The 5th Progress Report
on the Investigation and Examination of
Unconfirmed and Unresolved Issues
on the Development Mechanism
of the Fukushima Daiichi Nuclear Accident

December 25, 2017

Tokyo Electric Power Company Holdings, Inc.

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Unconfirmed and Unresolved Issues

Overview

1. Overview of the accident at Fukushima Daiichi Nuclear Power Station

To date, TEPCO Holdings has compiled the following documents to summarize the Fukushima Nuclear Accident:

Fukushima Nuclear Accident Investigation Report
(Clarifies the facts related to conditions before and after the Fukushima Nuclear Accident)

<u>Nuclear Safety Reform Plan</u>
(Analyzes organizational causes that served as a background for the accident, as well as the technical causes of the accident)

- ✓ Elucidated the root causes of the Fukushima Nuclear Accident
  → Kashiwazai-Kariwa NPS: Implemented safety countermeasures to prevent a severe accident occurring

- ✓ TEPCO Holdings compliance with new safety regulations
  → Nuclear Regulation Authority: Each measure discussed and confirmed at review meetings.

2. Positioning of the investigation/examination

Accident root causes (clarified)

Accident investigations to date have clarified that the accident occurred because of a widespread loss of safety function caused by the tsunami, which, in turn, occurred after all external power had been cut off by the earthquake and that subsequent escalation of the accident could not be halted due to the lack of advanced accident prevention preparation.

(After reviewing the details of various accident investigations conducted by other agencies and organizations, including TEPCO Holdings, the Nuclear Regulatory Agency's accident analysis review committee determined that the primary causes of the accident were the same as those above determined by TEPCO Holdings.)

⇒ The Kashiwazaki-Kariwa NPS has implemented safety countermeasures based on these results.

Scope covered by the accident investigation report and the nuclear safety reform plan

Efforts to ensure that an accident like the Fukushima Daiichi Nuclear Accident never recurs

Detailed accident development mechanism

Understanding the **unsolved issues of details of how the incident developed after the initial accident** is not only the responsibility of the parties involved in the accident but also important to:

- **predict the state of field debris and accumulate the knowledge required for decommissioning**
- **provide knowledge to help improve the precision of accident simulation models used by countries worldwide.**
- **continually improve nuclear power station safety technology**

Scope covered by deliberations and investigations into unsolved issues

Continuous efforts to further improve safety

This report compiles the results of investigations and deliberations conducted from the above perspectives. This is also the fifth progress report following those given in December 2013, August 2014 and May and December 2015.

3. Investigation/examination history and positioning of this report

- ✓ This study extracted 52 unconfirmed and unresolved issues on the detailed development mechanism after the accident occurred and published four reports concerning the progress of the investigation and examination.
- ✓ The fourth progress report included examination results of ten high-priority issues.

- ✓ In this study, TEPCO Holdings has effectively utilized information obtained onsite as the decommissioning progresses, for examination.
- ✓ As information near the actual field ***emerged*** after the fourth progress report ***was issued***, through investigation by muon measurement into the fuel debris location in Unit 2 and 3 reactor pressure vessels (RPV) and investigation inside Unit 1-3 primary containment vessels (PCV), assumptions regarding the status inside RPV and PCV were enhanced.

- ✓ **In cooperation with activities to identify the status inside the reactor implemented by the government*, distribution of fuel debris inside Unit 1-3 has been assumed since FY2016.**
- ✓ **Using direct onsite information inside RPV and PCV obtained as the decommissioning progresses, examination has been made by working together with the actual field.**
- ✓ **Examination will continue reflecting the ongoing quest to improve safety.**

* Subsidy for Decommissioning and Contaminated Water Management (Advancement in Comprehensively Identifying Status Inside the Reactor)

3. Investigation/examination history and positioning of this report

52 issues events related to the detailed development of the incident following the accident were identified as unsolved

In the fourth and previous progress reports, examination results of 30 issues, including ten high-priority issues to understand the development mechanism, were reported.

Issues examined in the fifth and subsequent reports (22 issues)

Issues that help elucidate the development mechanism
21 issues

Assumption of detailed fuel debris distribution based on onsite information
1 issue

Information obtained from onsite investigation

Self-propelled investigation device (Unit 1)

Underwater ROV (Unit 3)

Images provided by: IRID

The fifth report
3 issues
2 issues for additional examination

The fifth report
Assumption of fuel debris distribution (conducted in cooperation with the government project)

In the fifth progress report, as well as reporting examination results as in the previous reports, assumption is also provided regarding fuel debris distribution as the output of activities to identify the status inside the reactor, which has been implemented by the government.

4. Main points of the fifth progress report

Assumption of fuel debris distribution

1. Estimation diagram of fuel debris

TEPCO Holdings made Estimation diagrams of fuel debris in Fukushima Daiichi Nuclear Power Station Unit 1-3 in cooperation with the project of "Subsidy for Decommissioning and Contaminated Water Management (Advancement in Comprehensively Identifying Status Inside the Reactor)" since FY2016, the results of which are reported.

Issues that help *clarify* the development mechanism

2. Analysis of hydrogen explosion at Unit 1 reactor building

Regarding the Unit 1 Reactor Building, hydrogen generated in the accident was considered to have leaked onto the 5th floor of the building via the PCV head flange, triggering the explosion. However, another scenario was also pointed out. The explosion was analyzed in two assumed cases (hydrogen leaking onto the 5th floor and onto both the 4th and 5th floors respectively) and these cases were then compared with the damage status of the building. Since the damage status obtained correlated more closely to the assumption of leakage onto the 5th floor, the conventional assumption was considered more accurate.

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

Based on the previous investigations, the emergency AC power source loss was considered attributable to the tsunami. However, another analysis still cited the earthquake as a cause of the accident. In response, additional examination ensued to confirm the relation between the tsunami intrusion process and the loss of the emergency AC power for each components. We confirmed a clear correlation between path length of the tsunami intrusion route from sea to component and the time delay of functional loss from tsunami arrival at Fukushima Daiichi. Therefore, the estimation which accident was worsened due to tsunami was considered more accurate.

4. Estimation of reactor water levels at the time when core damage and core melt progressed at Unit-2

It is considered the Unit 2 water-level gauge was no longer indicating the reactor water level accurately after the RPV depressurization. To obtain information helping identify the accident development, readings as the core damage and meltdown progressed were analyzed based on characteristics of the water-level gauge to assume the range of changes in the actual reactor water level.

5. Evaluation of the fraction of Unit-3 vent gas that flowed into Unit-4 reactor building

To help understand the mechanism that triggered the Unit 4 hydrogen explosion, the rate of Unit 3 vent gas flowing into the Unit 4 Reactor Building was evaluated. Analysis showed that approx. 35% of vent gas containing a huge amount of hydrogen flowed into Unit 4. The high potential of this hydrogen inflow having triggered the explosion of the Unit 4 Reactor Building was reaffirmed.

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

Not all relations between the release of radioactive materials into the environment during the accident process and the accident development scenario causing the release have been identified. To understand the accident development leading to the release of radioactive materials, changes in the air-dose rate in Unit 1 were analyzed. The accident development scenario obtained from the analysis tallied with the conventional assumption.

5. Sharing insights and engaging in discussion with researchers from Japan and overseas

The Atomic Energy Society of Japan meetings/International meeting

We have given presentations on study results at academic and international meetings. We have been fortunate to receive awards for these presentations. We will continue our examination while considering the comments made and other achievements gained through these activities.

<Presentation>
AESJ meeting: Spring and Fall meetings, 2016 and 2017
International meeting:
NUTHOS (Nuclear Thermal Hydraulics, Operation and Safety) 10th meeting, 2014 and 11th meeting, 2016
International Workshop on Severe Accident Research, Tokyo Univ., 2014
NURETH (Nuclear Reactor Thermal Hydraulics) 17th meeting, 2017

Subsidy Project for Decommissioning and Contaminated Water Management (Advancement in Comprehensively Identifying Status Inside the Reactor)

In cooperation with this subsidy project, we have assumed the status inside the reactor and PCV such as fuel debris distribution in Fukushima Daiichi Nuclear Power Station Unit 1-3 since FY2016.

The status of fuel debris, nuclear fission products, etc. considered to be distributed inside the RPV and PCV has been assumed based on accumulated knowledge from across Japan and overseas, including cooperation with overseas organizations and by comprehensively analyzing and evaluating "various information obtained from onsite investigations, etc.," "measurement data during and after the accident," "knowledge obtained from experiments," "analytical results of accident development," etc.
OECD/NEA BSAF was implemented as part of this activity.

Nuclear Regulation Authority, Japan the Committee on Accident Analysis

We explained our evaluation of the tsunami arrival time and the cause of the loss of all power sources, as mentioned in the interim report made by the NRA. We will continue our examination using the results of field investigations and the analytical results from the Committee.

Niigata Prefecture Technical Committee

We have explained the issues regarding questions and points of interest from the governor and committee members during the discussion at the Niigata Prefecture technical committee meeting to verify the Fukushima Daiichi accident and safety measures at Kashiwazaki-Kariwa NPS.

We are continuing our investigation while considering discussions and opinions with and from various organizations and researchers.

(Reference) Ten high-priority issues

Group	Issue	Issue No.
Issues reported on in the second progress report	Factors in the shutdown of the reactor core isolation cooling system at Unit 3	Unit 3-1
	Evaluation of the HPCI system operational state at Unit 3 and its impact on the accident's progression	Unit 3-5
	Rise in reactor pressure following forced depressurization at Unit 2	Unit 2-7
	Improving the accuracy of our estimate of the volume of cooling water injections from fire engines into the nuclear reactor	Common-2
Issues reported on in the third progress report	Success or failure of Unit 2 containment vessel venting (Rupture disk status of Unit 2)	Unit 2-9
	Cause investigation of dose increase on around March 20th	Common-9
Issues reported on in the fourth progress report	Investigation into safety relief valve (SRV) operations after reactor core damage	Common-1
	Behavior of molten fuel when dropping to the lower plenum (Dropping of melted reactor fuel onto the lower plenum)	Common-6
	Thermal stratification in the suppression pool at Unit 3	Unit 3-3
	High-dose contamination measured around the vicinity of particular pipes in Unit 1 Reactor Building (Identification of causes of the high-dose contamination of pipes of the reactor cooling water (RCW) system in Unit 1)	Unit 1-9

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Specific Examination

1. Estimation diagram of fuel debris

This topic describes the output of the project supported by “Subsidy for Decommissioning and Contaminated Water Management (Advancement in Comprehensively Identifying Status Inside the Reactor).”

Overview

- The “Investigation and Examination of Unconfirmed and Unresolved Issues” conducted by TEPCO Holdings sets the assumption of fuel debris distribution in Fukushima Daiichi Nuclear Power Station Unit 1-3 as the subject issue. The previous progress reports provided an outline of the fuel debris distribution together with analytical results concerning accident development and status inside the reactor and PCV of each unit.

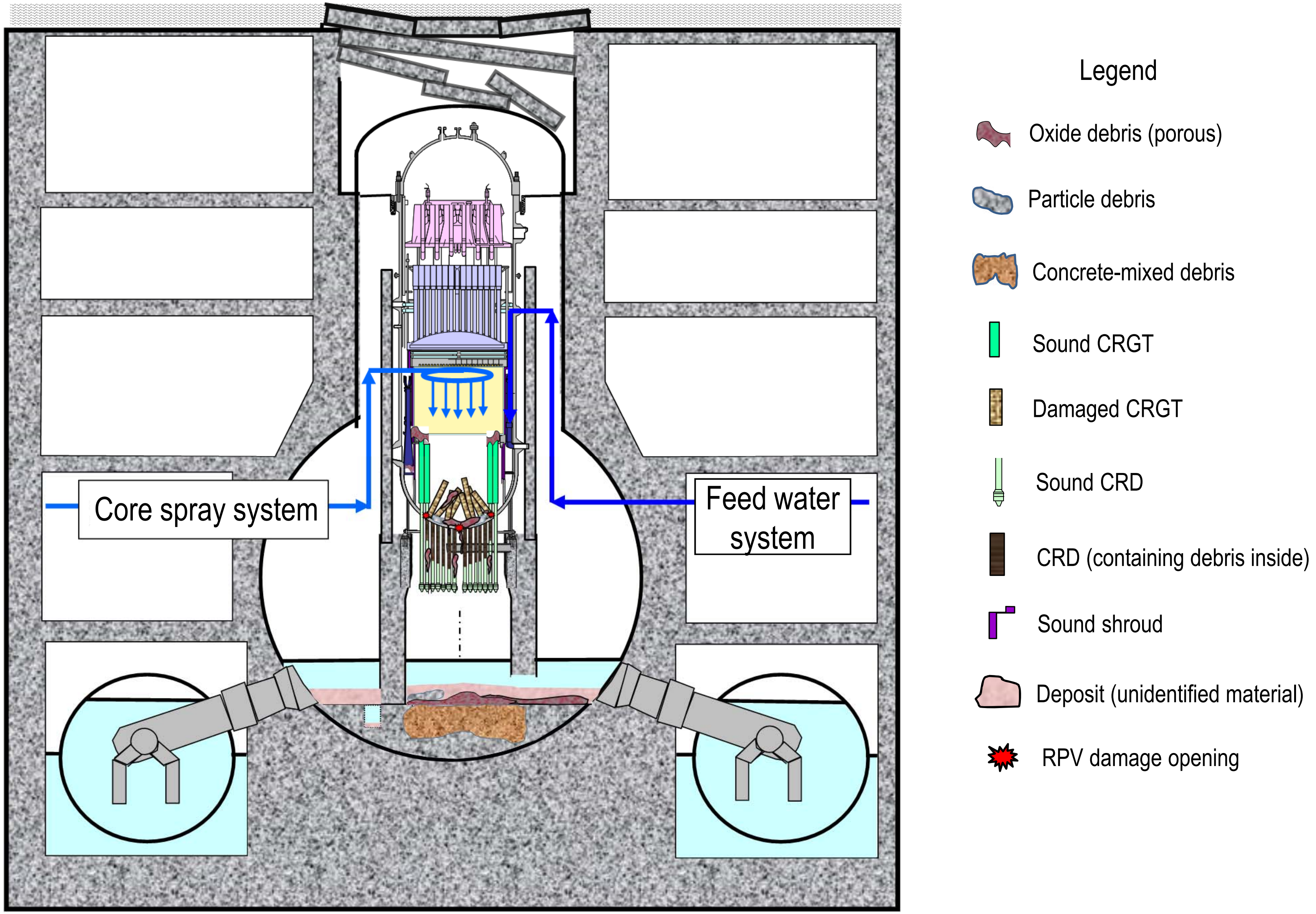
- As part of these efforts, TEPCO Holdings has assumed fuel debris distribution in Fukushima Daiichi Nuclear Power Station Unit 1-3 since 2016 in cooperation with the project of “Subsidy for Decommissioning and Contaminated Water Management (Advancement in Comprehensively Identifying Status Inside the Reactor)”. In this assumption, we have effectively utilized examination results, etc. regarding accident development provided previously “Investigation and Examination of Unconfirmed and Unresolved Issues.”

- We have been incorporating the views of outside engineers through cooperative relations with them fostered during this project. In addition, a quantity of information concerning the area inside the reactor and PCV was obtained as onsite investigations (*) progressed over the two years since the previous report was issued. The fuel debris distribution has been updated by actively using such information.

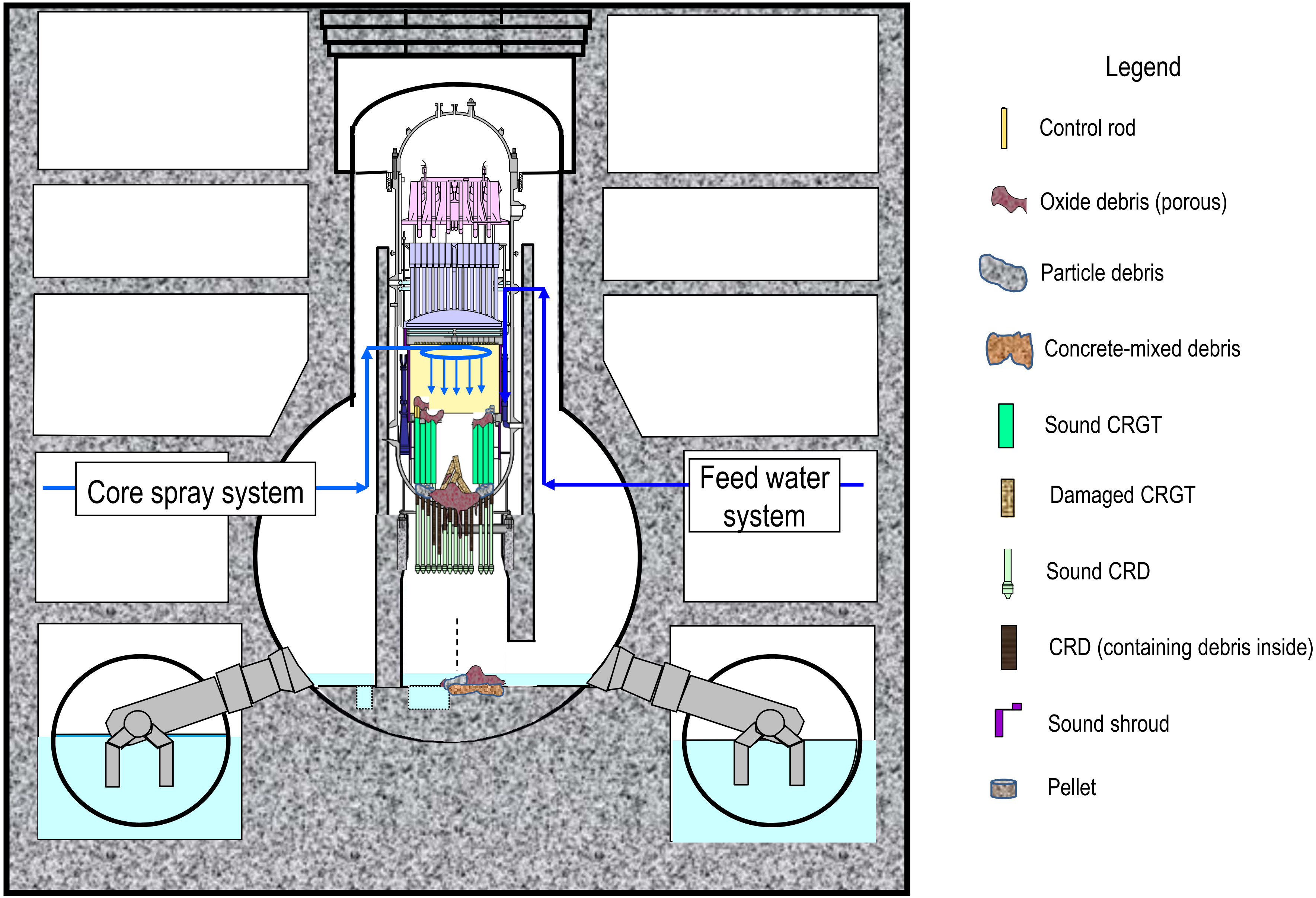
(*) Investigation inside Unit 1-3 PCV, muon measurement of Units 2 and 3, etc.

In the following pages, assumption of fuel debris distribution in Unit 1-3, the output of the project for Advancement in Comprehensively Identifying Status Inside the Reactor, is described.

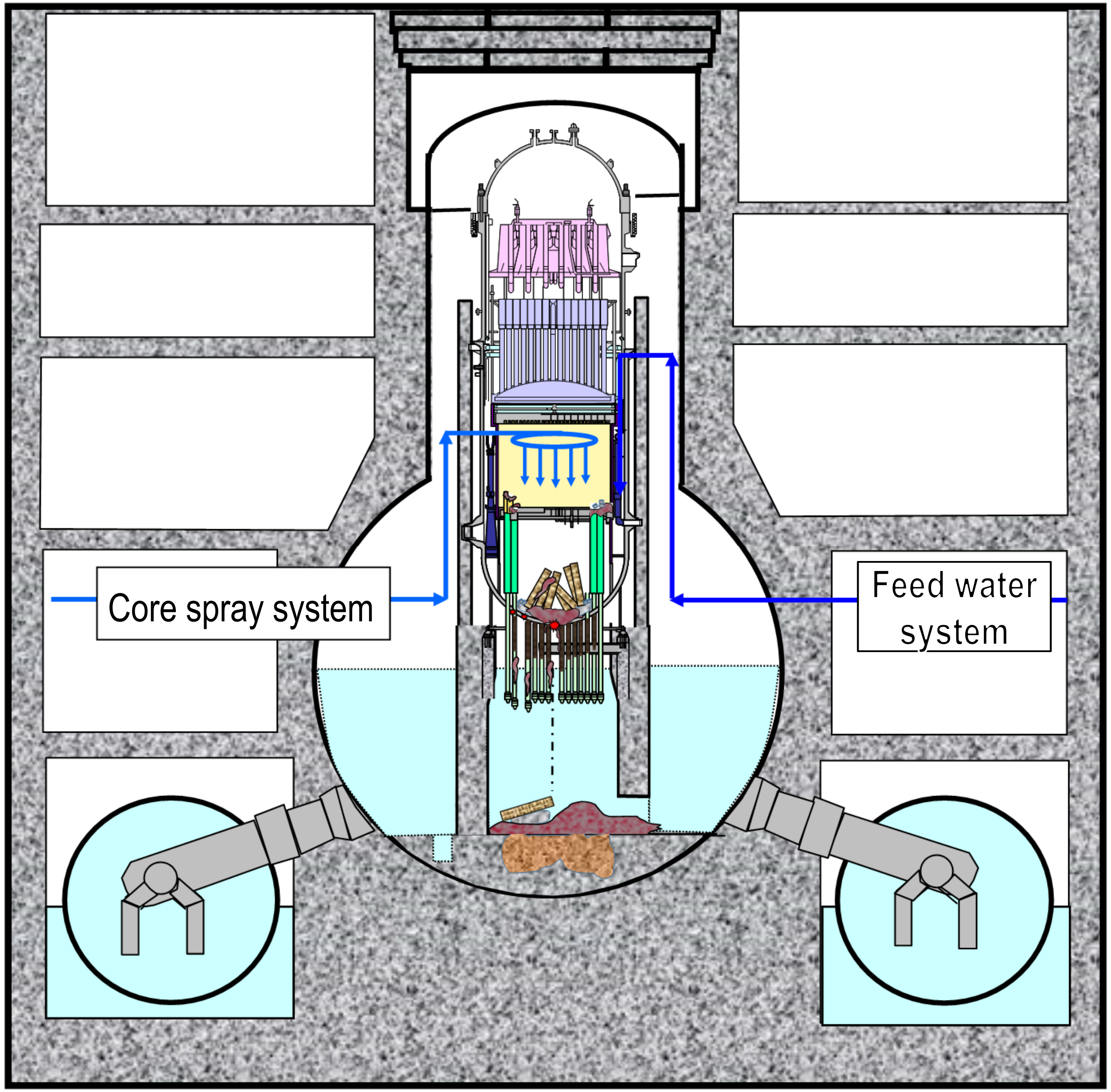
Assumption image of Unit 1 fuel debris distribution

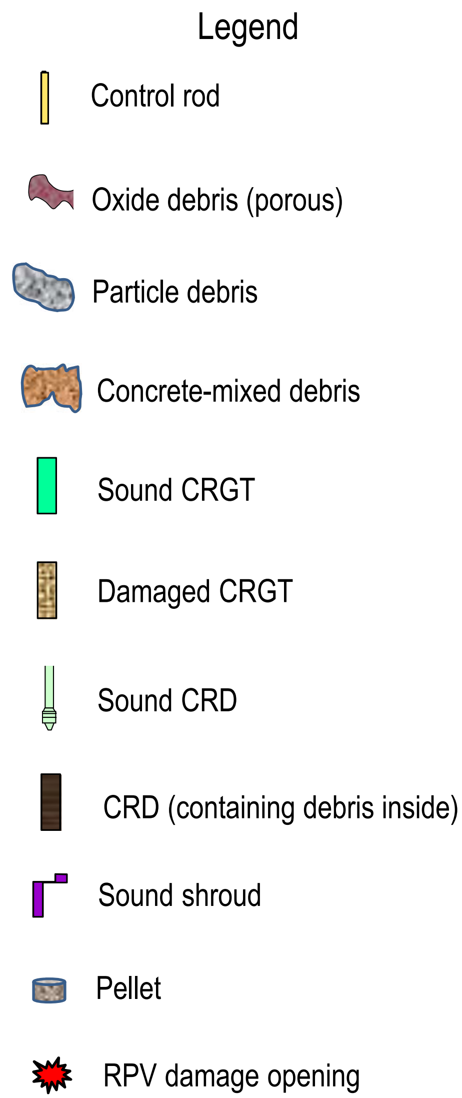


Assumption image of Unit 2 fuel debris distribution



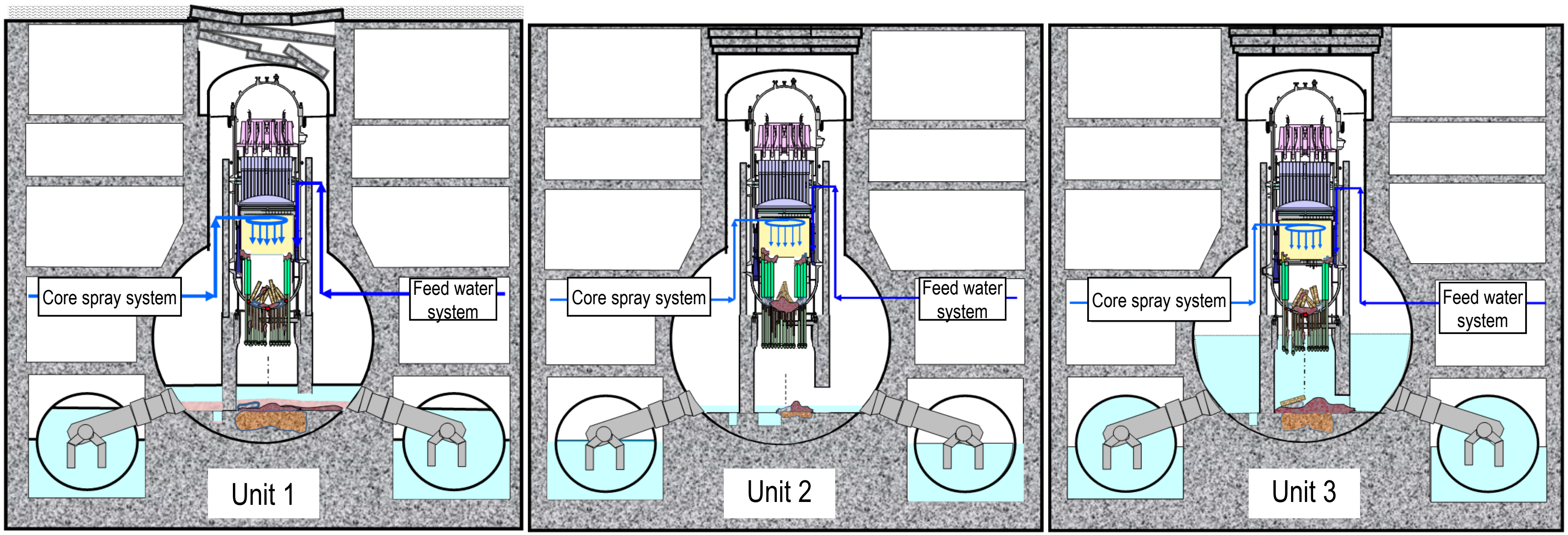
Assumption image of Unit 3 fuel debris distribution





Assumption summary of status inside reactor and PCV

	Core	Lower plenum	PCV	D/W water level	S/C water level
Unit 1	Almost none	Almost none	Most part	2m	Almost full
Unit 2	Little V	Lot V	Little Λ	0.3m	Middle
Unit 3	Little	Little	A certain level	6m	Full



2. Analysis of hydrogen explosion at Unit 1 reactor building

Overview

On March 12, 2011, a hydrogen explosion occurred at the Unit 1 Reactor Building. Based on the record indicating a relatively high dose rate on the 5th floor of the building, the explosion was considered attributable to hydrogen generated inside the reactor, which leaked onto the 5th floor through the PCV top head flange and eventually triggered the explosion. To confirm this assumption, additional examination was conducted regarding the hydrogen explosion.

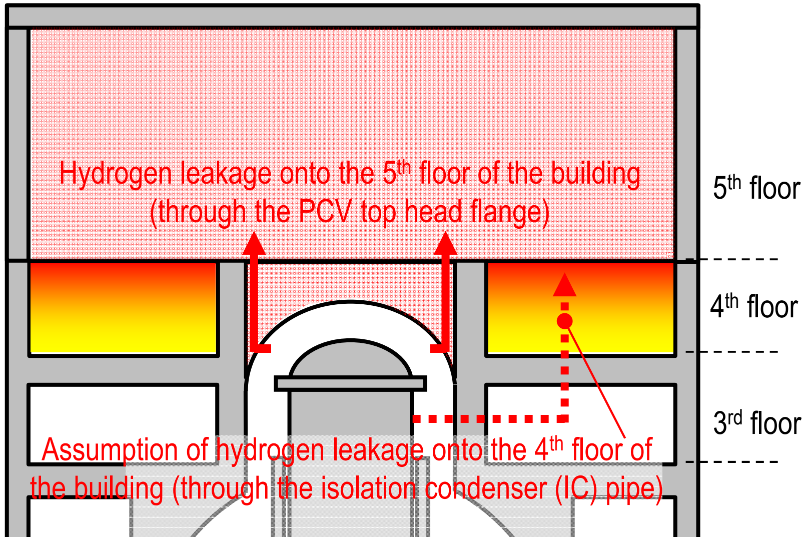


Figure: Image of hydrogen leakage route to the Reactor Building

<Analysis approach>

(1) Two potential scenarios leading to an explosion were assumed: hydrogen leakage onto the 5th floor only and leakage onto both the 4th and 5th floors.

(2) The process of hydrogen spreading inside the reactor building and eventually exploding was evaluated by analysis.

(3) Characteristics of the analytical results were summarized and compared with the damage status of the building, to determine which scenario was more accurate.

<u>The conventional scenario that hydrogen leaked onto the 5th floor of the building floor and triggered the explosion was considered more accurate.</u>

- In the case assuming leakage solely onto the 5th floor, the results were consistent with the damage status of the building.
- The results assuming leakage onto both the 4th and 5th floors were not consistent with the damage status of the building.

Results utilized in safety measures at the Kashiwazaki-Kariwa Nuclear Power Station:
Measures to prevent leakage from PCV / measures to prevent hydrogen explosion

Analysis of hydrogen explosion and its characteristics (overview)

Flow of hydrogen explosion analysis

Set leakage location → Analyze hydrogen distribution → Set the ignition location → Analyze hydrogen explosion → Summarize explosion characteristics

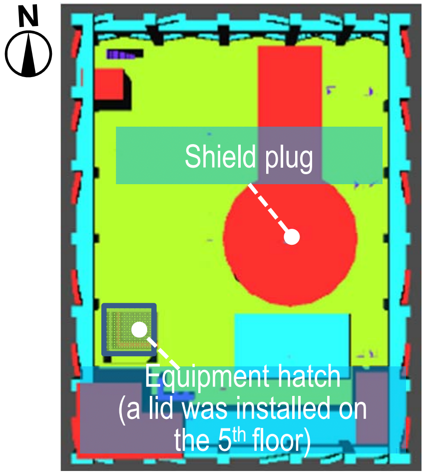


Figure 1: Building 5th floor plan view

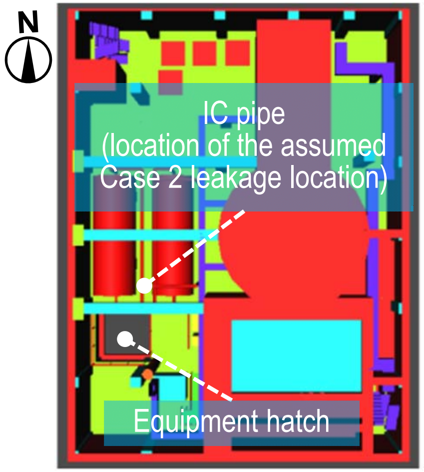


Figure 2: Building 4th floor plan view

	Case 1 **Leakage onto the 5th floor**	**Case 2** **Leakage onto the 4th and 5th floors**
Leakage location	5th floor shield plug	5th floor shield plug and 4th floor IC pipe
Amount of leaked hydrogen	134kg	154kg (Case 1 + 20kg from the 4th floor IC pipe)
Ignition location	5th floor shield plug	Near the 4th floor ceiling (immediately below the equipment hatch lid)
Major characteristics of analytical results	After the equipment hatch* lid was broken, the 5th floor side wall was broken. The blast was generated mainly on the 4th floor and above.	Pressure on the 4th floor rose sharply. An extreme blast was also generated on the 2nd and 3rd floors as well as the 4th and 5th floors.

* Equipment hatch: A hole penetrating from the 1st to 5th floors of the reactor building, used to transport equipment. When the accident occurred, the lid on the 5th floor was closed.

Analysis of hydrogen explosion and its characteristics (Case 1: leakage onto the 5th floor) (1/2)

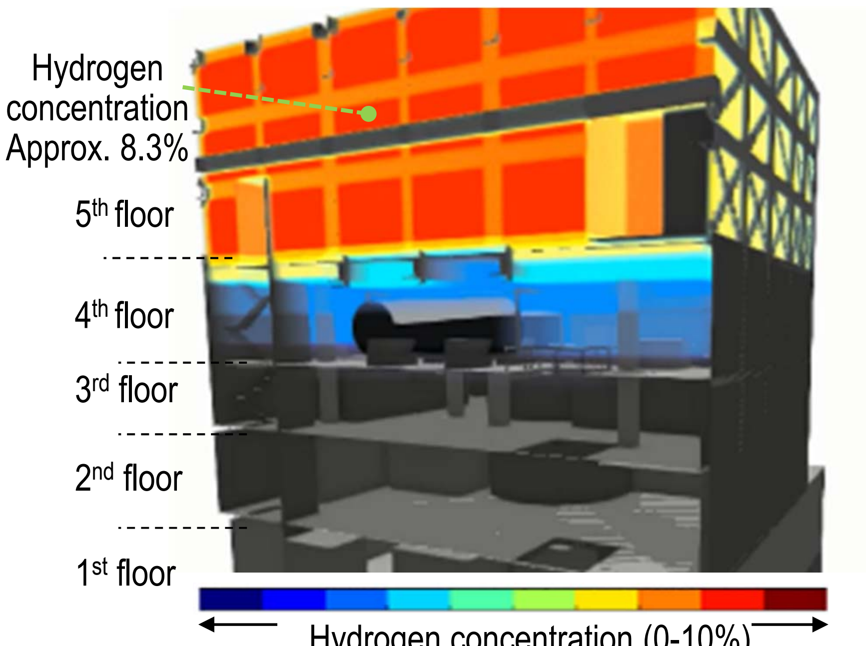


Figure 1: Distribution of hydrogen concentration immediately before explosion

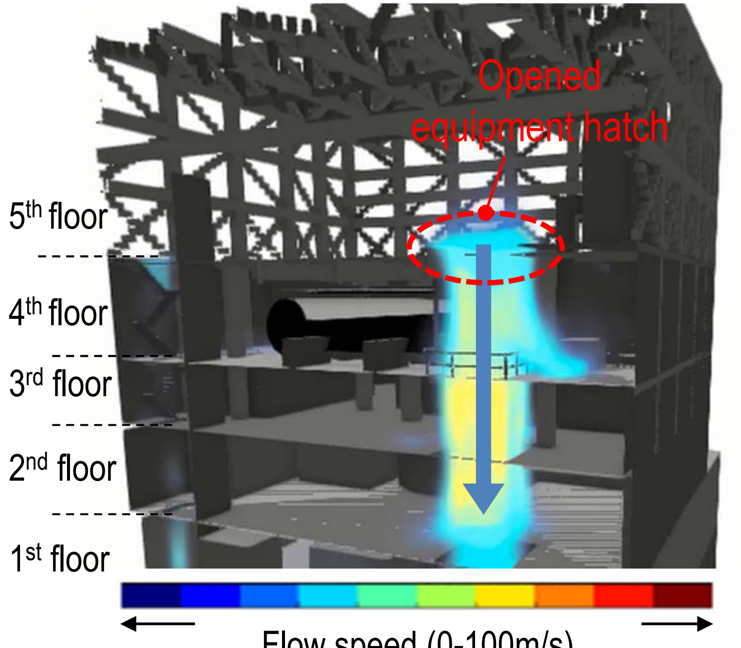


Figure 2: Distribution of blast speed (after the equipment hatch lid was opened)

- Hydrogen was distributed mainly on the 5th floor. Part of the hydrogen went from the 5th floor to the 4th floor via stairs, etc., though the concentration was low (Figure 1).
- The fire developed on the 5th floor and increased pressure on the 5th floor forced the equipment hatch* lid open.
- Through the opened equipment hatch, a downward blast was generated on the 4th floor and below due to pressure difference. The horizontal blast inflow from the equipment hatch was small on the 3rd floor and below (Figure 2).

Analysis of hydrogen explosion and its characteristics (Case 1: leakage onto the 5th floor) (2/2)

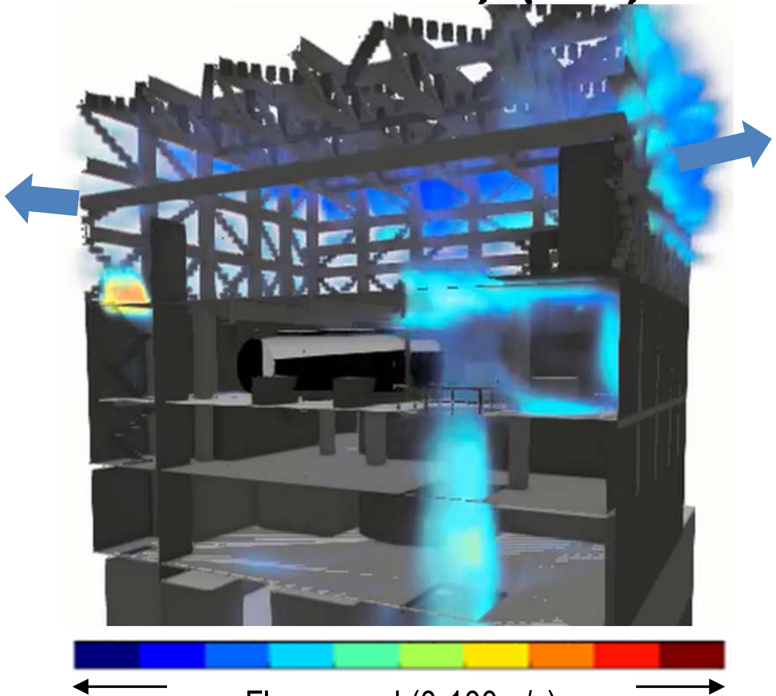


Figure 1: Distribution of blast speed
(when the 5th floor wall side was damaged)

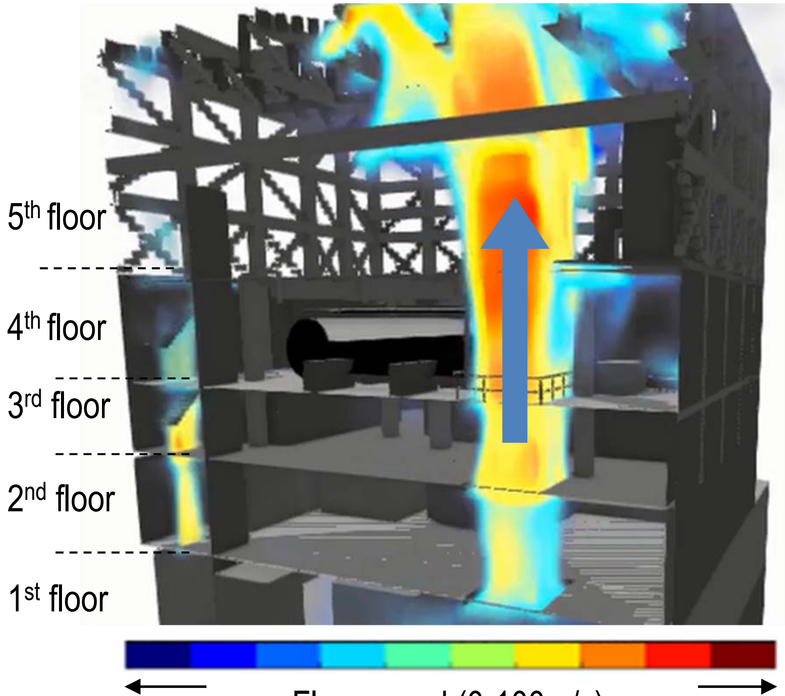


Figure 2: Distribution of blast speed
(after the 5th floor side wall was damaged)

- Increased pressure on the 5th floor subsequently broke the side wall on the 5th floor, which was the weakest load-bearing wall and a horizontal blast was generated (Figure 1).
- After the side wall was broken, pressure on the 5th floor declined, pressure on the 4th floor and below relatively increased and later an upward flow was generated through the equipment hatch (Figure 2).

Analysis of hydrogen explosion and its characteristics (Case 2: leakage onto 5th + 4th floors) (1/2)

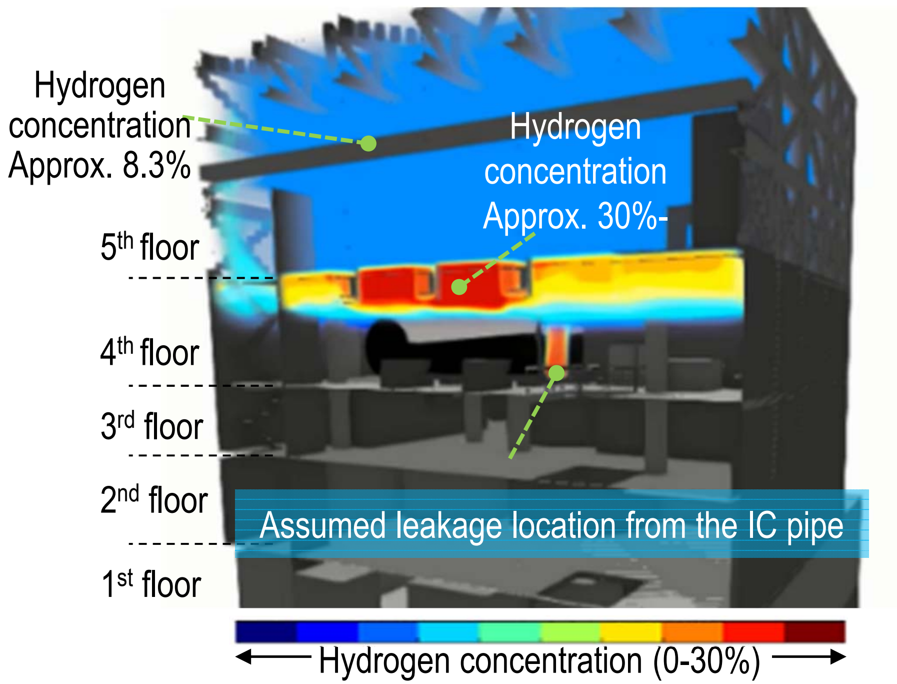


Figure 1: Distribution of hydrogen immediately before the explosion

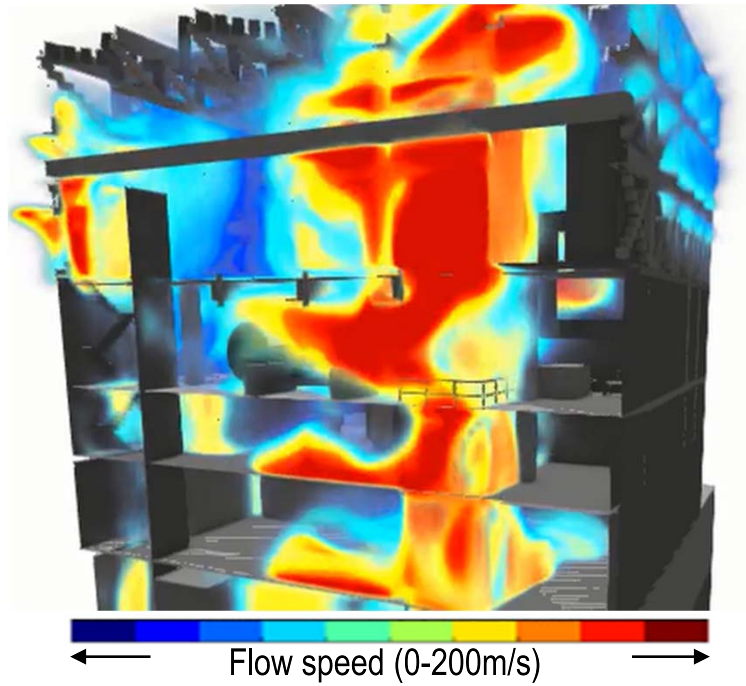


Figure 2: Distribution of the blast speed (when the 5th floor side wall was damaged)

- High hydrogen concentration around the ceiling near the leakage point on the 4th floor. Hydrogen distribution on the 5th floor was similar to that in the leakage case on the same floor (Figure 1).
- A significant fire took hold on the west side of the 4th floor, where a high hydrogen concentration was recorded, which caused pressure to soar. Consequently, the blast speed exceeded that in Case 1 (Figure 2).

Analysis of hydrogen explosion and its characteristics (Case 2: leakage onto the 4th and 5th floors) (2/2)

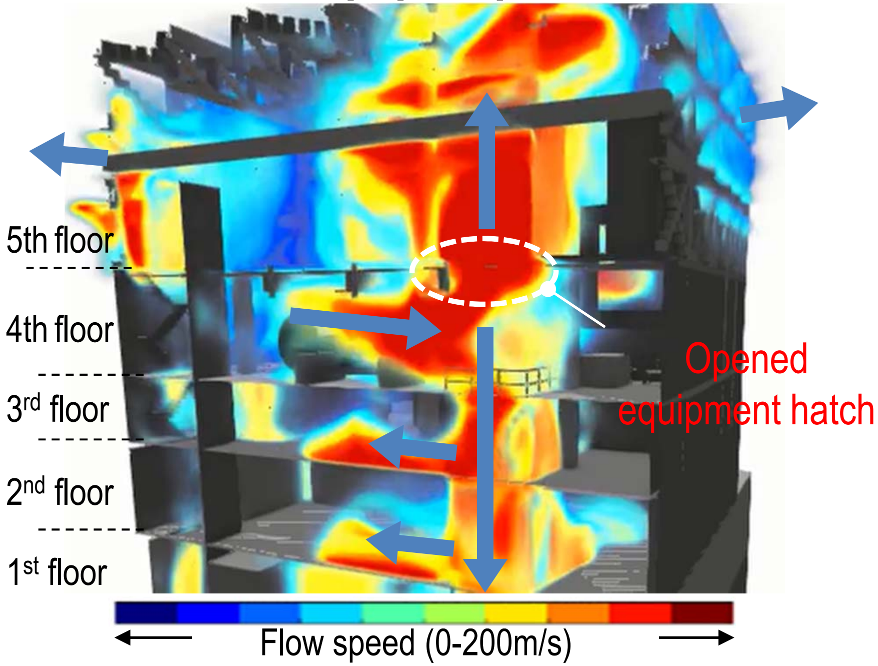
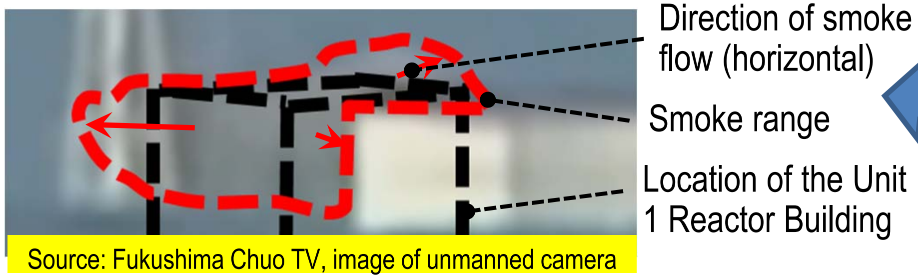


Figure: Distribution of the blast speed (when the 5th floor side wall was damaged)

- On the 5th floor, the equipment hatch lid and the side wall broke simultaneously, creating simultaneous horizontal and upward flows.
- On the 4th floor, where an ignition point was located, following an increase in pressure, a strong blast was generated toward the equipment hatch, which provided an escape route for the air.
- On the 3rd floor and below, a strong horizontal inflow blast was generated from the equipment hatch.

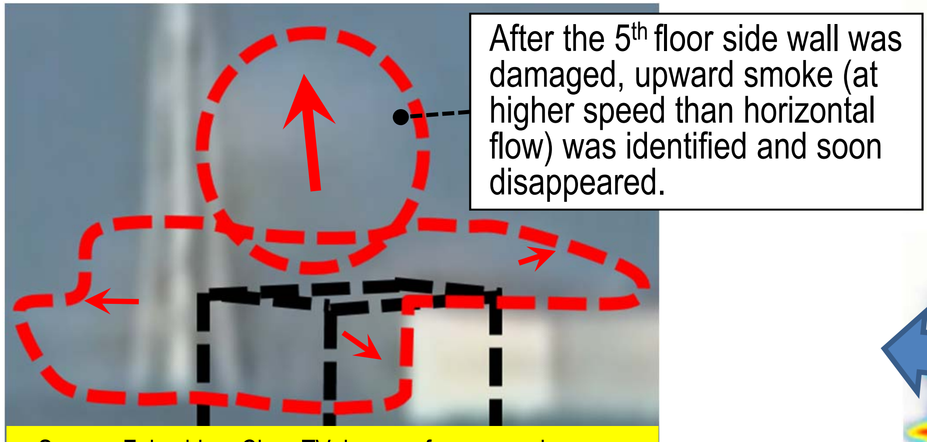
Comparison between analytical results and the damage status (5th floor)

- Images of the unmanned camera showed a horizontal blast when the 5th floor side wall was damaged and later, an upward high-speed blast.
- The analytical results showed that following damage to the 5th floor side wall, a horizontal flow followed by a subsequent upward high-speed flow were generated in Case 1 (the same tendency in the images), while horizontal and upward flows were generated simultaneously in Case 2.



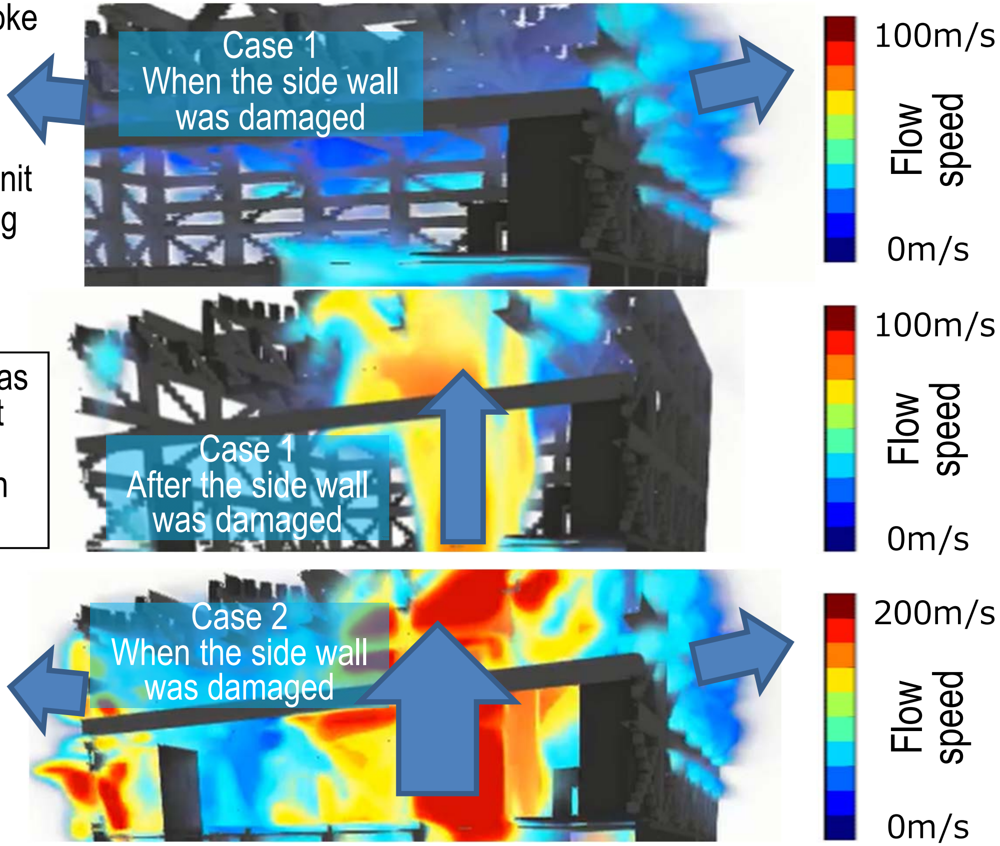
Source: Fukushima Chuo TV, image of unmanned camera

Smoke when the 5th floor side wall was damaged



Source: Fukushima Chuo TV, image of unmanned camera

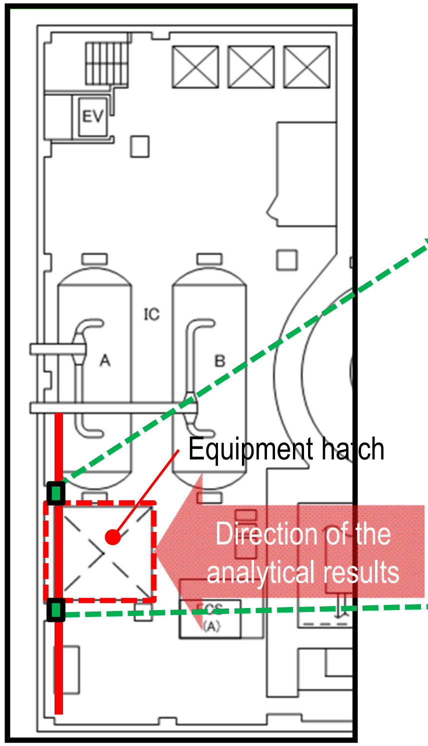
Smoke after the 5th floor side wall was damaged



The images captured the potential upward flow after the side wall was broken, which was indicated in the case of the leakage on the 5th floor.

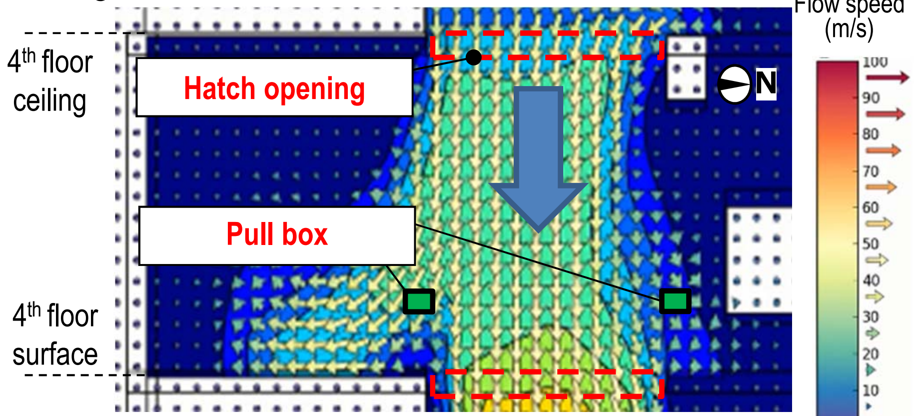
Comparison between analytical results and damage status (4th floor)

- An inspection of pull boxes (metal boxes that integrate and split cables) around the equipment hatch on the west side of the 4th floor identified distortion which was likely due to downward crushing.
- The analytical results concluded that a downward blast was generated at the above point in Case 1 (correlating with the damage status) while a strong horizontal blast was generated in Case 2.

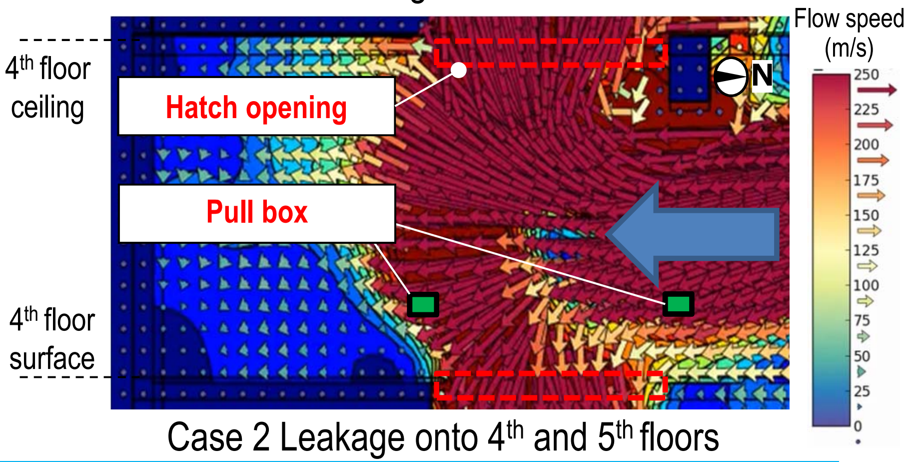


Plan view on the west side of the 4th floor

Distorted pull box (downward distortion)



Case 1 Leakage onto 5th floor



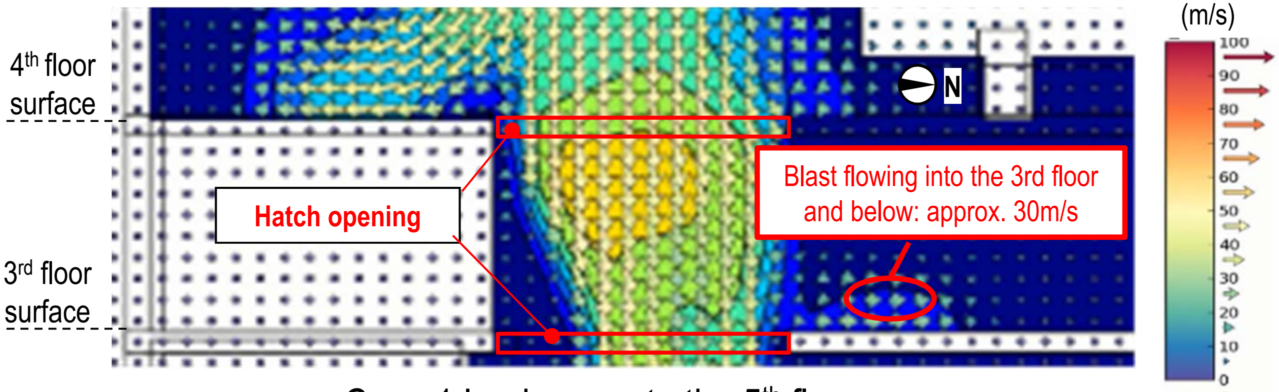
Case 2 Leakage onto 4th and 5th floors

The distortion direction of the pull boxes around the 4th floor equipment hatch was consistent with the direction of the blast in the case of leakage onto the 5th floor. The blast direction in the case of leakage onto the 4th and 5th floors differed from the distortion direction of the pull boxes.

Comparison between analytical results and damage status (3rd floor)

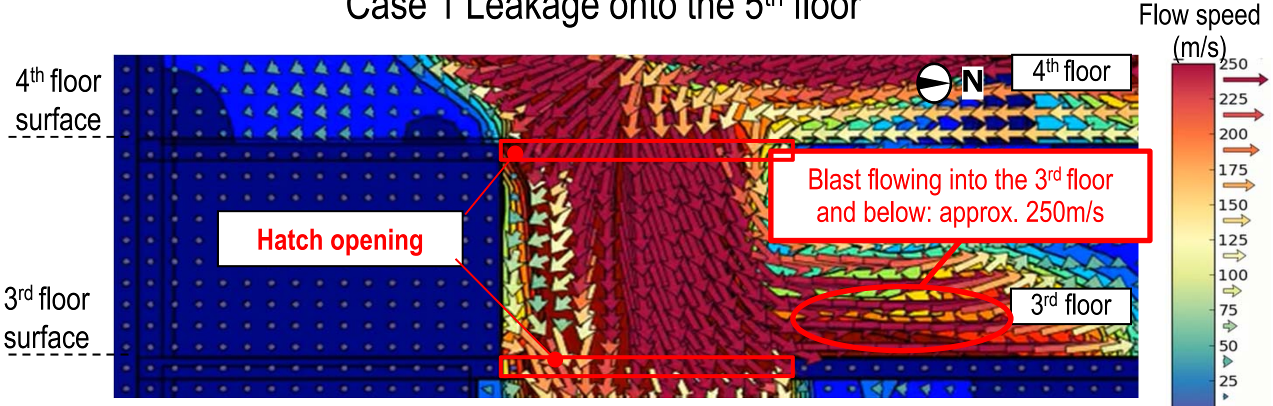
- Minor damage was identified on the 3rd floor and below compared to the 4th floor, with no sign of damage from the strong blast.
- Analytical results showed that the maximum blast speed flowing into the 3rd floor and below was relatively low in Case 1, but significantly high in Case 2.

3rd floor equipment hatch north side



Case 1 Leakage onto the 5th floor

(Comparison)
4th floor equipment hatch north side



Case 2 Leakage onto the 4th and 5th floors

Minor damage was identified on the 3rd floor and below and in terms of the inflow blast speed, it was consistent with the case of leakage onto the 5th floor. The blast in the case of leakage onto the 4th and 5th floors was likely to be excessive.

Comparison between analytical results and damage status (summary)

Floor	Compared damage status*	Consistency between analytical results and damage status	
		Case 1 Leakage onto the 5th floor	Case 2 Leakage onto the 4th and 5th floors
5th	Images of the unmanned camera showed a horizontal blast when the 5th floor side wall was damaged, followed by a subsequent upward high-speed blast.	After the horizontal flow, an upward high-speed flow was generated.	Horizontal and upward flows were generated simultaneously (unlike the actual status)
4th	Distortion which was likely due to downward crushing was identified in the pull boxes around the equipment hatch.	A downward blast was generated at the location.	A horizontal strong blast was generated at the location (unlike the actual status)
3rd and below	No sign of damage from the strong blast was identified on the 3rd floor and below.	The horizontal blast speed flowing into the 3rd floor and below was relatively low	The horizontal blast speed flowing into the 3rd floor and below was significantly high (unlike the actual status)

* A comparison with the analytical results, made separately for damages not indicated in the above table, confirmed that the analytical results in Case 1 were consistent with the damage status.

The conventional scenario of hydrogen leakage onto the 5th floor of the building was considered more accurate.

- Potential leakage from PCV top head flange was again suggested.
  ⇒ Measures to prevent leakage from PCV need to be implemented.
- In addition, measures to keep the hydrogen concentration inside the reactor building, including local hydrogen accumulation, appropriately low in the event of leakage need to be implemented to prevent hydrogen explosion.

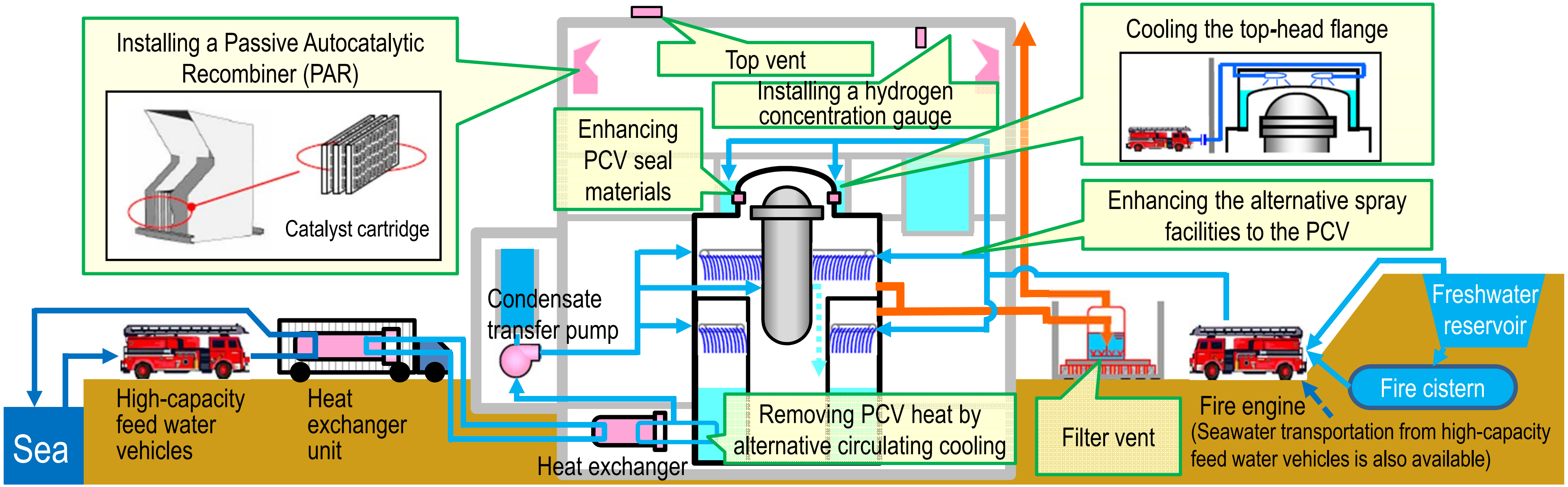
Safety measures in the Kashiwazaki-Kariwa Nuclear Power Station

Measures to prevent leakage from the PCV

- To prevent damage to the PCV by overheating and over pressure, measures are implemented, including enhancing PCV seal materials, cooling the top head flange, strengthening alternative spray facilities to the PCV, alternative circulating cooling and filter vent.
- Pipes penetrating the PCV are designed to be isolated automatically by the isolation valve in the event of an accident.

Measures to prevent a hydrogen explosion

- To detect any hydrogen leakage onto the building promptly, hydrogen concentration gauges are installed at potential leakage points (the building top floor at the end of the PCV top head and small rooms housing the equipment hatch and air lock). Measures to reduce PCV pressure by a filter vent when the hydrogen concentration on the building top floor exceeds the limit value are implemented to prevent further hydrogen leakage onto the building,
- To maintain the hydrogen concentration appropriately low, measures are implemented including installing ***a*** Passive Autocatalytic Recombiner (PAR) on the building top floor and releasing hydrogen from the building by ***the*** top vent. The evaluation concluded that in the event of hydrogen leaking from the equipment hatch and air lock, hydrogen would ingress onto the building top floor via the ventilation duct, etc. and the hydrogen concentration would not reach the flammability limit.



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

Overview

The investigation to date suggests that the loss of emergency AC power supplies was caused by the tsunami. Conversely, some point out the possibility that it was caused by the earthquake because the relationship between the intrusion of the tsunami and the loss of emergency AC power supplies remains unclear. Therefore, for further consideration and to enhance the plausibility of our deduction, we investigated the relationship between the intrusion process of the tsunami and the loss of emergency AC power supplies.

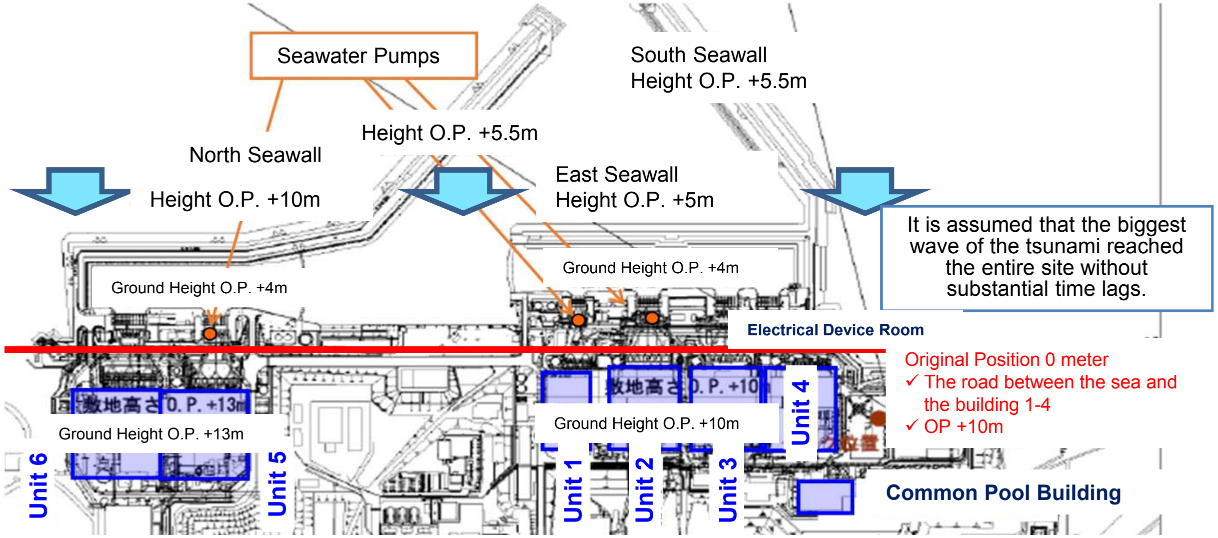


Figure Assumption of Each Reactor Building, the Location of the Seawater Pump, and Tsunami Runup

Assuming that the tsunami caused the loss of power supplies, the time of the water ingress to the power supply equipment of each reactor correlates to the time of its loss of function.
➔As the time of water ingress is considered to correlate with the path length of the tsunami ingress to each power supply system, we investigated the relationship between the location of each system (path length) and the time of its function loss.

The investigation revealed a trend: The longer the path, the later the time of the loss of function.
➔This made the theory that the tsunami caused the loss of power supply systems more plausible.

Reflection on safety measures at the Kashiwazaki-Kariwa nuclear power plant:
Anti-tsunami measures and securing the power supply

See “9. Supplement” of the report for O.P.

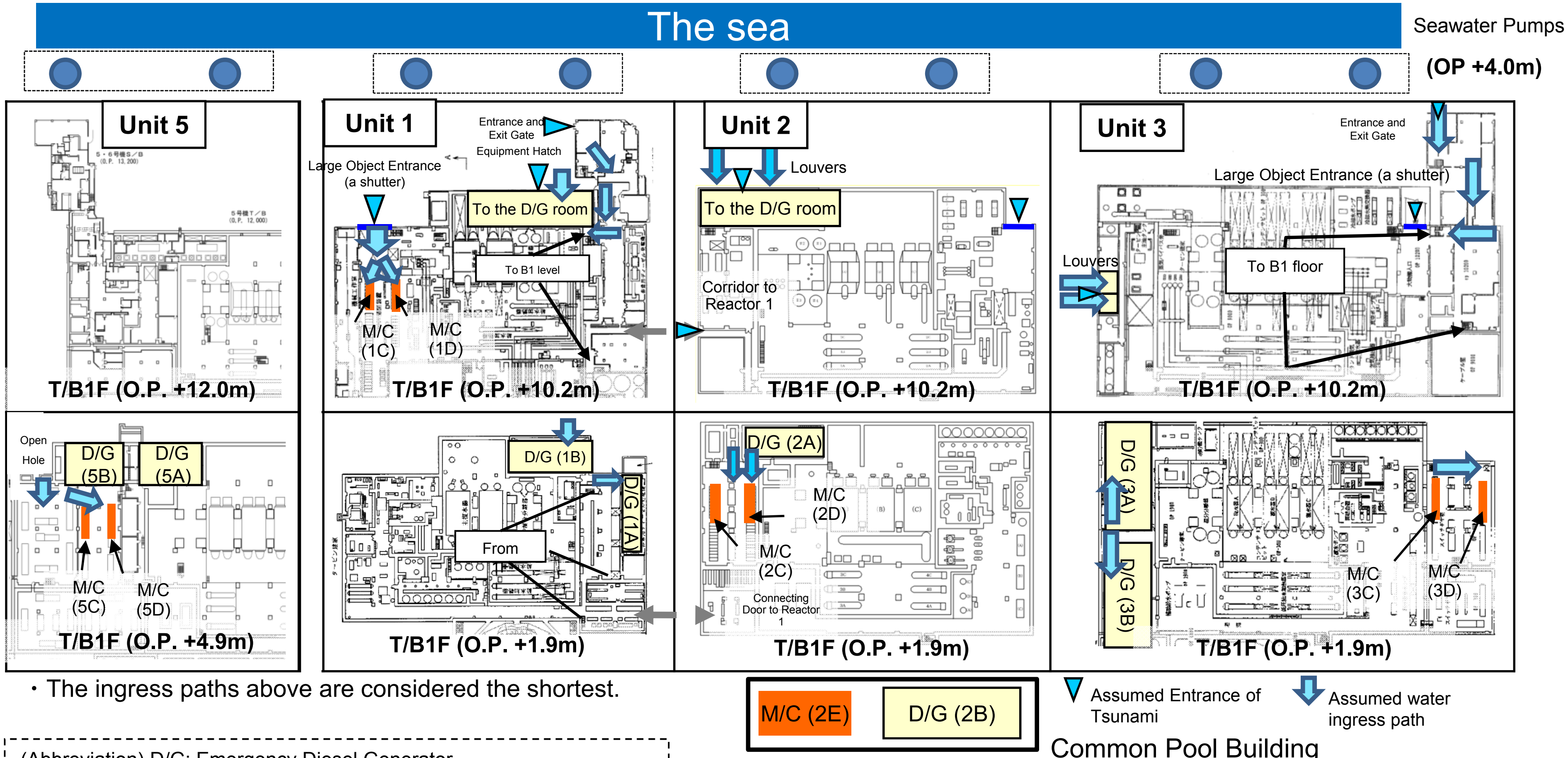
The slide includes contents published in ATOMOΣ, the Journal of the Atomic Energy Society of Japan, in November 2017.

Settings of main Tsunami intrusion path to each system and the time of function loss

The path length to system: Calculated based on drawings with setting reference line, 0m, at the sea side road near the unit 1-4.

✓ To consider the tsunami ingress into the buildings, the path lengths are calculated from the assumed length of the tsunami ingress, rather than straight lines from the standard.

The time of function loss: Obtained from the records of alarm typers



・The ingress paths above are considered the shortest.

(Abbreviation) D/G: Emergency Diesel Generator,
M/C: Emergency High-voltage Power Panel

Figure Main Intrusion Path to Each Unit

(Reference) Assumed main Tsunami intrusion path to each system

<Example path lengths> M/C3C and 3D of unit 3

(The shortest path lengths to the systems are calculated along the red arrow.)

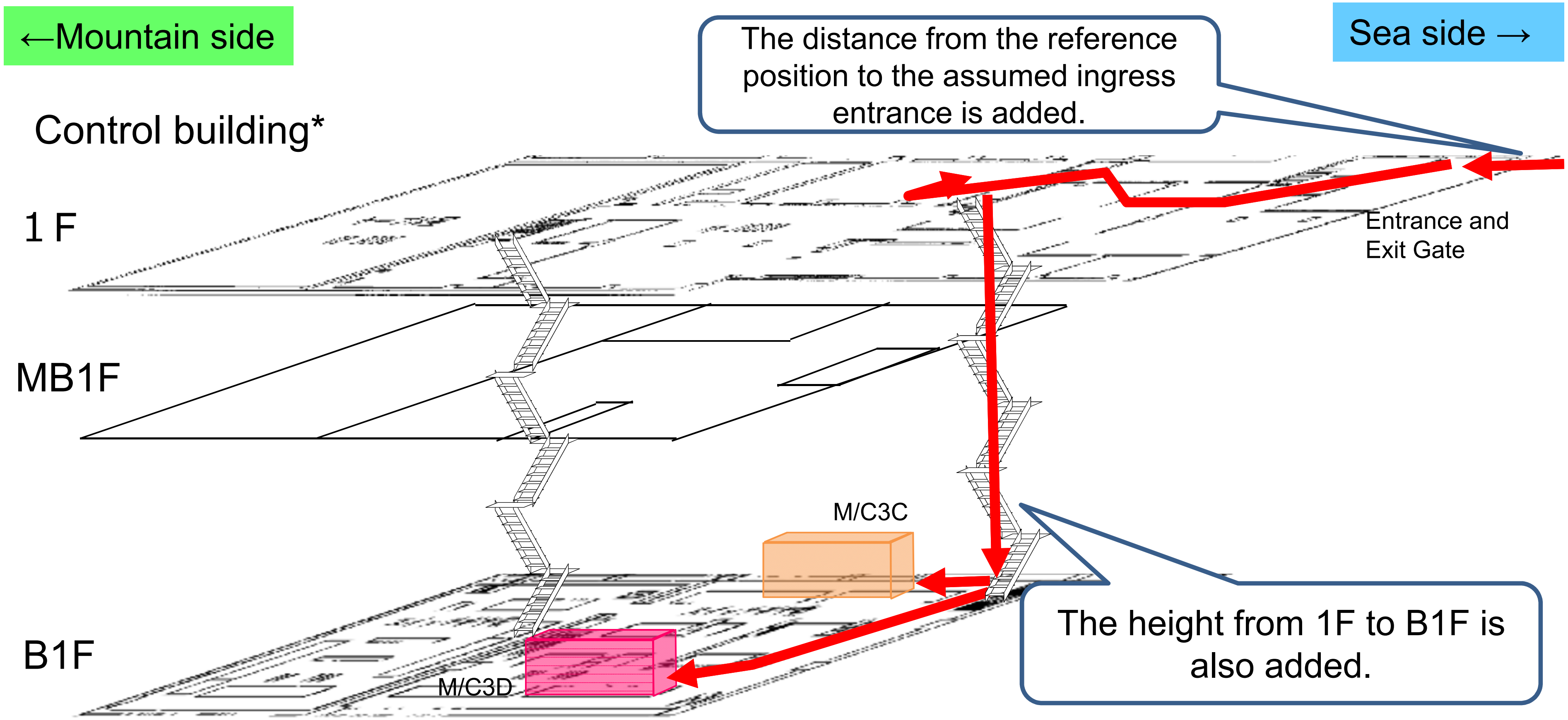


Figure Assumed Main Tsunami Intrusion Paths to M/C3C and 3D of Unit 3.

* The figure shows only the control building because tsunami intrusion path to the M/C3C and 3D of Unit 3 was only in this building. The control building is located next to the turbine building.

Investigative results <the relationship between the location of each system (path length) and the time of its function loss>

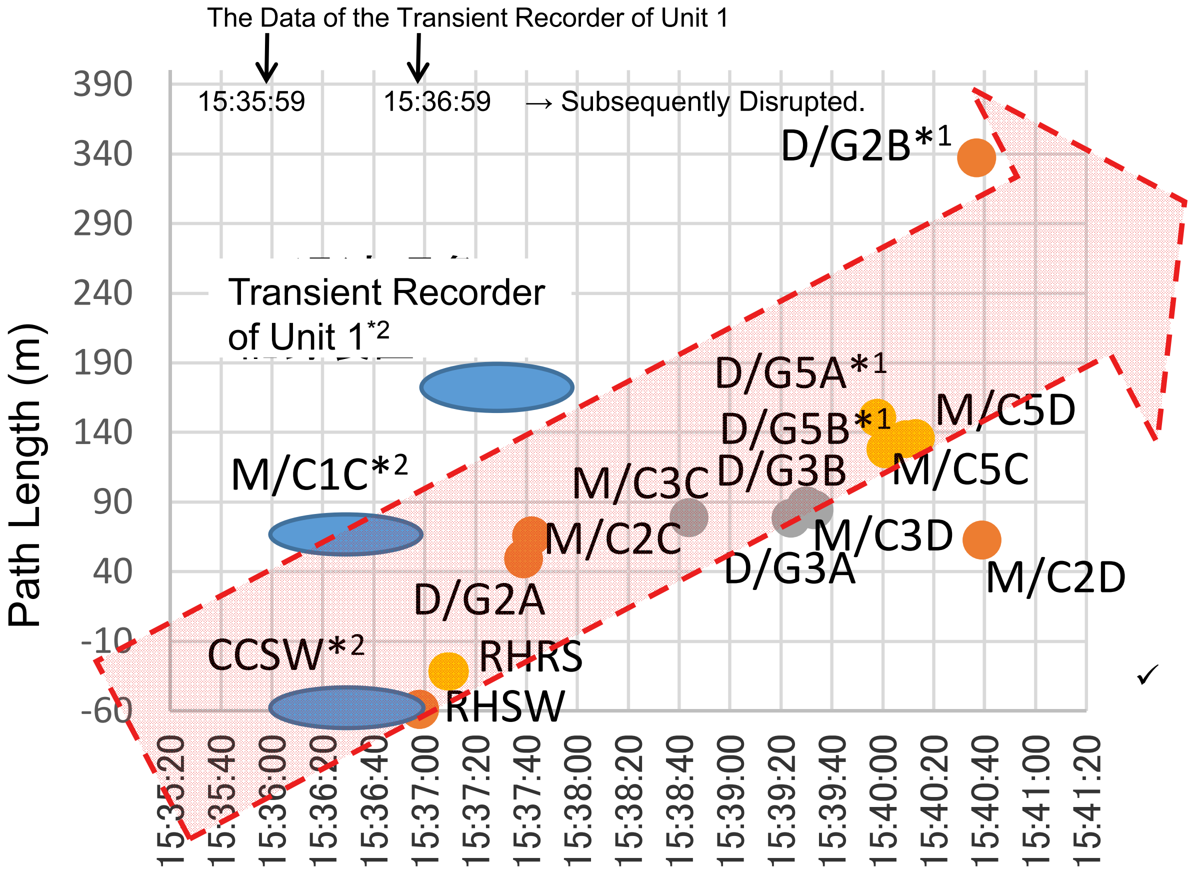


Figure The Relationship between the Location of Each System (Path Length) and the Time of Its Function Loss

*1 The device on which flood marks cannot be confirmed.
*2 The M/C1C, the CCSW, and the transient recorder of Unit 1 are considered to have lost their function within the period of time shown with the plots, because data of the transient recorder of unit 1 was recorded at 1-minute intervals.

✓ The figure shows a tendency: The longer the path length of the tsunami ingress to the emergency power supply system, the later the time at which the function was lost.

The investigation found that the tsunami causing the loss of power supply systems was a more plausible scenario.

(Abbreviation) D/G: Emergency Diesel Generator,
M/C: Emergency High-voltage Power Panel

(Abbreviation) CCSW: Containment vessel Spray Seawater system
RHSW: Residual Heat removal Sea Water system
RHRS: Residual Heat Removal Seawater system

Safety (anti-Tsunami) measures at the Kashiwazaki-Kariwa Nuclear Power Plant

Measures at units 6 and 7 of the Kashiwazaki-Kariwa nuclear power plant as an example

- Anti-tsunami measures are implemented to prevent accidents.
  - ① Tsunami ingress prevention (ground height > the maximum run-up height of tsunamis)
  - ② Prevention of tsunami ingress into the buildings (intake chamber closing plates)
  - ③ Prevention of tsunami ingress into critical equipment areas (watertight doors, waterstop installation on openings, etc.)
  - ④ Seawater securing in backwash (A water-restoring weir)
  - ⑤ Portable equipment storage on high ground
  - ⑥ Installation of tsunami monitoring cameras etc.

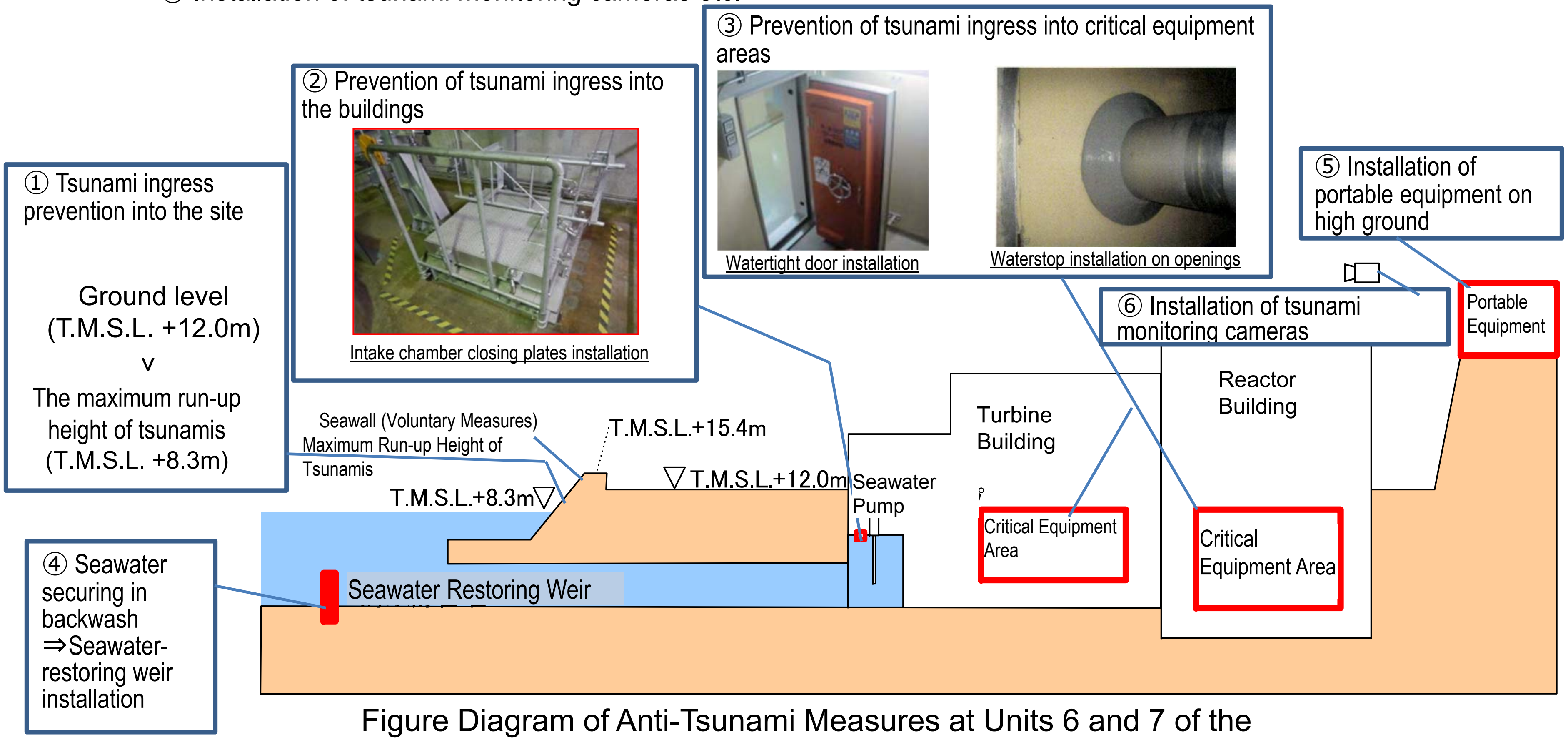


Figure Diagram of Anti-Tsunami Measures at Units 6 and 7 of the Kashiwazaki-Kariwa Nuclear Power Plant

Safety measures at the Kashiwazaki-Kariwa Nuclear Power Plant (power supply securing)

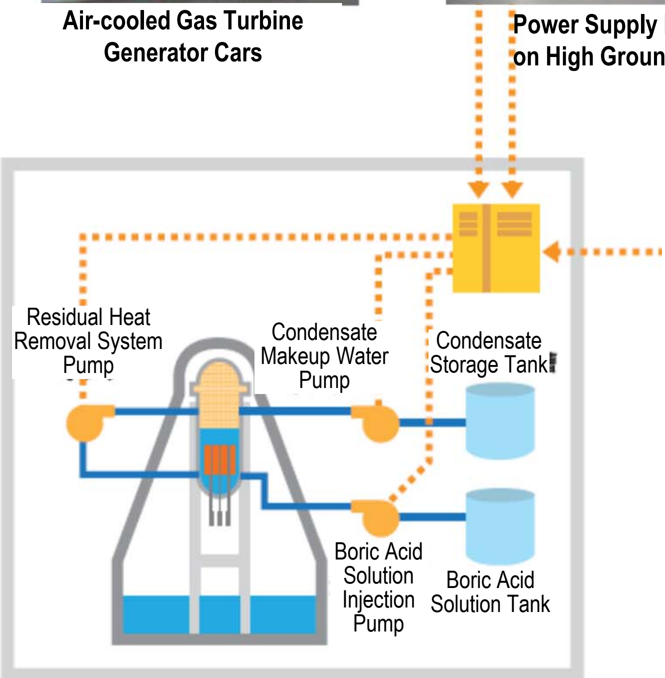
- Power supply means were reinforced to prevent post-accident core damage.
  - Air-cooled gas turbine generator cars, power supply equipment including switchboards, power supply cars deployed on high ground
  - Deployment of spare batteries and additional DC power supply equipment equipped at elevation in the reactor buildings.


Air-cooled Gas Turbine Generator Cars


Power Supply Equipment on High Ground


Training of Connecting a Power Supply Car


DC Power Supply Equipment (Equipped at Elevation in the Reactor Buildings)




Deployment of Power Supply Cars

Figure Overview of Power Supply Securing Measures at the Kashiwazaki-Kariwa Nuclear Power Plant

4. Estimation of reactor water levels at the time when core damage and core melt progressed at Unit-2

Overview

The reactor water level is a key benchmark to determine the accident progression. Although the water level gauge may have failed to indicate an accurate value in the process where the accident caused the temperatures in the reactor and containment vessel to soar, the actual reactor water levels can be estimated by analyzing the indicated values based on the features of the water level gauge. Here, we have estimated how the reactor water level actually changed based on the water level gauge readings during the period of progressive core damage and meltdown (the night of March 14, 2011), on which we have focused to date.

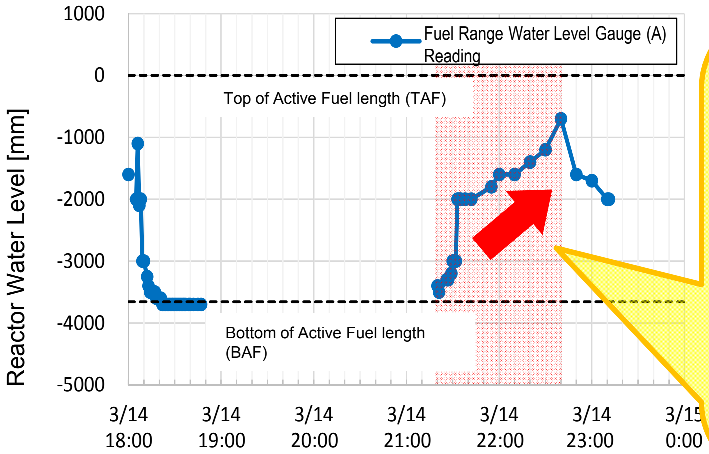


Figure Water Level Gauge Readings of Unit 2

What does the observed increase in water level gauge readings mean?

- The water level could have actually risen because fire engines injected water during the period.
- However, judging from the features of the water level gauge, the reading could also have increased without any increase in the water level.

<The analytical method>

(1) Deduce a scenario of the actual reactor water level change from the values of water level and pressure measured

(2) Estimate the reactor water level ranges by changing the amount of water injection according to the reactor pressure and based on the deduced water level change scenario.

We evaluated the reactor water level ranges from 18:00 to 22:40 on March 14, which corresponds to the period of progressive core damage and meltdown, as follows*:

- The fuel range water level gauge indicated higher values than the actual reactor water levels.
- The increase in the reactor water level wasn't sufficient to fill the core, although fire engines injected water.

* We have already considered this matter for unit 1. The matter for unit 3 was partly considered this time, and we reached the same conclusion as above.

Reflection in safety measures at the Kashiwazaki-Kariwa nuclear power plant

- Measures to inject sufficient water into the reactors without fail (Reinforcement of the depressurization-maintaining function and diversification of water injection means)
- Measures to determine accurate water levels (Thermometer installation to reference rigs, development of water level estimation method used when water levels are uncertain)

Structures of the fuel range water level gauge

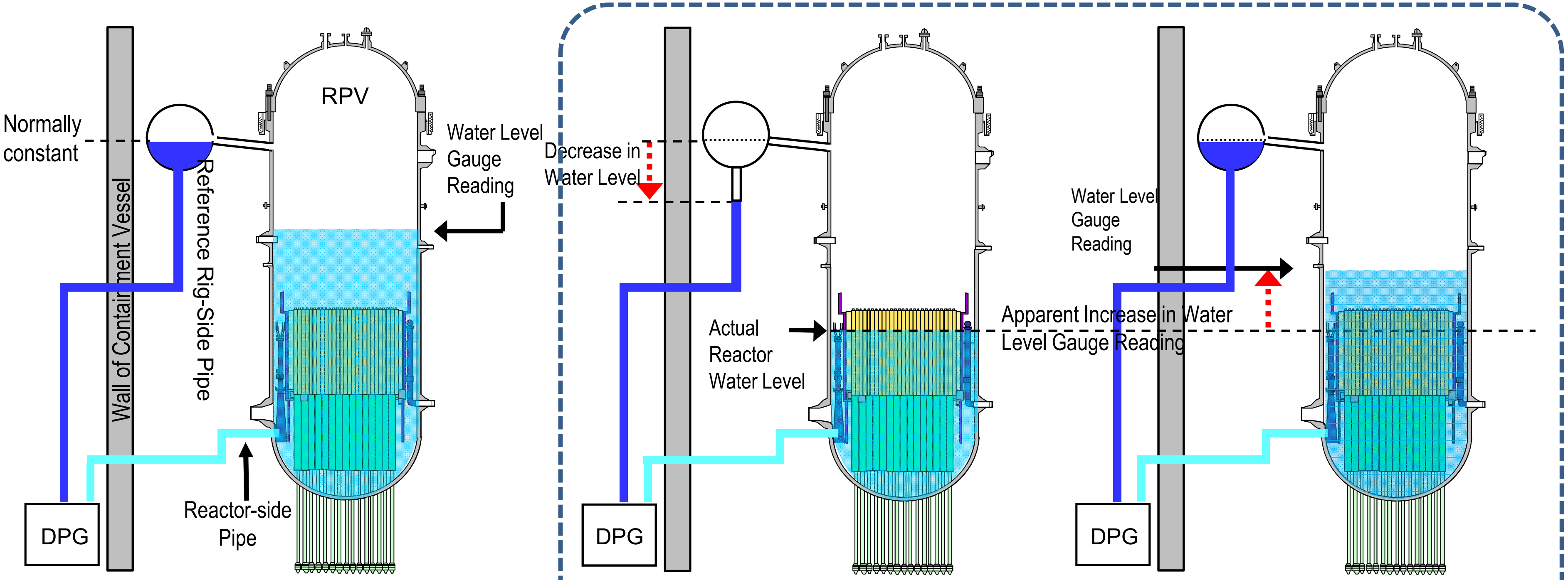


Figure 1 Structures of the Water Level Gauge

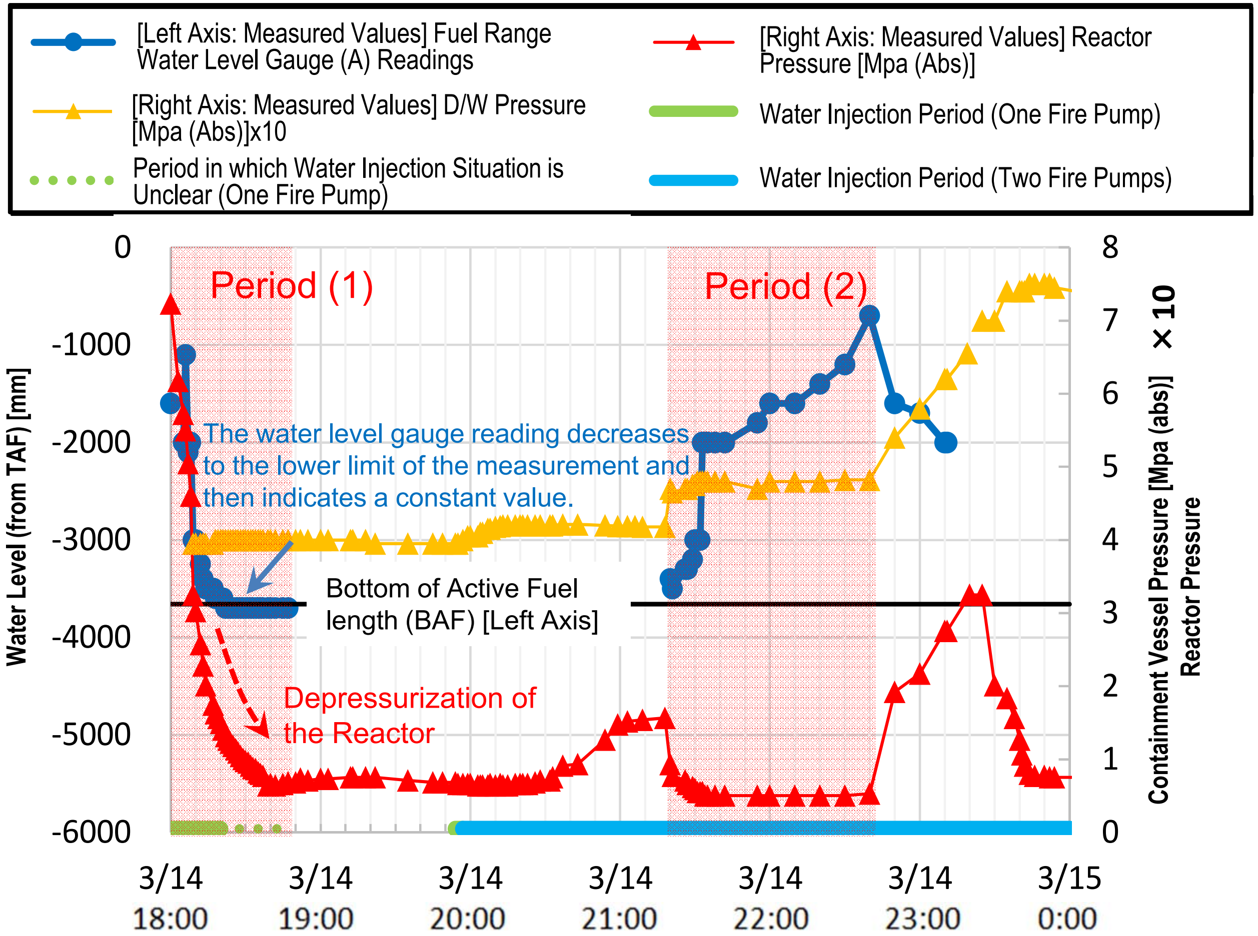
Figure 2 Actual Reactor Water Level

Figure 3 Water Level Gauge Reading

- Reactor water level is calculated from the difference in water head pressure between the reference rig-side pipe and the reactor side-pipe (Figure 1).
- Normally, the water level in a reference rig is always kept constant. The change in the reactor water level change is detected by the water head pressure change of the reactor-side pipe (Figure 1).
- Under grueling conditions, with progressive core damage and meltdown and soaring temperatures in the reactor and containment vessel, as occurred in the Fukushima Daiichi Nuclear Disaster, the water in the reference rig-side pipe evaporates and the water level in the pipe, which is normally constant, decreases (Figure 2).
- Consequently, an accurate reactor water level cannot be obtained. (When the water in the reference rig evaporates, the reactor water level is estimated to be higher than the actual water level (Figure 3).

Scenario deduction of the reactor water level change from the measured values (1/2)

During the period in which the water level decreased as the RCIC system came to a halt, the reactor was depressurized and water was injected under low pressure. We focused on period 1, in which the water level gauge reading decreased and period 2, in which it increased.



Period 1: Decrease in water level gauge reading

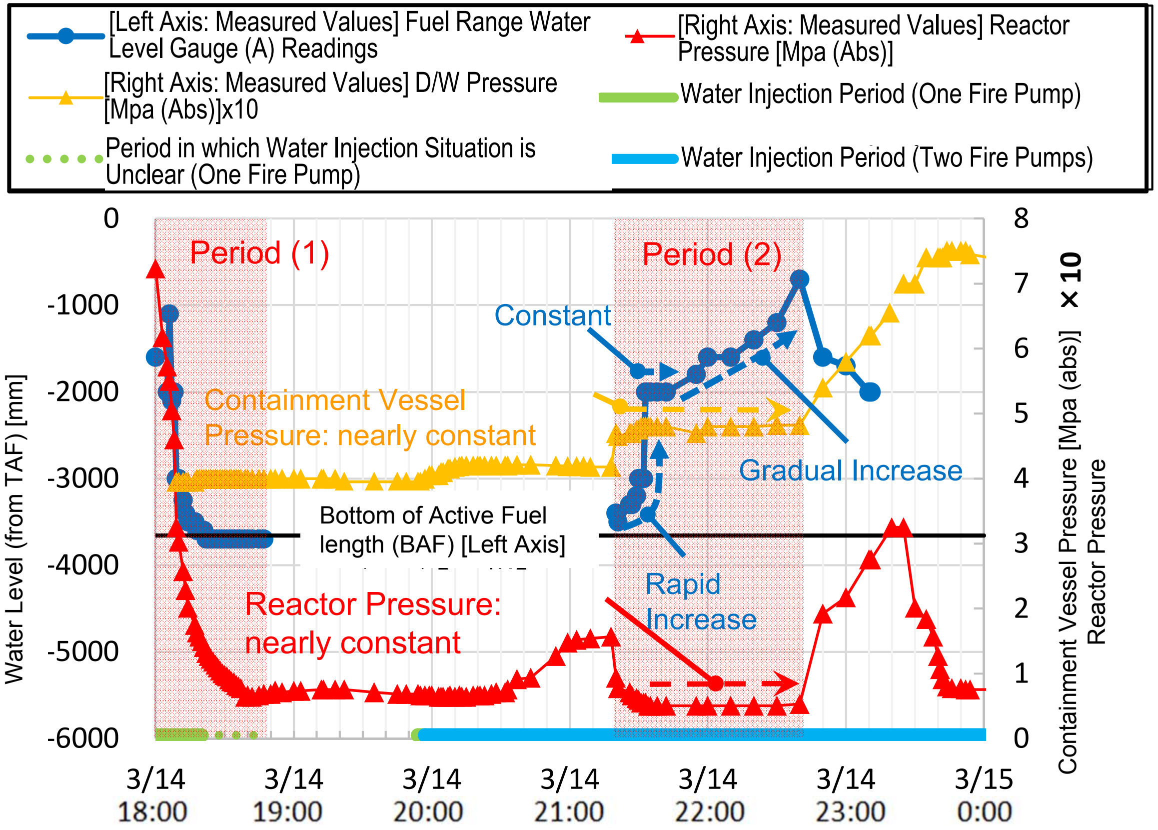
・The water level gauge reading decreases to the lower limit of the measurement (TAF-3700 mm) and then indicates a constant value.

Deduced situations:

・The rapid depressurization caused decompression boiling and around 30 percent of water in the reactor evaporated.

・As estimated from the fuel range water level gauge readings before depressurization, **the actual reactor water level decreased to a level even lower than the water level gauge reading** due to decompression boiling.

Scenario deduction of the reactor water level change from the measured values (2/2)



Period 2: Increase in water level gauge reading

・The water level gauge reading indicates a constant value right after the rapid increase, whereupon it gradually increases.

・Reactor pressure and containment vessel pressure are nearly constant.

Deduced situations:

・Although the amount of water injection increases with decompression of the reactor, the pressure behavior shows no rapid change in the rate of increase of the water level during period 2. Therefore, **we have deduced that water evaporation in the reference leg resulted in the rapid increase of the water level gauge reading.**

・Based on the fact that reactor pressure and containment vessel pressure are nearly constant and that the temperature in the containment vessel shows no sign of rapid increase, **we have conducted that the gradual increase in the water level gauge reading shows an increase in the reactor water level increase by water injection, rather than water evaporation in the reference rig-side pipe.**

・Based on the fact that no increase in pressure due to the generation of water vapor and hydrogen, which is supposed to happen when water comes into contact with a high-temperature fuel, was seen, **we have deduced that the water level didn’t rise to the fuel (BAF: Bottom of Active Fuel length) during the period.**

Deducted scenario of the reactor water level change from the measured values

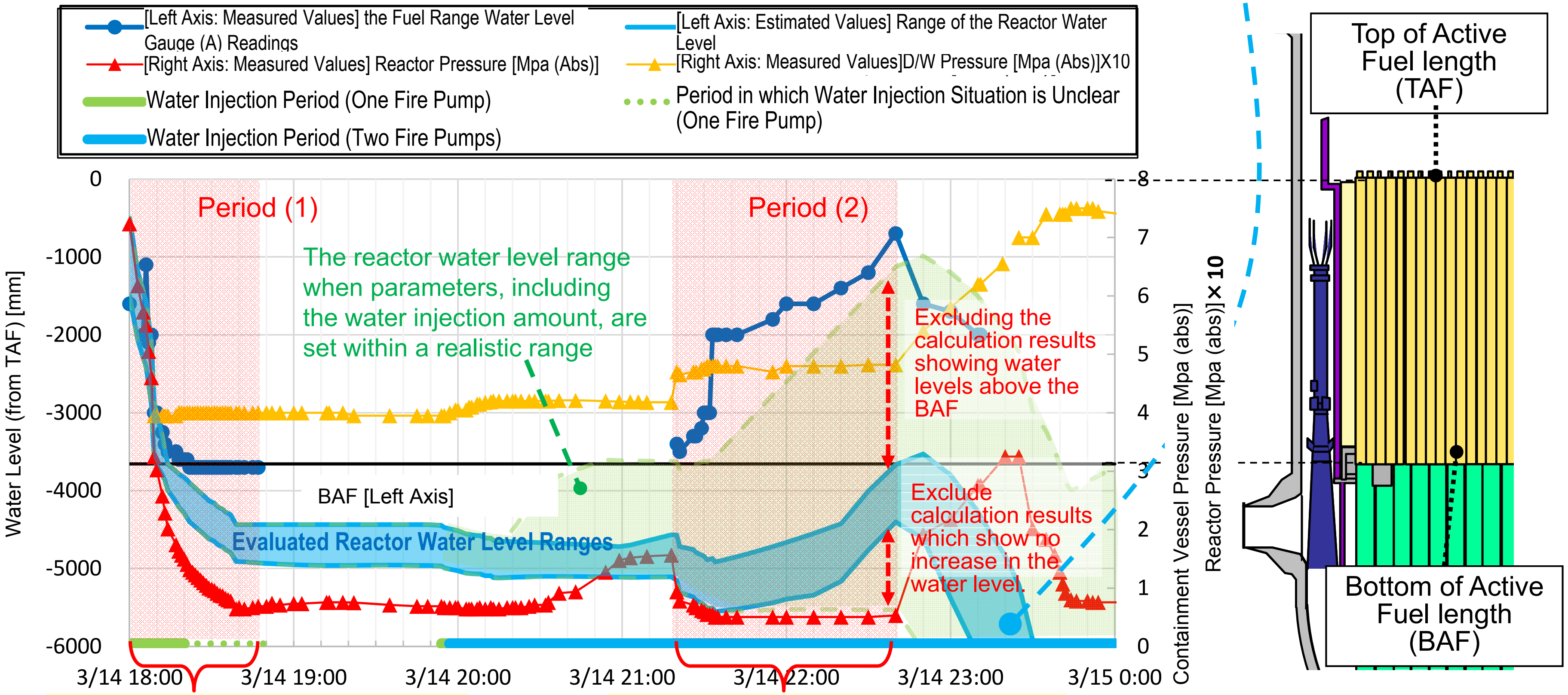
- During period 1, the reactor water level decreased to BAF or below due to decompression of the reactor.
- During period 2, the reactor water level increased because of water injection, but didn’t reach BAF.

Evaluation of the reactor water level ranges (1/2)

4. Estimation of reactor water levels at the time when core damage and core melt progressed at Unit-2

➢ The range of reactor water levels was evaluated by the following procedure:

(1) Set the parameters affecting the water levels, such as the water injection amount, within a realistic range → (2) Calculate the reactor water level → (3) Plot the calculation results which correlate with the deduction*, based on the analysis of measured values over the range of reactor water levels.



* The reactor water level decreased to BAF or below due to the decompression of the reactor.

* The reactor water level increased because of water injection, but did not reach the BAF.

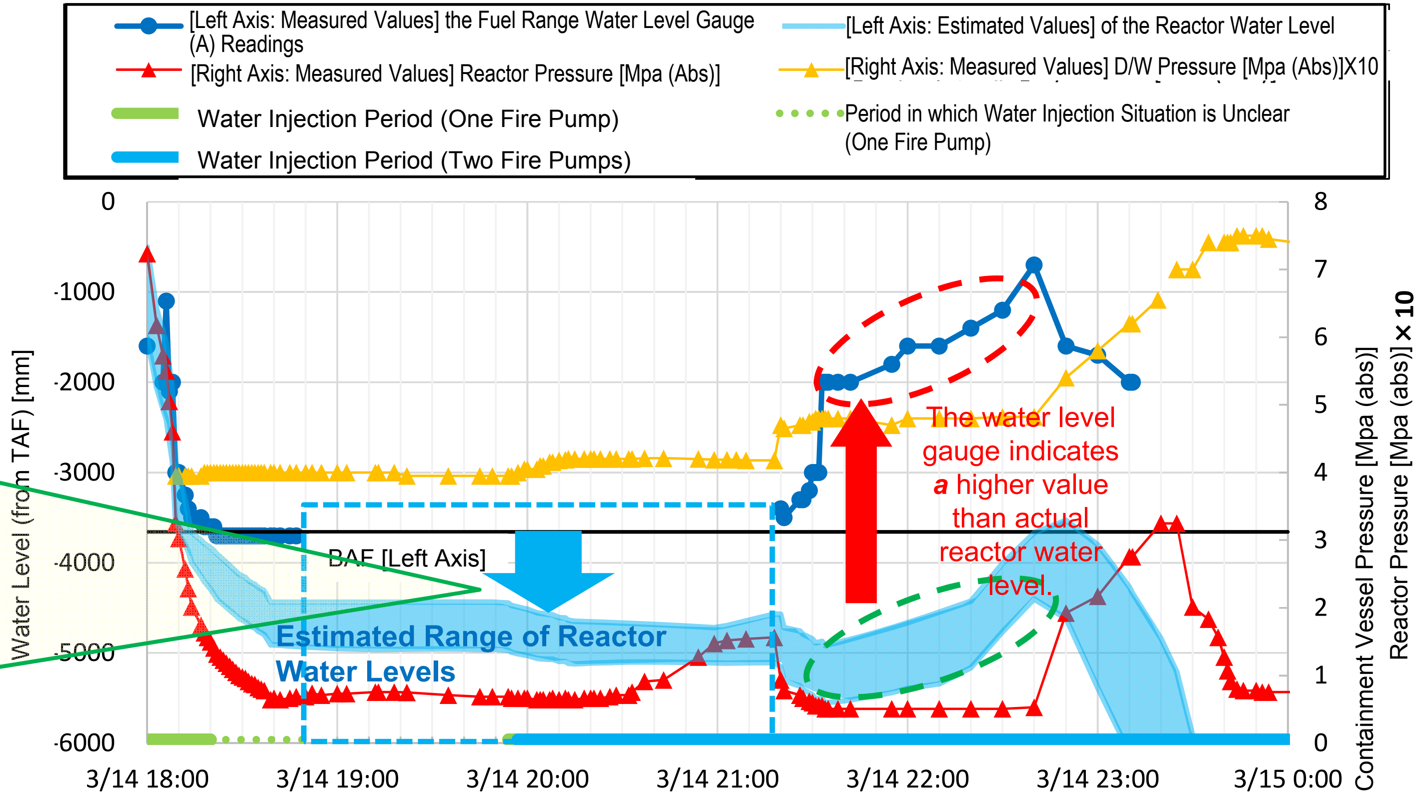
Evaluation of the reactor water levels range (2/2)

4. Estimation of reactor water levels at the time when core damage and core melt progressed at Unit-2

The results show that the reactor water level doesn't reach the BAF either during the period without the measured values.

⇒ Discussion:

- Although a reactor pressure increase was observed from 20:30 to 21:20, no clear explanation can be determined under circumstances where the water level did not reach the BAF.
- Also, under the water level condition used in the evaluation, the potential remains that molten material having dropped to the lower plenum increased the pressure.
- We regard the consideration results as **a scenario with a relatively low reactor water level**.



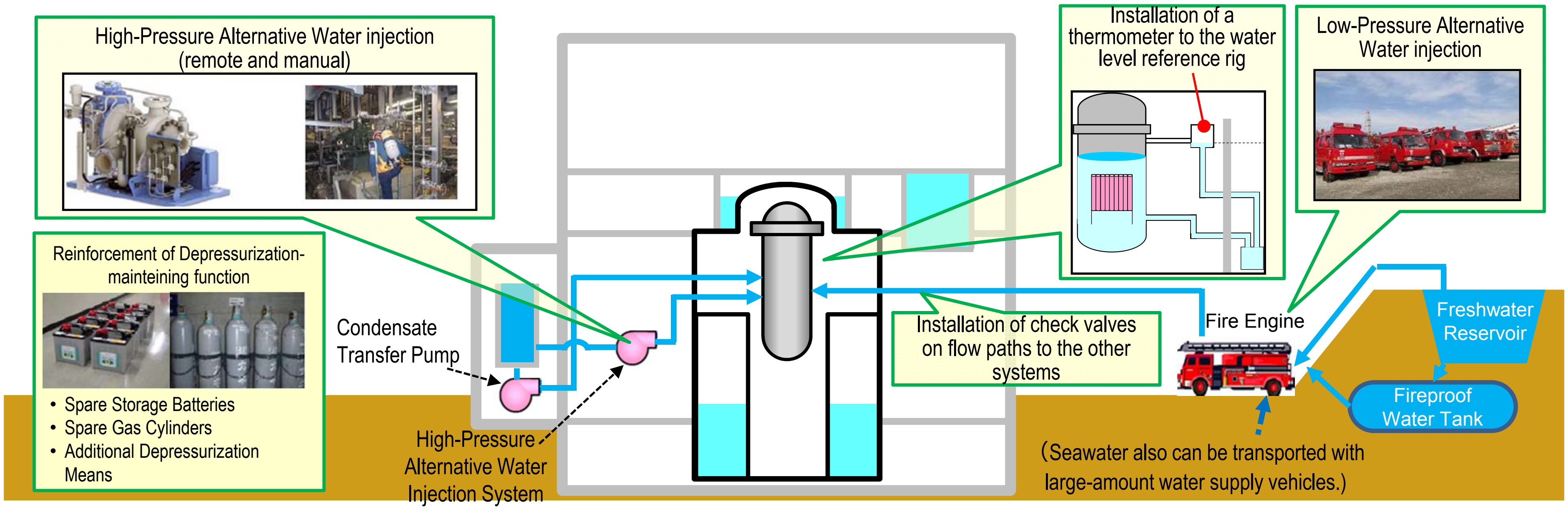
<u>We evaluated the reactor water level ranges from 18:00 to around 22:40 on March 14, which corresponds to the period of progressive core damage and meltdown as follows*:</u>

- The fuel range water level gauge indicated higher values than the actual reactor water level.
- The increase in the reactor water level wasn't sufficient to fill the core, although fire engines injected water.
- The reactor water level (amount of water in the pressure vessel) is a key benchmark to evaluate the generation of hydrogen, melting behavior of fuel and cooling status of fuel debris having dropped into the lower plenum. We will deduce the accident progression based on the estimated water levels.

Safety measures at the Kashiwazaki-Kariwa Nuclear Power Plant

4. Estimation of reactor water levels at the time when core damage and core melt progressed at Unit-2

- We deduced that although water was injected by fire engines, the water level increase was limited.
  ⇒Measures to inject enough amount of water into the reactors without fail are needed.
  - Reinforcement of depressurization-maintaining function: Power supply securing, additional means of nitrogen supply and depressurization
  - Diversification of water injection means: High-pressure alternative water injection (remote and manual) and low-pressure alternative water injection （permanent and portable)
  - Prevention of water injected into the reactor from flowing into other systems: Installation of check valves on flow paths to the other systems
- We deduced that the water level gauge indicated higher values than the actual reactor water level.
  ⇒ Measures to determine accurate water level are needed.
  - Judgement of the credibility of reactor water level gauges: Install thermometers to water level reference rigs (condensation tanks) and when water level in a reference rig is considered not to be maintained, response to the situations where the water level is uncertain.
  - Development of water level estimation methods: Water level estimation using contributing information such as the amount of water injection and thermometers around the reactor.



5. Evaluation of the fraction of Unit-3 vent gas that flowed into Unit-4 reactor building

Overview

5. Evaluation of the fraction of Unit-3 vent gas that flowed into Unit-4 reactor building

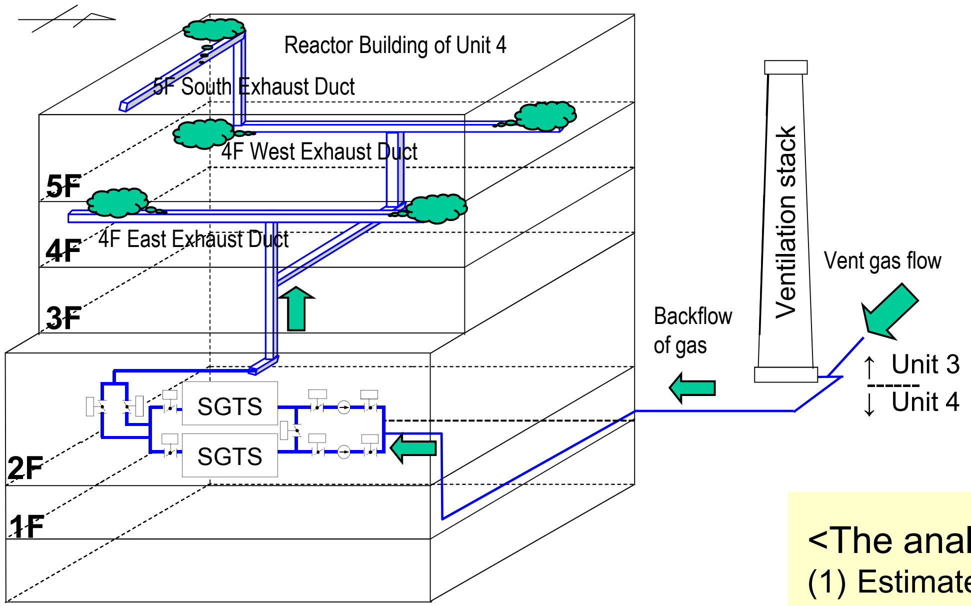


Figure　Inflow route of vent gas to Unit 4

- We deduced that the hydrogen explosion in the reactor building of Unit 4 occurred because some vent gas, including hydrogen, flowed through the pipes of the Standby Gas Treatment System (SGTS) of Unit 4 into the reactor building of Unit 4 in the venting of the containment vessel of Unit 3. (It was already reported in our accident investigation report).
- To understand the hydrogen explosion in Unit 4, we estimated the percentage of vent gas having flown into the reactor building of Unit 4 with an analysis based on the design information, including the vent lines of the Units.

<The analytical method>

(1) Estimate conditions in the containment vessel of Unit 3, including the hydrogen amount based on the plant parameters in the accident.

(2) Evaluate the percentage of vent gas inflow to Unit 4 by the analysis.

We evaluated that about 35% of the vent gas in Unit 3 flowed into the reactor building of Unit 4

- From the amount of hydrogen included in the vent gas, we deduced that a considerable amount of hydrogen flowed into the reactor building of Unit 4 and caused the hydrogen explosion.

\- Reflection in safety measures at the Kashiwazaki-Kariwa nuclear power plant -

- Securing independence of vent pipes (as a measure to prevent vent gas backflow)

Inflow route of vent gas to Unit 4

5. Evaluation of the fraction of Unit-3 vent gas that flowed into Unit-4 reactor building

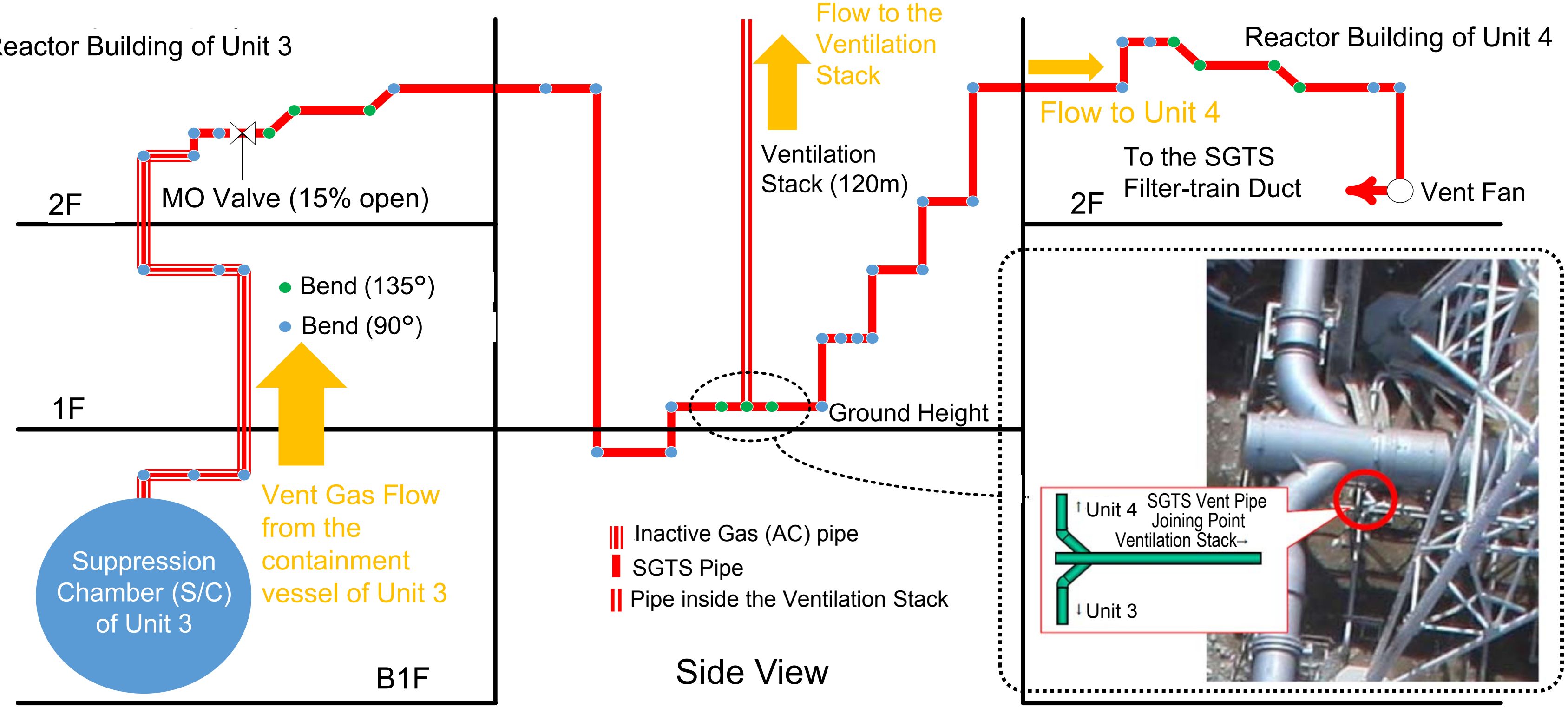


Figure Diagram of Vent Pipes and SGTS Pipes

Causes of vent gas inflow from Unit 3 to the reactor building of Unit 4

① As accident management measures, vent lines were installed in the construction by using a large portion of existing SGTS pipes, which were connected to the duct on the building side and had an opening inside the building.

② SGTS pipes of Units 3 and 4 were connected to the shared ventilation stack.

③ All the valves of the SGTS pipes of Unit 4 were open because all AC power supplies were lost. (Fail-open design)

* Note that no dampers to prevent backflow had been installed downstream of the SGTS filter train. Although dampers do not eliminate backflow completely, their absence is considered to have increased the amount of inflow to the building.

Estimation of conditions in the containment vessel based on measured values

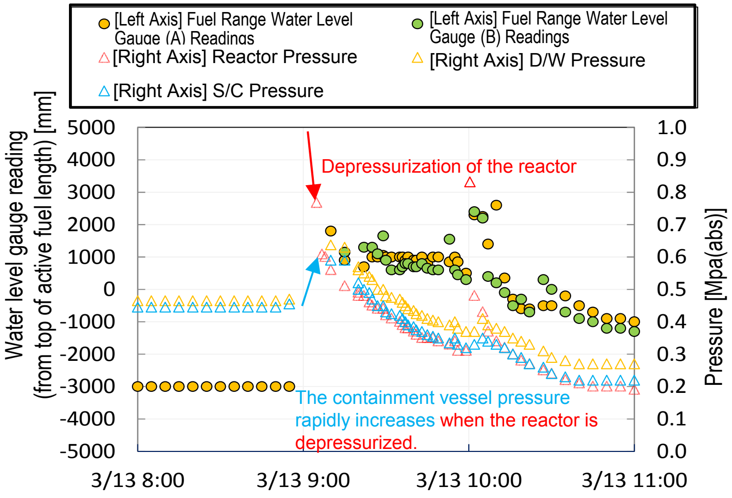


Figure 1　Parameters at the Start of Containment Vessel Vending

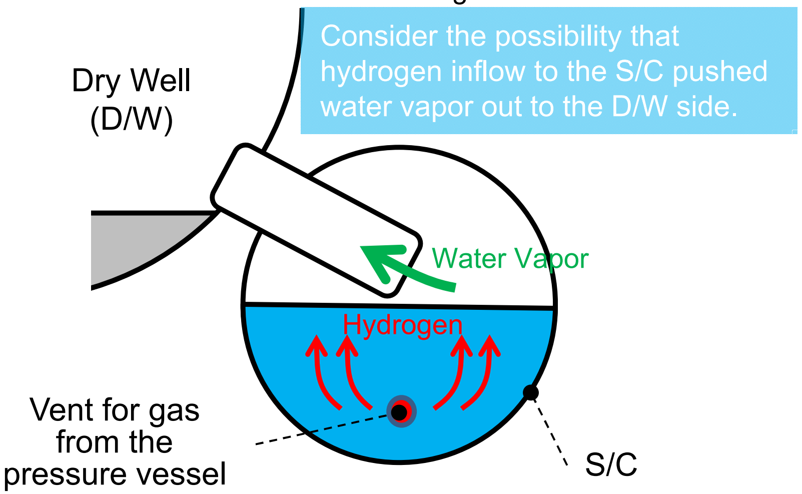


Figure 2　Vapor behavior in the S/C in hydrogen inflow

- Settings of the initial state in the containment vessel of Unit 3
  - ➢ We evaluated the conditions (temperature and gas composition) in the containment vessel of Unit 3 at the start of venting based on the plant parameters before and after containment vessel venting (Figure 1: reactor pressures, containment vessel pressures and fuel range water level gauge readings)

- Settings of the range of hydrogen in the containment vessel of Unit 3
  - ➢ Judging from the rapid pressure increase in the containment vessel during the reactor depressurization, it is likely that hydrogen was transferred from the RPV into the S/C, increasing the hydrogen concentration in the S/C (Figure 2).
  - ➢ Based on the measured containment vessel pressure, we simulated 2 extreme cases with the ratio of hydrogen to water vapor in the S/C as parameters as follows:*
  - ◆ <u>Case A: The hydrogen amount in the S/C is small.</u> Water vapor at saturation vapor pressure remains in the S/C after hydrogen inflow (the amount of hydrogen in the containment vessel is about 910 kg).
  - ◆ <u>Case B: The hydrogen amount in the S/C is large.</u> The S/C is filled with only hydrogen due to hydrogen inflow (the amount of hydrogen in the containment vessel is about 1410 kg)

* We assumed the S/C water level in the venting to be half the height of the S/C, which is close to the normal level, because the actual S/C water level is unknown.

Analysis of vent gas inflow to Unit 4

Taking pipe lengths, pipe diameters and the influence of bends into consideration, we analyzed the percentage of vent gas which flowed from Unit 3 to Unit 4 as a proportion of total vent gas in Unit 3 and the amount of hydrogen inflow by using the thermal hydraulic analysis code GOTHIC in both the case A (the hydrogen amount in the containment vessel of Unit 3 at the start of venting is small) and case B (the amount is large).

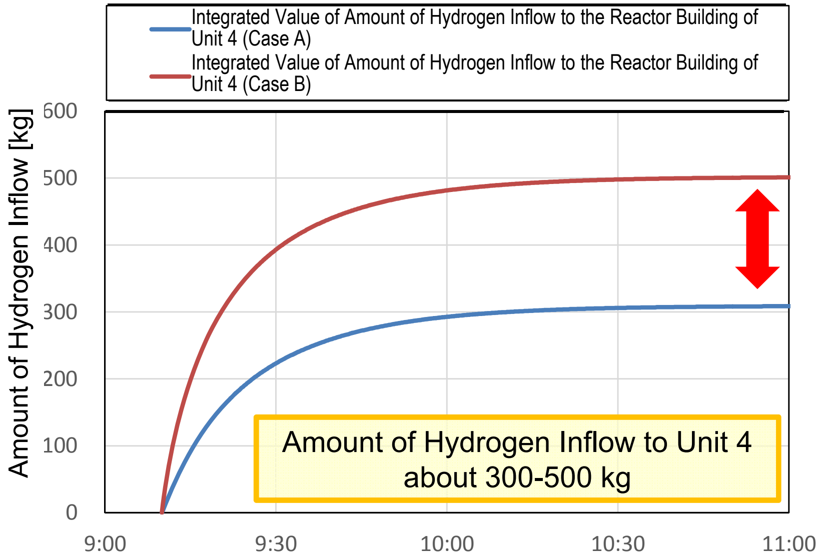


Figure 1 Amount of Hydrogen which Flowed into the Reactor Building of Unit 4

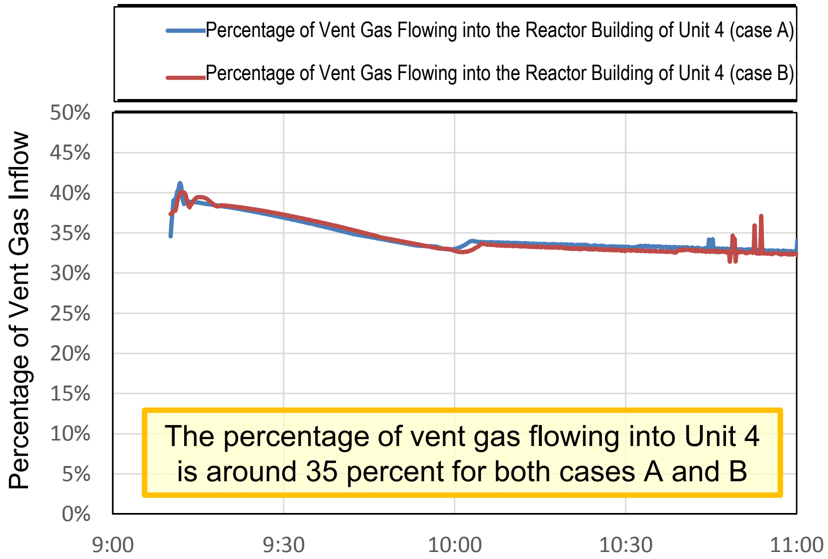
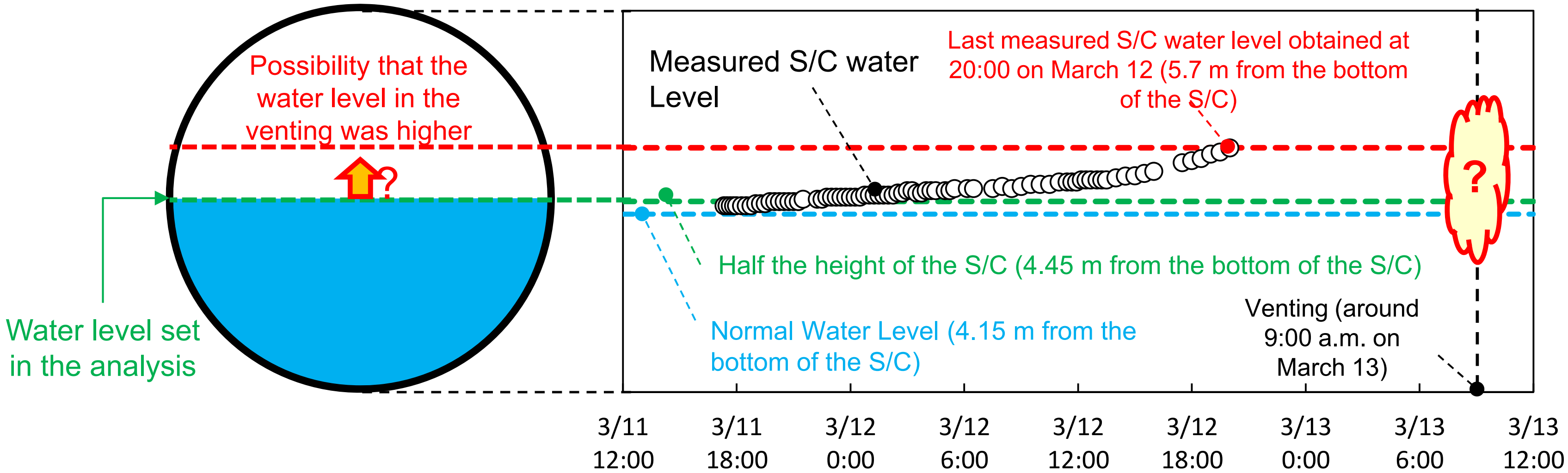


Figure 2 Percentage of vent gas flowing into the reactor building of Unit 4

Regardless of the partial pressure of gas in the containment vessel, the percentage of vent gas flowing into the reactor building of Unit 4 is almost the same (about 35 %).

Uncertainty in the analytical results of the amount of hydrogen inflow



- In the analysis, we assumed that the S/C water level at the time of venting was half the S/C (≒normal water level). However, although the measured values don't exist, there is a possibility that the water level in the venting was higher because of the S/C spray and for other reason***s*** (Figure above).
- If the S/C water level in the venting is as high as the last obtained measured value, the space in the S/C will be around 35 percent smaller than determined in the analysis, which means the hydrogen amount is estimated to be smaller correspondingly. Therefore, the analytical result, which shows a hydrogen inflow to Unit 4 of about 300-500 kg, is uncertain.
- We consider that if the plausibility of the S/C water level changes and the amount of hydrogen generation in the accident increases in future, this will help reduce uncertainty.

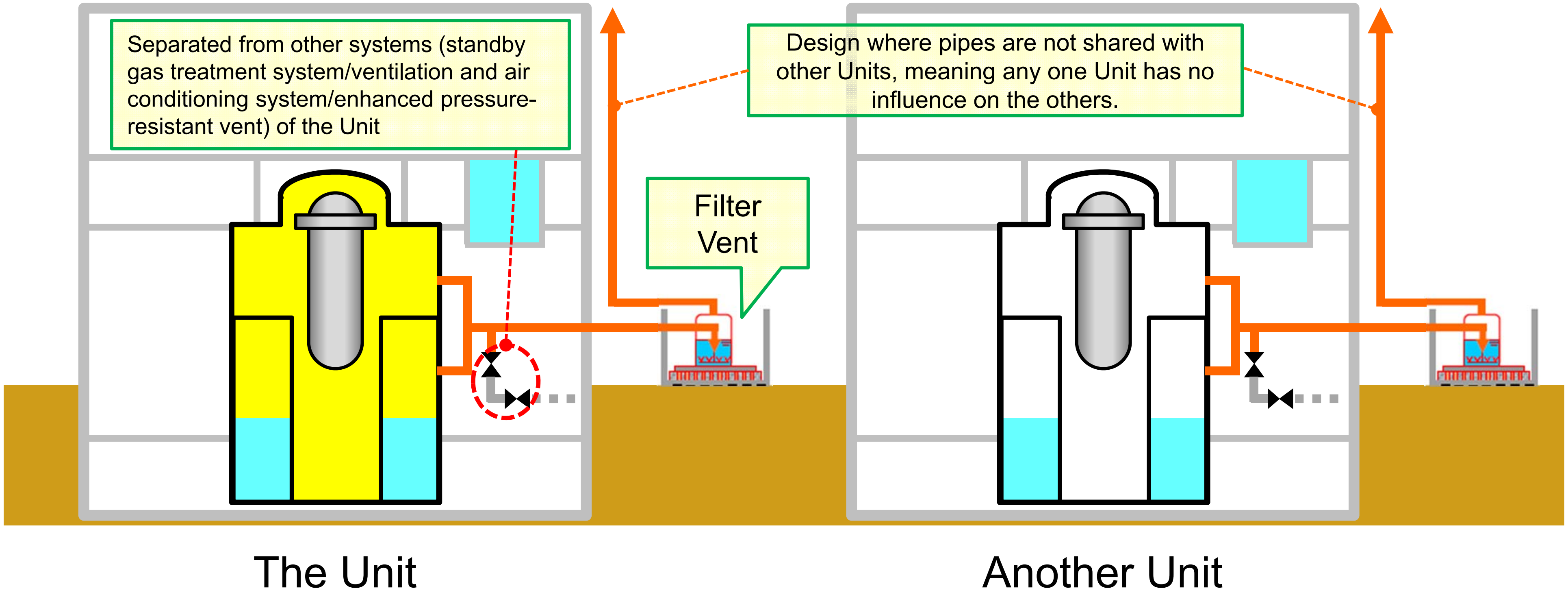
We deduced that ***a*** considerable amount of hydrogen flowed into the reactor building of Unit 4 and resulted in the hydrogen explosion in Unit 4.

Measures to prevent gas generated in the venting of the containment vessel from flowing into the building (securing independence of vent pipes) are needed

Safety measures at the Kashiwazaki-Kariwa Nuclear Power Plant

<u>Securing independence of vent pipes from other Units</u>

- Filter vents newly installed to the Kashiwazaki-Kariwa Nuclear Power Plant don't share pipes with other Units. Moreover, vent pipes are separated from other systems of the Unit.
- As seen above, the nuclear power plant takes measures to secure the vent pipes' independence and prevent any vent gas generated in the venting of the containment vessel from flowing into the reactor building.



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

Overview

Radioactive material emitted from the fuel as the accident progressed was emitted into the atmosphere because of a direct leak from the containment vessel vent and containment vessel, the explosion in the reactor building and other reasons. Regarding the emission behavior of radioactive material and the accident progression scenario, not all associations have been identified to date.

Variation in the air dose rate monitoring data observed within and outside the power plant site during the accident is expected to shed light on how the accident progressed. Therefore, we started by focusing on Unit 1 and proceeded to analyze the change in the air dose rate monitoring data to determine the accident process leading to radioactive material emissions.

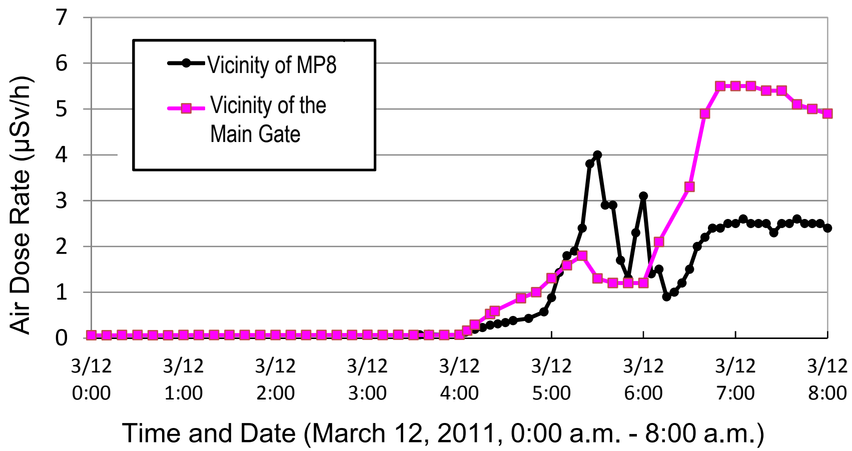


Figure Example of the air dose rate monitoring data in the Power Plant site

< The analytical method >

(1) Deduct the emission behavior of radioactive material and the accident progression scenario, focusing on changes in the air dose rate monitoring data from 0:00 a.m. to around 8:00 a.m. on March 12 and in the containment vessel pressure.

(2) Compare the accident progression scenario obtained in (1) with the existing scenario based on data such as reactor pressures, containment pressures and reactor water levels, ***as*** reported in the third progress report.

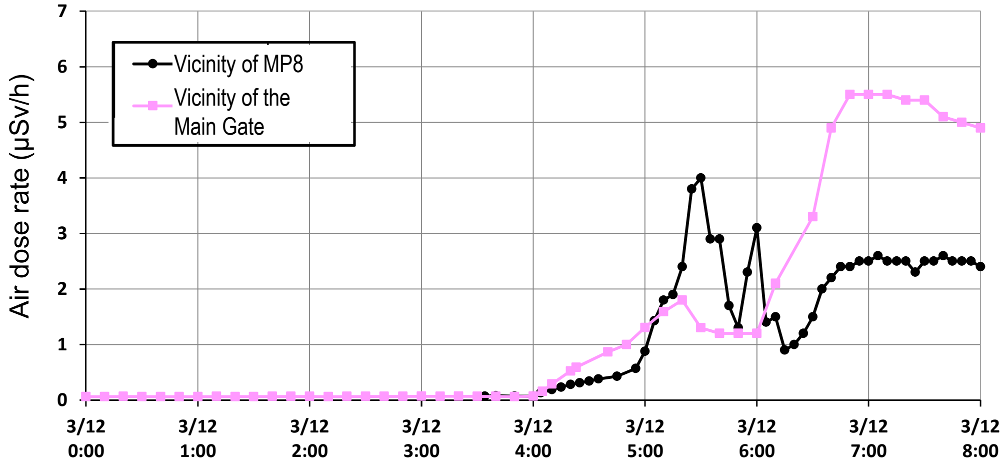
The accident progression deduced from the emission behavior of radioactive material correlates with the existing scenario.

The air dose rate monitoring data within and outside the Power Plant site

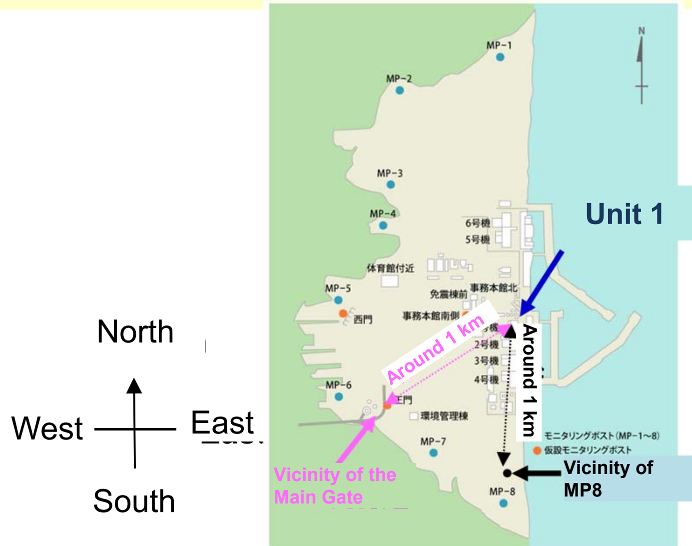
6. Estimation of accident progression at Unit-1 based on the air dose rate monitoring data

Fuels in Unit 2 and 3 remained undamaged as of around 8:00 a.m. on March 12, hence the monitoring data during the period is considered to show the transfer and emission behavior of radioactive material from Unit 1. The air dose rate within and outside the power plant site is as follows:

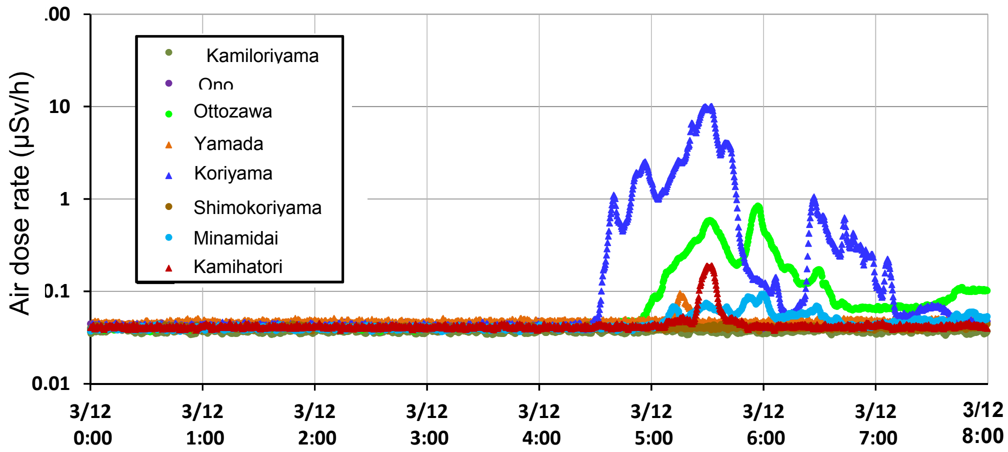
Air Dose Rate Monitoring Data in the Power Plant Site



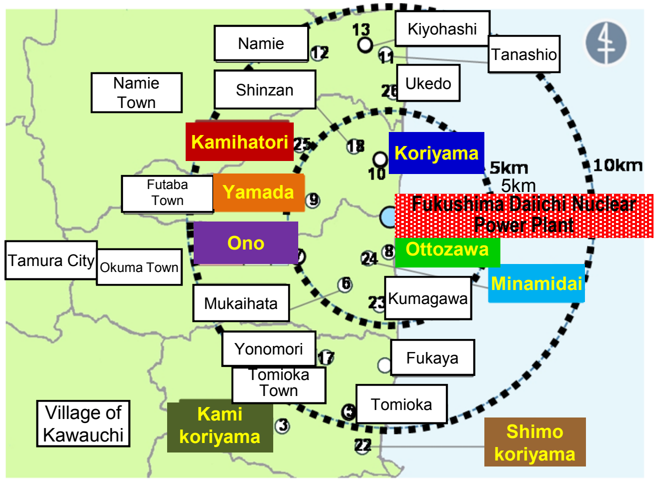
Time and Date (March 12, 2011, 0:00 a.m. - 8:00 a.m.)



Air dose rate monitoring data outside the Power Plant site



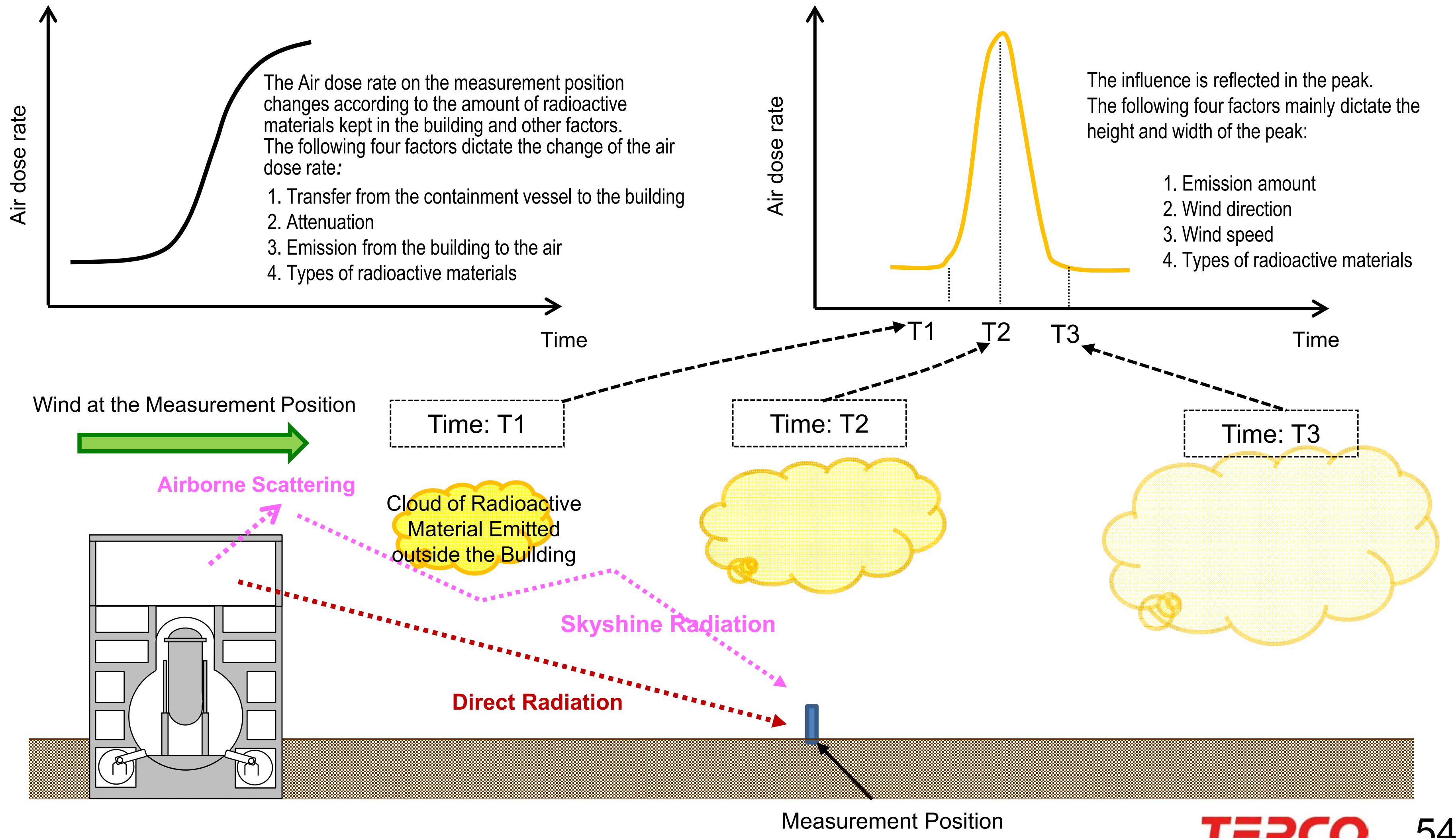
Time and Date (March 12, 2011, 0:00 a.m. - 8:00 a.m.)



Relationship between transfer and emission of radioactive materials and behavior of air dose rate

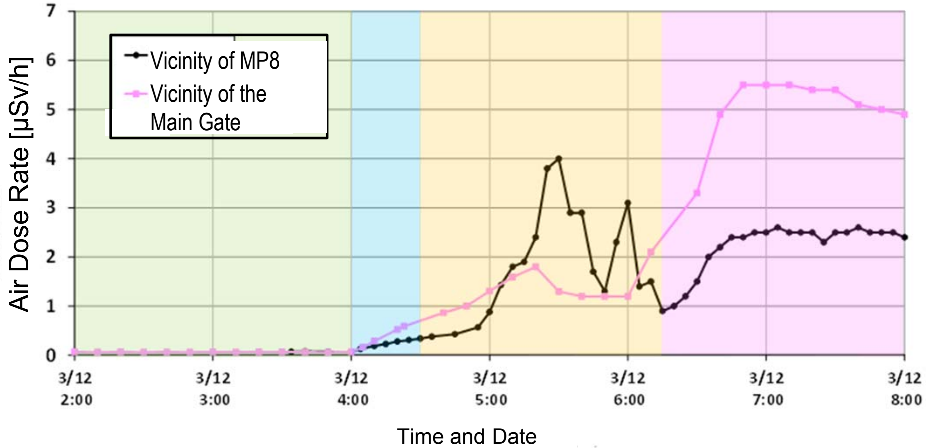
<Behavior of the air dose rate: pattern A>
Influence of direct and skyshine radiation exerted by radioactive material kept in the building

<Behavior of the air dose rate: pattern B>
Influence of the cloud of radioactive material emitted outside the building

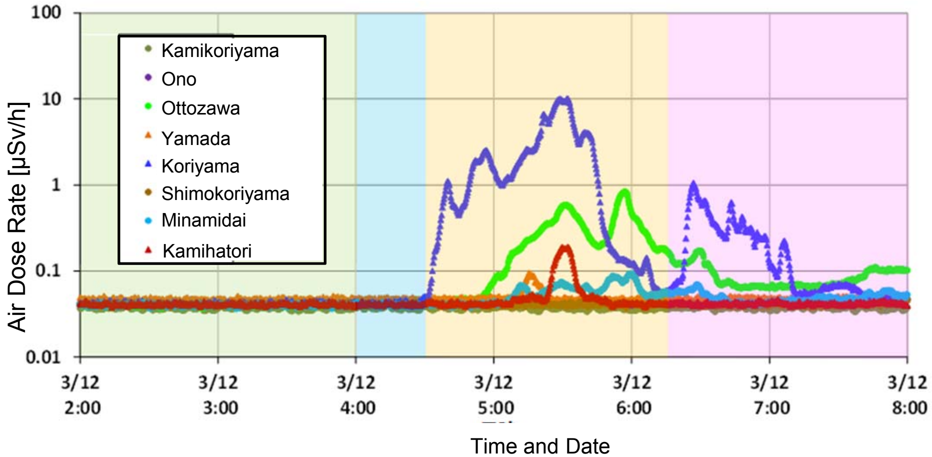


Deduction of the transfer and emission behaviors of radioactive material from the change in the air dose rate monitoring data

Air dose rate monitoring data in the Power Plant site



Air dose rate monitoring data outside the Power Plant site



At around 4:00 a.m. on March 12

The air dose rate within and outside the power plant site shows no remarkable change.

<Deduction>

The transfer of radioactive materials from the containment vessel to the reactor building and to the environment was not significant enough to be observed outside the reactor building.

From around 4:00 a.m. to around 4:30 a.m. on March 12

The air dose rate increases only in the power plant site. No peaks are visible. It is considered to reflect the features of pattern A (influence of direct and skyshine radiation).

<Deduction>

The transfer of radioactive materials from the containment vessel to the reactor building was significant enough to be observed outside the reactor building.

From around 4:30 a.m. to shortly after 6:00 a.m. on March 12

The peaks of the air dose rate, which are characteristic of pattern B (influence of the cloud of radioactive material), can be seen within and outside the power plant site. The influence of pattern A is also visible during the period.

<Deduction>

Radioactive material leaked from the reactor building into the environment.

From shortly after 6:00 a.m. to around 8:00 a.m. on March 12

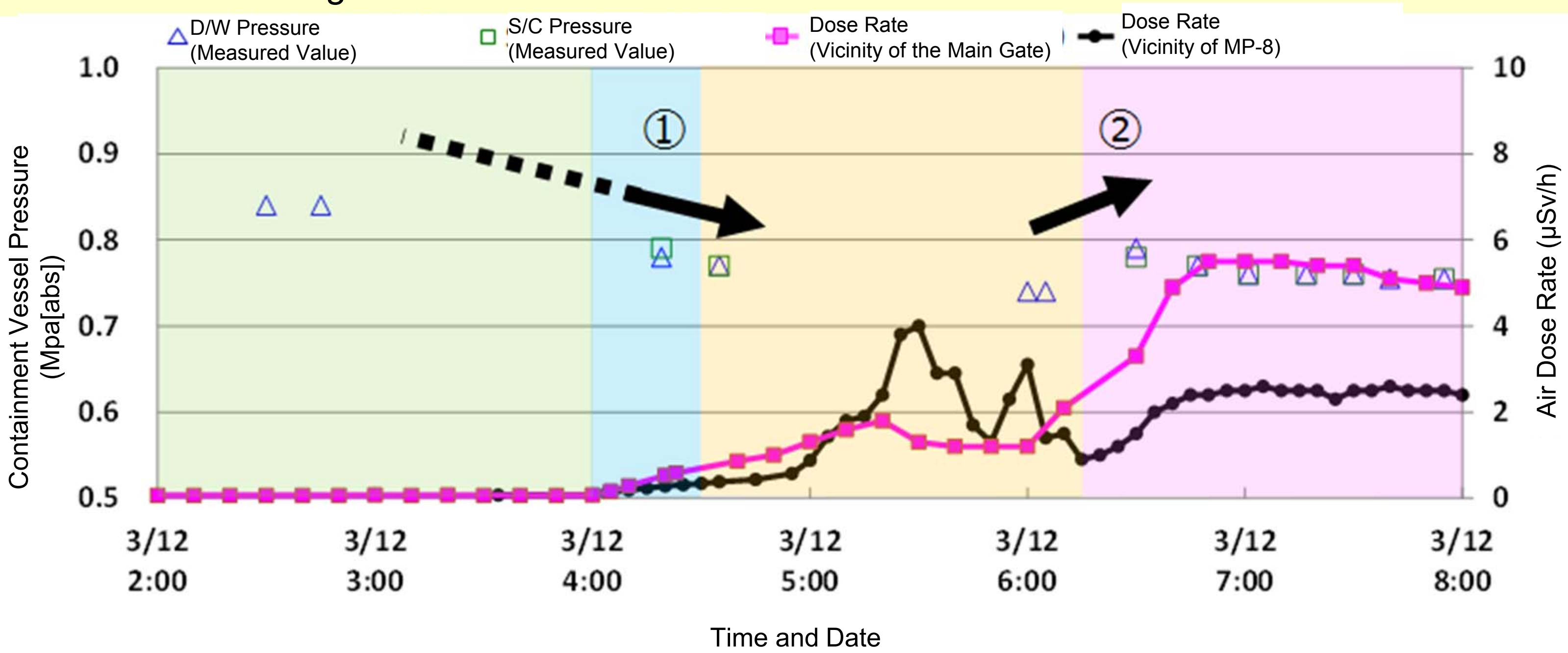
The air dose rate in the power plant site increases and then remains flat. It is considered to reflect the features of pattern A. The air dose rate outside the power plant site has peaks, which are characteristic of pattern B.

<Deduction>

Judging from the high air dose rate in the site compared to the data before this period, more radioactive material transferred from the containment vessel to the reactor building.

Estimation of the containment vessel pressure of Unit 1 based on the measurement data

Relationship between the Measured Values of the Containment Vessel Pressure of Unit 1 and the Air Dose Rate Monitoring Data in the Power Plant Site



・The air dose rate increased during both ① and ② shown in the figure, in which the containment vessel pressure changes. Therefore we deduced that radioactive material was transferred from the containment vessel to the reactor building.

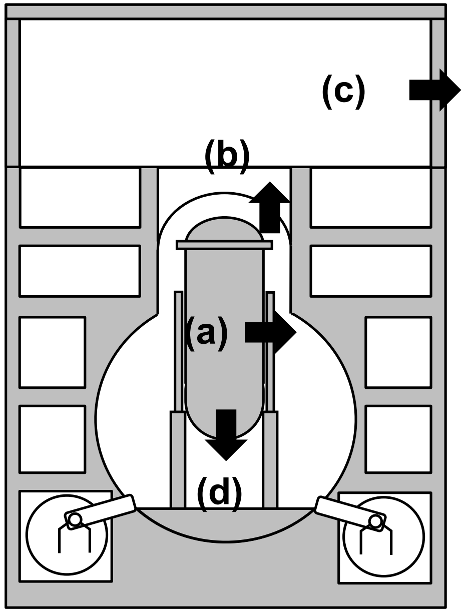
・Conversely, the containment vessel pressure showed different behavior; it decreased shortly before 4:30 a.m. (① in the figure) and increased shortly after 6:00 a.m. (② in the figure).

<Deduction>

Judging from the difference in the way the containment vessel pressure behaved during ① and ②, the rapid increase in the air dose rate shortly after 6:00 a.m. does not indicate an increase in radioactive material transfer simply because of the increased leakage area of the containment vessel. Instead, some event which increased the containment vessel pressure occurred and increased the radioactive material transfer to the reactor building.

Summary (comparison with the existing accident progression scenario)

Example transfer path of the radioactive material



(a) Pressure Vessel (Connecting pipes) → Containment Vessel
(b) Containment Vessel → Reactor Building
(c) Reactor Building → Environment
(d) Pressure Vessel (Bottom Head) → Containment Vessel

Time and Date	Existing Accident Progression Scenario [1]	Accident Progression Scenario Based on the Air Dose Rate Behavior and Containment Vessel Pressure
Mar. 11	・Fuel melted and radioactive material was transferred from the pressure vessel to the containment vessel. ((a) in the figure). ・Molten fuel was transferred from the reactor core to the pressure vessel (bottom head).	<Deduction (1)> Until around 4:00 a.m. on March 12, the leakage of radioactive material from the containment vessel into the reactor building and the environment was not significant enough to be observed outside the reactor building.
Mar. 12 around 4:00 a.m.		<Deduction (2)> As of around 4:00 a.m., the transfer of radioactive material from the containment vessel to the reactor building was significant enough to be observed outside the reactor building ((b) in the figure).
around 4:30 a.m.		<Deduction (3)> The radioactive material leaked from the reactor building into the environment by 4:30 a.m. at the latest ((c) in the figure).
around 6:00 a.m.	・The pressure vessel (bottom head) was damaged ((b) in the figure)	<Deduction (4)> Some event which increased containment vessel pressure occurred at around 6:00 a.m. and caused an increase in radioactive material transfer to the reactor building ((d) in the figure).

[1] Tokyo Electric Power Co., Inc. (December, 2015). "The fourth Progress Report on the Estimation of the Situations in the Reactor Core and the Containment Vessel of Unit 1-3 of the Fukushima Daiichi Nuclear Power Plant and Consideration of the Open Issues"

・The deduction that radioactive material transferred to the containment vessel as of March 11 in the existing accident progression scenario matches deductions (1) and (2) of the deduced scenario.
・The deduction that the pressure vessel (bottom head) was damaged at around 6:00 a.m. is also consistent with deduction (4).

➔The accident progression scenario deduced from the air dose rate behavior correlates to the existing scenario.