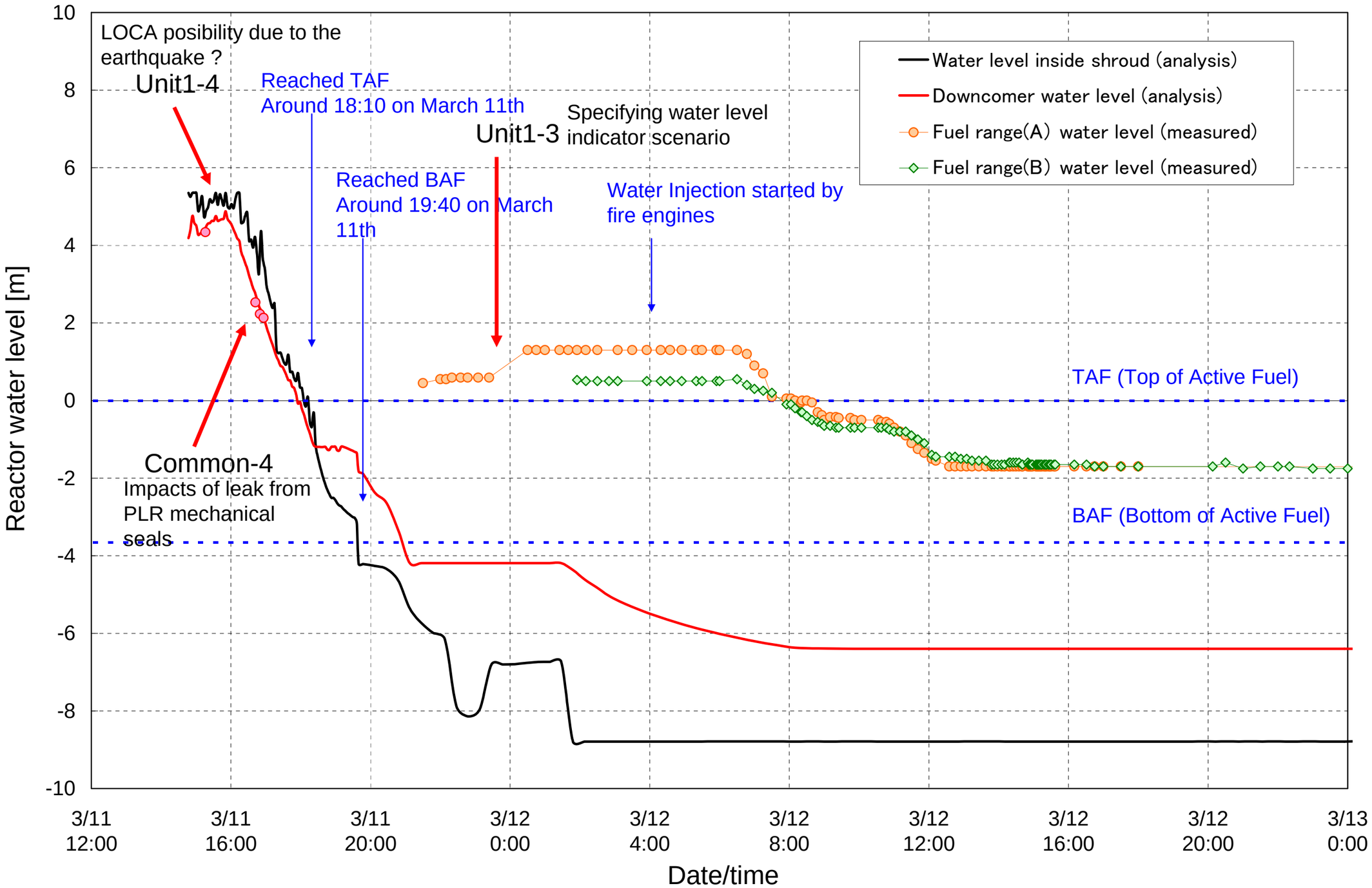

Figure 3.2.1 Issues derived from reactor water level change of Unit-1

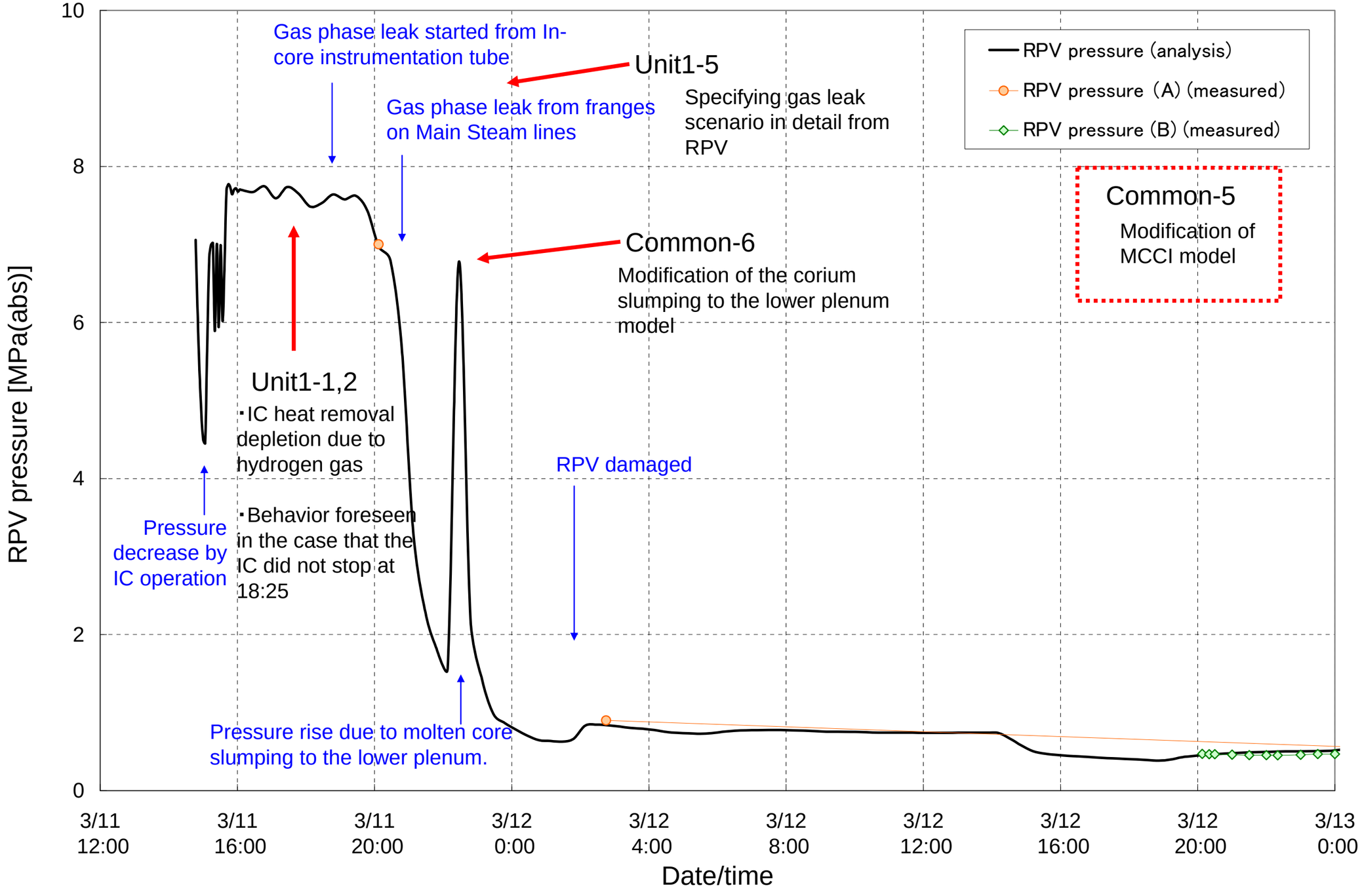

Figure 3.2.2 Issues derived from reactor pressure changes of Unit-1

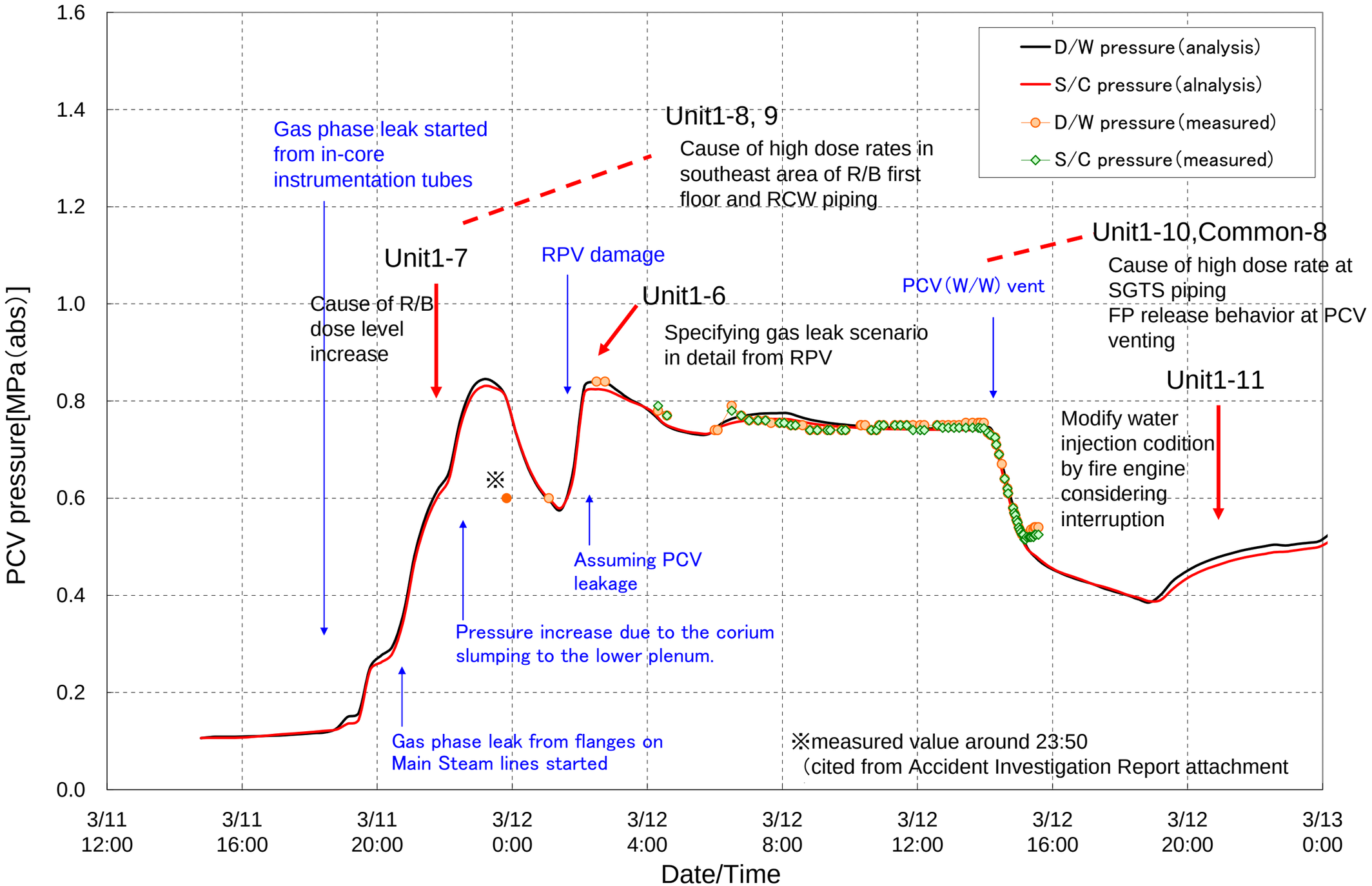

Figure 3.2.3 Issues derived from PCV pressure changes of Unit-1

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 (see 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 (see 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．　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 (see 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”). Fig. 4.2.1 shows the reactor water level changes, while Fig. 4.2.2 shows the reactor pressure changes and Fig. 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 (separate Volume 1) 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, start-up 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 generated 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/Issue1)

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).

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

After the reactor water level increased, no accurate water levels can be estimated, because the fuel range reactor water level indicator had reached their maximum limit of measurement. The reactor pressure, on the other hand, started to decrease after the RCIC started up (MAAP gave a later time for the pressure to start decreasing). 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, can 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 3-1 of the separate Volume 1.

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 depressurization boiling by opening SRV. The core was completely uncovered and core damage started (see Figure 3-1 in separate Volume 1). 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 (see Figure 3-2 in separate Volume 1). 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).

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).

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 (Common/Issue-2) from the injection lines of the fire engines.

Around the time when the core became uncovered and fuel cladding temperatures started to rise, a big amount of hydrogen was generated by the water – zirconium reactions (see Figure 3-6 in separate Volume 1).

The PCV pressure increased to 0.75 MPa[abs], thereafter, due to hydrogen generation 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 generation. 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).

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. The reactor pressure changes on this occasion need to be examined (Unit-2/Issue-7, 8).

At Unit-2 preparation was underway for the S/C venting, but no decisive evidence exists whether or not the rupture disk got 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). The occasional increase in reactor pressure around this time was at most about 1.5 MPa[abs] and noncondensable 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 noncondensable gas not being condensed is another issue to examine (Unit-2/Issue-10).

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).

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. The question of what incidents could have occurred at this time needs to be investigated (Unit-2/Issue-12).

The analysis predicted that the total amount of hydrogen generated over about a week after the earthquake was about 456 kg (see Figure 3-6 in separate Volume 1). 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 generation rate of Unit-2 as compared to Units-1 and 3. The reasons need to be examined (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 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. Whether the reactor vessel was damaged or not is highly influenced by the amount of water injection by fire engines. Therefore, 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) indication at 15:25 on March 15th were assumed to be due to the debris falling when the reactor vessel was damaged, MAAP would not be able to reproduce the reactor vessel damage around this time period, as far as the current MAAP results indicate. 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 reactor 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-8).

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).

The issues derived above are shown in Figures 4.2.1 to 4.2.3. They are also described in parallel in Attachment 2.

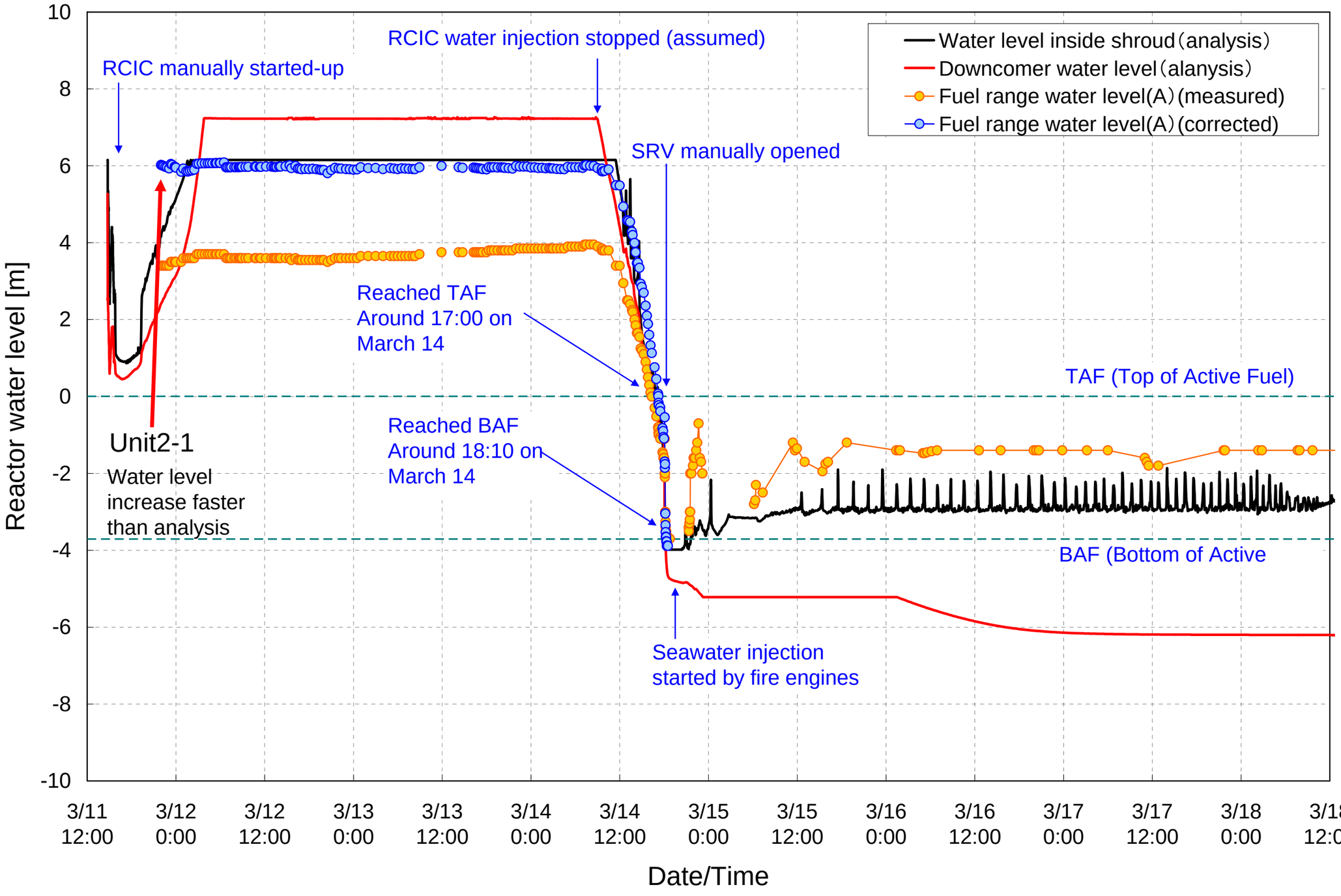

Figure 4.2.1　Issues derived from the reactor water level changes at Unit-2

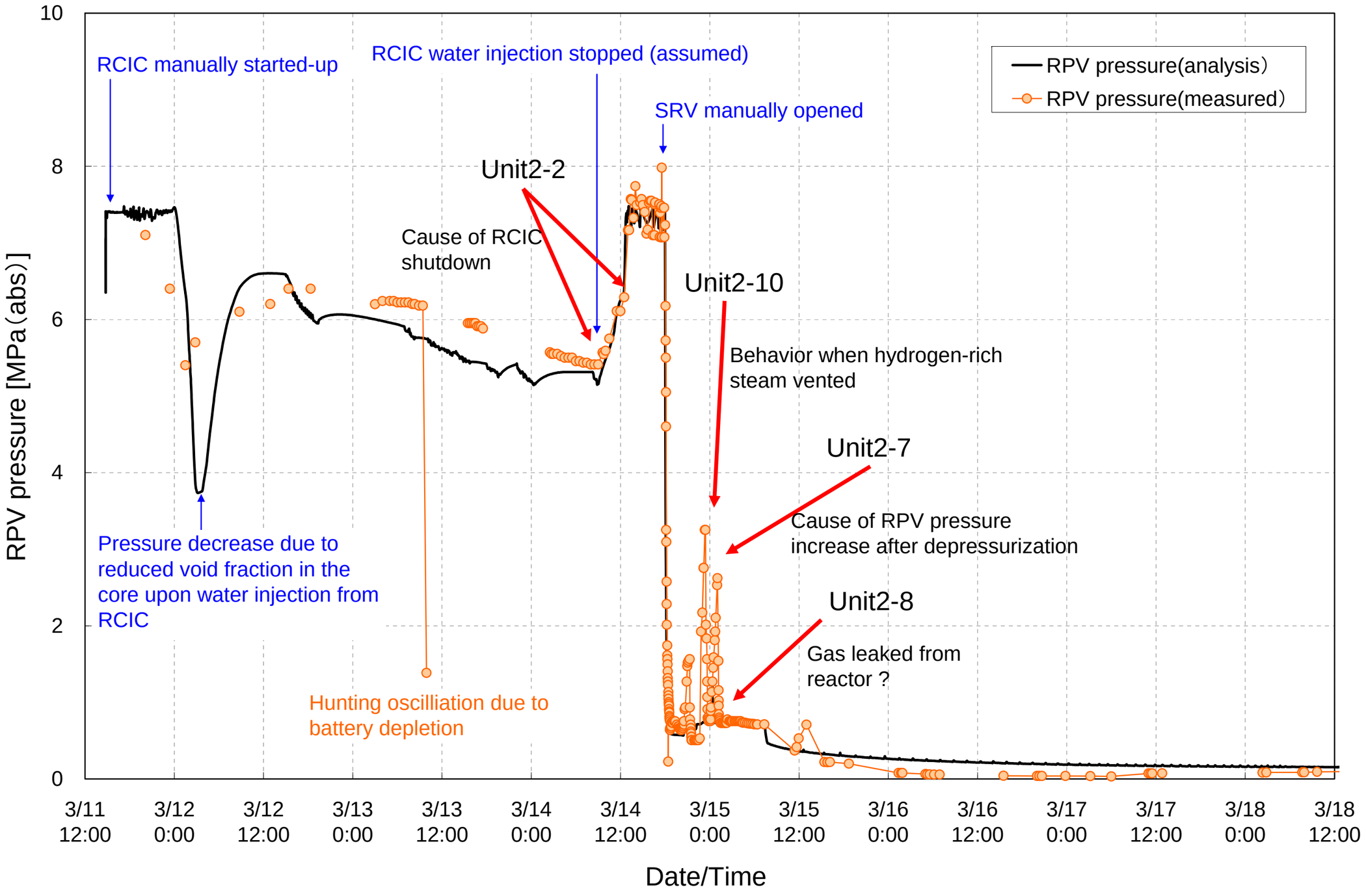

Figure 4.2.2 Issues derived from the reactor pressure changes at Unit-2

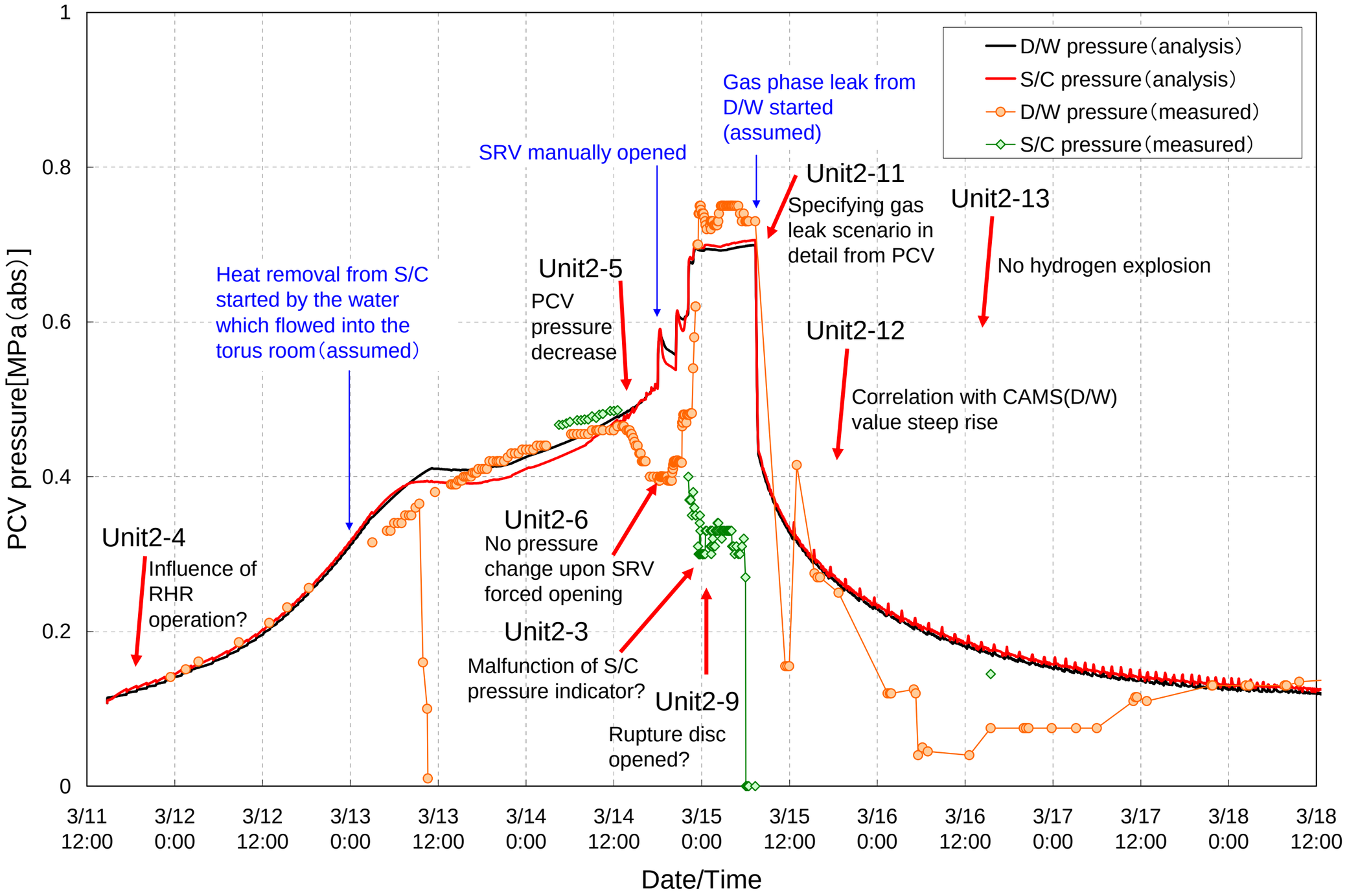

Figure 4.2.3 Issues derived from the containment pressure changes at Unit-2

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 (see 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 (see Attachment 2-5).

It has turned out through reinvestigation of operators' actions 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 was examined concerning the PCV pressure decrease after RCIC system shutdown (see 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 quantative 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 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 quantative reproduction calculations are necessary for verification.

4．3．4． 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 RCIC 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. Behavior of RCIC operation withoutat the time of loss of RCIC 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”.

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 (see 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”). Fig. 5.2.1 shows the reactor water level changes, while Fig. 5.2.2 shows the reactor pressure changes and Fig. 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 (separate Volume 1) 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.

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 tsunami arrived. As the DC power supply was available at Unit-3, the RCIC was manually started at 16:03. 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 RCIC), which would prevent automatic shutdown due to high reactor water levels and avoid battery depletion due to RCIC re-activation, 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 analysis value 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).

The RCIC stopped automatically at 11:36 on March 12th and thereafter its status of shutdown was confirmed on-site but its restart-up failed after all. 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 and the valve had been closed, but the background to this and reasons remain unknown and are subjects for continued examination (Unit-3/Issue-1).

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. 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, 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 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. Alter 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 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 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 need to be examined (Common/Issue-1).

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).

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 TAF-2m.

According to the analysis, the reactor water level decreased after the HPCI shutdown and the core got uncovered upon rapid reactor depressurization at about 09:00 on March 13th and core damage started. However, the water level was kept above TAF, overestimating the reactor water levels. This indicates a possibility, judging from the overestimated amount of water injection during the HPCI in operation as mentioned earlier and the HPCI having been operating under low reactor pressure conditions, that the reactor water level had started to decrease earlier than the HPCI manual shutdown. Thus examinations are needed into the reactor level behavior after the HPCI manual shutdown (Unit-3/Issue-5).

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 needs to be examined because it is known not to be by manual SRV operation (UnIt-3/Issue-6).

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).

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 generated 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 generation, but its details remain unknown and are left for further examination (Unit-3/Issue-7).

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).

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

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 generated 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 (Unit-3/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).

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).

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 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. On the other hand, there is a possibility that the amount of water injected by the HPCI was overestimated. Whether the reactor vessel was damaged or not is highly influenced by the amount of water injection by fire engines. Therefore, uncertainties in analytical conditions have big effects on the analysis results.

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).

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).

The issues derived above are shown in Figures 5.2.1 to 5.2.3. They are also described in parallel in Attachment 2.

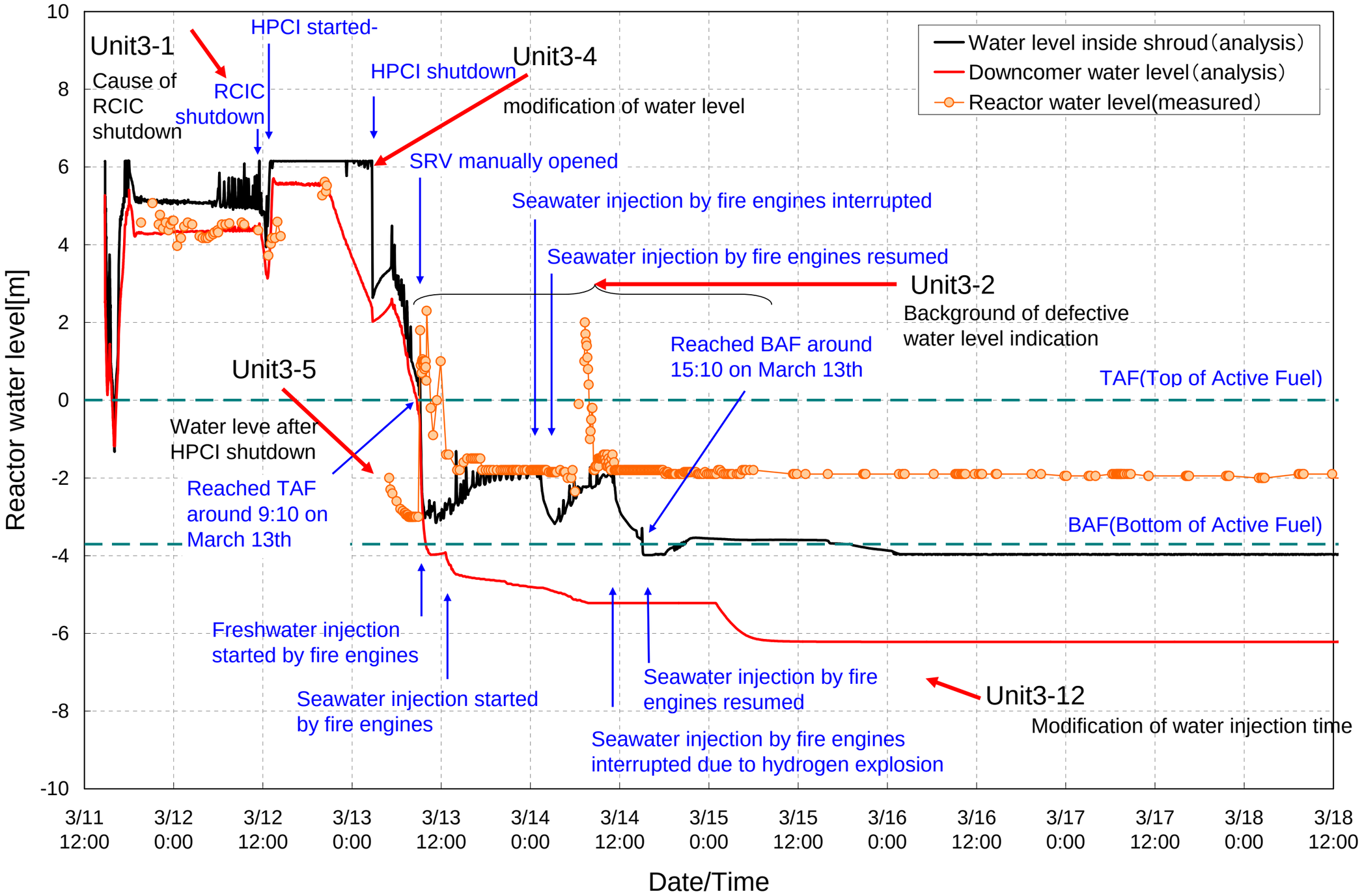

Figure 5.2.1 Issues derived from the reactor water level changes at Unit-3

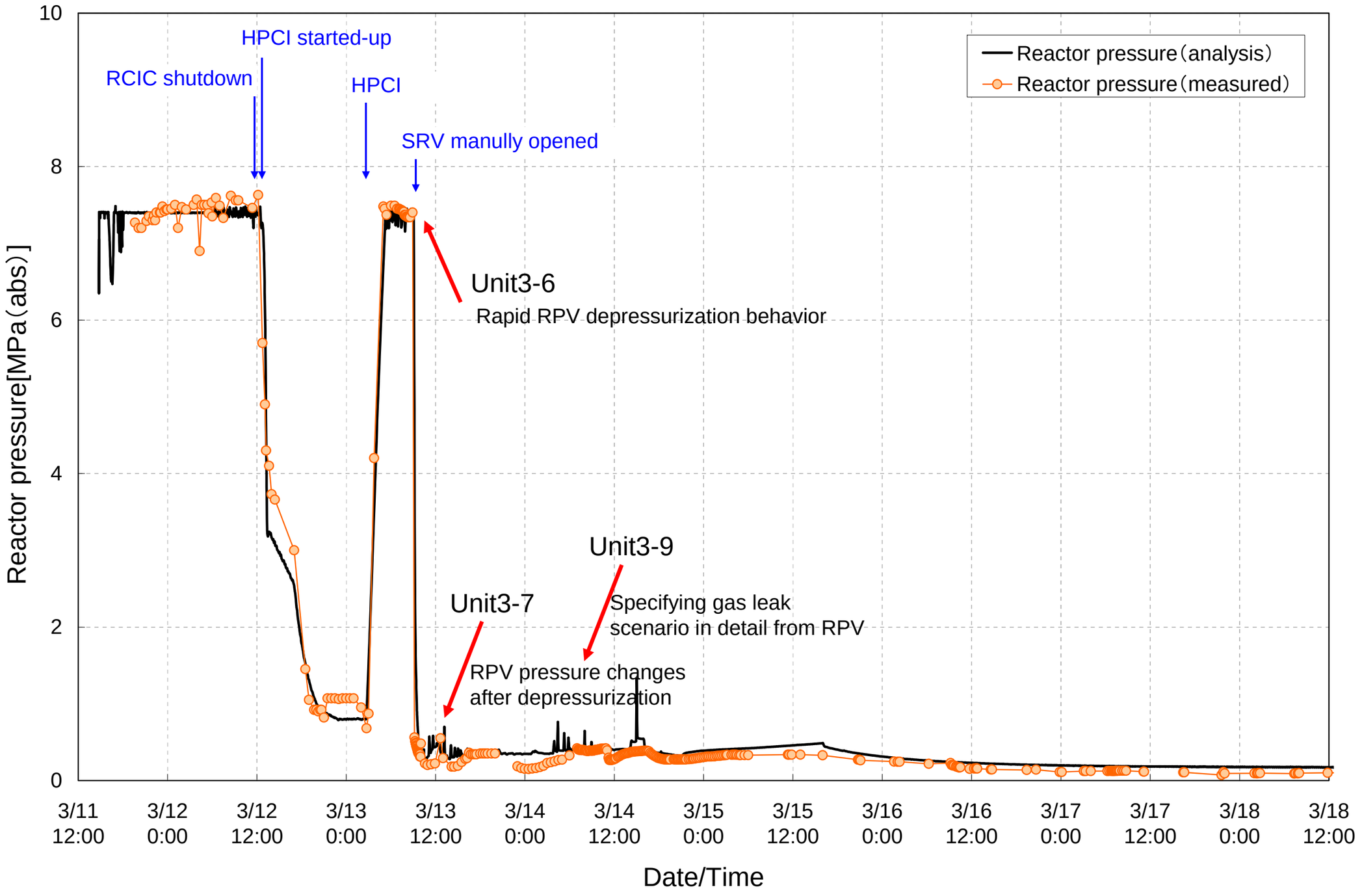

Figure 5.2.2. Issues derived from the reactor pressure changes at Unit-3

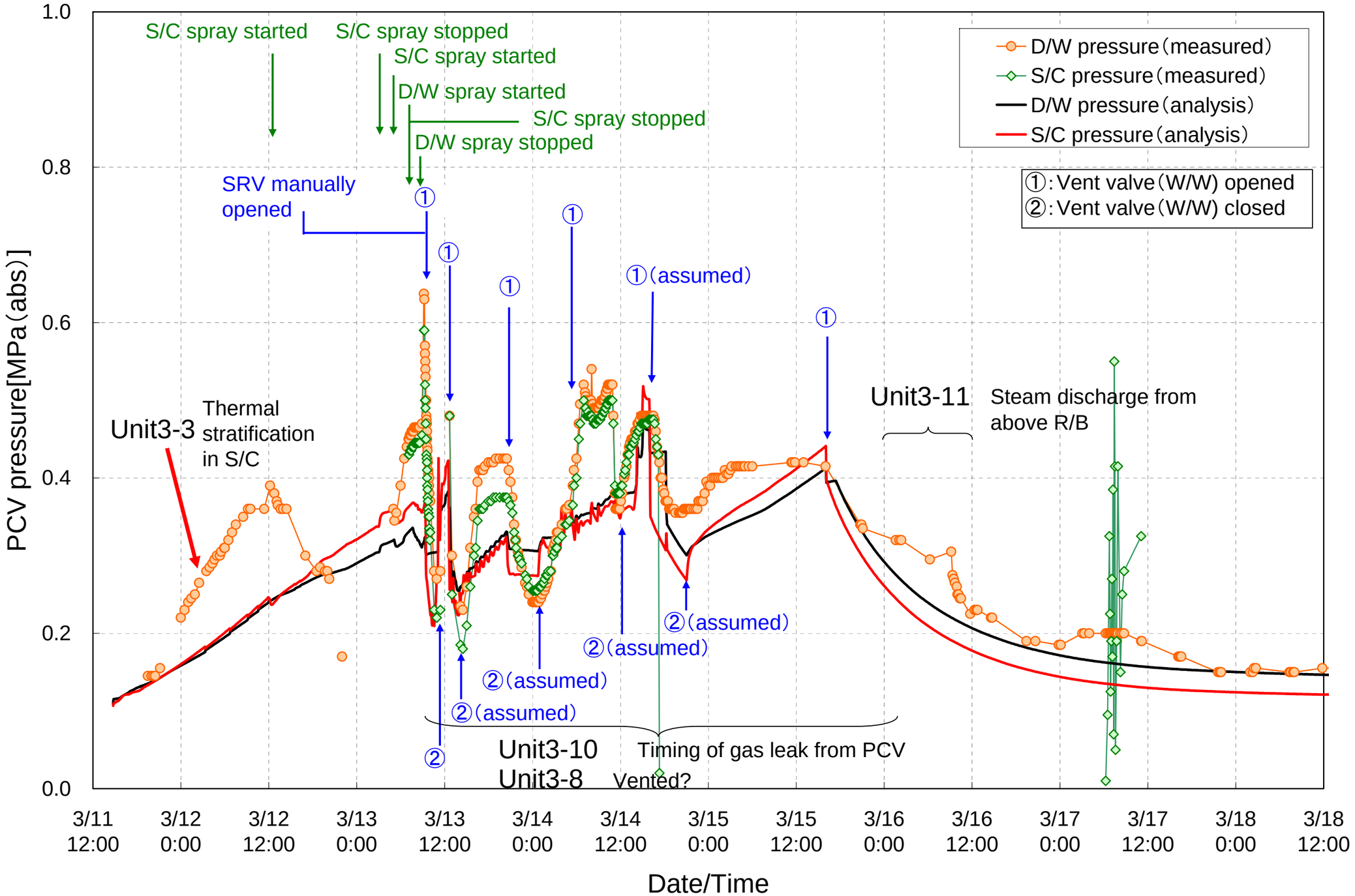

Figure 5.2.3. Issues derived from the PCV pressure changes at Unit-3

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 HPCI from the night of March 12th and of 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 that pressure decrease had been too rapid for one or two SRVs to cause by manual opening operation.

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 start-up 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 over this time period were also examined in parallel (Attachment 3-4).

5．3．2．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”.

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

6. Estimation of the present situation of core and PCV of Unit-1 to Unit-3

6. 1. The present situation of core and PCV of Unit-1

Water is being injected to Unit-1 from the CS and feedwater system, as shown in Figure 6.1.1. Water 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. The reactor level is confirmed to be below TAF-5m, based on the calibrated results of the water level indicators, that is, no sufficient water exists on the core region.

The status of Unit-1 core was estimated based on the above facts and aforementioned examination results, and is illustrated in Figure. 6.1.1. As can be seen in the figure, most of the molten fuel generated 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 PCV.

At the in-containment investigation in October 2012, the level of residual water in the D/W was checked by cameras. It was about 2.8m above the D/W floor (as of October 10th, 2012) (Attachment 4).

Concerning the status in the S/C, the nitrogen gas injection experiment in September 2012 demonstrated a mechanism that Kr-85 and hydrogen generated 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).

The water leak paths from the S/C have not been located yet. The internal investigation by cameras in the torus room in February 2013 showed that at least one vacuum breaker valve (out of eight) confirmed no water leak (Attachment 4).

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

Water is being injected to Unit-2 from the CS and feedwater system, as shown in Figure 6.2.1. Water 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 reference water level side piping shown by the water level indicators, the reactor water level is estimated to be below TAF-5m, meaning there is not sufficient water for covering the core. 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 generated 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 in-containment investigation in March 2013, the level of residual water in the D/W was checked by cameras. It was about 60cm above the D/W floor (as of March 26th, 2013).

The nitrogen gas injection experiment to the S/C conducted in May 2013 showed the S/C pressure of 3kPag (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 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 (Attachment 4).

6．3． The present situation of core and PCV of Unit-3

Water is being injected to Unit-3 from the CS and feedwater system, as shown in Figure 6.3.1. Water 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. 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. MAAP predicted no damage at the Unit-3 reactor vessel, which is contradictory to the observation, probably due to uncertainties in analysis.

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 generated 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.

No measured values are available so far concerning the D/W water level. But it could be estimated to be about 5.5 to 7.5m above the floor by converting the S/C pressure to water head. The S/C pressure was obtained from its existing pressure indicators, not calibrated since the accident, so they do not have high accuracies but they could be reliable as a trend to a certain extent because they have followed the pressure changes according to the water injection.

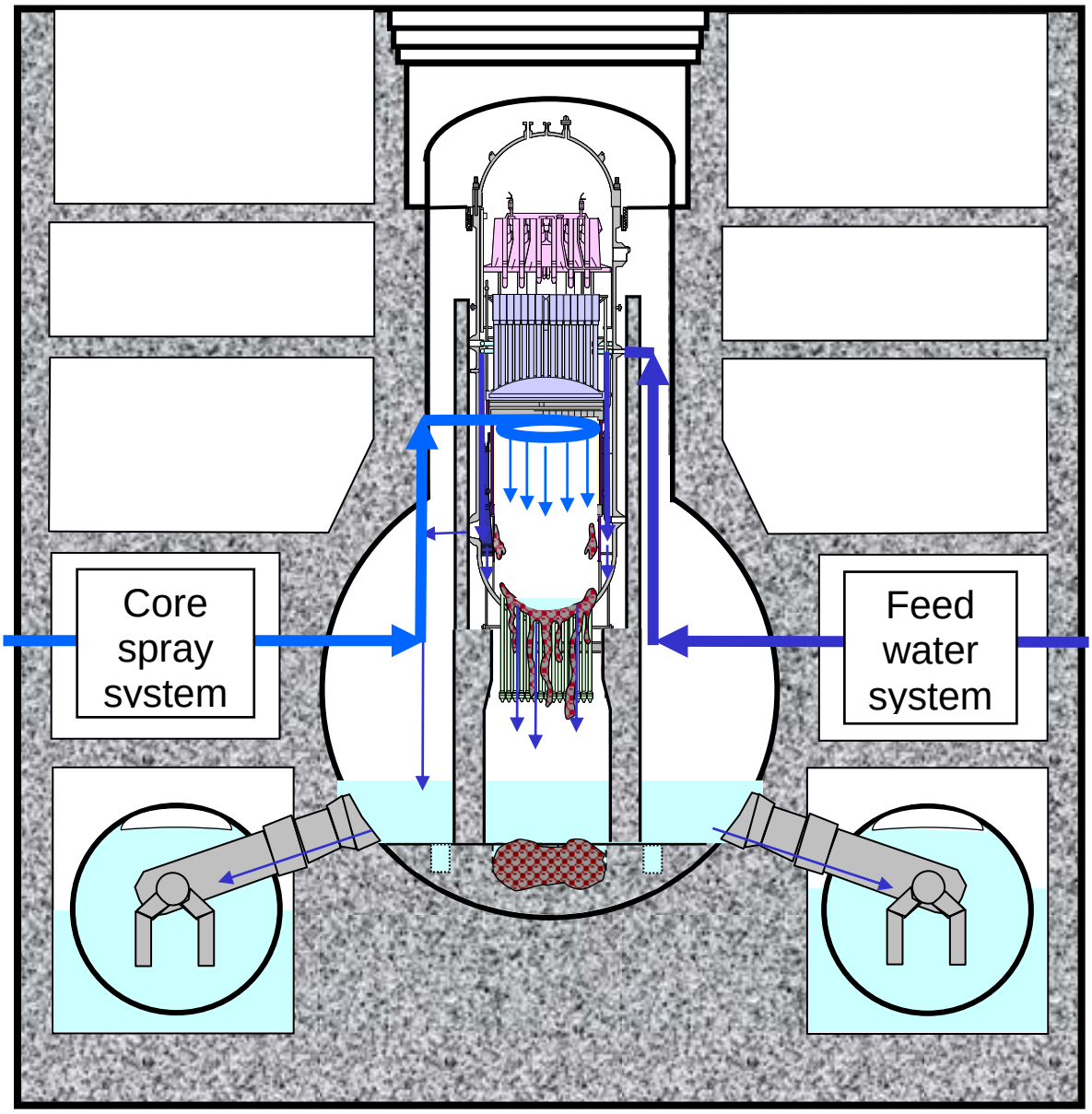

Figure 6.1.1 Estimated conditions of the core and PCV of Unit-1

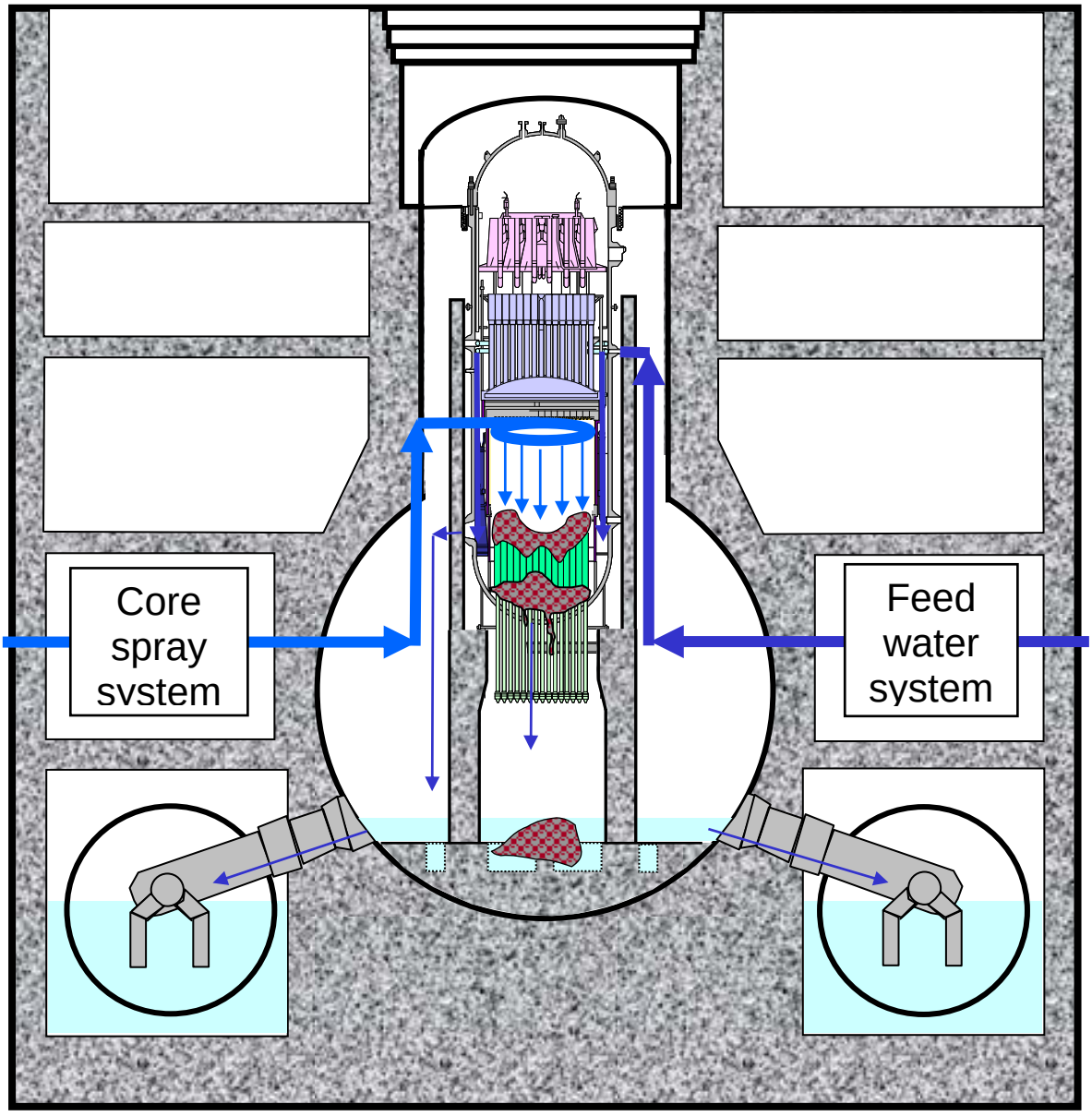

Figure 6.2.1 Estimated conditions of the core and PCV of Unit-2

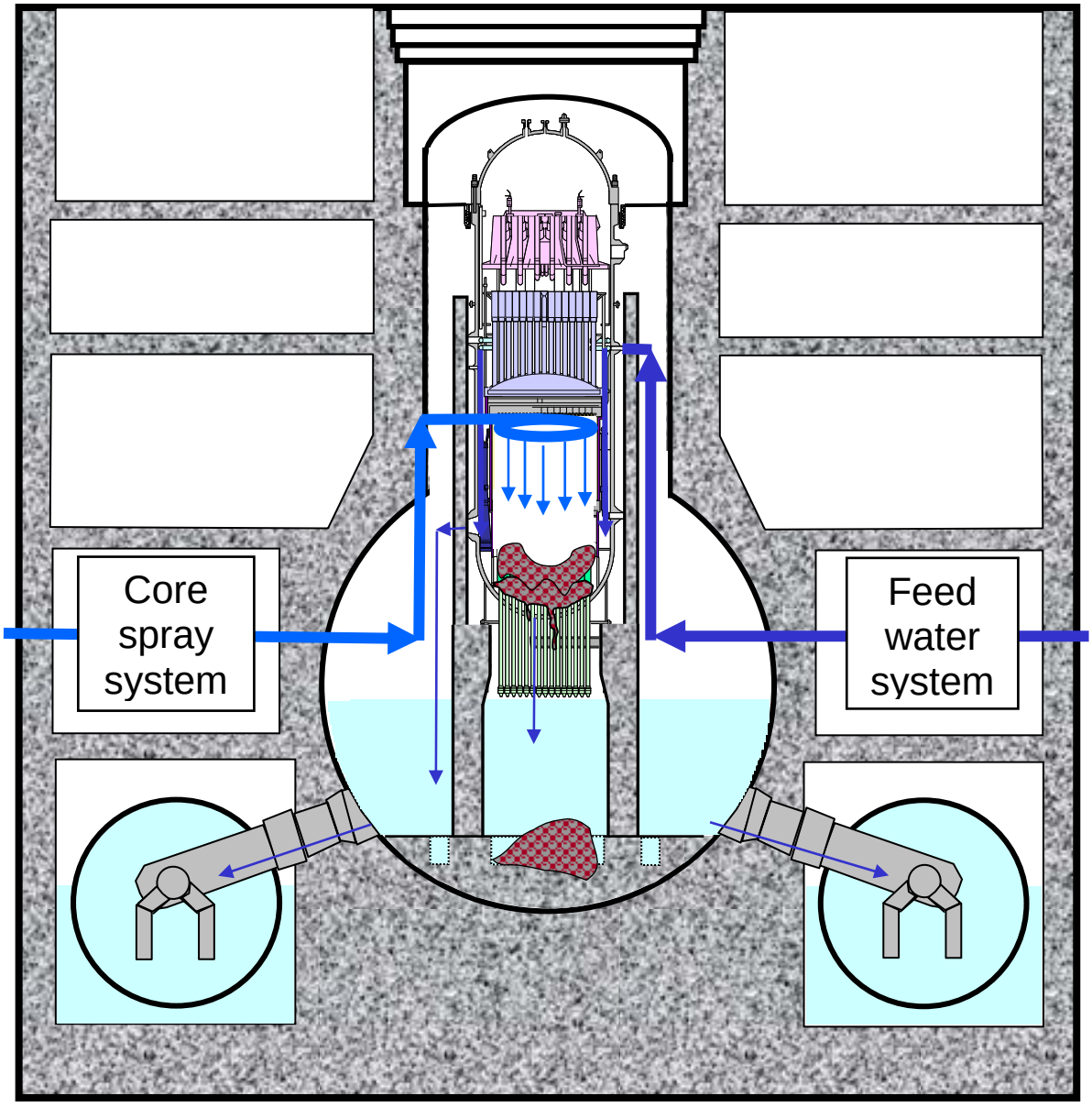

Figure 6.3.1 Estimated conditions of the core and PCV of Unit-3

7. Connection with safety measures

7. 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 of 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).

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 by 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.

7．2．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 reportmention 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.

8. Conclusions

The issues, which still remain 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 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.

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

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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 volumes

[Separate Volume 1] MAAP analysis results as of March 12th, 2012

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[Attachment 2] List of issues

[Attachment 3] Latest results of MAAP5 analysis (to be published)

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[Attachment 3-3] Reactor pressure decreasing behavior at about 9:00 on March 13th in Unit- 3

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