The 6th Progress Report

on the Investigation and Examination of Unconfirmed and Unsolved Issues on the Development Mechanism of the Fukushima Daiichi Nuclear Accident

November 10, 2022

Tokyo Electric Power Company Holdings, Inc.

TEPCO

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The 6th Progress Report

on the Investigation and Examination of Unconfirmed and Unsolved 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)

Nuclear Safety Reform Plan

(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
→ Kashiwazaki-Kariwa NPS: Implemented safety countermeasures to prevent severe accident occurrence

✓ 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 functions 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 accident developed after the initial incident** is not only the responsibility of the parties involved in the accident but also important to:

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

Scope covered by deliberations and investigations into unsolved issues

Efforts to continuously improve safety without being satisfied with current safety levels

This report compiles the results of investigations and deliberations conducted from the above perspectives. This is also the sixth progress report following those given in December 2013, August 2014, May 2015, December 2015 and December 2017.

3. Investigation/examination history and positioning of this report

- ✓ The study extracted 52 unconfirmed and unsolved issues on the detailed development mechanism after the accident occurred and has published five 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 effectively utilized information obtained on-site as the decommissioning has progressed, for examination.
- ✓ With decommissioning progress, information close to the site center has been obtained, such as results of an investigation inside primary containment vessels (PCVs) of Units 1-3 and analysis of collected samples. We can now focus on estimating current conditions inside RPVs (reactor pressure vessels) and PCVs.

- ✓ **Coordinating with activities to clarify status inside the reactors made by the government*, in FY2016 and FY2017, distribution of fuel debris inside Units 1-3 was estimated.**
- ✓ **By working together in the field, direct on-site information about status inside RPVs and PCVs obtained by the decommissioning progress is used in examinations.**
- ✓ **Examinations will continue reflecting the ongoing quest to improve safety.**

* Subsidy for Decommissioning and Contaminated Water Management (Upgrading Level of Grasping State inside Reactor Status)

3. Investigation/examination history and positioning of this report

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

In the fifth and previous progress reports, examination results of 34 issues, including 10 high-priority issues to understand the development mechanism, were reported.

Unreported Issues

Issues that help elucidate the development mechanism
18 issues

The sixth report
4 new studies

Reported issues（34 issues)

Issues that help to understand progress mechanisms
23 issues

Of the above the sixth report adds 5 studies*

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

The sixth report estimation of fuel debris distribution*

High-priority issues to understand development mechanism
10 issues

Included in fourth report

In the sixth progress report, the study details are reported and the current estimates for fuel debris distribution are presented.

*Examinations related to issues for which results have been reported in the past

4. Main points of the sixth progress report

※【 】Report attachment No.

Estimation of fuel debris distribution

1. Estimation diagram of fuel debris【Attachment 4】

TEPCO Holdings has continued to estimate the distribution of fuel debris in Fukushima Daiichi Units 1-3 even after the completion of the "Subsidy for Decommissioning and Contaminated Water Management (Upgrading Level of Grasping State inside Reactor)", and to present the results. The changes of the estimations that we have been working on since 2011 are summarized separately.

Issues that help clarify the development mechanism

2. Identification of causes of high dose rate observed in the southeast area on the 1st floor of Unit 1 Reactor Building【Attachment 1-12】

At Unit 1, a high dose rate was observed in the southeast area of the 1st floor of the reactor building. As a result of identifying scenarios that could cause high dose rates and examining the impact on the southeast area, it was determined that the cause was contamination in the piping used for containment vessel venting in the southeast area.

3. Estimation of the reason for high dose rate not being observed in Unit 2 Reactor Building Cooling Water System【Attachment 2-15】

The reason for the high radiation dose rate observed around the reactor coolant cooling water (RCW) system of Unit 1 is thought to be that the RCW piping was damaged by fuel debris that fell to the bottom of the containment vessel, causing radioactive materials to diffuse into the system. On the other hand, no high contamination was observed in the RCW system of Unit 2, where it was believed that fuel debris fell to the bottom of the containment vessel as well. Based on the results of an investigation of the inside of the containment vessel of Unit 2, it was assumed that the fuel debris did not damage the RCW piping in the containment vessel because of the low temperature at the time the fuel debris fell from the pressure vessel.

4. Decrease in Unit 2 containment vessel pressure in the morning of March 15【Attachment 2-16】

The D/W pressure in Unit 2 remained above 0.7MPa[abs] from around 23:30 on March 14 to 7:20 on March 15, after which measurement was temporarily interrupted, and when measurement resumed at 11:20 on March 15, the pressure had dropped to 0.155MPa[abs]. The reason for this large drop in pressure was examined based on plant parameter readings and observations, and it was estimated that, in addition to the gas-phase leakage from the containment vessel, the torus room where the S/C was housed was flooded, which might have contributed to vapor condensation in the S/C gas phase section.

5. Behavior of S/C pressure gauge in Unit 2 after 21:00 on March 14【Attachment 2-17】

The S/C pressure gauge for AM, one of the containment pressure gauges used in Unit 2 at the time of the accident, showed a low indicator value that deviated from D/W pressure, although the battery was connected, and the power was restored at 3:00 on March 13. It was estimated that the room where the pressure gauges were installed was flooded by the tsunami and the gauges were submerged, resulting in electrical failures and abnormal readings.

4. Main points of the sixth progress report

※【 】Report attachment No.

Issues that help clarify the development mechanism

6．Evaluation method of core damage ratio of Mark-I containment vessel【Attachment 2-18】

In Unit 2, CAMS measurements were obtained as power was restored. It was found that the evaluation map, which was supposed to evaluate the core damage ratio conservatively, tended to underestimate the core damage ratio when using S/C CAMS. This was presumed to be due to the fact that it did not properly reflect the effects of the geometry of the Mark-I containment vessel and the location of the CAMS detector.

7. Examination of water level in Unit 3 Suppression Chamber Attachment 3-11】

S/C water level data were collected at Unit 3 from 17:15 on March 11 to 20:00 on March 12. Information on the S/C water level could be used to estimate the amount of hydrogen generated and the state of water storage when fuel debris fell to the floor of the containment vessel. This is important information for estimating the cooling state of fuel debris. In this study, the S/C water level at around 9:00 on March 13 was examined based on the measured values of the S/C water level and changes in the pressure of the containment vessel. As a result, it was estimated that the S/C water level was about 7 m from the bottom of the S/C, higher than the vacuum break valve.

8. Accident progress after the Unit 3 reactor depressurization【Attachment 3-12】

As a further study of accident progress scenarios for Unit 3 from 9:00 on March 13 to 0:00 on March 14, the possible ranges of important parameters for accident progress scenarios, such as the area of gas-phase leakage from the pressure vessel and the number of SRV open valves, were evaluated through analysis. From the study results, it was determined that the open/close status of SRVs and the gas-phase leakage from the pressure vessel to D/W after the ADS activation around 9:00 on March 13 could be used in accident progress scenarios that quantitatively reproduce the trend of the measured values.

9．Examination of plant conditions during RCIC operation of Unit 3【Attachment 3-13】

The RCIC operation of Unit 3 after the arrival of the tsunami was based on adjusting the amount of water injected into the reactor, including utilizing the return line to the CST, the water source, in order to reduce battery consumption due to startup shutdown. The behavior of the reactor pressure during this period was peculiar in that it changed at a pressure different from the pressure at which the SRVs were set to operate, and it was recognized that this reflected the complex situation in which the SRVs were opened and closed under special operation of the RCIC. In this examination, the validity of this qualitative explanation was confirmed by conducting a reproducible analysis simulating water injection into the reactor by this RCIC operation and the opening and closing of SRVs.

10. Sample analysis to determine accident progress【Attachment 5】

Radioactive particles were detected in samples taken inside and outside the containment vessels of Units 1-3 and in environmental samples, and their formation process was estimated by focusing on their composition and crystal structure. Based on the results, the status of fuel debris and the accident progression process were discussed.

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

The Atomic Energy Society of Japan meetings/International meetings

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 investigation while considering the comments made and other achievements gained through these activities.

<Recent Major Presentations>

- Atomic Energy Society of Japan Spring 2018 (Sample analysis)/Autumn (Unit 2 RCW), Fall 2019 (Sample analysis, Unit 2 PCV pressure)
- FDR 2019 (International Topical Workshop on Fukushima Decommissioning Research 2019) (Unit 2 PCV pressure, sample)
- 4th International Forum on Decommissioning (Sample poster)
- Geochemical Society Annual Meeting 2018 (Insoluble Cs particles)

Subsidy Project for Decommissioning & Contaminated Water Management (Upgrading Level of Grasping State inside Reactor)

In cooperation with this subsidy project, in FY2016 and FY2017, we estimated the status inside the reactors and PCVs, including the fuel debris distribution in Fukushima Daiichi Nuclear Power Station Units 1-3. And TEPCO continues to study the issue.

Status of fuel debris, nuclear fission products, etc. considered to be distributed inside RPVs and PCVs has been estimated based on accumulated knowledge from Japan and overseas, cooperation with overseas organizations, and comprehensive analysis and evaluation of "information obtained from onsite investigations, etc.", "measurement data during and after the accident", "knowledge obtained from experiments", and "analytical results of accident development", etc. OECD/NEA BSAF was implemented as part of this activity.
The project to estimate status of the in-vessel and containment vessel from the analysis results of samples collected at the site is being carried out in cooperation with the "Development of Analysis and Estimation Technology for Characterization of Fuel Debris," a project funded by the subsidy for decommissioning and contaminated water countermeasures.

Nuclear Regulation Authority, Japan the Committee on Accident Analysis

We have participated in discussions as an observer at the restarted accident analysis review committee. 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 related to questions and points of interest from the governor and committee members during discussion at the Niigata Prefecture technical committee meeting to elucidate the Fukushima Daiichi accident and verify safety measures at Kashiwazaki-Kariwa NPS.

We are continuing our investigation through discussions and exchanging opinions with researchers from various organizations.

(Reference) Ten high-priority issues

		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 operation status of Unit 2)	Unit 2-9
	• Cause investigation of dose increase on or around March 20, 2011	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 in the vicinity of particular pipes in Unit 1 reactor building (<u>Identification of causes of high dose contamination of pipes of the reactor cooling water (RCW) system in Unit 1</u>)	Unit 1-9

(Ref.) Status of efforts to address 52 issues (1/2)

Red: Newly added

Issue No.	Issue name	Study	Related reference
Common-1	SRV operations after core damage	○	Ref. 1-3, 2-12, 3-4, 3-13
Common-2	Amounts of water injected to the reactor by fire engines	○	Ref. 1-4, 1-5, 2-14
Common-3	Water evaporation in the reference leg of water level indicators	○	Ref. 1-6, 2-14, 3-9
Common-4	Water leaks from PLR pump mechanical seals	–	–
Common-5	Core-concrete reactions	–	–
Common-6	Molten core behavior on falling to the lower plenum	○	Ref. 1-8
Common-7	Correlation between the timing of a large amount of radioactive materials released to the air and the monitoring data	○	Ref. 1-11
Common-8	Radioactive materials release behavior at the time of PCV venting	–	–
Common-9	Air dose increases on around March 20th	○	Ref. 3-6
Common-10	Core damage and the location of core debris	○	本文, Ref. 4, 5
Common-11	Reactor building hydrogen explosions	○	Ref. 1-10, 3-10
Common-12	Knowledge about massive synchronized earthquakes with accompanying tsunami	–	–
Common-13	Intensified seismic activities in the southern area of Hama-dori in Fukushima Prefecture	–	–
Common-14	Exact timing of the tsunami wave arrival s at major buildings of the Fukushima Daiichi NPS and their inundation routes	○	Ref. Earthquake/Tsunami-1, and -2
Common-15	Impacts of tsunami wave forces	–	–
Common-16	Investigation from the viewpoint of human factors	–	–
Unit 1-1	Deterioration of IC heat removal performance due to hydrogen gas in Unit-1	○	Ref. 1-7
Unit 1-2	Plant behavior if the Unit-1 IC s had functioned	○	Ref. 1-7
Unit 1-3	RPV water level indicator readings at Unit-3 after loss of true value indications	○	Ref. 1-6
Unit 1-4	LOCA possibility at Unit-1 due to the earthquake	○	Ref. 1-3
Unit 1-5	Leaks in gaseous phase from Unit-1 RPV	–	–
Unit 1-6	Leaks in gaseous phase from Unit-1 PCV	–	–
Unit 1-7	Dose rate increase in Unit-1 reactor building on March 11th	–	–
Unit 1-8	Causes of high contamination in the southeast area of the ground floor in the Unit-1 reactor building	○	Ref. 1-12
Unit 1-9	Causes of high dose contamination around the Unit-1 RCW piping	○	Ref. 1-9, 2-15
Unit 1-10	High dose rates contamination near the Unit 1 SGTS piping	–	–

(Ref.) Status of efforts to address 52 issues (2/2)

Red: Newly added

Issue No.	Issue name	Study	Related reference
Unit 1-11	Impacts of water injection interruptions on the accident progression	○	Ref. 3
Unit 2-1	RCIC flow rates after loss of DC power supply at Unit 2	○	Ref. 2-4
Unit 2-2	Cause of RCIC shutdown at Unit 2	–	–
Unit 2-3	Behavior of S/C pressure indicator of Unit 2 after 21:00 on March 14th	○	Ref. 2-17
Unit 2-4	Unit-2 RHR system operating conditions after tsunami arrival	○	Ref. 2-5
Unit 2-5	PCV pressure behavior at Unit 2 after about 13:00 on March 14th	○	Ref. 2-6
Unit 2-6	PCV pressure behavior upon forced SRV opening at Unit 2	○	Ref. 2-6
Unit 2-7	RPV pressure increase after forced depressurization at Unit 2	○	Ref. 2-7, 2-9
Unit 2-8	Leaks in gaseous phase from Unit 2 RPV	○	Ref. 2-10
Unit 2-9	Consideration of possible rupture disc actuation at Unit 2	○	Ref. 4
Unit 2-10	Condensation behavior upon hydrogen rich steam release at Unit 2	○	Ref. 2-8, 2-13
Unit 2-11	Leaks in gaseous phase from the Unit 2 PCV	○	Ref. 2-16
Unit 2-12	Sharp increase of CAMS readings on March 15th at Unit 2	○	Ref. 2-10, 2-11, 2-18
Unit 2-13	Grounds for no hydrogen explosion at Unit 2	–	–
Unit 3-1	Causes of repeated shutdown of RCIC at Unit 3	○	Ref. 3-5
Unit 3-2	RPV water level indicator readings at Unit 3 after loss of true value indications	○	Ref. 3-9
Unit 3-3	Thermal stratification in the S/C of Unit 3	○	Ref. 3-7
Unit 3-4	Reactor water le vel behavior during HPCI in operation at Unit 3	○	Ref. 3, 3-3
Unit 3-5	Reactor water le vel behavior after the loss of function of HPCI at Unit 3	○	Ref. 3, 3-3, 3-4, 3-9
Unit 3-6	Rapid depressurization at about 09:00 on March 13th at Unit 3	○	Ref. 3-3, 3-4
Unit 3-7	RPV pressure behavior after rapid depressurization at Unit 3 on March 13th	○	Ref. 3-3, 3-4
Unit 3-8	PCV pressure behavior upon venting operations at Unit 3	○	Ref. 3-8, 3-11, 3-12
Unit 3-9	Leaks in gaseous phase from Unit 3 RPV	○	Ref. 3-11, 3-12
Unit 3-10	Leaks in gaseous phase from Unit 3 PCV	○	Ref. 3-8, 3-11, 3-12
Unit 3-11	Large amount of steam discharge from the top of Unit 3 R/B	○	Ref. 3-8
Unit 3-12	Impacts of water injection interruptions on the accident progression	–	–

The 6th Progress Report
on the Investigation and Examination of
Unconfirmed and Unsolved Issues

Specific Examination Topics

1. Estimation diagram of fuel debris

This topic includes the output of the project supported by "Subsidy for Decommissioning and Contaminated Water Management (Upgrading Level of Grasping State inside Reactor)."

Overview

- The “Investigation and Examination of Unconfirmed and Unsolved Issues” conducted by TEPCO Holdings sets estimation of the fuel debris distribution in Fukushima Daiichi Nuclear Power Station Units 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.

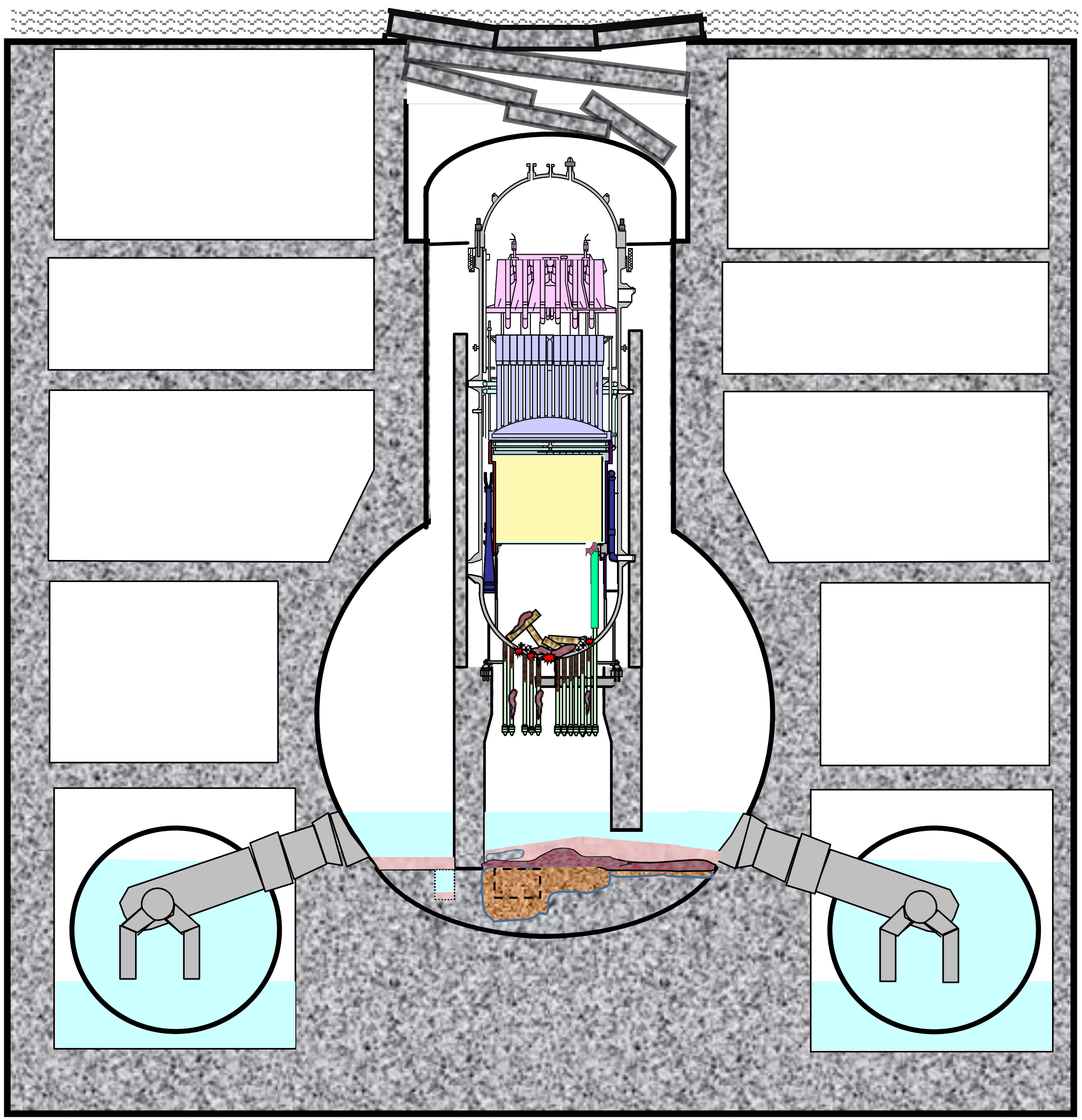
- As part of these efforts, in FY 2016 and FY 2017, TEPCO Holdings estimated the fuel debris distribution in Fukushima Daiichi Nuclear Power Station Units 1-3 in cooperation with the project “Subsidy for Decommissioning and Contaminated Water Management (Upgrading Level of Grasping State inside Reactor Status).”

- TEPCO Holdings is continuing this effort even after the project completion. As a result of the progress of on-site investigations* since the previous report, information on the inside of the reactors and containment vessels has been obtained, and the fuel debris distribution has been updated by proactively incorporating such information.
  *Investigation inside the containment vessels of Units 1 to 3, etc.

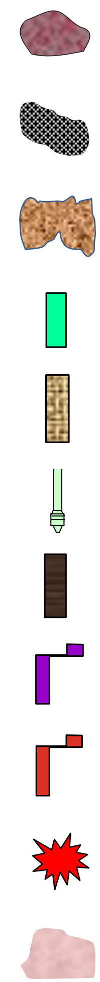
- Past estimates have already been published in July 2021 as the report "Estimation of Conditions in the Reactor Pressure Vessel and Containment Vessel after the Accident at the Fukushima Daiichi Nuclear Power Station." In the future, it will be updated as part of the "Investigation and Examination of Unconfirmed and Unsolved Issues.”

In the following pages, estimation of the fuel debris distribution in Units 1-3 is described.

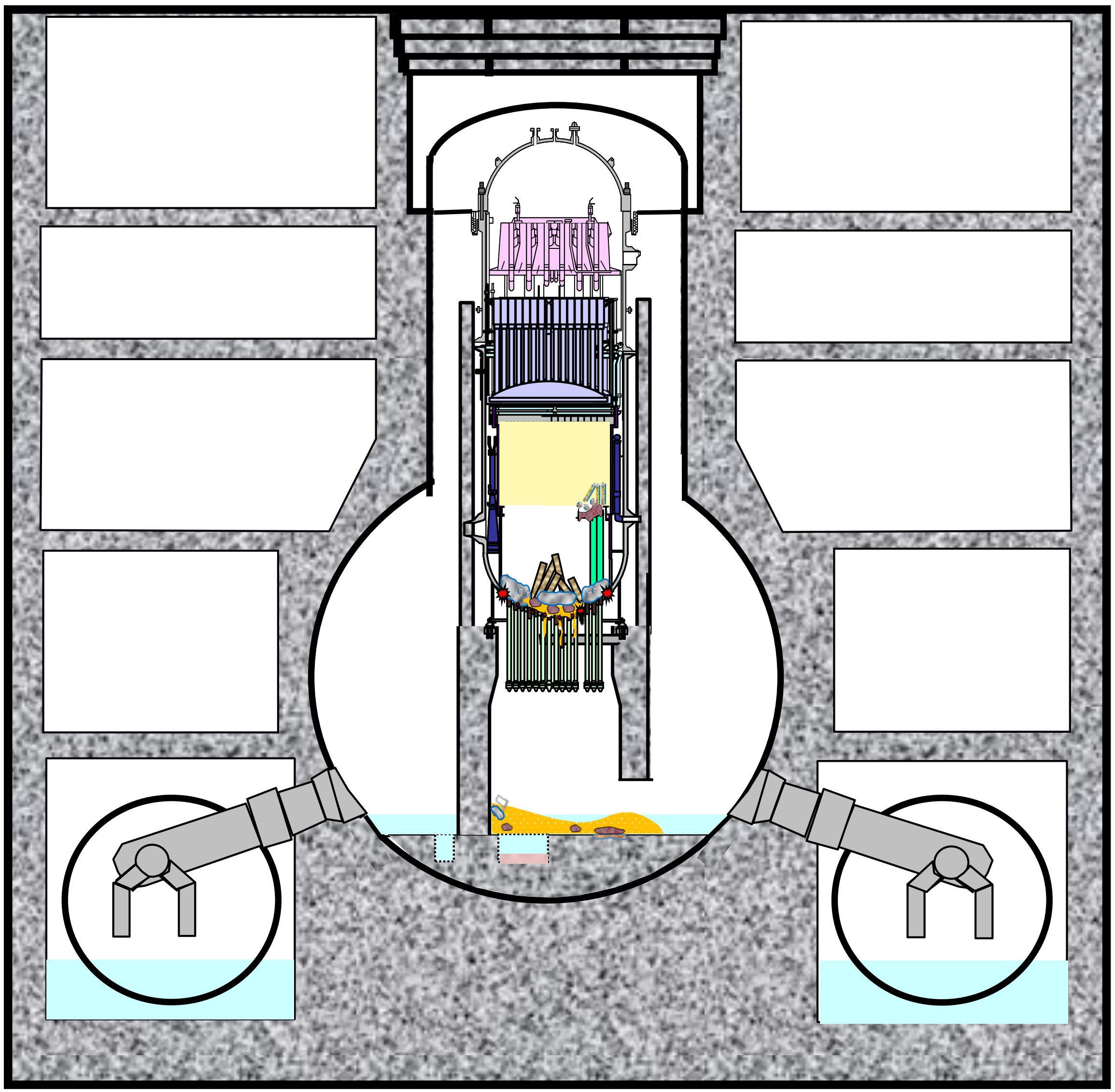
Estimation image of Unit 1 fuel debris distribution



Legend



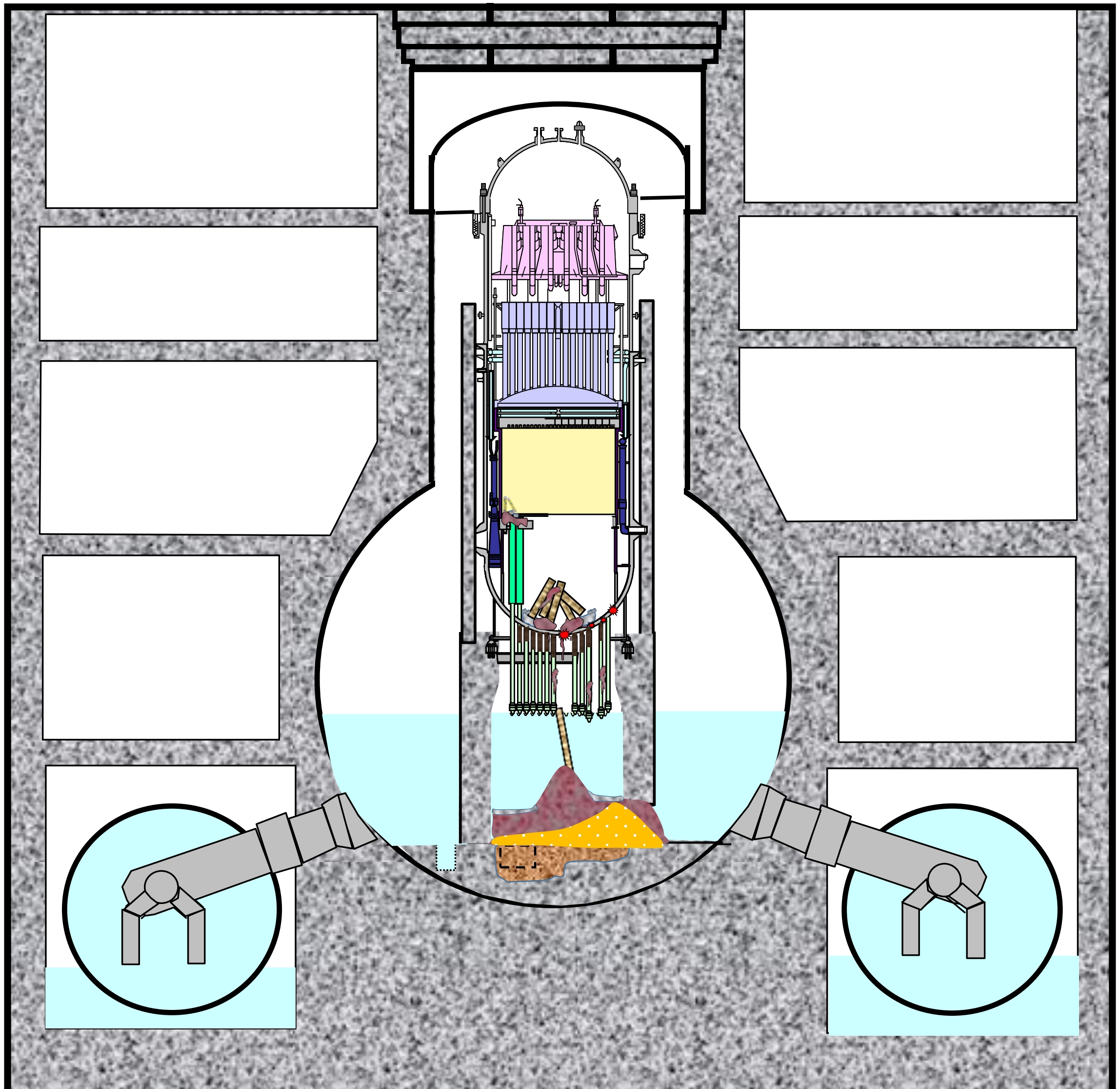
Estimation image of Unit 2 fuel debris distribution



Legend

- Residual fuel rods and remnants
- Oxide debris (porous)
- Particle debris
- Fuel debris (containing much metal)
- Concrete-mixed debris
- CRGT
- Damaged CRGT
- CRD
- CRD (containing debris inside)
- Shroud
- Pellet
- RPV damage opening
- Upper tie plate
- Deposit (unidentified material)

Estimation image of Unit 3 fuel debris distribution

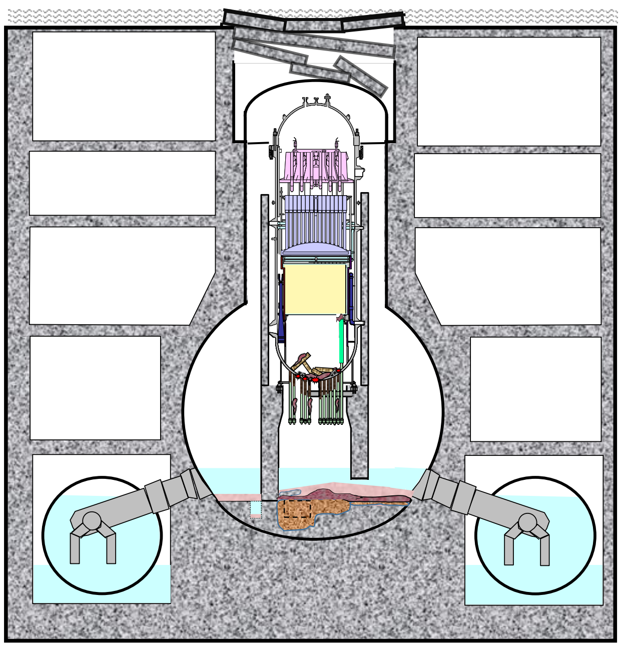


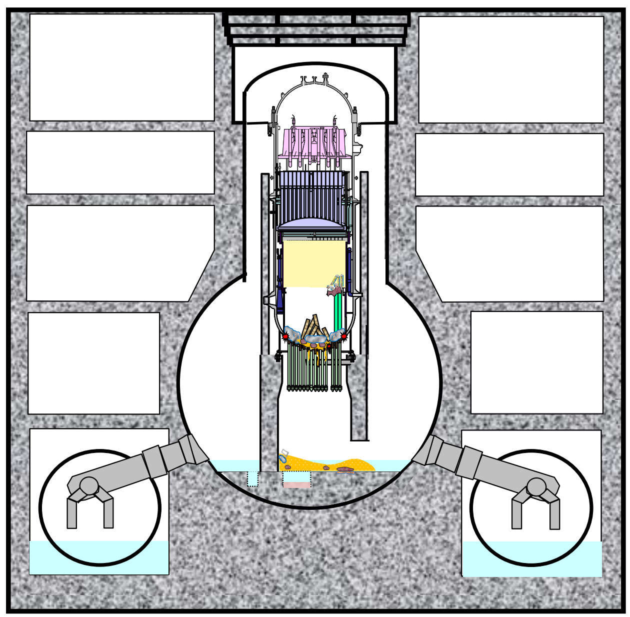
Legend

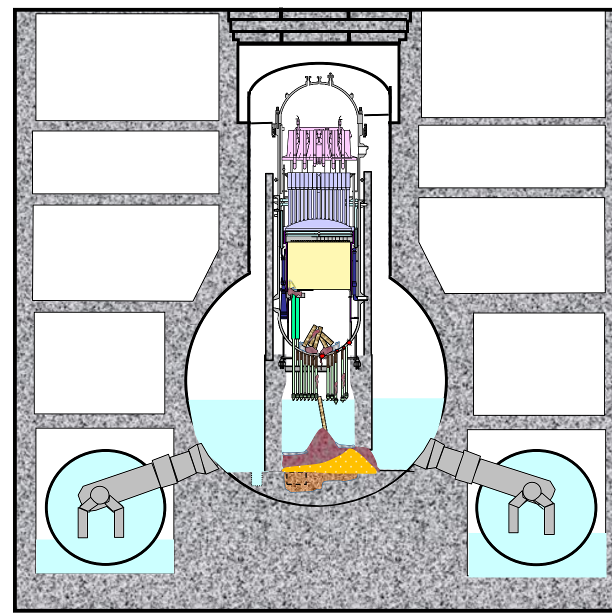
- Residual fuel rods and remnants
- Oxide debris (porous)
- Particle debris
- Fuel debris (containing much metal)
- Concrete-mixed debris
- CRGT
- Damaged CRGT
- CRD
- CRD (containing debris inside)
- Shroud
- Pellet
- RPV damage opening

Summary of estimated debris status inside reactors and PCVs

	Core	Lower plenum	PCV	D/W water level	S/C water level
Unit 1	Almost none	Almost none	Almost all	About 2m	Almost full
Unit 2	A little	A lot	A little	About 0.3m	Low level
	∨	∨	∧		
Unit 3	A little	A little	Some	About 5m	Full


Unit 1

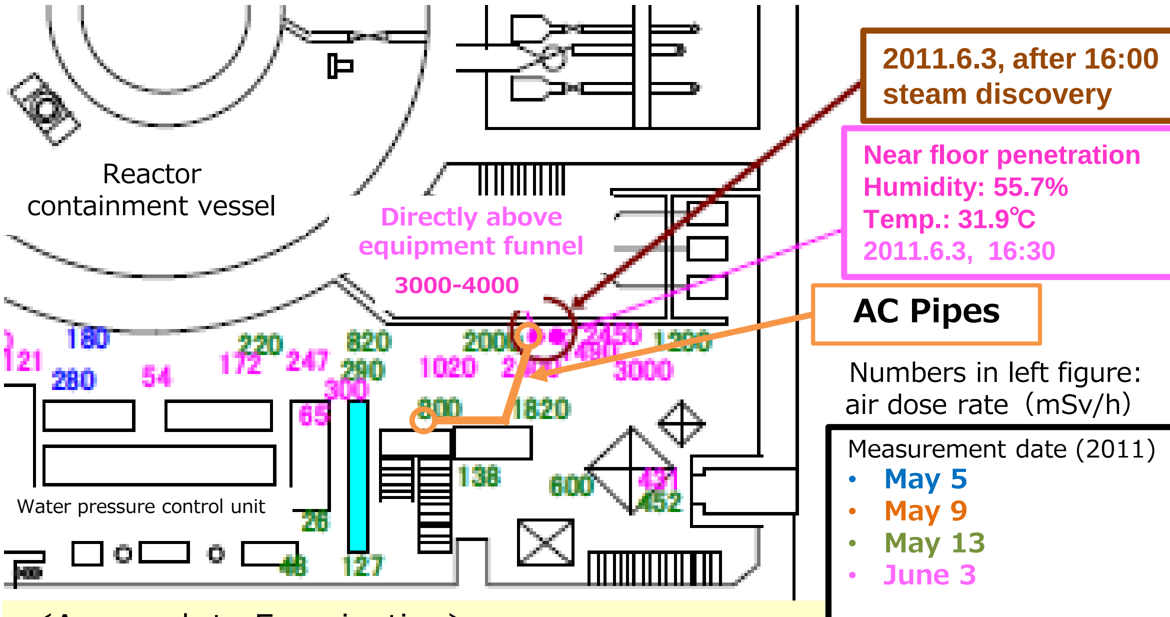

Unit 2


Unit 3

2. Identification of causes of high dose rate observed in the southeast area on the 1st floor of Unit 1 Reactor Building

Overview

- In Unit 1, a high radiation dose rate > 1000mSv/h was observed in the southeast area on the 1st floor immediately after the accident.
- It was confirmed in June 2011 that steam was leaking from a floor penetration in this area.
- AC piping used for PCV venting was laid in the vicinity of the area, and this review focused on this and other possible causes of the high radiation dose rate observed in the southeast area.



Steam observed at a floor penetration in the southeast area (June 2011 photo)

<Approach to Examination>
- Identify contamination sources that may cause high radiation dose and their impacts on the southeast area
- Evaluate the impacts of the identified contamination sources on the area from the following perspectives:

(1) Causes of contamination; (2) Effects of radiation from the contamination sources; (3) Whether or not radioactive materials have migrated from the contamination sources

The high dose rate observed in the southeast area was identified as being dominated by radiation from the AC piping used for the PCV venting

Status of Reflection on Safety Measures at Kashiwazaki-Kariwa NPS: Measures to reduce the impact of radiation from vent lines on accident response operations

Identification of contamination sources to be examined

2. Identification of causes of high dose rate observed in the southeast area on the 1st floor of Unit 1 Reactor Building

①Contamination from the steam and torus room

・During June 2011 survey, steam was observed flowing from the floor penetration connecting the southeast area to the torus room on 1st basement floor.
・High radiation dose rate of several thousand mSv/h was observed in the vicinity of the floor penetration.
・High radiation dose rate >1000mSv/h was observed in the torus room.

②Contamination of inert gas system (AC) piping

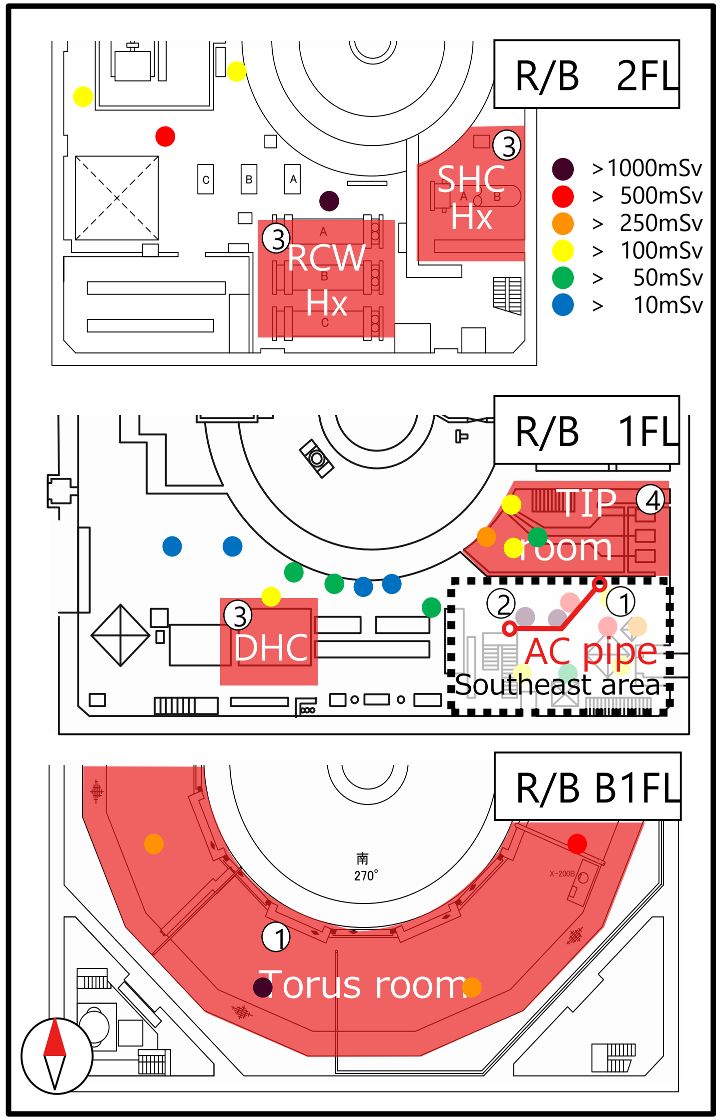
・AC piping used for PCV venting had been laid in the southeast area, and radioactive materials in the vent gas may have contaminated the piping and increased the dose rate in the area.

③Contamination of reactor auxiliary cooling water (RCW) piping

・High dose rate observed near the RCW piping in Unit 1
⇒ It was estimated that molten fuel falling to the bottom of the PCV damaged RCW piping, and radioactive materials migrated through the RCW piping (shown in the 4th report).
・RCW heat exchanger (RCW Hx), RHR shutdown cold system heat exchanger (SHC Hx), and dry well dehumidification system (DHC) are present nearby as RCW system loads.

④Contamination of traversing in-core probe system (TIP) room

・Possibility that TIP instrument dry tube was damaged by molten fuel, contaminating the inside of the TIP instrument and increasing the dose rate in the surrounding area.



□ Evaluate the impacts of the selected contamination sources on southeast area from the following perspectives: (1) Causes of contamination; (2) Effects of radiation from the contamination sources; (3) Whether or not radioactive materials have migrated from the contamination sources

Examination of possible sources of high dose contamination
①Contamination from the steam and torus room

2. Identification of causes of high dose rate observed in the southeast area on the 1st floor of Unit 1 Reactor Building

・Steam that leaked out was most likely from stagnant water in the PCV, and the possibility of the southeast area of the 1st floor being contaminated from the steam should be taken into consideration.
・The torus room itself has also been observed to have a high dose rate, but since the concrete body provides sufficient shielding, impact on the southeast area of the 1st floor is not considered to be a dominant factor.

(1) Causes of contamination (causes of torus room with high dose)

・Contamination inside vent line, inside vacuum break line and inside S/C
・Liquid leakage was observed from the sand cushion drainpipe (lower part of X-5B vent pipe) and from a broken expansion joint (upper part of X-5E vent pipe).
→ Radioactive materials were transferred to torus room as a result of leakage.
(As of June 2011, PCV water level was at the level of the expansion joint failure point, from which steam might have leaked out.)

Discussion on next page

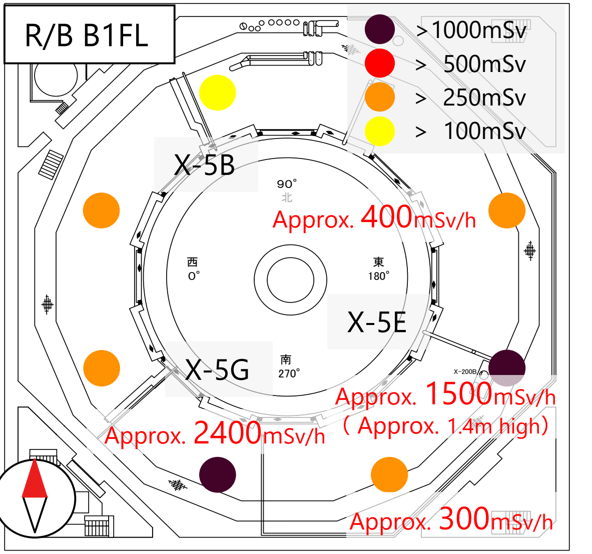
(2) Radiation effects from contamination sources (effects of contamination of torus room)

・Several hundred to 2,400mSv/h dose rates were observed on the catwalk.
→The concrete shielding (650mm thick) was sufficient to attenuate the radiation in the southeast area of the 1st floor, so the impact was not significant.

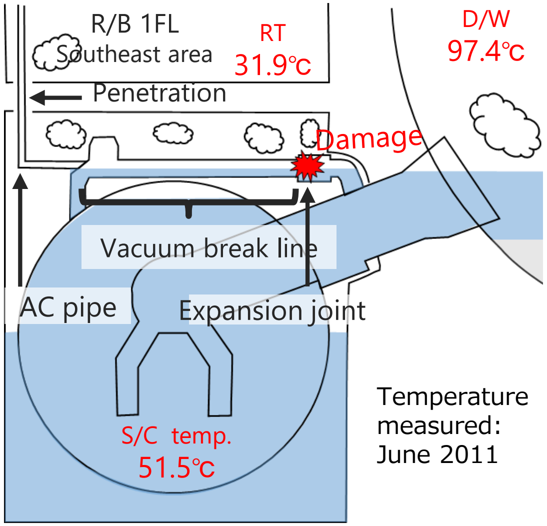
(3) Migration of radioactive materials from contamination sources (causes of steam generation)

Possible steam generation from stagnant water in torus room	×
・Because steam was not confirmed at other penetrating parts, and water temperature in torus room was not high, torus room stagnant water was not thought to be the steam source.	
Possible steam generation from stagnant water in PCV	○
・The AC pipe floor penetration was located almost directly above the expansion joint of the vacuum break line where the leak was confirmed. ・PCV water level at the time was about the level of the expansion joint breakage point, and there was a possibility that gas (steam) inside the PCV was leaking out.	

Discussion on next page



Dose rate in the torus room (on the catwalk; measured in May 2014)



Estimated steam generation path

Examination of possible sources of high dose contamination
①Contamination from the steam and torus room

The results of γ camera measurements and floor sample analysis in the southeast area indicated that there was no significant contamination from the steam and its impact on the southeast area was not considered dominant.

γ camera measurement results

- High dose was confirmed at the AC piping in the photo center.
- No contamination of surrounding structures expected to be contaminated by steam.
- No significant contamination at floor penetrations where steam was leaking.

Floor sample analysis

Sample A

- Near X-6 penetration
- No water marks on surface

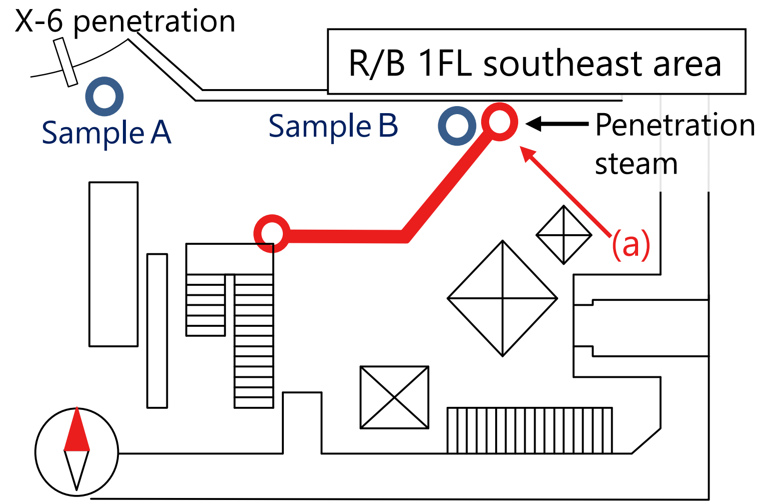
Surface dose rate: 0.14mSv/h

Sample B

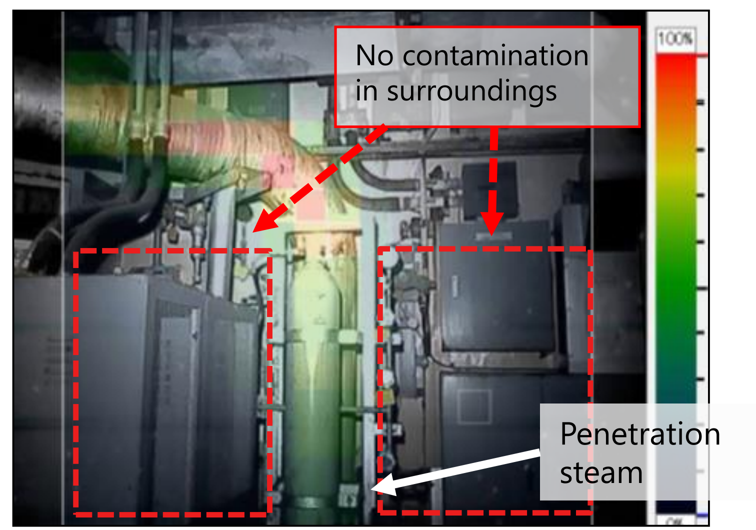
- Near the penetration area where steam leaked out
- Water marks on surface

Surface dose rate: 0.38mSv/h

→Dose rate of B was 2.7 times A dose rate, but not significantly high enough to affect the high dose rate observed in the southeast area.



γ camera measurement position and sampling position



γ camera photo (direction (a); December 2013)

Examination of possible sources of high dose contamination
②Contamination from AC piping

The results of γ camera measurements showed contamination along the AC piping, and although there was no transfer of radioactive materials from inside the piping to the southeast area, the effect of contamination of the AC piping was considered to be dominant.

(1) Causes of contamination

・AC piping used for venting was contaminated by radioactive materials that passed inside piping during venting (contamination was confirmed along AC piping)

(2) Radiation effects from contamination sources

・Based on γ camera measurement results, air dose rate at 150cm above the floor due to AC piping was evaluated to be about 900mSv/h

→The air dose rate in the southeast area (>1000mSv/h) was generally consistent with air dose rate at 150cm above the floor.

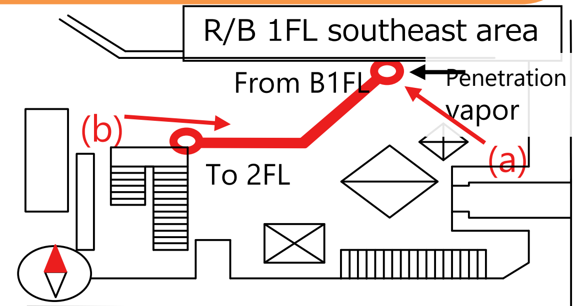
・AC piping came from the penetration where steam was flowing out, passed 200cm above the floor in the southeast area, and exited to the second floor.

→High dose rate at the piping bend at high elevation, consistent with the fact that the dose rate was higher at 150cm above the floor than at 5cm above the floor.

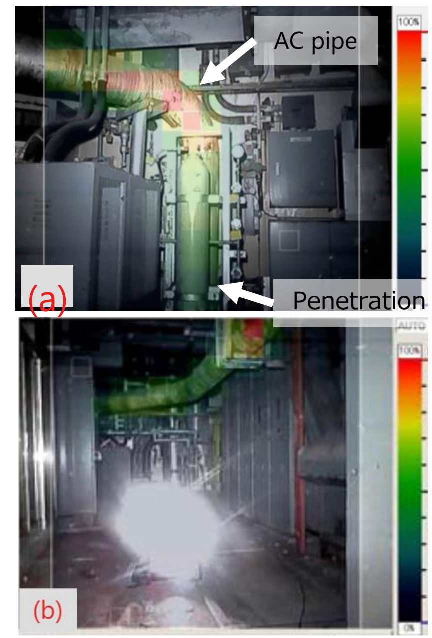
(3) Migration of radioactive materials from contamination sources

・γ camera measurement confirmed contamination along AC piping.

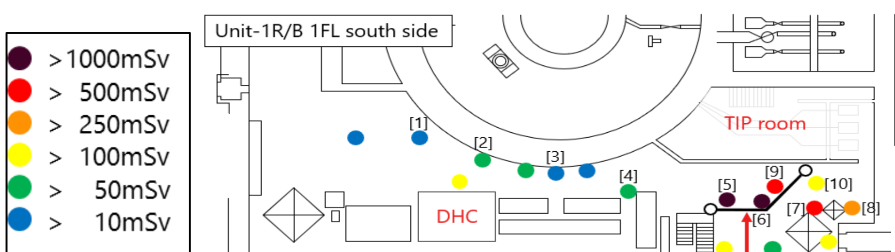
→Contamination remained within the piping and there was no transfer of radioactive materials to the southeast area.



γ camera location in southeast area



γ camera photo (Dec. 2013)



※Measurement at 150cm above floor

Position	1	2	3	4	5	6	7	8	9	10
150 cm above floor	31	62	26	52	1662	1097	511	314	659	203
5 cm above floor	20	41	32	31	186	331	274	244	158	144

Air dose rate (mSv/h) south side (Dec. 2013)

Examination of possible sources of high dose contamination
③Contamination from RCW piping

Radiation from the 2nd floor RCW-Hx and 1st floor SHC pump room, where high dose rates were observed in the RCW system, was sufficiently shielded by the concrete body, and there was no evidence of RCW system water leakage in the vicinity of the 1st floor southeast area, so the impact of contamination in the RCW piping on this area was not considered dominant.

(1) Causes of contamination

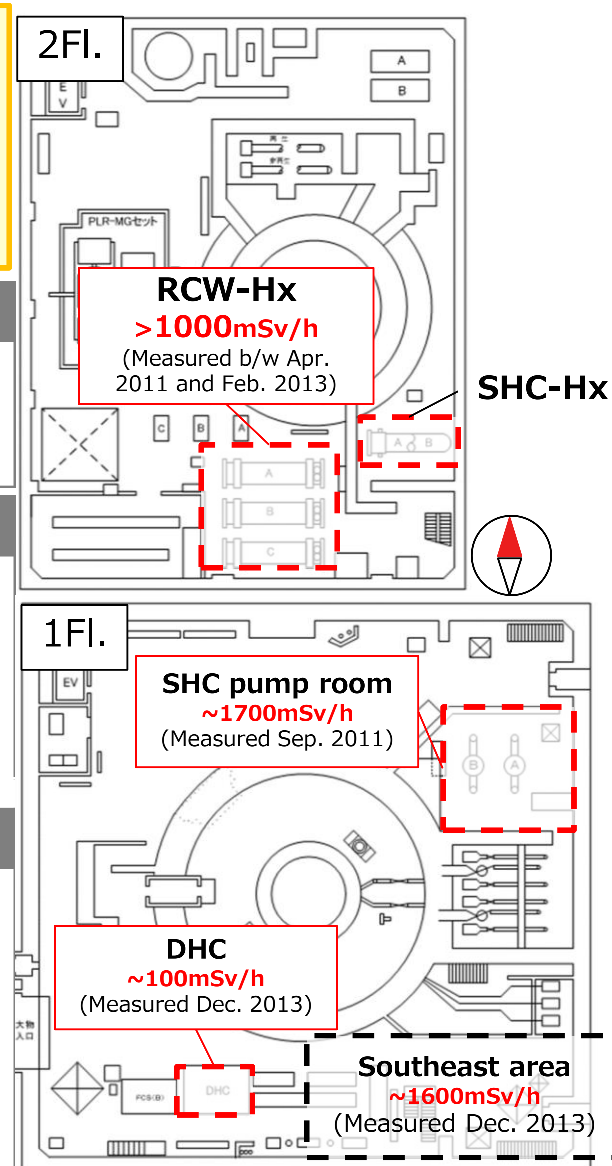
- Molten fuel that fell to the PCV bottom damaged RCW piping, and radioactive materials migrated through the RCW piping and were estimated to have been retained in the system.

(2) Radiation effects from contamination sources

- More than 1000mSv/h was confirmed around RCW-Hx on the 2nd floor and in the SHC pump room on the 1st floor.
  →The concrete shielding (600mm or thicker) of the floor and walls was sufficient to attenuate the radiation in the southeast area of the 1st floor, so the impact was not significant.

(3) Migration of radioactive materials from contamination sources

- Possible residual RCW system water in DHC, RCW-Hx, and SHC-Hx, which could cause high dose rate if leaked to the southeast area.
- Water traces were observed near the AC system piping floor penetration in the southeast area.
  →The surface dose rate of the collected floor sample (0.38mSv/h) was not significantly different from the surface dose rate of the area without the water traces (0.14mSv/h), suggesting that there was no leakage of RCW system water.



RCW system loads around the southeast area

Examination of possible sources of high dose contamination
④ Contamination from TIP room

Since radiation from high dose areas in the TIP room was sufficiently shielded by the concrete body and there was no transfer of radioactive materials into the TIP room, the effect of contamination in the TIP room on the southeast area was not considered to be dominant.

(1) Causes of contamination

・It was presumed that the TIP instrumentation dry tube that contacted with the molten fuel was damaged and radioactive material migrated into the TIP instrumentation.

(2) Radiation effects from contamination sources

・Approximately 300mSv/h dose rate was observed near the X-31 penetration area using the γ camera.

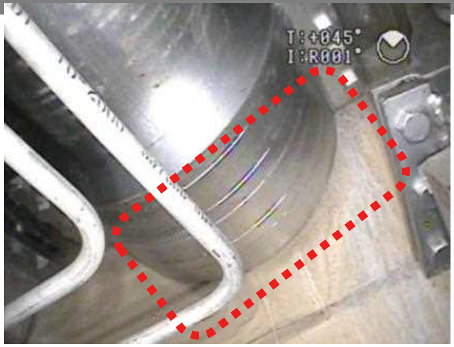
→Concrete shielding (750mm thick) was sufficient to attenuate radiation in the southeast area, so the effect was not significant.

(3) Migration of radioactive materials from contamination sources

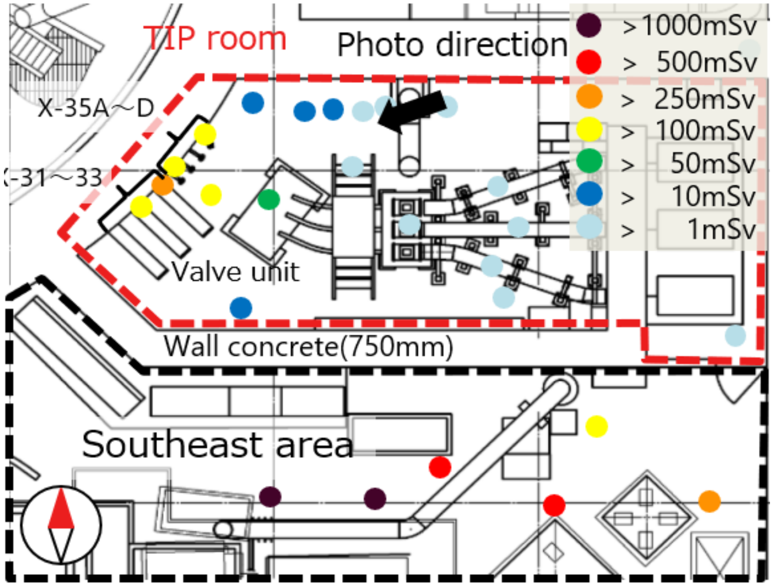
・No evidence of leakage in the X-31 penetration area.

・Dose rate in the room was low (several tens of mSv/h).

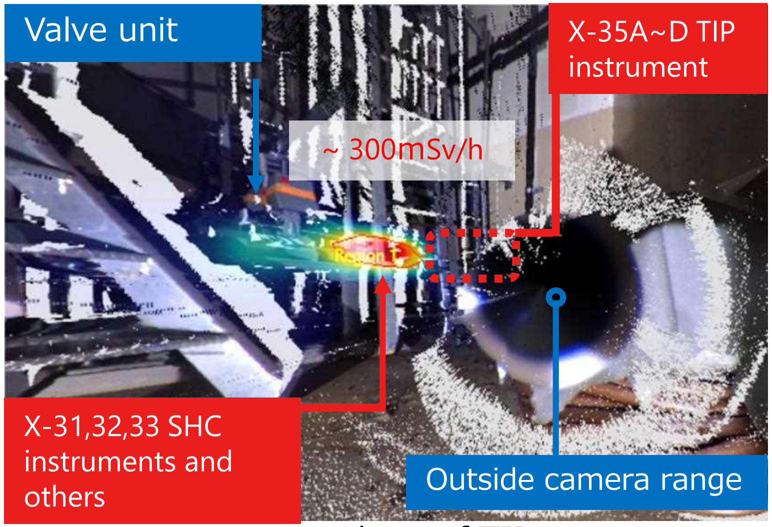
→Contamination was considered to have remained inside the penetration area and did not migrate into the TIP room.



X-31 penetration



Dose rate in the TIP room (Sep. 2015) and dose rate in the southeast area (Dec. 2013)



γ camera photo of TIP room

Summary of examination

・The possible sources of the high radiation dose rate observed in the southeast area of the first floor of Unit 1 were identified.
・The contamination sources were examined in terms of (1) the causes of contamination, (2) radiation effects, and (3) radioactive material transfer to the southeast area.
・As a result of the review, the radiation impact from the AC piping used for the PCV venting was identified as the dominant factor.

Possible contamination sources	Result	Details of the examination results		
		Radiation effects	Migration	Examination contents
①Contamination from the steam and torus room	✕	×	×	・Steam from PCV water was blowing out, but it was not a noticeable contamination. ・Attenuation due to shielding of concrete body.
②Contamination from AC piping	〇	○	×	・Dose rates similar to those in the southeast area were observed around the AC piping. ・Contamination was distributed along the pipe, no leakage.
③Contamination from RCW piping	✕	×	×	・Attenuation due to shielding of frame concrete. ・No leakage of RCW water to the southeast area.
④ Contamination from TIP room	✕	×	×	・Attenuation due to shielding of frame concrete. ・No leakage into TIP room from X-31 penetration where high dose rate was observed.

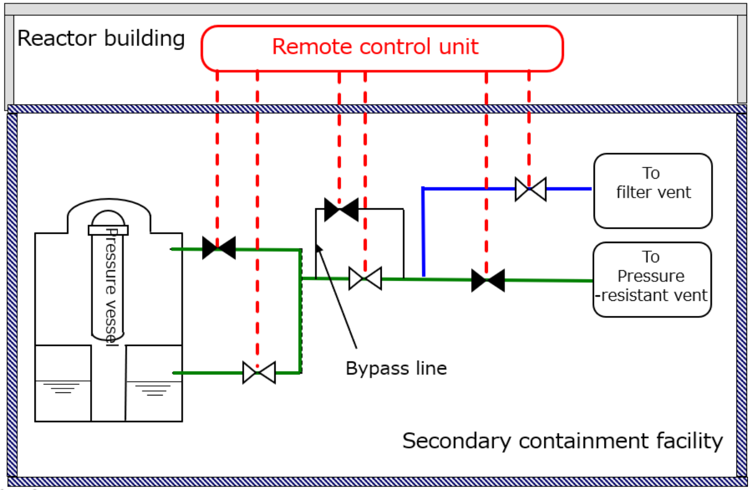
Safety measures in the Kashiwazaki-Kariwa NPS

2. Identification of causes of high dose rate observed in the southeast area on the 1st floor of Unit 1 Reactor Building

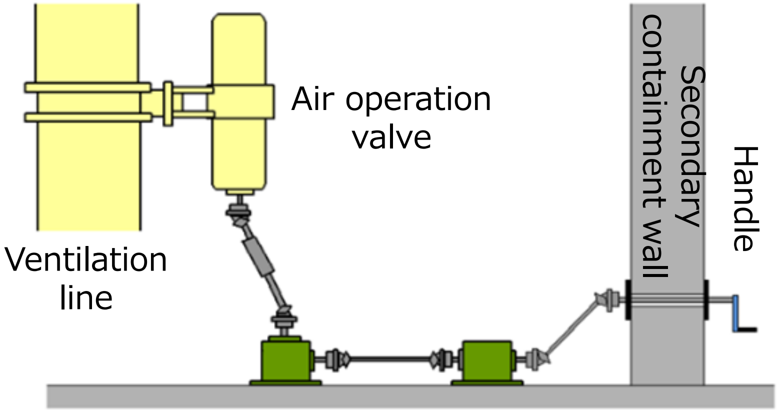
Lesson Learned: Measures need to be taken to prevent radiation from the vent line from affecting accident response operations

In addition to measures to prevent core damage and to remove heat from the PCV while maintaining the PCV boundary using alternative circulation cooling, the following measures to reduce exposure from the filter vent system are implemented.

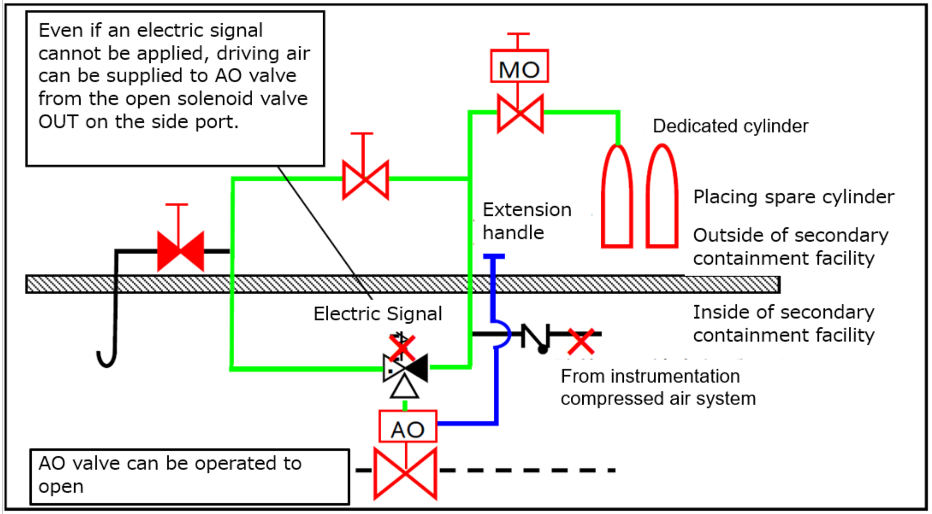
- Valves that need to be opened during venting can be remotely operated electrically from the central control room. A bypass line is provided for the secondary isolation valve to prevent remote electric operation failure from the central control room due to valve failure itself (right figure).
- If remote electric operation is disabled due to loss of power, etc., the vent line valves can be remotely operated from outside the secondary containment facility by human power (lower left figure) or by a dedicated cylinder (lower right figure).
- Remote manual operation during venting after core damage reduces exposure.
- Shielding is installed on the filter system, iodine filter, and outdoor piping connected to the filter system to reduce exposure during outdoor operations.
- The system is evaluated as operable from the viewpoint of radiation dose.



Outline of vent line



Outline of remote manual control device

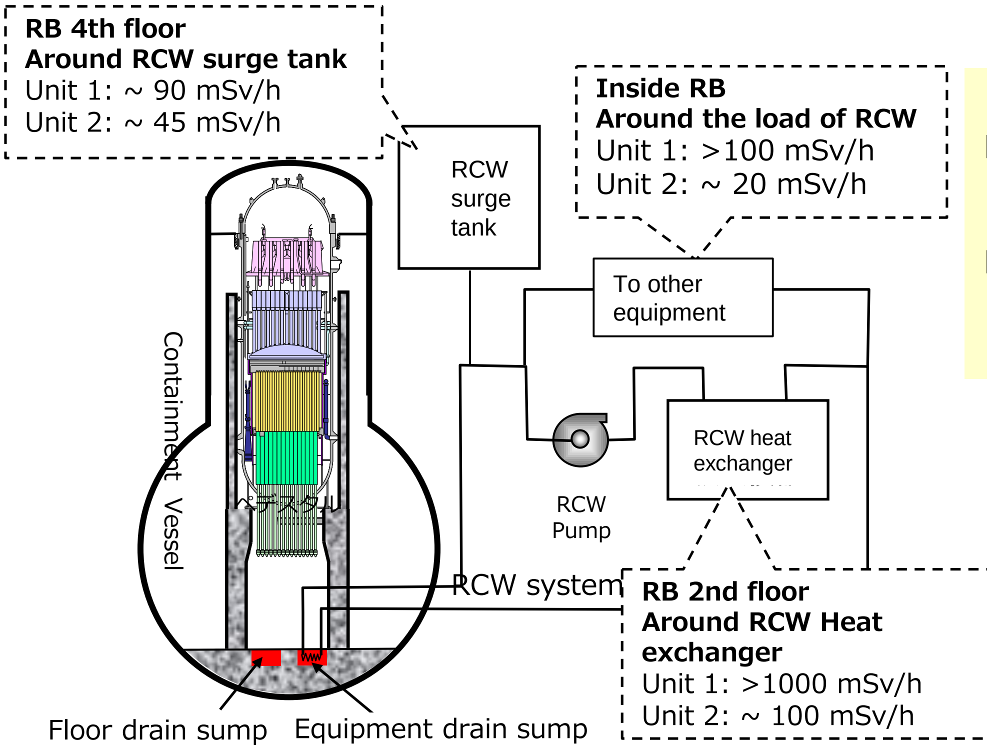


Outline of AO valve operating mechanism with dedicated cylinder

3. Estimation of reasons for high dose rate not being observed in Unit 2 Reactor Building Cooling Water System

Overview

- In Unit 1, a high dose rate was observed around equipment that was a load on the reactor building cooling water system (RCW)*. It was presumed that the RCW pipe in the equipment drain sump was damaged and that contamination spread throughout the RCW system (already presented in the 4th progress report).
- On the other hand, it was assumed that some of the fuel fell from the reactor pressure vessel to the containment vessel in Unit 2, but no significant traces of contamination were found in the RCW system.
- Since clarifying this difference will contribute to estimating the distribution of fuel debris as well as progress of the accident, this study estimates why high dose rates were not observed in the Unit 2 RCW system.



Units 1 and 2 RCW system and image of contamination

< Approach to Examination >

- Consideration of the RCW piping condition in containment vessel based on results of investigation inside Unit 2 containment vessel.
- Consideration of whether the plant data at the time of the accident indicated a situation where contamination could have spread into the system.

Based on the results of the internal investigation of the containment vessel, it was estimated that why the high dose rate was not observed in the RCW system of Unit 2 was because, unlike Unit 1, the RCW piping was not damaged.

Status of Reflection on Safety Measures at Kashiwazaki-Kariwa NPS: Measures to prevent the spread of contamination due to piping damage in the containment vessel

*System for cooling equipment in the reactor building and other locations. Closed-loop design with no openings to the reactor pressure vessel or containment vessel.

Comparison of Units 1 and 2 situations

- Only Unit 1 had significant contamination in the RCW system.
  - High radiation dose rate was observed around RCW system only in Unit 1.

  (Example around the RCW heat exchanger: Unit 1, > 1000mSv/h; Unit 2, ~ 100mSv/h)

- In both units, fuel was estimated to have fallen into the containment vessel.
  - In Unit 1, a small amount of fuel debris might be present in the RPV, but most of it was estimated to have fallen into the containment vessel.
  - In Unit 2, most of the fuel debris was estimated to be at the bottom of the RPV, and some of it had fallen into the containment vessel.

- In both units, the containment isolation valves in the RCW system were estimated to have been open after the accident.
  - The containment isolation valve in the RCW system was an electrically operated valve.
  - They were not designed to automatically isolate (close) the containment.
  - Units 1 and 2 lost all power when the tsunami hit, and the valves could no longer operate.
  - There was no record of any operation to close the containment isolation valves in the RCW system during the response to the accident.

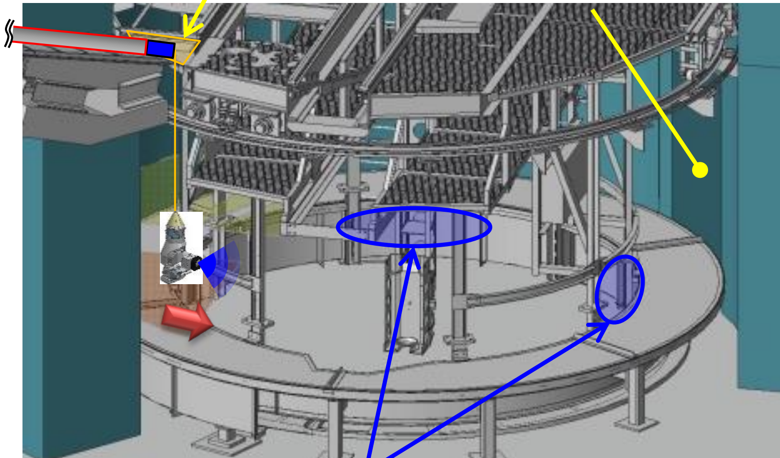
■ The situations in Units 1 and 2 were similar in that fuel had fallen into the containment vessel of both units and contamination could have spread within the RCW system.

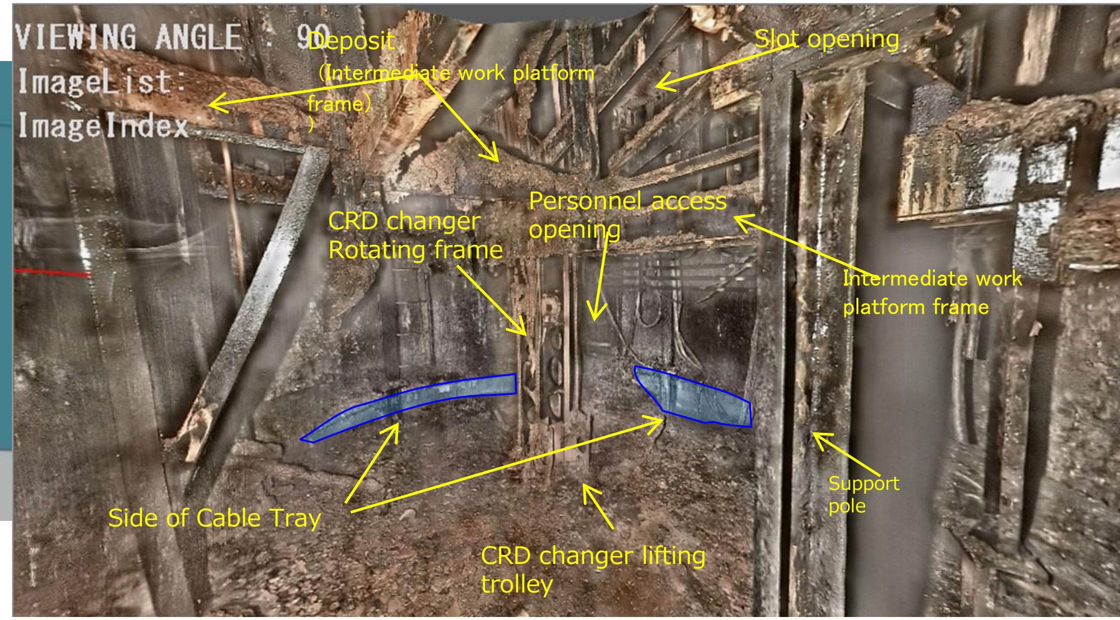
→The results of the in-containment vessel investigation were used to determine why there was a difference in contamination.

State of the Unit 2 containment vessel bottom

3. Estimation of reasons for high dose rate not being observed in Unit 2 Reactor Building Cooling Water System

- January 2018 investigation inside Unit 2 containment vessel confirmed state of the containment vessel bottom.

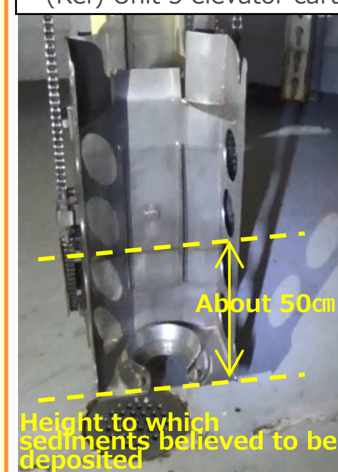




- Deposits looking like pebbles and clay were all over the pedestal bottom.
- It was confirmed that there was no major deformation or damage to structures such as the rotating frame of the CRD exchange machine, the frame of the intermediate work platform, the struts, and the cable tray.
- No deformation of the cable tray (stainless steel, 4mm thick) was observed, although it had a deposit that looked like solidified melt.

⇒It was possible that when deposition on the cable tray started, the temperature of the deposited material was not at the temperature causing the tray thermal deformation.

(Ref) Unit 5 elevator cart



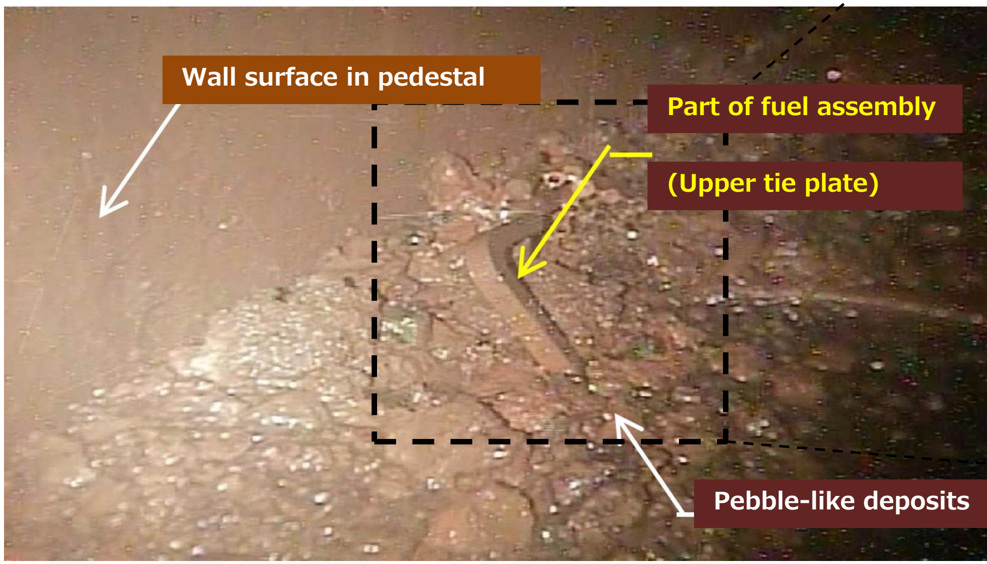
(Ref) During Unit 2 regular inspection

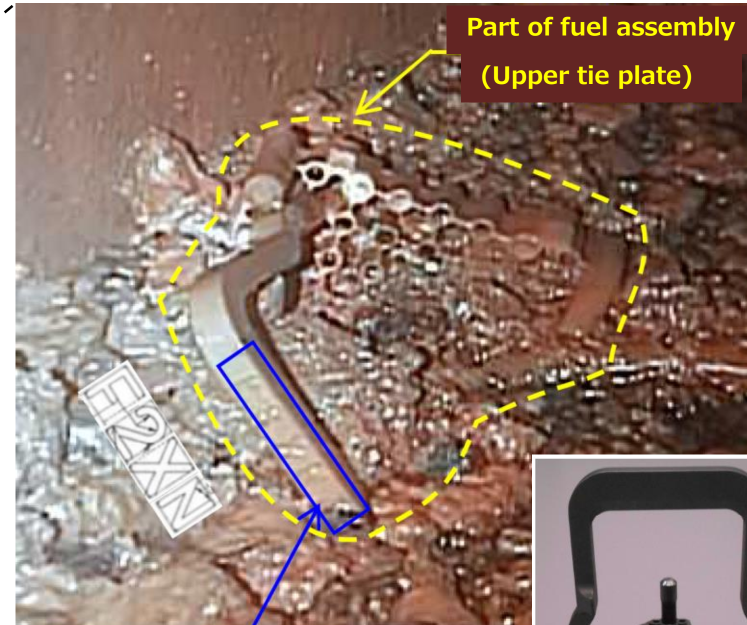
※Structures are removed from PCV during operation

Image provided courtesy of the International Nuclear Decommissioning Research and Development Organization (IRID)

State of the Unit 2 containment vessel bottom

- January 2018 investigation inside Unit 2 containment vessel confirmed state of the containment vessel bottom.





Part of fuel assembly
(Upper tie plate)

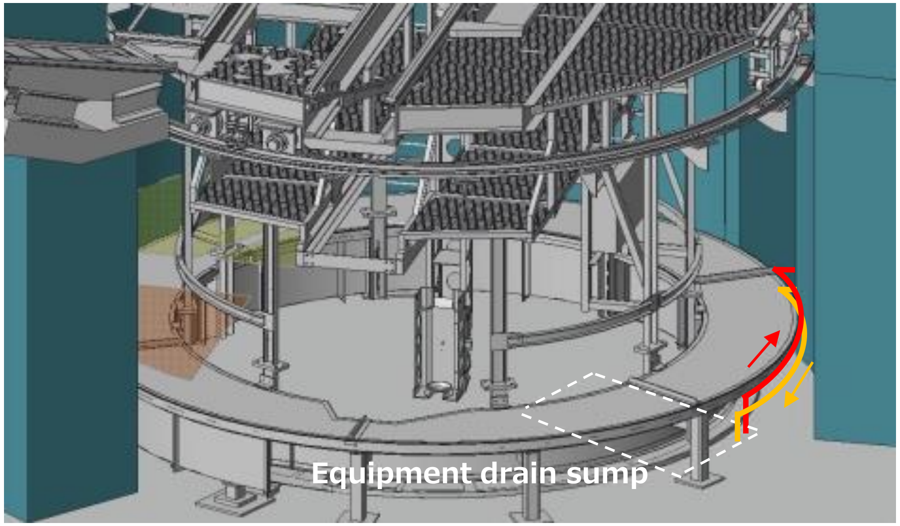
■ A part of the fuel assembly (upper tie plate) from the reactor core was confirmed to be at the bottom (periphery) of the containment vessel inside the pedestal.

⇒It was estimated that at least a hole was made in the reactor pressure vessel to the extent that the upper tie plate fell through. Sediments around the upper tie plate and other areas were presumed to contain fuel components.

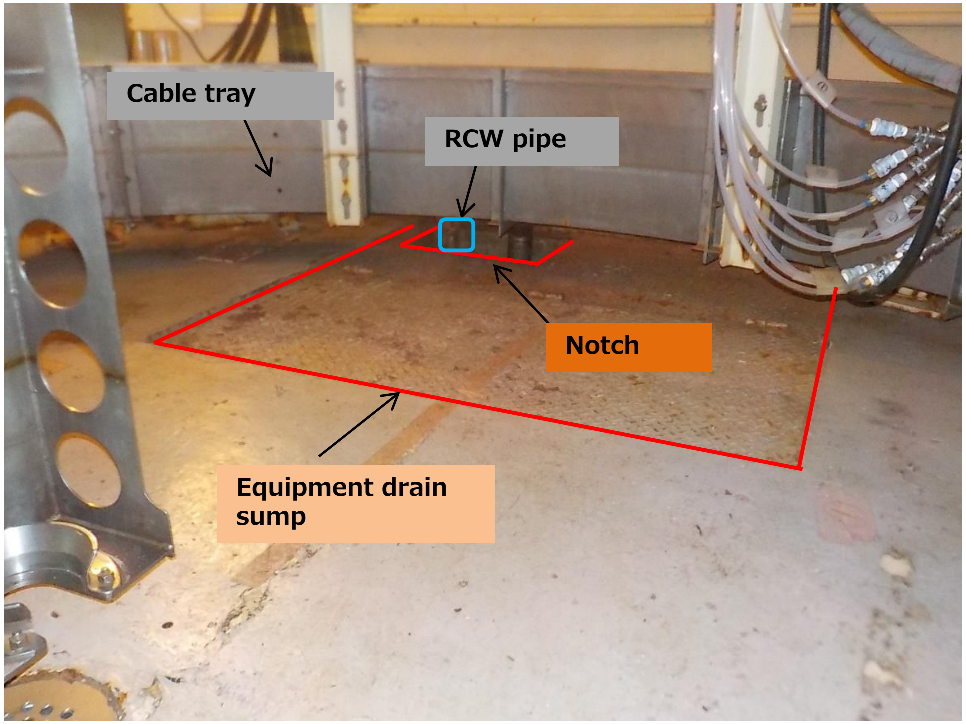
⇒However, since no damage was observed in the structure at the bottom of the containment vessel, fuel debris was presumed to contain a large amount of metal.

Image provided courtesy of the International Nuclear Decommissioning Research and Development Organization (IRID)

Discussion of reasons why high dose rates were not observed in the Unit 2 RCW system



RCW piping in pedestal
(Arrows: Normal cooling water flow)



Equipment drain sump in Unit 5 (similar structure in Unit 2)

	Thickness	Material	Melting point
Cable tray	~ 4mm	Stainless steel	~ 1450℃
Equipment drain sump lid	~ 3mm	Carbon steel	~ 1500℃
RCW Piping	~ 3.7mm	Carbon steel*	~ 1500℃*

*Material near the equipment drain sump was stainless steel (m. p. ~1450°C).

■ Melting points of RCW piping and cable tray materials were close.
⇒Equipment drain sump lid, RCW piping, and the cable tray might not have been damaged in Unit 2.
⇒Presumed reason why high dose rate was not observed in the Unit 2 RCW system.

Summary

- **Estimation of reasons why high dose rates were not observed in the Unit 2 RCW system**

From results of the investigation inside the Unit 2 containment vessel, it was presumed that the reason why a high dose rate was not observed in the RCW system of Unit 2 was that the RCW piping was not damaged unlike Unit 1.

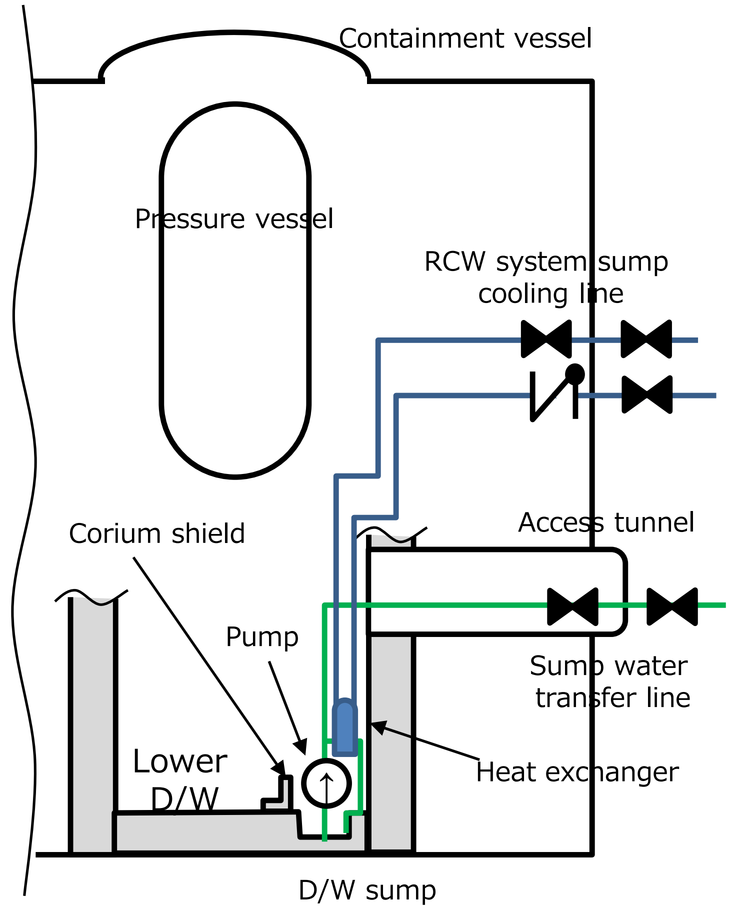
- **[Supplemental] The state of Unit 3**

The RCW system in Unit 3 showed no signs of contamination, as in Unit 2. However, in Unit 3, a reasonable amount of fuel debris was estimated to have fallen into the containment vessel, as about 2 to 3m deep deposits were observed on the bottom of the containment vessel.
The situation in Unit 3 was different from that in Unit 2, and the reason why high dose rates were not observed in the RCW system in Unit 3 is not clear.
⇒
Estimating the causes of the accident is important for understanding the distribution of fuel debris in Unit 3 and the progress of the accident. Therefore, we will continue to examine these causes based on the results of future investigations.

Safety measures in the Kashiwazaki-Kariwa NPS

Lesson learned: Preventing the spread of contamination caused by damage to piping in the containment vessel, including RCW piping, is critical.
(Unlike Unit 2, contamination spread to the RCW system in Unit 1, affecting restoration work.)

- For RCW piping that penetrates the containment vessel, in the original design, containment vessel isolation valves (or check valves) are installed both inside and outside the containment vessel at positions close to the penetration. These valves are designed to automatically isolate themselves when the reactor water level drops or the D/W pressure rises, and to close before the RPV is damaged (the check valves prevent reverse flow from inside the containment vessel to outside). This prevents spread of contamination to the piping outside the containment vessel.
- The drive power supply for the containment vessel isolation valves is strengthened by gas turbine generators, power interchange between units, and power supply vehicles.
- Before the accident at the Fukushima Daiichi NPS, the procedure had been adopted that the lower D/W would be filled with water before the RPV was damaged, and the water level was maintained to cool down the fallen molten fuel. At Kashiwazaki-Kariwa NPS Units 6 and 7, in addition to the MUWC system as a means of injecting water into the lower D/W, fire engines will be used to inject water to reduce the risk of damage to the pipes that act as routes for the spread of contamination.
- The D/W sump has a line that transfers the sump water to the outside of the containment vessel, and isolation valves with an automatic isolation function are installed inside and outside the containment vessel penetration like in the RCW system. In addition, a corium shield is installed on the lower D/W to prevent melted fuel from entering the sump.



Isolation valves in RCW system and sump water transfer line

4. Decrease in Unit 2 containment vessel pressure in the morning of March 15

Overview

- The D/W pressure in Unit 2 remained above 0.7MPa[abs] from around 23:30 on March 14 to 7:20 on March 15, and had dropped to 0.155MPa[abs] by 11:20 on March 15, when the once interrupted measurement resumed.
- Since the decrease in PCV pressure is related to the release of radioactive materials, it is important to elucidate the behavior of this pressure decrease. In this study, we examined scenarios consistent with the indicated values of plant parameters such as RPV pressure and PCV pressure and the observed facts.

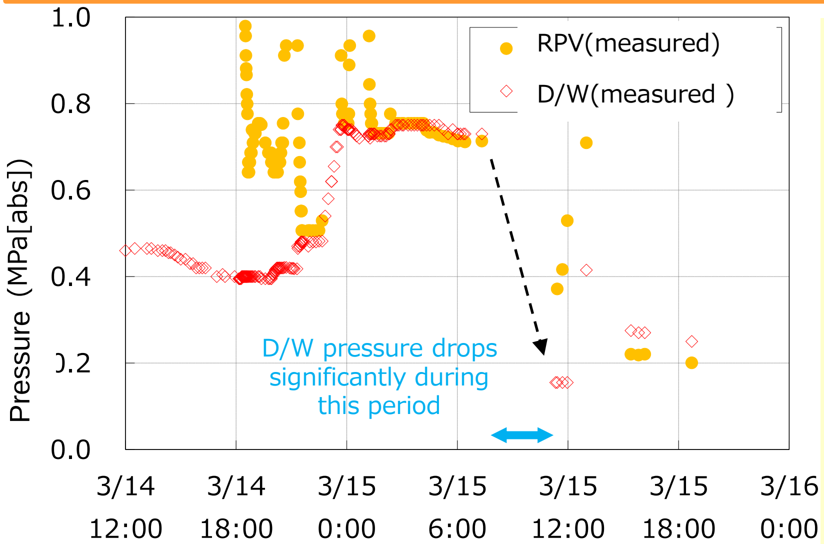


Fig. Changes in RPV and PCV pressures

< Approach to Examination >

The feasibility of the following two scenarios was examined.

① **Depressurization due only to large-scale gas phase leakage from the PCV**
- Evaluate the PCV vapor phase leakage area that reproduces depressurization
- Examine the feasibility of the scenario based on observed facts, etc.

② **In addition to gas phase leakage from PCV, decompression occurred due to condensation of water vapor inside the PCV**
- Assume a scenario in which condensation in the PCV was accelerated
- Evaluate depressurization behavior in the assumed scenario
- Examine feasibility of the scenario based on observed facts, etc.

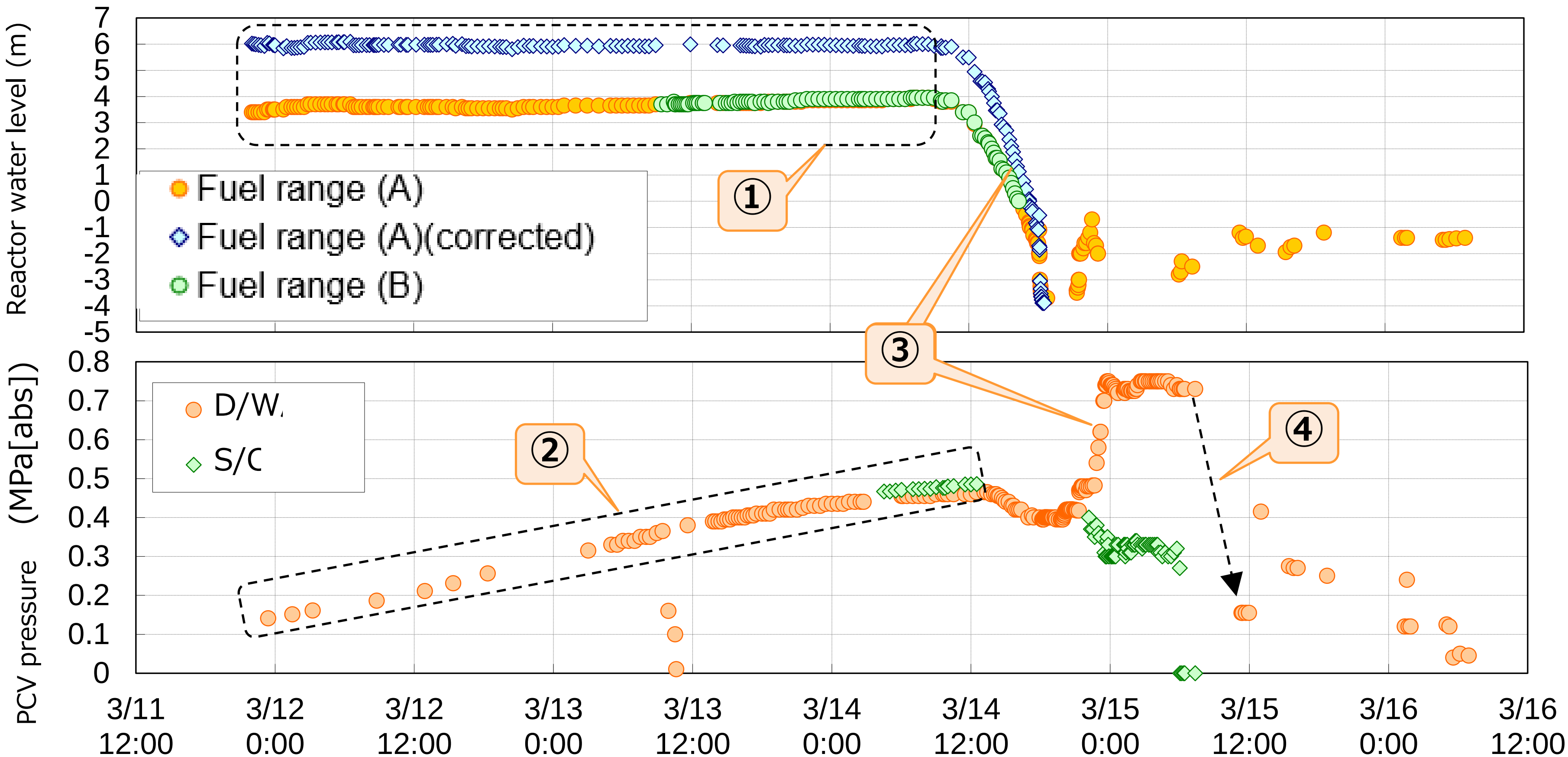
- The depressurization scenario based solely on a large gas phase leak from the PCV was inconsistent with some observed facts.

(Relatively high airtightness of the PCV in Unit 2 after the accident, and relatively small amount of contamination in the building outside the operation floor although leakage from other than top head flange must also be considered)

- Considering that condensation of water vapor contributed to depressurization in addition to a small leak, there were many points that were consistent with observed facts. However, since the effect of condensation greatly depends on the state inside the PCV, we will continue to examine whether the accident progressed in such a way.

Overview (Accident Progress at Unit 2)

4. Decrease in Unit 2 containment vessel pressure in the morning of March 15



【Approximate accident progress】

①Reactor water level maintained by continuing RCIC operation.

②Rise in PCV pressure during this period was slower than that expected from the decay heat.
⇒It was assumed that the torus room where S/C was housed was inundated by the tsunami and the S/C was cooled from the outside.

③Loss of RCIC water injection function at around 9:00 on March 14 caused the reactor water level to drop, presumably leading to fuel meltdown during the night of the same day.
⇒Rise in PCV pressure due to hydrogen generation.

④PCV pressure dropped significantly in the morning of March 15←**Examination of this declining behavior**

TEPCO

Examination of decompression scenario due to gas phase leakage from PCV

Evaluation of PCV gas phase leakage area that reproduces decompression

Possible decompression factors ①: Decompression due to gas leakage from PCV
Analysis (using the GOTHIC code) evaluated the PCV gas phase leakage area that reproduced the depressurization after 7:20 on March 15, and the results indicated that large-scale leakage must continue throughout the depressurization period.

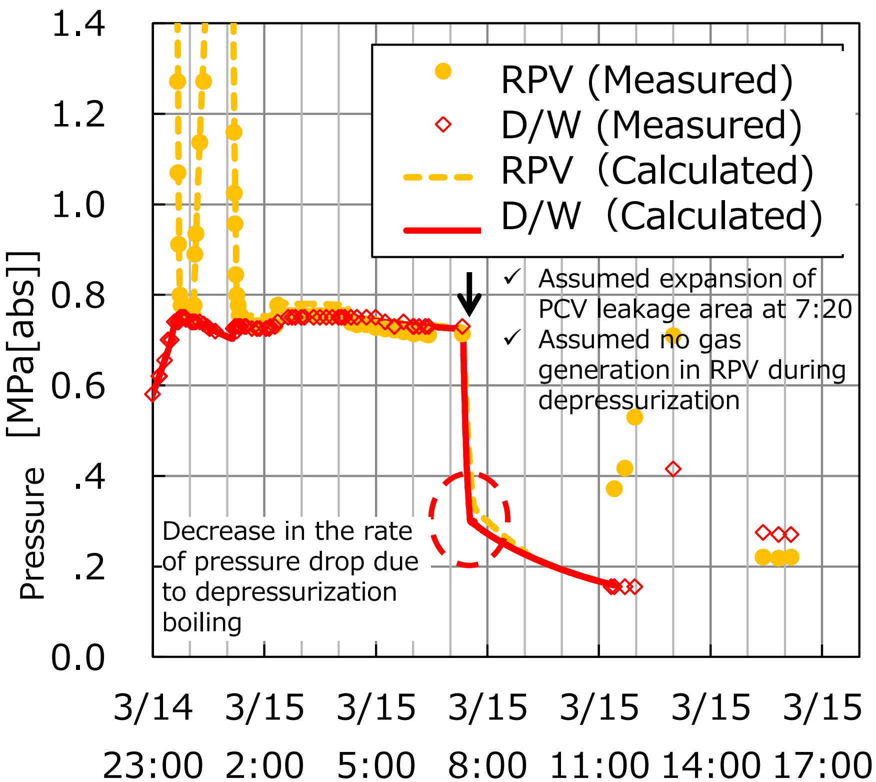


Fig. Analysis results given PCV leakage area that reproduced D/W depressurization

* Japan Nuclear Technology Institute, "Work on MARK I Reactor Containment Vessel Elasto-Plasticity Analysis for Severe Accident Response Standard Development, FY2011 Report," (2012)

The PCV leakage area required to reproduce the pressure drop was confirmed by analysis.
⇒Set 300cm^2 throughout depressurization (left figure).

- While the S/C was externally cooled, the water temperature in the S/C pool rose due to the effects of long-term RCIC operation and fuel melting.

⇒In the analysis, it was assumed that the water temperature of the entire S/C pool had risen uniformly. As a result of decompression boiling of the S/C pool, it became difficult to decompress.

⇒ A large leak area was required to reproduce the actual measurement value of the D/W pressure.

- The first candidate for the leakage point was the PCV top head flange.

⇒Possible leakage mechanism
①Change in clearance area according to PCV pressure
②Thermal degradation of silicone rubber in the sealing part

⇒①alone could not reproduce the measured depressurization behavior because the leakage area decreased during the depressurization process.

⇒According to the structural analysis* of the MARK-I containment vessel, even under high-pressure and high-temperature conditions assumed before depressurization, and even if the silicone rubber of the seal was considered missing, which could not be reproduced in ②, the opening area of the top head flange was less than 300 cm^2.

⇒**If depressurization due to vapor phase leakage from PCV is the main cause, leakage from other than PCV top head flange should also be considered.**

Examination of decompression scenario due to gas phase leakage from PCV

Evaluation of PCV gas phase leakage area that reproduces decompression

It is necessary to consider that there was a reasonable amount of leakage besides the top head flange due to thermal damage, etc. However, it is difficult to explain the consistency with the observed facts, such as the relatively airtight PCV in Unit 2 after the accident and the relatively small contamination in the building outside the operation floor.

⇒Depressurization after 7:20 on March 15 was unlikely to be caused solely by gas phase leakage from PCV.

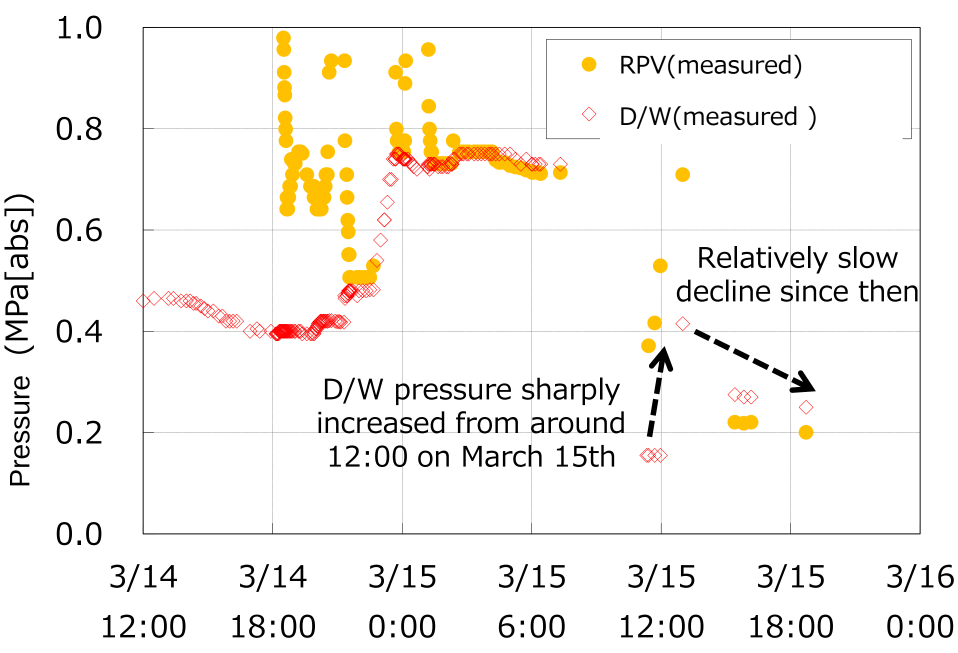


Fig. Changes in RPV and PCV pressures

- To reproduce the rapid increase in D/W pressure from around 12:00 on March 15 and subsequent relatively gradual decrease (left figure), more energy than the amount of heat that could be released from the fuel (decay heat + heat storage) is required.

⇒**Difficult to explain the behavior of D/W pressure when assuming a large leakage.**

- After the accident progress had settled down, Unit 2 PCV was airtight compared to other units, and the leakage area calculated from pressure balance was less than 1 cm^2.

⇒To reproduce the D/W pressure in the left figure, it is necessary for the leakage opening, which once opened wide, to shrink.

⇒**It is unlikely the leakage opening area, which was thermally damaged and maintained during depressurization, will shrink significantly thereafter.**

- In the reactor building, except for the top head flange, high doses have been observed in some PCV boundaries such as X-6 penetration, but no particularly high doses have been observed in areas (stairs, etc.) that are migration pathways for radioactive materials.

⇒**It is difficult to assume the main gas phase leakage point is other than the top head flange.**

Examination of decompression scenario due to condensation in PCV

Assuming a scenario that promotes condensation in the PCV after 7:20

Possible decompression factors ②: Decompression by water vapor condensation

One possible scenario for increased condensation after 7:20 compared to earlier times is that the water level in the torus room rose and exceeded the S/C pool water level, which might have accelerated cooling of the S/C gas phase and increased condensation of water vapor.

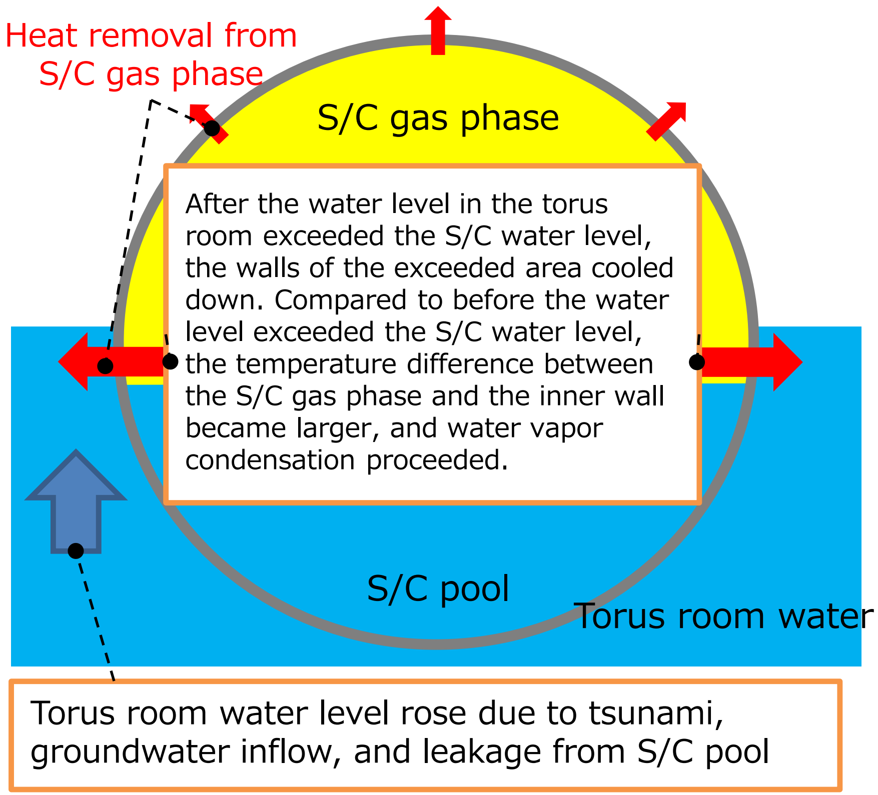


Fig. Image of rising water level in torus room

【Possibility of S/C water level < torus room water level】

- From the behavior of the PCV pressure during RCIC operation, it was estimated that the torus room was flooded with tsunami water from the accident beginning.
- The reactor building and other buildings were connected, and groundwater was flowing into the reactor building based on the behavior of the stagnant water level in that building after the accident.

⇒**Possibly, water level in the torus room was rising.**

- It was estimated that a small leak was concurrently occurring in the lower part of the S/C (or piping connected from the lower part; timing of leakage start is unknown).

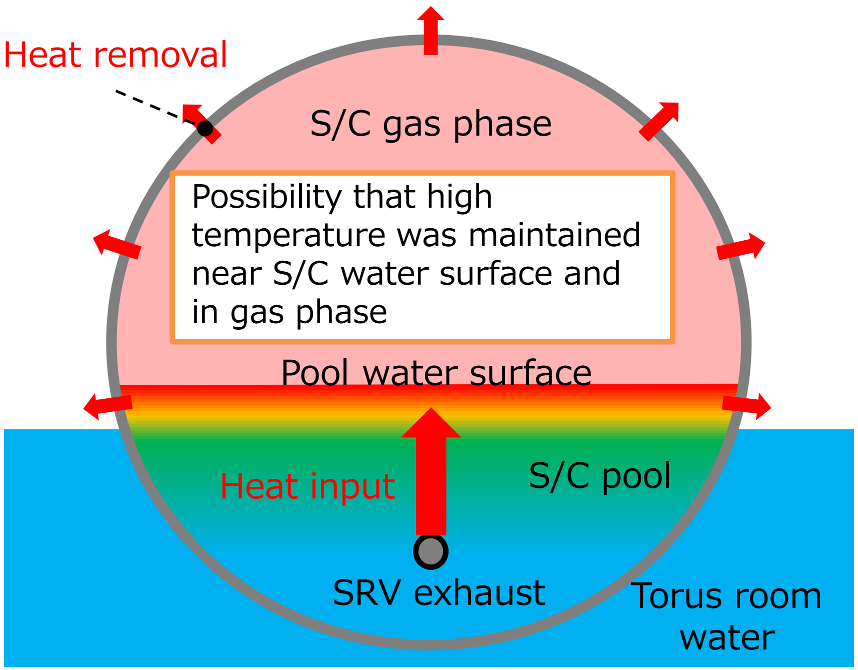
⇒**If there was leakage from the S/C pool during accident progression, the S/C water level would have dropped, and the water level in the torus room might have risen due to leaked S/C pool water.**

【Necessary prerequisites for scenario establishment】

- Most of the non-condensable gases that inhibited depressurization had been vented out of the PCV prior to depressurization.
- The PCV was almost filled with water vapor, the PCV pressure before depressurization was maintained, and the amount of boiling under reduced pressure was small, so only the surface layer of the S/C pool was in a high temperature state.

Examination of decompression scenario due to condensation in PCV
Examination of feasibility of scenario prerequisites

Prerequisites
①Most of the non-condensable gas in PCV had been released
②It was presumed possible that only the surface temperature of the S/C pool was high before depressurization.



Assumes torus room water level was lower than S/C pool water level before depressurization

Fig. Image of temperature stratification of S/C pool before depressurization

【Feasibility of prerequisite ①】

- Before depressurization, gases (mainly steam) generated in RPV were conducted to S/C via SRV.
- Part of the heat was transferred to the S/C pool water surface, and water surface temperature was maintained, resulting in continuous generation of water vapor from the water surface and maintenance of the PCV pressure.

⇒Continuous water vapor generation might have led to most of the non-condensable gas being discharged from the PCV through the top head flange.

It was evaluated that if the pool water surface temperature was maintained at the saturation temperature (168°C), PCV pressure before depressurization of 750kPa[abs] could be achieved even if the gas in the PCV was only water vapor.

【Feasibility of prerequisite ②】

- The S/C lower part was presumed to be in a cooling state due to water thought to have been in the torus room.
- The pressure difference between the RPV and PCV before depressurization was relatively small, and the stirring effect of the S/C pool water due to SRV exhaust might have been limited.

⇒It was possible that only the surface of the S/C pool was at a high temperature (thermal stratification).

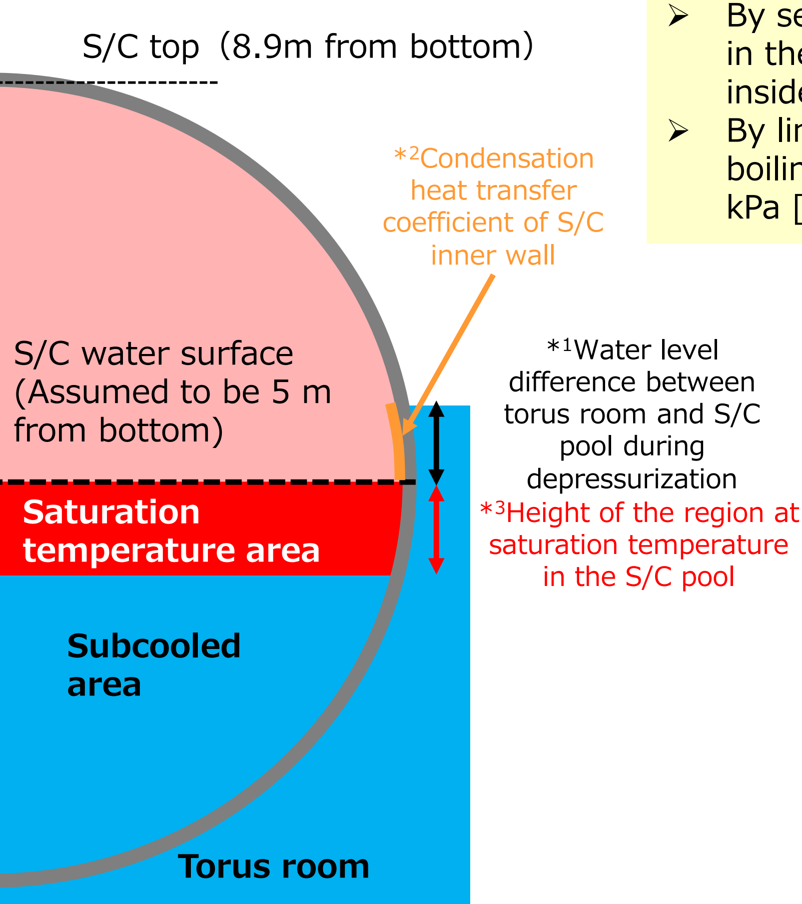
Examination of decompression scenario due to condensation in PCV

Evaluation of depressurization behavior in assumed scenarios

The depressurization behavior, assuming the effect of condensation, was evaluated based on the energy change inside the PCV before and after the pressure change. Assuming a situation where there were few high-temperature regions in the S/C pool and few non-condensable gases in the PCV, the results showed that even small leakage from the PCV could reproduce depressurization.

Leakage area required for depressurization after 7:20 on the 15th, including effect of condensation on S/C wall, was evaluated and found to be significantly reduced from the area of 300cm² required for depressurization due to gas phase leakage alone (below figure).

- By setting the torus room water level > S/C pool water level, S/C wall in the submerged area was cooled and condensation of water vapor inside the S/C was accelerated.*1,*2
- By limiting the saturation temperature region, the amount of vacuum boiling that occurred during the process of PCV decompression to 155 kPa [abs] was reduced.*3



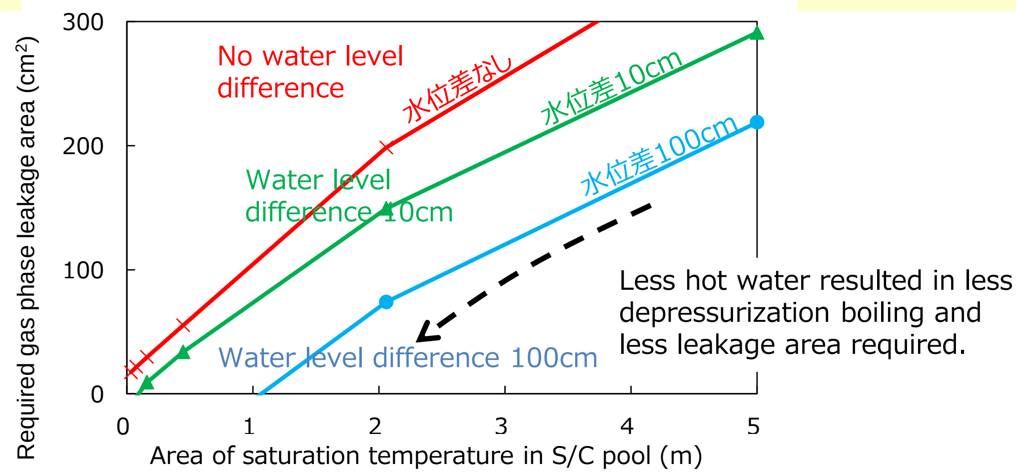


Fig. Variation of required leakage area with respect to height of saturated temperature region for S/C pool water. (Assuming no non-condensable gas in the PCV)

Examination of decompression scenario due to condensation in PCV

Examination of the feasibility of scenarios based on observed facts, etc.

Considering that decompression due to condensation made a contribution, it was possible to explain the consistency with the observed facts, which was difficult in the decompression scenario due to large-scale gas phase leakage.

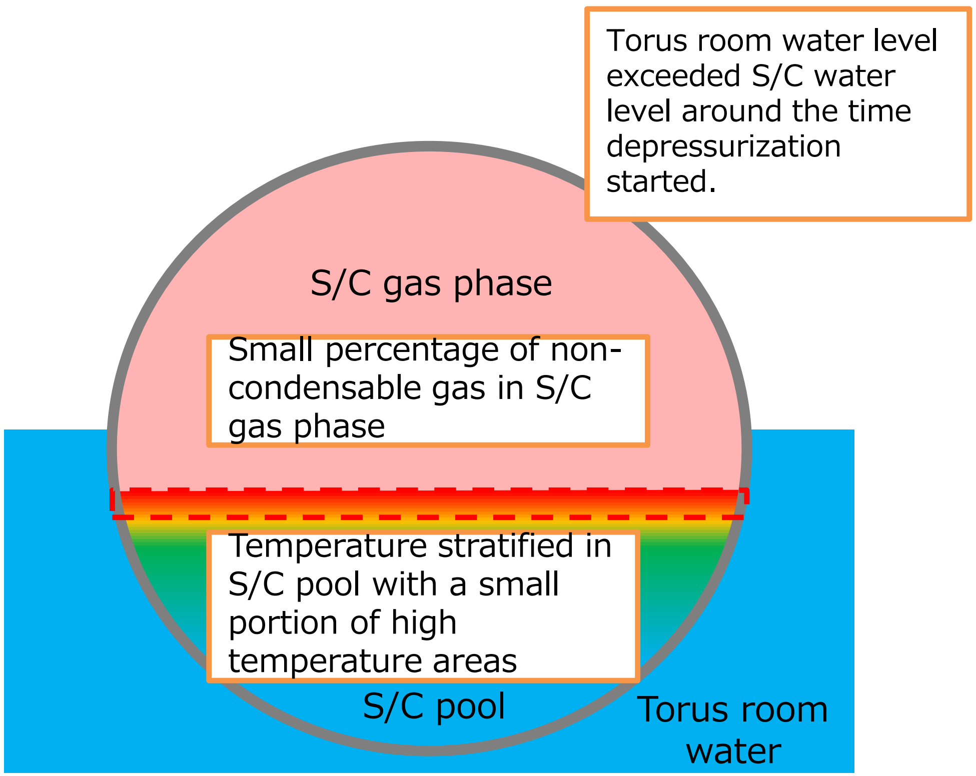


Fig. Image of the state of the S/C pool before depressurization in the depressurization scenario with condensation

Explanation of the consistency with the observed facts when considering that the pressure was reduced due to condensation

- **The rapid increase in D/W pressure from around 12:00 on March 15, followed by a relatively gradual decrease**

⇒If the leak area was small, the energy required for this pressure behavior would decrease, and the possible explanation was that the pressure increased or decreased due to changes in evaporation of the water that cooled the fuel.

- **The current containment vessel of Unit 2 was airtight compared to other units.**

⇒Consistent with the explanation that pressure could be reduced even if the leakage area was small.

- **No particularly high radiation doses could be confirmed at locations in the reactor building where radioactive materials were considered to have migrated.**

⇒It was possible to say that there were no major leaks other than the top head flange.

Summary

➢ Regarding the depressurization behavior of D/W pressure after 7:20 on the 15th, it was indicated that condensation of water vapor might have contributed to it, in addition to a small leakage from the PCV.

(We believe that there was a leakage from the Unit 2 PCV after the core damage, because a white gas, which was thought to be steam, was confirmed to be emitted from the blowout panel of Unit 2 before 9:00 on March 15, and soil contamination in the direction of Iidate Village was thought to have originated from Unit 2.)

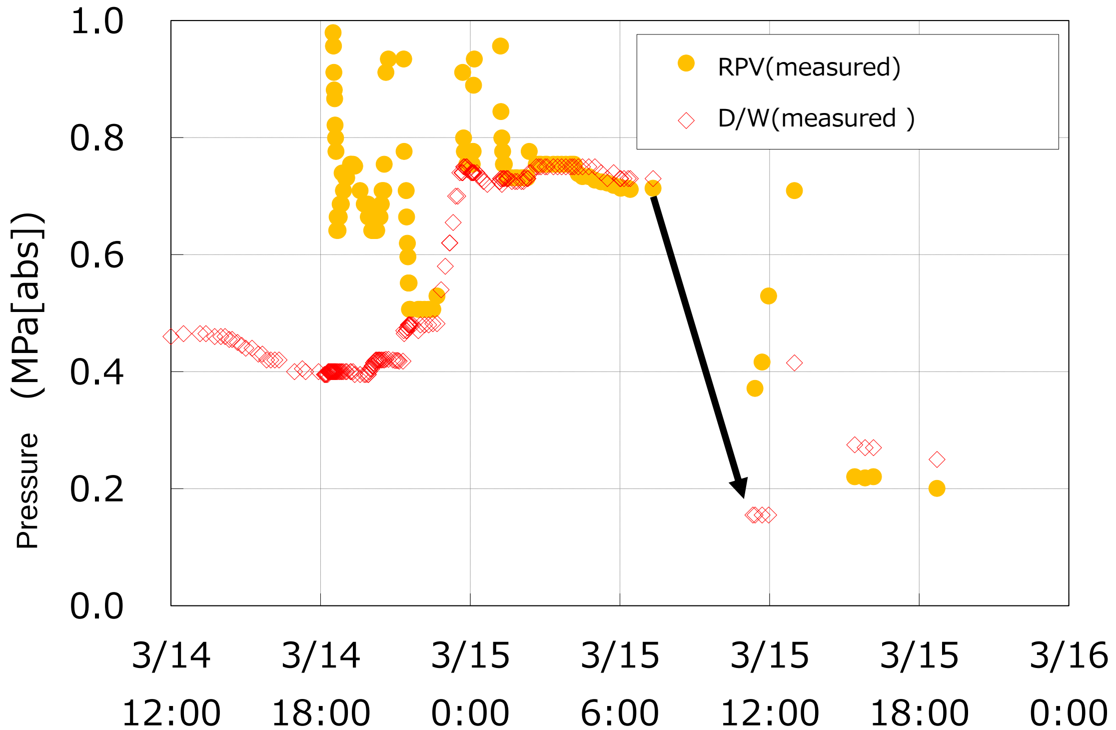
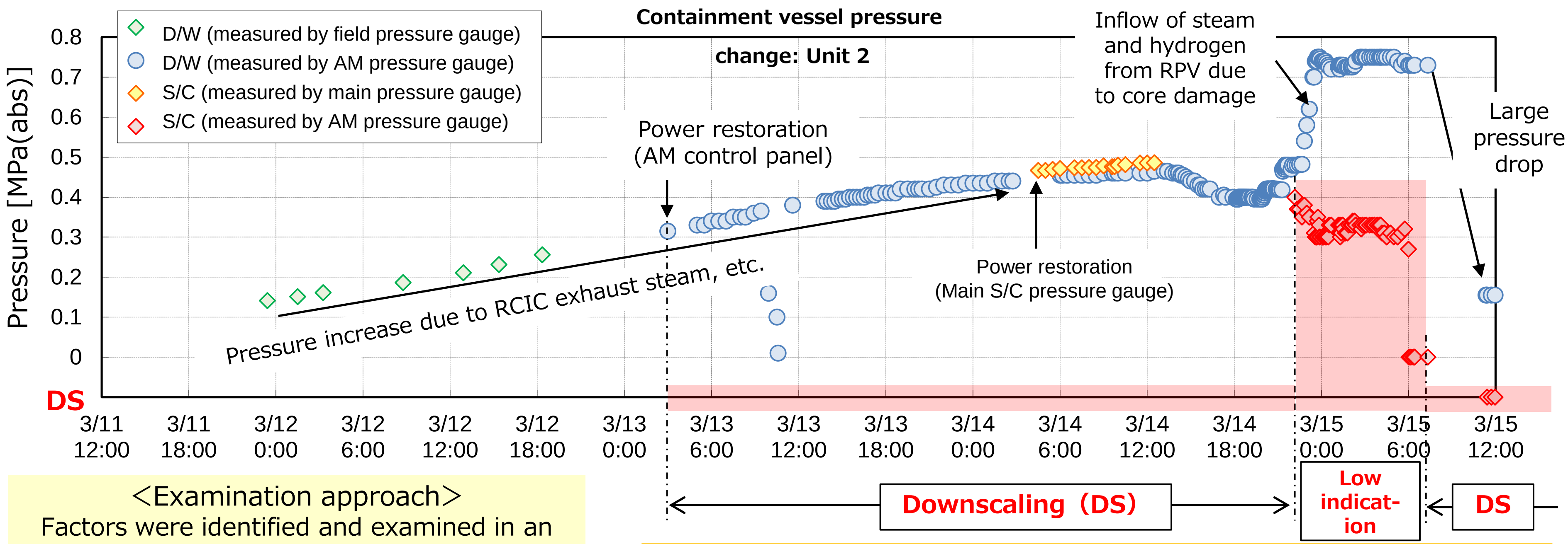


Fig. Changes in RPV and PCV pressures

5. Behavior of S/C pressure gauge in Unit 2 after 21:00 on March 14

Overview

- Of the containment pressure gauges used in Unit 2 at the time of the accident, the S/C pressure gauge for accident management (AM) was connected to the battery at 3:00 on March 13 and power was restored, but it showed values that differed significantly from those of other pressure gauges, including downscaling (hereafter referred to as DS) and an indicated value approximately 400kPa lower than the D/W pressure.
- Such a large discrepancy between D/W pressure and S/C pressure does not occur due to the structure of the containment vessel, and it was extremely likely that the S/C pressure gauge for AM was not indicating the actual pressure, since it was indicating DS.
- Since containment pressure is a very important parameter in accident response, we examined the factors that caused the S/C pressure gauge for AM to show an abnormal indication value.



<Examination approach>
Factors were identified and examined in an elimination process

Factor classification
①Mechanical factors
②Measurement principle factors
③Electrical factors

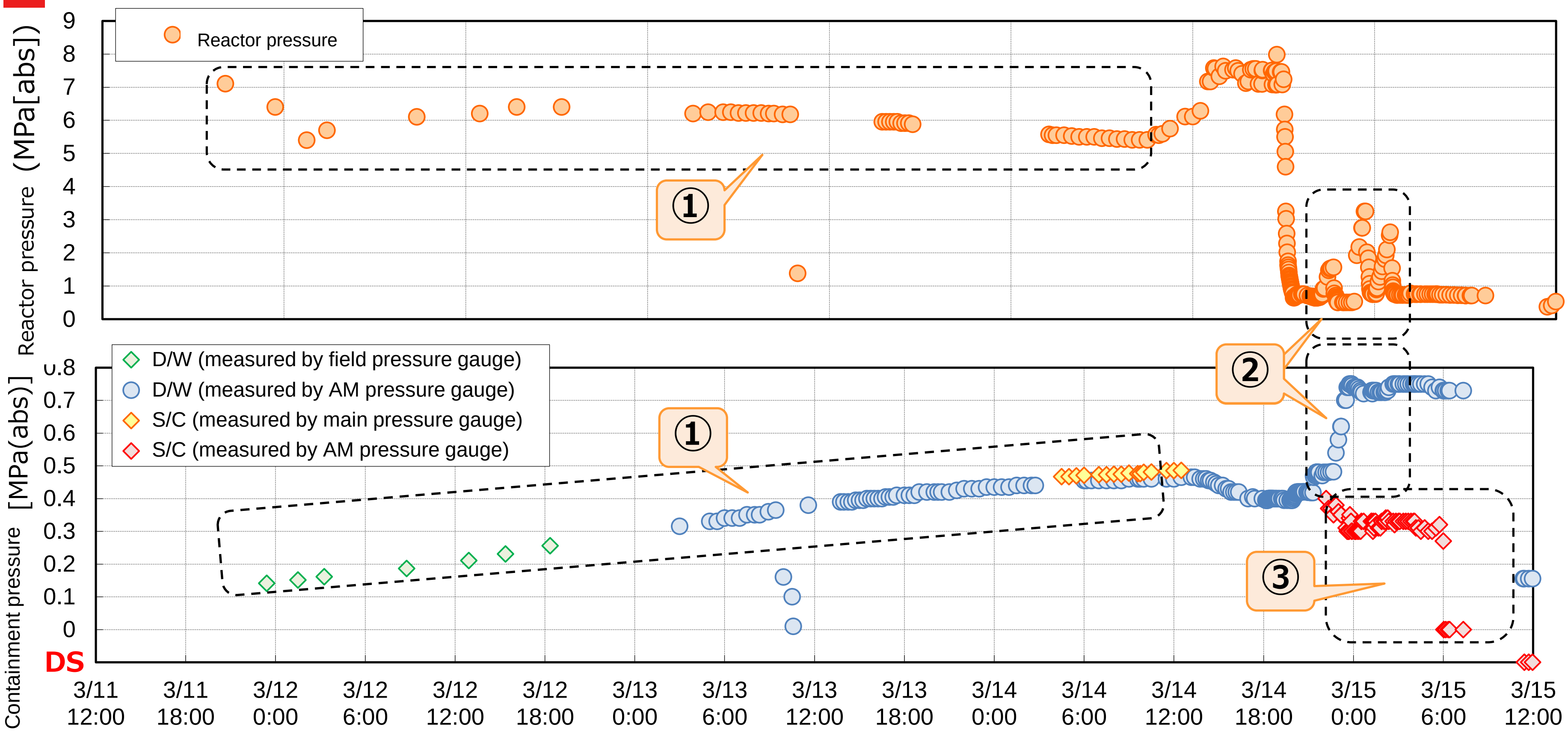
The S/C pressure gauge for AM was presumed to have shown an abnormal indication value that was far from the actual value due to electrical failure caused by submersion in water.

Status of Reflection on Safety Measures at Kashiwazaki-Kariwa NPS: Countermeasures against instrument submersion due to overflow

TEPCO

Overview (Accident progression at Unit 2)

5. Behavior of S/C pressure gauge in Unit 2 after 21:00 on March 14



【Approximate accident progression and containment pressure gauge readings】

①Due to continued RCIC operation, reactor pressure remained lower than during normal operation
⇒Containment pressure during this period (D/W pressure gauge (on-site), D/W pressure gauge for AM, and main S/C pressure gauge) rose more slowly than the rise expected from decay heat
⇒Estimated that torus room where the S/C was housed was flooded by the tsunami, and S/C was cooled from the outside.

②Estimated that RCIC lost its water injection function at around 9:00 on March 14, leading to fuel meltdown on the same night.
⇒D/W pressure gauge readings for AM increased due to inflow of steam and hydrogen from RPV as a result of core damage.

③S/C pressure should have increased in conjunction with D/W pressure, but the S/C pressure gauge for AM showed a low value that deviated from the D/W pressure gauge for AM, eventually leading to DS ⇒ Investigation of these factors.

Specifications and installation environment of S/C pressure gauge for AM

Location: R/B basement floor, southeast triangular corner
60 cm from the floor (floor: T.P. -3496)

Measurement method: Measure water pressure with condensation tank of the S/C water level gauge for AM as the detection source

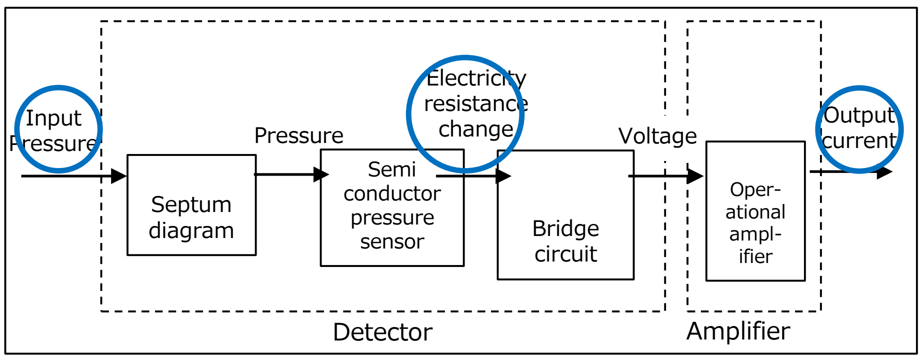
Measurement principle: Diaphragm type

Indication method: Absolute pressure [MPa(abs)] is indicated by S/C pressure indicator for AM installed in the AM control panel

Index for dust and water resistance, Ingress Protection (IP) code, IP67, where,

6: Dust resistance...Dust and dirt cannot penetrate into the instrument.

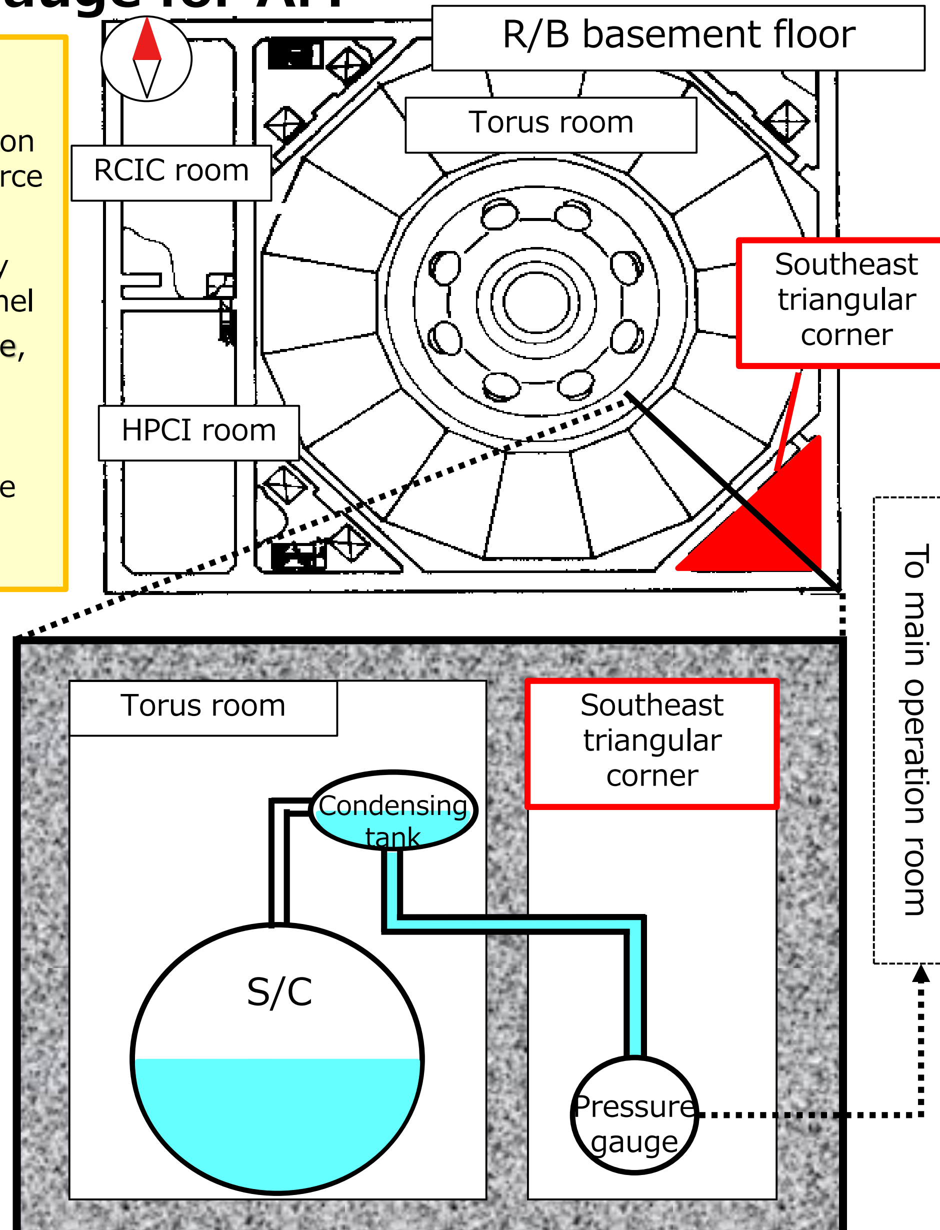
7: Submersion resistance...Water does not penetrate into the instrument even if it is temporarily immersed in water.



S/C pressure gauge, AM Measuring principle

(Ref.) Location of other D/W and S/C pressure gauges

Gauge	D/W (site)	D/W (for AM)	S/C (Main)
Location	R/B 2nd Fl. Northeast area (Floor: T.P. 17264)	R/B 3rd Fl. Northeast area (Floor: T.P. 25464)	R/B middle lower ground Fl. Northeast triangular corner (Floor: T.P. 2564) *In case of Unit 4



Investigation the cause of the abnormal reading of the S/C pressure gauge for AM

Possible factors that might have caused the S/C pressure gauge for AM to show a "drop in indicated value" and "DS" were identified, and each factor was examined.

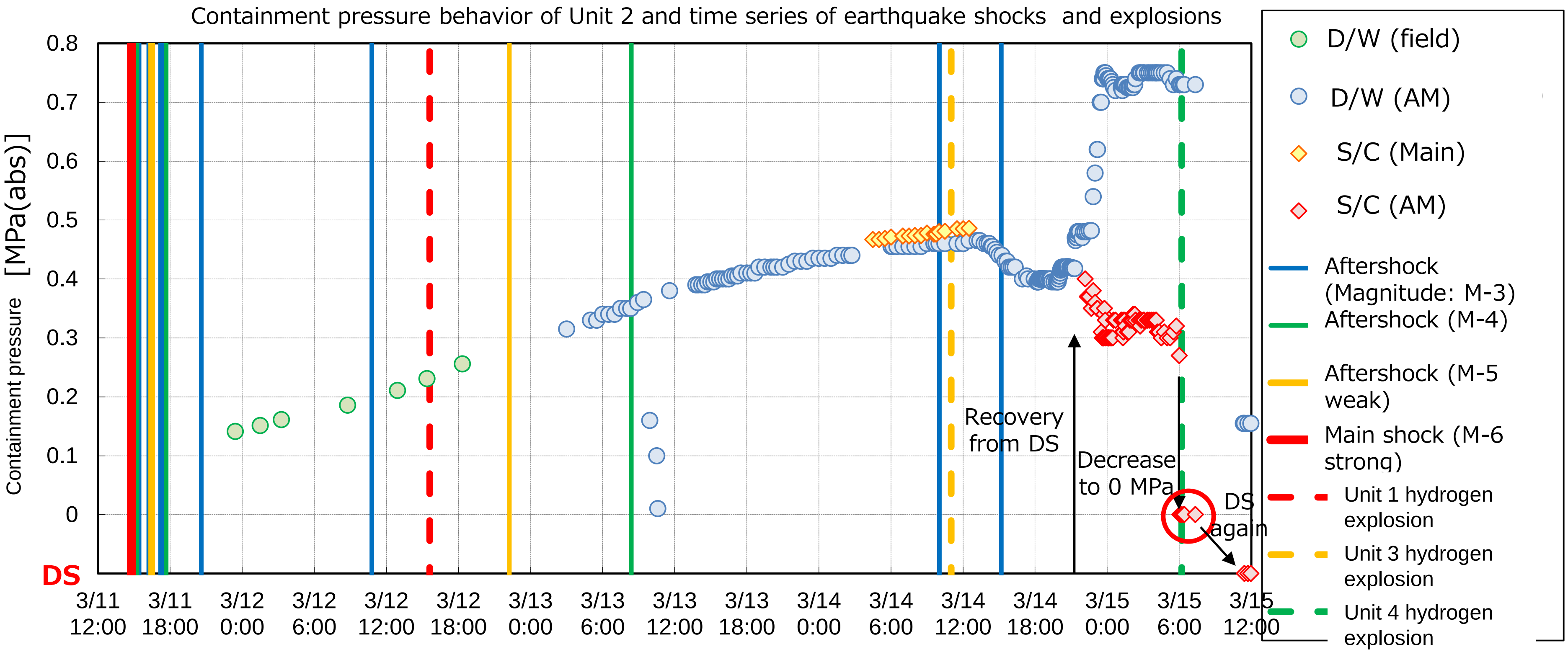
Classification of factors	Possible factors
① Mechanical factors*	・Earthquake ・Explosion of other units ・Tsunami Damage to pressure gauge body (pressure receiver and terminals) by these causes
② Factors related to measurement principle	・Decrease or loss of water in condensate tank piping to be measured
③ Electrical factors	・Insufficient or depleted battery voltage ・Water leakage into pressure gauge body or cable

*Although the possibility of damage due to mechanical factors is considered low when the indicated value is restored from DS, it was identified and examined as a factor of DS in accordance with the examination policy.

①Examination of mechanical factors
(1) Earthquake and explosion impacts

5. Behavior of S/C pressure gauge in Unit 2 after 21:00 on March 14

After examining the possibility that the main body of the S/C pressure gauge for AM was damaged by the earthquake shocks and explosions of other units, it was considered unlikely that either of these factors was a direct cause of the drop in the indicated values at DS and around 6:00 on March 15.



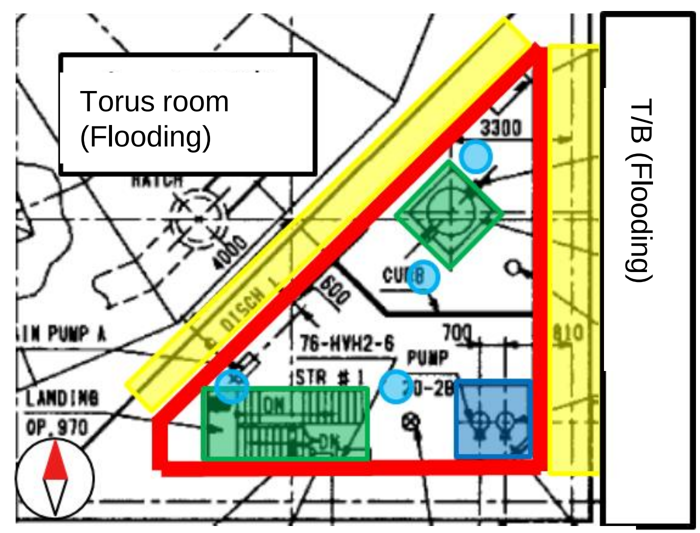
- If earthquake shocks or explosions damaged the pressure gauge itself or ruptured the cable, it is unlikely that the indicated value would be restored from DS, so it is unlikely that the gauge was damaged by this factor before March 15.
- The indicated value of the S/C pressure gauge for AM dropped sharply to 0 MPa at 6:02 on March 15, but no earthquake occurred around that time.
- The hydrogen explosion at Unit 4 occurred very close by, but the explosion was not a direct cause of the drop in the indicated value because it occurred after the 0 MPa reading was reached. (0 MPa: March 15, 6:02. Unit 4 hydrogen explosion: March 15, 6:12)

①Examination of mechanical factors
（2）Tsunami impact

After examining the possibility that the main body of the S/C pressure gauge for the AM was damaged and showed DS when the tsunami arrived on March 11, it was considered unlikely that the pressure gauge was damaged by the impact of the tsunami arrival.

Tsunami ingress routes	Possibility of ingress	
Openings to the outdoors (direct)	×	No openings directly connected to the outdoors
Wall penetration (from side)	○	Ingress through the torus room and T/B connection
Ground floor and middle lower ground floor (from above)	○	Ingress through hatch above CS pump or stairwell
Funnel (from below)	○	Reverse flow via sump from ground floor and southwest triangular corner of basement floor
Floor drain sump (from below)	×	Tsunami flowed backward through the piping of the waste treatment system and entered (unlikely due to check valve on discharge side of sump pump)

There are multiple tsunami entry routes to the installation site of the S/C pressure gauge for the AM, but all of them are difficult to reach with the momentum of the tsunami still intact.
⇒It is unlikely that the impact of the tsunami caused the DS.



Middle lower ground Fl. Southeast triangle corner

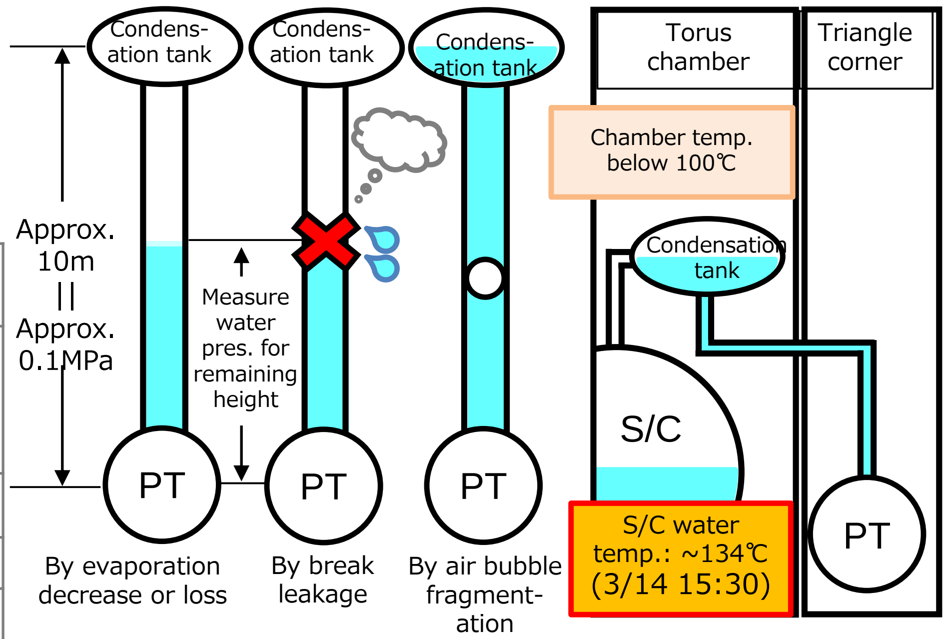
②Examination of factors related to measurement principle

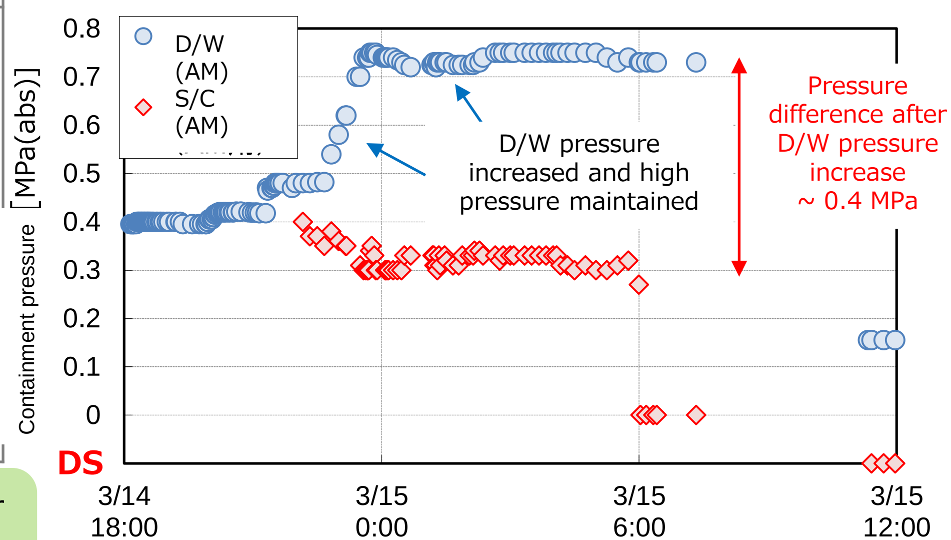
5. Behavior of S/C pressure gauge in Unit 2 after 21:00 on March 14

As a result of examining the decrease in indicated value due to decrease, loss, and fragmentation of water in condensate tank piping as a factor related to measurement principle, even assuming the maximum decrease in the indicated value, the deviation from the D/W pressure is difficult to explain and this is not considered to be the main cause of the decrease in indicated value.

Possible decrease due to evaporation of water in piping	×
・Torus room was under atmospheric pressure condition (≤100℃) ・High pressure in piping → Saturation temperature > 100°C ・D/W pressure increased → Water in piping did not depressurize and boil	
Possible leak due to pipe rupture	**×**
No change in dose in the torus room and triangular corner	
Effect of air bubbles separating water in pipes	**×**
・Uncertain conditions → Difficult to estimate indicated value change ・Difference in height between condensate tank and pressure gauge was ~ 10m. → Maximum decrease in the indicated value was ~ 0.1 MPa. ・Pressure difference between D/W and S/C was ~ 0.4 Mpa (right figure).	

In the period during which the indicated value of the S/C pressure gauge for AM was decreasing
Difference between D/W and S/C pressure > Maximum amount of indicated value drop
and it is difficult to explain the deviation from D/W pressure by this factor alone.





It was unlikely that the decrease, loss, or fragmentation of water in condensate tank piping damaged the pressure gauge body or broke the cable, and it was unlikely that this factor caused DS.

③ Examination of electrical factors
(1) Possible submergence of pressure gauges due to tsunami

5. Behavior of S/C pressure gauge in Unit 2 after 21:00 on March 14

S/C pressure gauge for AM indicated DS at 3:00 on March 13, when measurement began. By this time, the water level at the southeast triangular corner had likely risen so that the AM S/C pressure gauge (60cm above the floor) was submerged.

Interlocking of stagnant water level in the basement

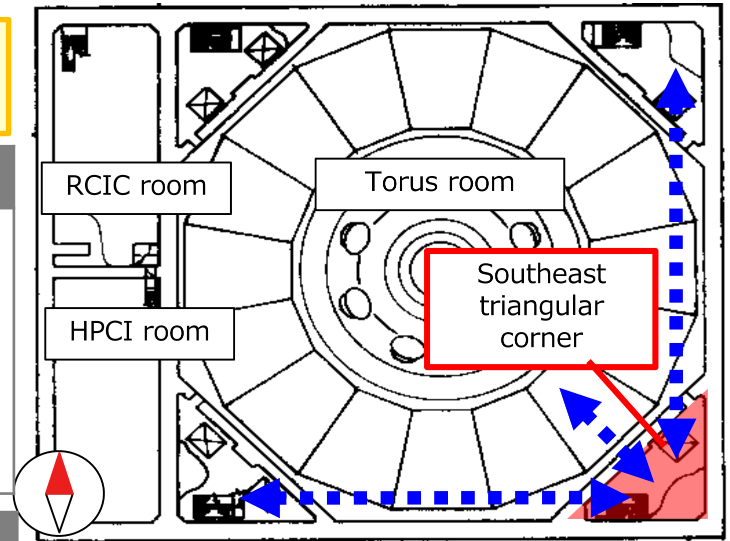
・The rooms on the basement floor, including the southeast triangular corner, were connected via funnels.
・The wall between the southeast triangular corner and the torus room had a penetrating part from 5cm to 5m above the floor.
・The water level in each room on the basement floor changed in tandem over a long period after the accident.
⇒Water levels in each room might have been changing in tandem from the beginning of the accident.

Flooding in the basement floor confirmed at the time of the accident

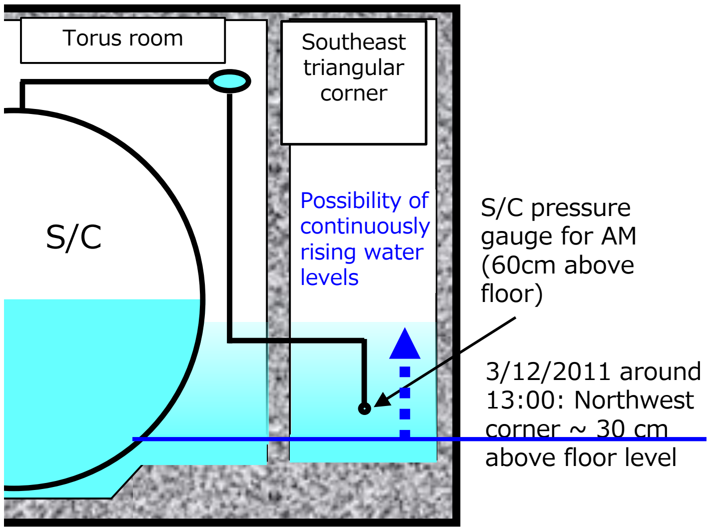
・At around 1:00 on March 12, the water level was observed to be about boot-high at the northwest triangular corner (in front of the RCIC room door), and when the door was opened, water flowed out of the RCIC room (at this point, water might have been about 30cm above the floor).
・At 2:12 on March 12, water level in front of the RCIC room door was confirmed to be rising, and when the door was opened, water slowly leaked out.
⇒It was possible that the water level in the basement floor gradually rose from about 30cm above the floor at 1:00 on March 12.

Flooded torus room inferred from plant behavior

・During RCIC operation (to about 9:00 on March 14), D/W pressure increased slowly
→Torus room was flooded, and S/C was presumed to be cooled from outside.
・In the morning of March 15, D/W pressure dropped significantly.
→In addition to leakage from the containment vessel, water level in the torus room exceeded the level of the S/C pool, and the condensation of water vapor in the S/C was accelerated by cooling of the S/C gas phase, which might have contributed to the depressurization (see "4. Decrease in Unit 2 containment vessel pressure in the morning of March 15").
⇒Possibility that torus room water level was continuously rising.



Water connection to each room through funnel/pipe penetrations at the southeast triangular corner (image)



Water level change in the basement Fl. of Unit 2 R/B (image)

③ Examination of electrical factors
(2) Examination of electrical factors

S/C pressure gauge for the AM was located in an environment where it could be submerged in water, and there was a high possibility that the inside of the unit was flooded.*
The terminal section might have been flooded, resulting in a combination of short circuit, ground fault, and insulation degradation, which might have caused the S/C pressure gauge for the AM to show DS or a low indication value.

Effects of battery depletion /under-voltage	×

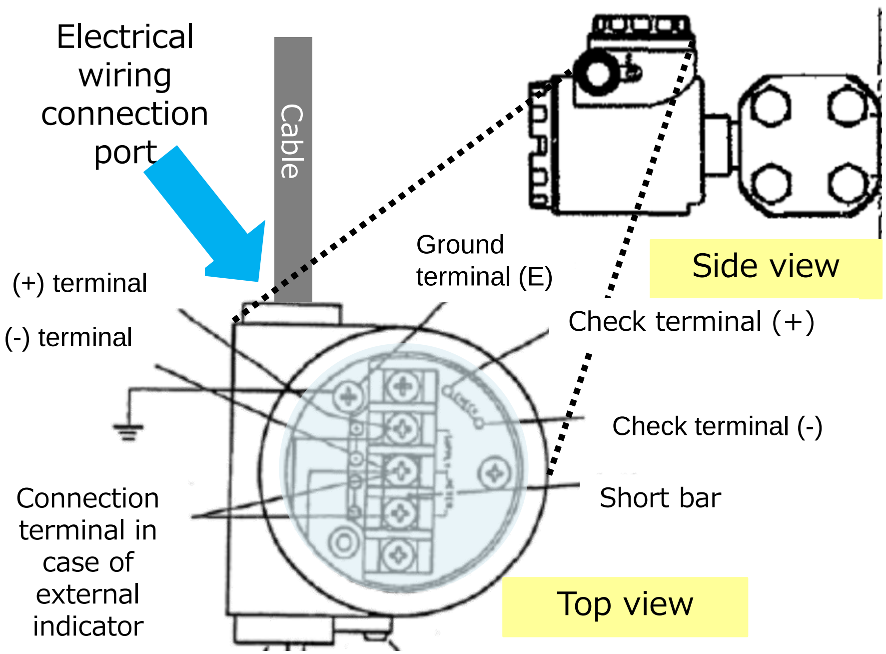
・Insufficient voltage might cause a drop in the indicated value or DS.
・The battery was connected to the control panel, and shared with the D/W pressure gauge for the AM and the S/C pressure gauge for the AM.
・Specified voltage was checked before the connection.
⇒Low possibility of battery depletion or under-voltage

Short circuit effects	△

・A short circuit would work to increase the indicated value, which might cause equipment failure in some cases.
・There was a high possibility that the pressure gauge was submerged in water and became flooded inside.
⇒A short circuit was quite possible.

Effects of ground fault or insulation loss	○

・If the current leaked out of the electric circuit and the current value decreased, this might result in DS or a decrease in the indicated value.
・Insulation resistance of the control panel was not measured.
・There was a high possibility that the pressure gauge was submerged in water and became flooded inside.
⇒There was a strong possibility of ground fault or insulation degradation.



Schematic diagram of S/C pressure gauge and terminals for AM

*Although the S/C pressure gauge for the AM was IP67 (dust-proof and immersion-proof), water might enter the interior of the instrument if submerged conditions were persisting.

Summary of examination results

・Factors causing the S/C pressure gauge for Unit 2 AM to show abnormal indicated values (DS, decreased indicated value) at the time of the accident were examined.
・As a result of identifying the factors and examining the possibilities using a process of elimination approach, the possibility of an electrical abnormality due to submergence of the pressure gauge body remained as the main factor.

Classification of factors	Result	Details of the result	
① Mechanical factors	×	×	Damage to the main unit due to impacts of earthquakes or explosions of other units
		×	Damage to the main unit due to impact of the tsunami
② Factors related to measurement principle	×	×	Water decrease due to evaporation in condensate tank piping
		×	Water leakage from piping due to break of condensate tank piping
		×	Separation of water in condensate tank piping due to air bubbles
③ Electrical factors	○	×	Battery depletion or power shortage
		○	Electrical abnormality due to seawater intrusion into the main unit (short circuit, ground fault, insulation loss)

Lesson learned: Measures need to be taken against inundation of instruments due to water flooding.

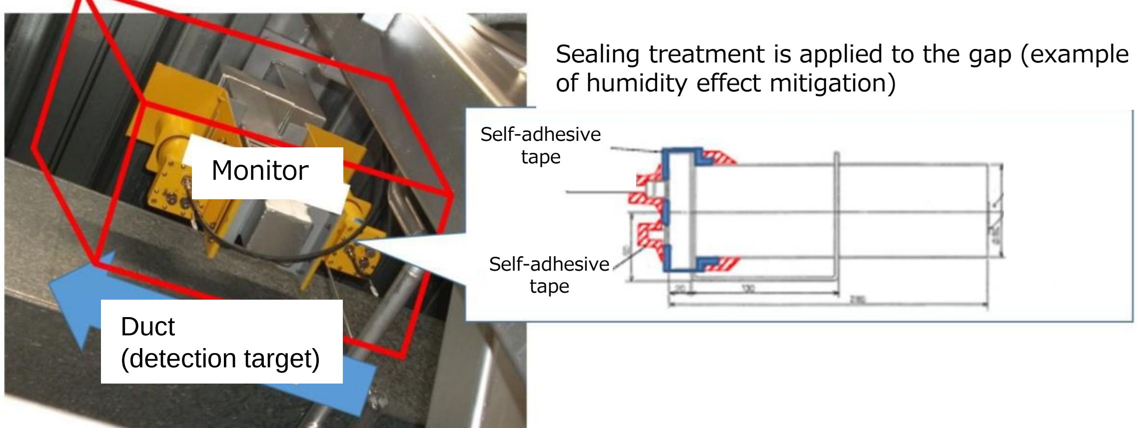
Safety measures in the Kashiwazaki-Kariwa NPS

Lesson learned: Measures need to be taken against inundation of instruments due to water flooding.

- Tsunami (external overflow) countermeasures
  - External protection: Prevention of tsunami run-up and inflow by site elevation, water intake tank closing plates, etc.
  - Inner protection: Prevention of tsunami inflow into the areas of focus for flooding protection in the event of seawater pipe breakage and flooding protection in the event of damage to outdoor tanks through watertight doors, watertight penetrations, and other measures.
  - Ensuring water intake: Ensuring water intake of seawater pumps in the event of receding waves by installing seawater storage weirs, etc.
- Internal overflow countermeasures
  - Prevention of occurrence: Isolation and draining of overflow sources, relocation of overflow sources, ensuring earthquake resistance of overflow sources, etc.
  - Prevention of expansion: Waterproofing of doors, penetrations, hatches, etc., construction of drainage guidance routes, etc.
  - Prevention of impact: Improvement of drip-proof specifications by sealing, relocation of facilities (e.g., raising the installation height), etc.

Installation of thermal insulation such as heat shields (example of temperature impact mitigation)

Sealing treatment is applied to the gap (example of humidity effect mitigation)



Example: Countermeasure against internal overflow of water into the exhaust monitor in an area of the gas waste treatment system facilities

Safety measures in the Kashiwazaki-Kariwa NPS

- Response to loss of instrument function
  - If it becomes difficult to measure parameters that need to be monitored to deal with a major accident, etc. (main parameters), a means to estimate such parameters (alternative parameters) is provided.
  - The evaluation confirmed that the internal overflow countermeasures described on the previous page do not make it impossible to monitor the main parameters and the alternative parameters at the same time due to internal overflows.

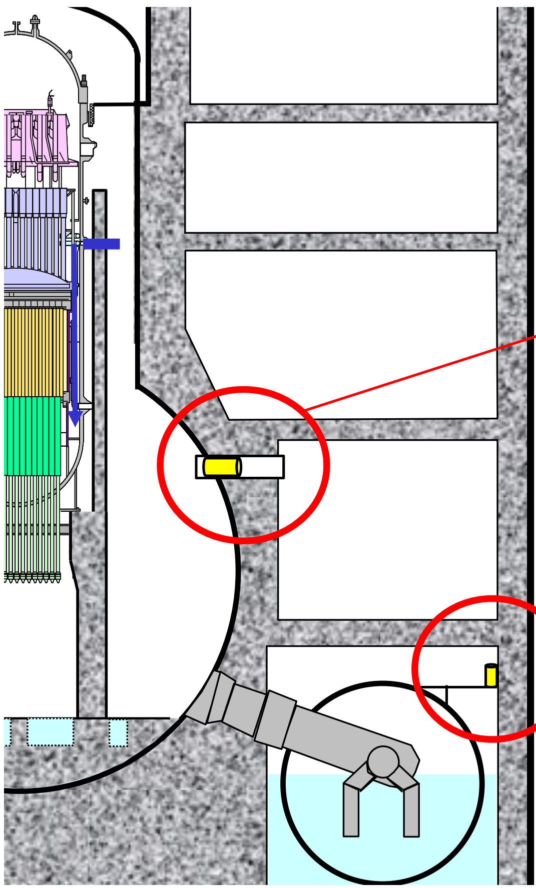
    (Example) Alternative parameters for S/C pressure
    ①D/W pressure (using D/W and S/C vent pipe or vacuum break valve to equalize pressure)
    ②S/C gas temperature (estimated from saturation temperature/pressure relationship)
    ③Regularly used monitoring instrument for S/C pressure

- Education and training of emergency response personnel and operators
  - Emergency response personnel (including operators) are provided with education on the basics of accident management according to their roles, and education on physical and parameter behavior during a major accident, in order to provide them with a broad knowledge of the phenomena of a major accident.
  - For operators, simulator training is conducted to simulate the failure of monitoring instruments used to make judgments in operations at the central control room, in order to improve their ability to judge events based on relevant parameters and other response skills.

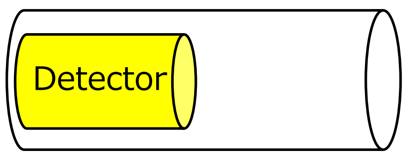
6. Evaluation method of core damage ratio of Mark-I containment vessel

Overview

- In Unit 2, core damage occurred after D/W and S/C CAMS (containment atmosphere monitoring system) measurements resumed due to the power supply restoration. In the third and fourth progress reports, accident progress was estimated based on these measurements, and the FP presence rate was also evaluated during the time period when core damage and fuel meltdown progressed.
- Furthermore, focusing on the CAMS measurements, differences were found between the trends of the actual data and the time-dose map for the evaluation of the core damage fraction (hereinafter referred to as "evaluation map").
- Since the CAMS measurements are important data for understanding the accident progression, their factors and the validity of the evaluation map were discussed.



D/W CAMS



Penetration

S/C CAMS

Detector (Unit 5)

<Examination approach>

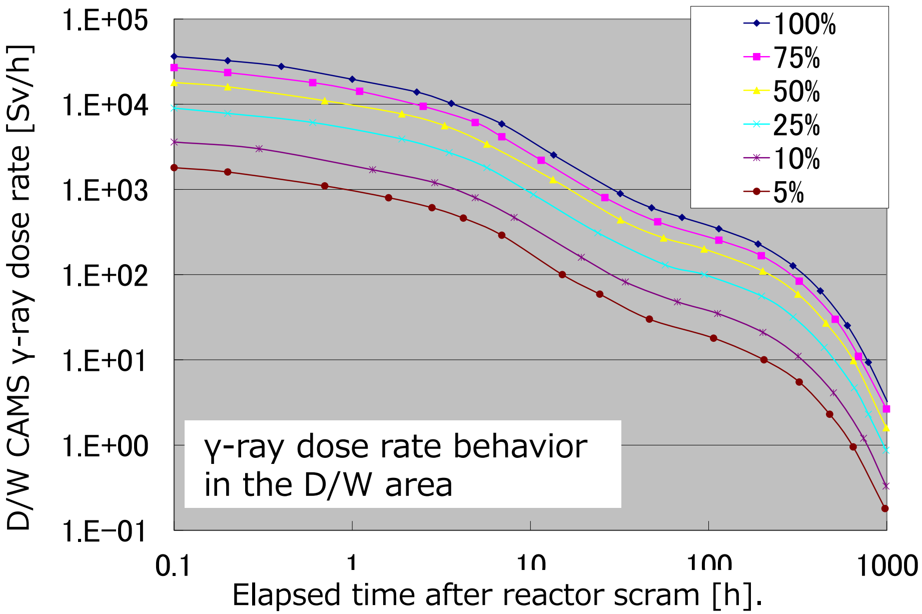
- Comparison of D/W and S/C CAMS measurements of Unit 2 with the evaluation map and evaluation of the map reproducibility
- Examination from viewpoints of evaluation method and geometric factors (location of D/W and S/C CAMS detectors, etc.)

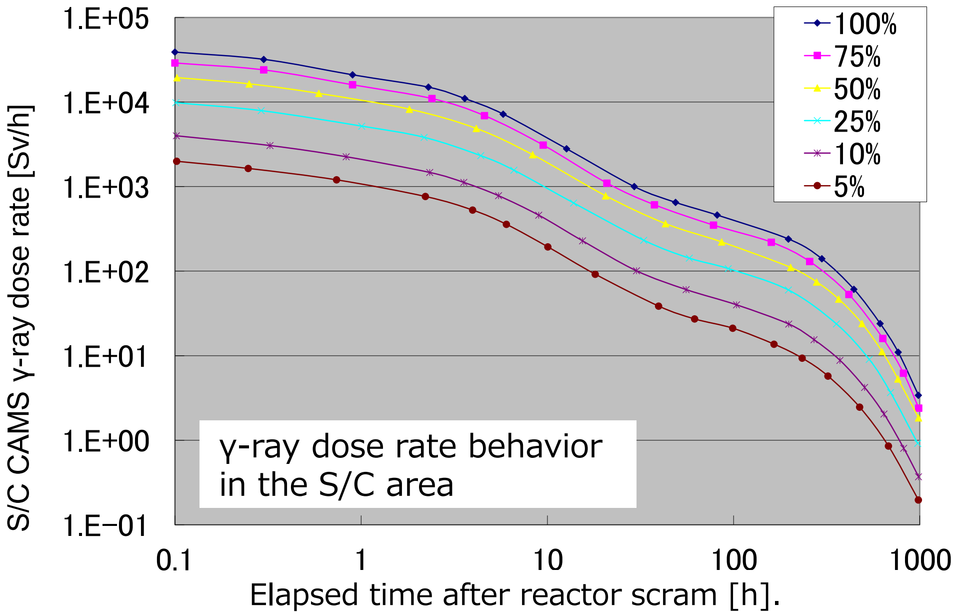
There was a discrepancy between the evaluation map and reality, and the main reason was thought to be that the evaluation map assumed the distance between the S/C wall (shielding) and the CAMS detector (distance from the radiation source) was smaller than the actual distance.

Status of Reflection on Safety Measures at Kashiwazaki-Kariwa NPS:
Validation of procedures for estimating core conditions from CAMS dose rates

Fukushima Daiichi BWR4 core damage ratio map

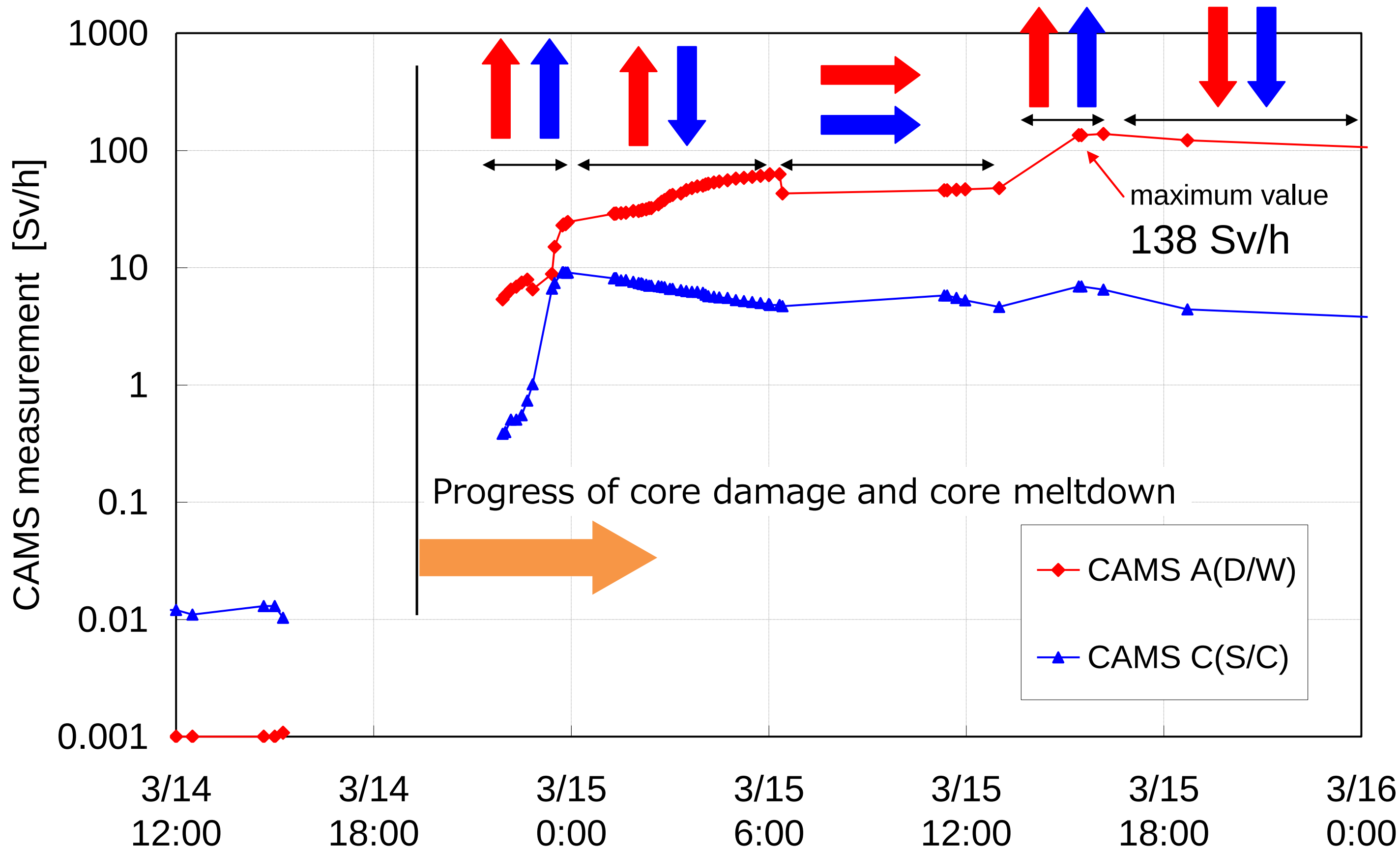
- In the evaluation map for the Mark-I containment used in the Fukushima Daiichi NPP accident, there was no significant difference in the CAMS dose rate for each damage ratio between the D/W and S/C.
- The evaluation map was prepared by considering the radiation from noble gases released from the fuel only, and it has been considered that the core damage ratio was conservatively evaluated when iodine and other gases were released at the same time.





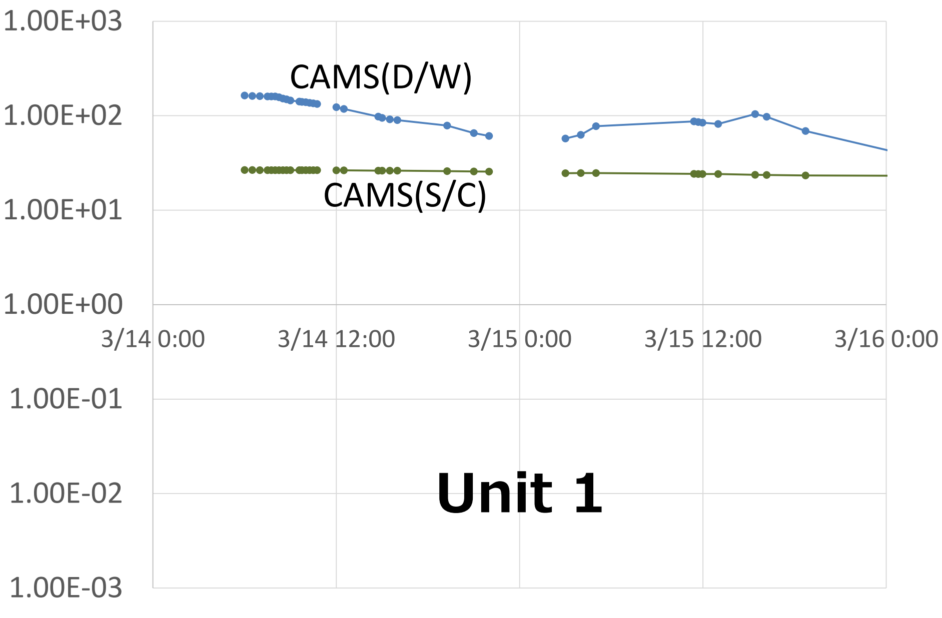
At the time of the accident, the core damage ratio was evaluated and presented on an evaluation map. However, while the current knowledge indicates 100% core damage in all units, the evaluation at that time gave smaller numbers of < 100% (Unit 1, 55%; Unit 2, 35%; Unit 3, 30%; published on 2011/4/27).

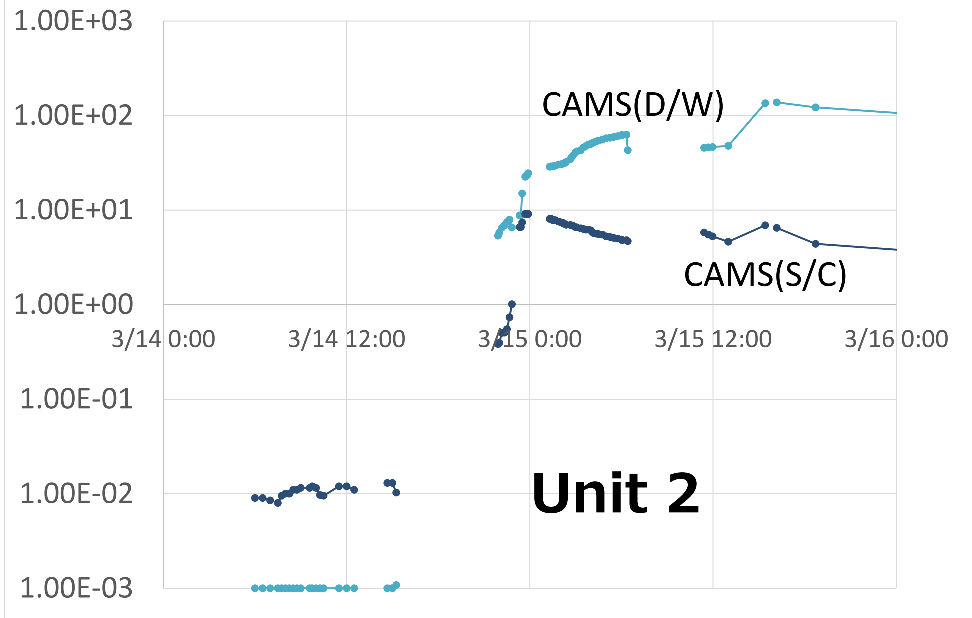
CAMS measurements for Unit 2

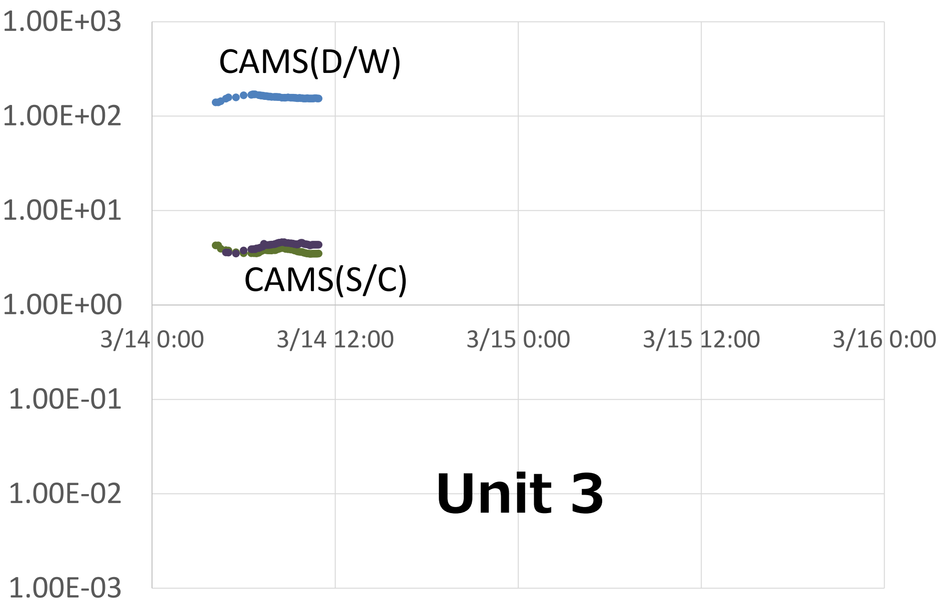


- After the core was damaged, CAMS measurements were resumed on the night of March 14, and they were always about one order of magnitude lower in the S/C.
- At the start of core damage, FPs moved via SRVs from S/C to D/W, and large amounts of FPs would be expected to be present in S/C, but the measured values were different.

(Ref.) CAMS measurements for each unit





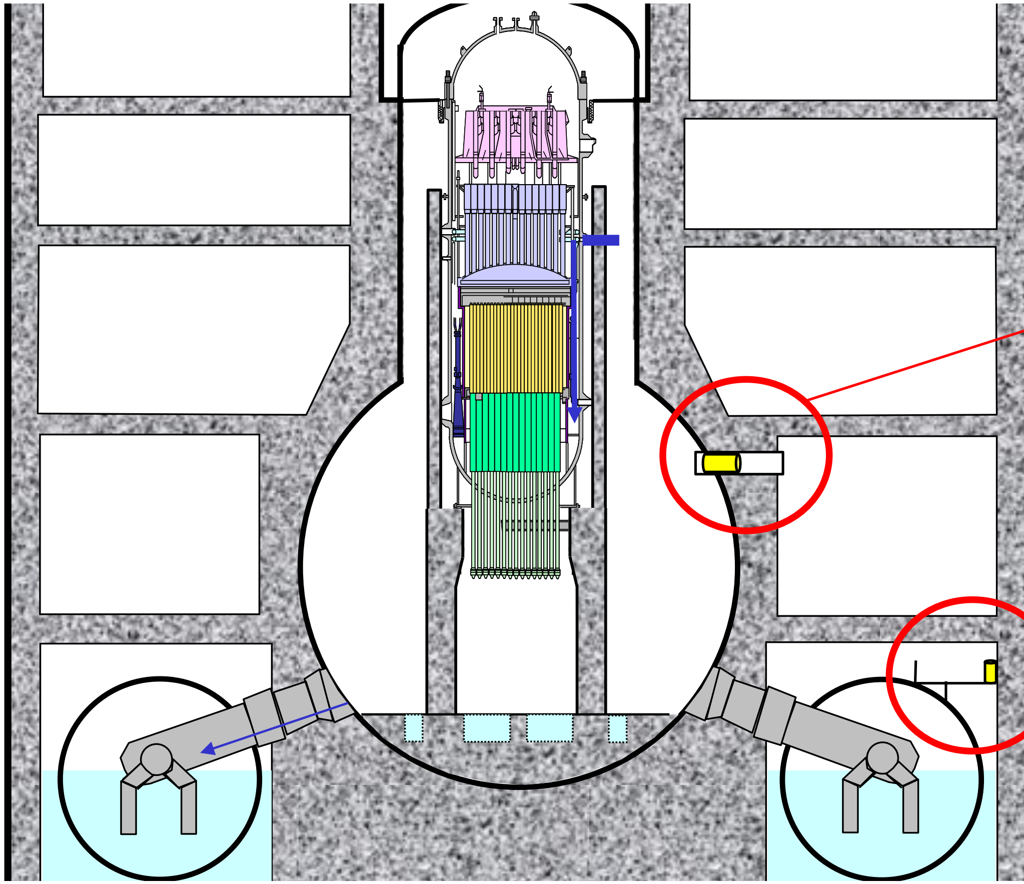


[Dose rate unit: Sv/h]

Although accident progression differed in Units 1 to 3, the trend of CAMS measurements (D/W > S/C) was the same

Note: Measured values for Units 1 and 3 after venting

Locations of CAMS detectors



Detector（Unit 5）

D/W CAMS: Located immediately adjacent to the D/W inside

S/C CAMS: Installed on the wall of the torus room a little away from the S/C

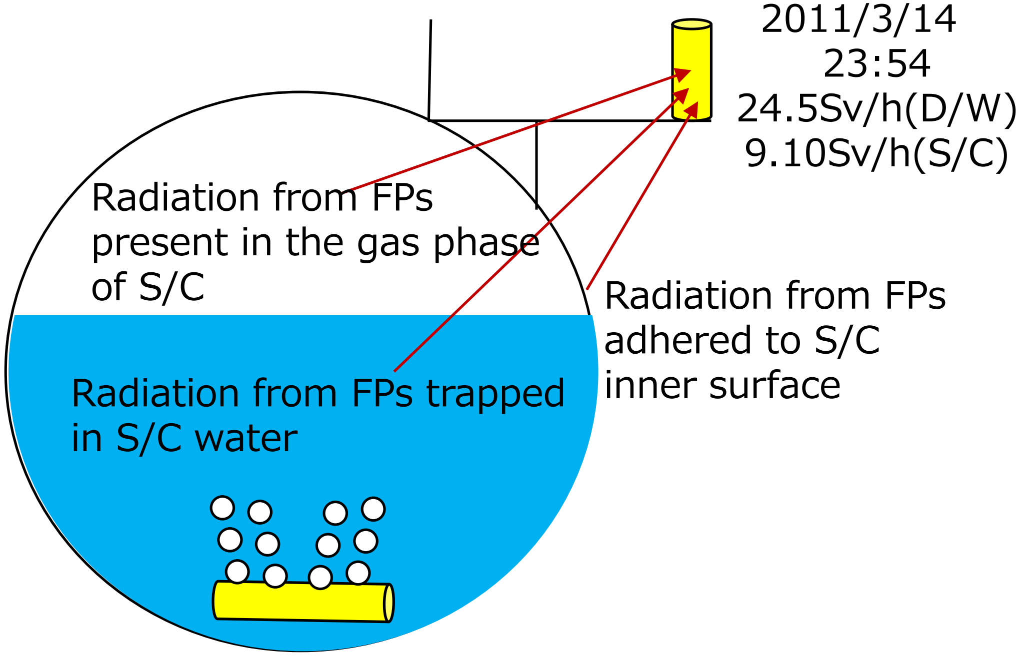
⇒Since the distance from the radiation source was different, it was thought that the absolute value of the dose rate was affected.

Migration and distribution of FPs to S/C at midnight on March 14

When FPs released from the fuel were released into the S/C through the SRVs
- FPs of noble gases migrated directly to the gas phase of S/C
- Most volatile FPs such as iodine and Cs were trapped in water
- Some of the volatile FPs that migrated to the gas phase adhered to the inner surface of the S/C

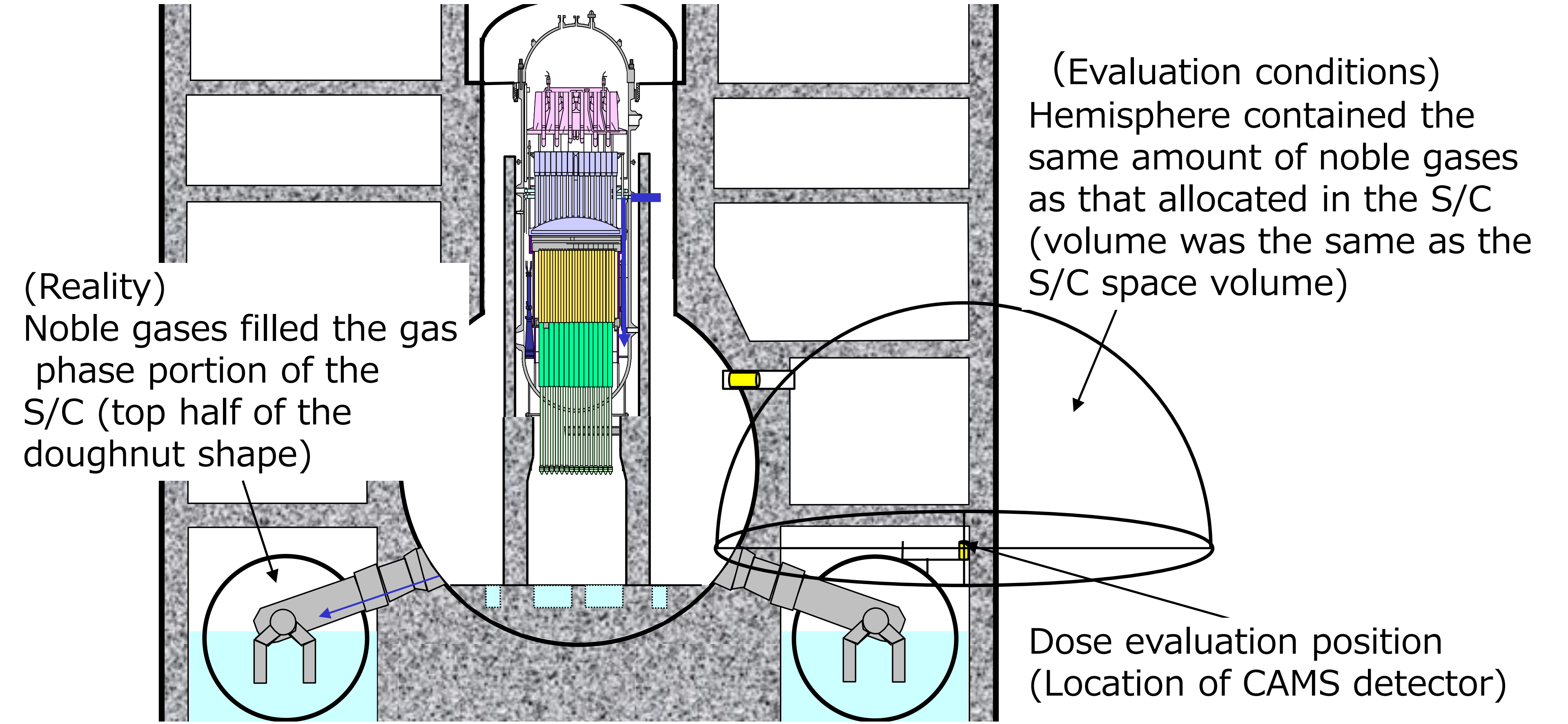
The measured value of 9.1Sv/h on March 14 at 23:54 was the sum of radiation from 3 sources

Assessed as having reached the point of core meltdown during this time period (no PCV leakage)



<u>The evaluation map</u> conservatively assumed that only the contribution of noble gases be taken into account, which was **<u>equivalent to 460Sv/h after 81.8h and 240Sv/h after 197.4h in the case of 100% total emission,</u>** while the contribution of noble gases was estimated to be **<u>about 1.2 Sv/h at most</u>** out of the **<u>measured value</u>** of 9.1Sv/h.
→In other words, the S/C CAMS measurements and the evaluation map were inconsistent due to the influence of the S/C CAMS location.

Dose rate calculation method used in the methodology for evaluating core damage ratio



The method used to obtain the core damage ratio was based on a simplified dosimetry calculation method that assumed a hemispherical plume. An inconsistency was caused by the distance between S/C wall (shielding) and CAMS location (distance from the source) not being considered. (However, at the time of the Fukushima Daiichi NPS accident, core meltdowns were also evaluated using D/W CAMS values, so the effect of underestimation was small.)

Summary

Examined the validity of the evaluation map to assess the core damage ratio, etc., using the CAMS measurements for Unit 2

- It was found that the evaluation map, which was supposed to conservatively evaluate the core damage ratio, tended to underestimate the core damage ratio when evaluated using S/C CAMS.
- This was presumed to be due to improperly reflecting the effects of the Mark-I containment vessel geometry and the CAMS detector location.

Lesson learned: When estimating core conditions from CAMS dose rates, attenuation due to shielding and distance between the source and the CAMS detector must be properly considered.

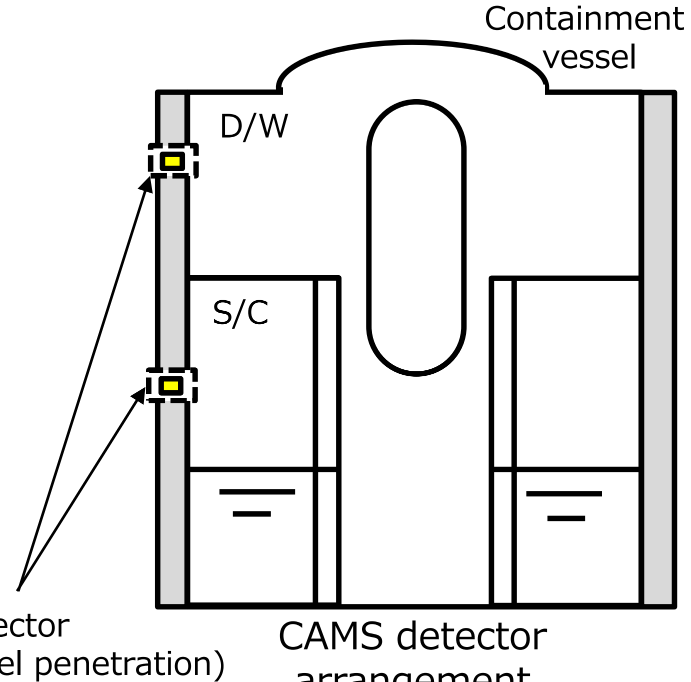
■ **Confirm the validity of the procedure for estimating core conditions using CAMS dose rates (①② below).**

①Determination of core damage
The following items confirm that there are no obstacles to judgment.

- In Kashiwazaki-Kariwa NPS Units 6 and 7, CAMS detectors are located inside the containment vessel penetrations for both D/W and S/C.
- The dose rate to determine core damage is conservatively low to avoid delay in judgment.
- Since the dose rate increases significantly in a short time period at the time of core damage, the influence of the uncertainty of the core damage determination curve on the determination time of core damage is small.

②Estimation of core damage ratio

- The core damage ratio is not used by operators to determine the operation.
- The conventional practice of calculating the core damage ratio has been discontinued in the manuals referred to by organizations that provide technical support to operators.



CAMS detector arrangement

7. Examination of water level in Unit 3 Suppression Chamber

Overview

- It is important to understand the containment pressure data in order to estimate the accident progress (containment vessel venting, gas phase leakage from the pressure vessel and containment vessel, hydrogen explosion, etc.) and the cooling status of fuel debris since the reactor depressurization at around 9:00 on March 13 in Unit 3.
- S/C water level data were collected at Unit 3 from 17:15 on March 11 to 20:00 on March 12.
- These data are useful for estimating the amount of hydrogen generated and whether or not water flowed back from the S/C to the D/W. This information is important for understanding the accident progression as described above. In this study, we focused on the S/C water level at the start of the S/C venting (hereinafter "first venting") at around 9:00 on March 13, and we estimated this level.

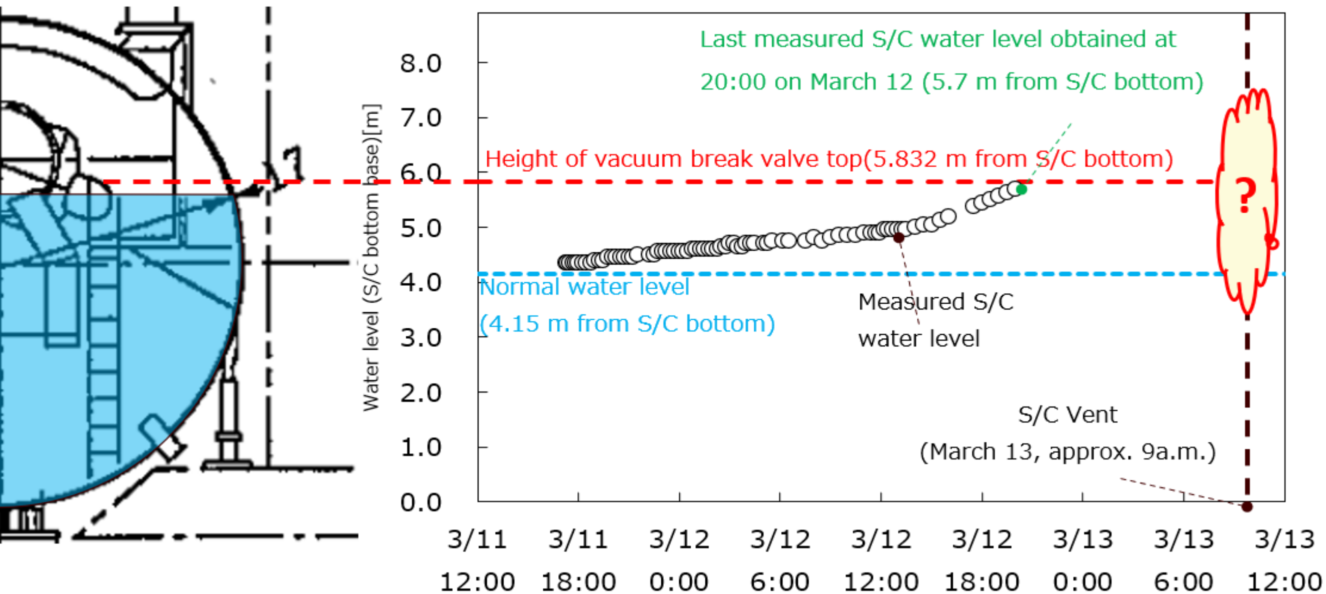


Fig. Actual measured S/C water level

<Examination approach>

S/C water level at the time of first venting was evaluated by two independent methods below, and the S/C water level was estimated by unifying the results of both evaluations

(1) Evaluation based on available S/C water level data

(2) Evaluation based on containment pressure data

The S/C water level at the start of the first venting was estimated to be around 7m from the bottom of the S/C, which was higher than the vacuum break valve.

⇒When the D/W was depressurized after 20:40 on the 13th, water may have flowed back from the S/C to the D/W and contributed to cooling the falling fuel debris.

Status of Reflection on Safety Measures at Kashiwazaki-Kariwa NPS:
Submersion measures for vacuum break valves

Overview of accident progress and (1) Overview of the evaluation based on actual measurements of S/C water level

Based on the S/C water level data that have been obtained, the water level at the time of the first venting was evaluated.

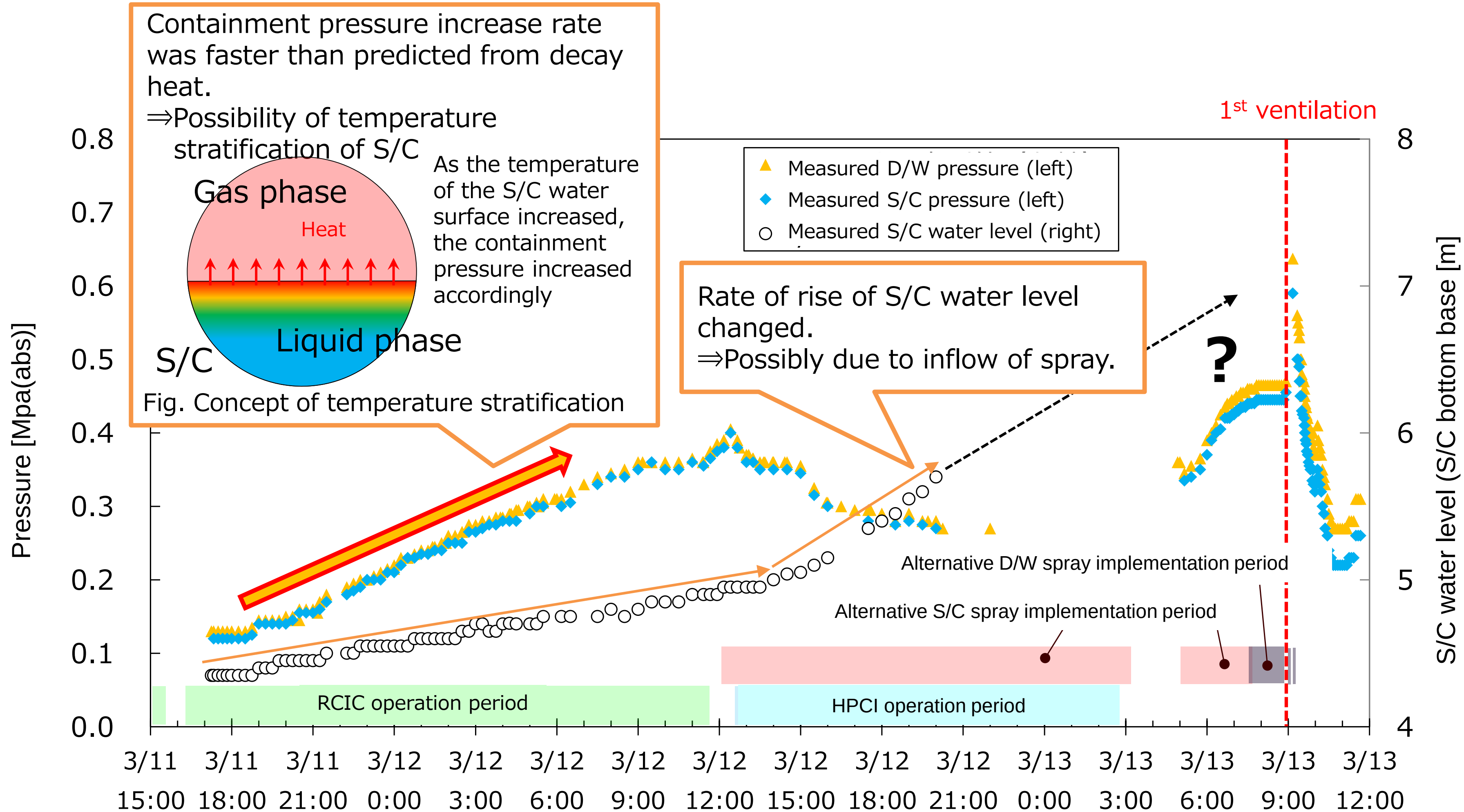
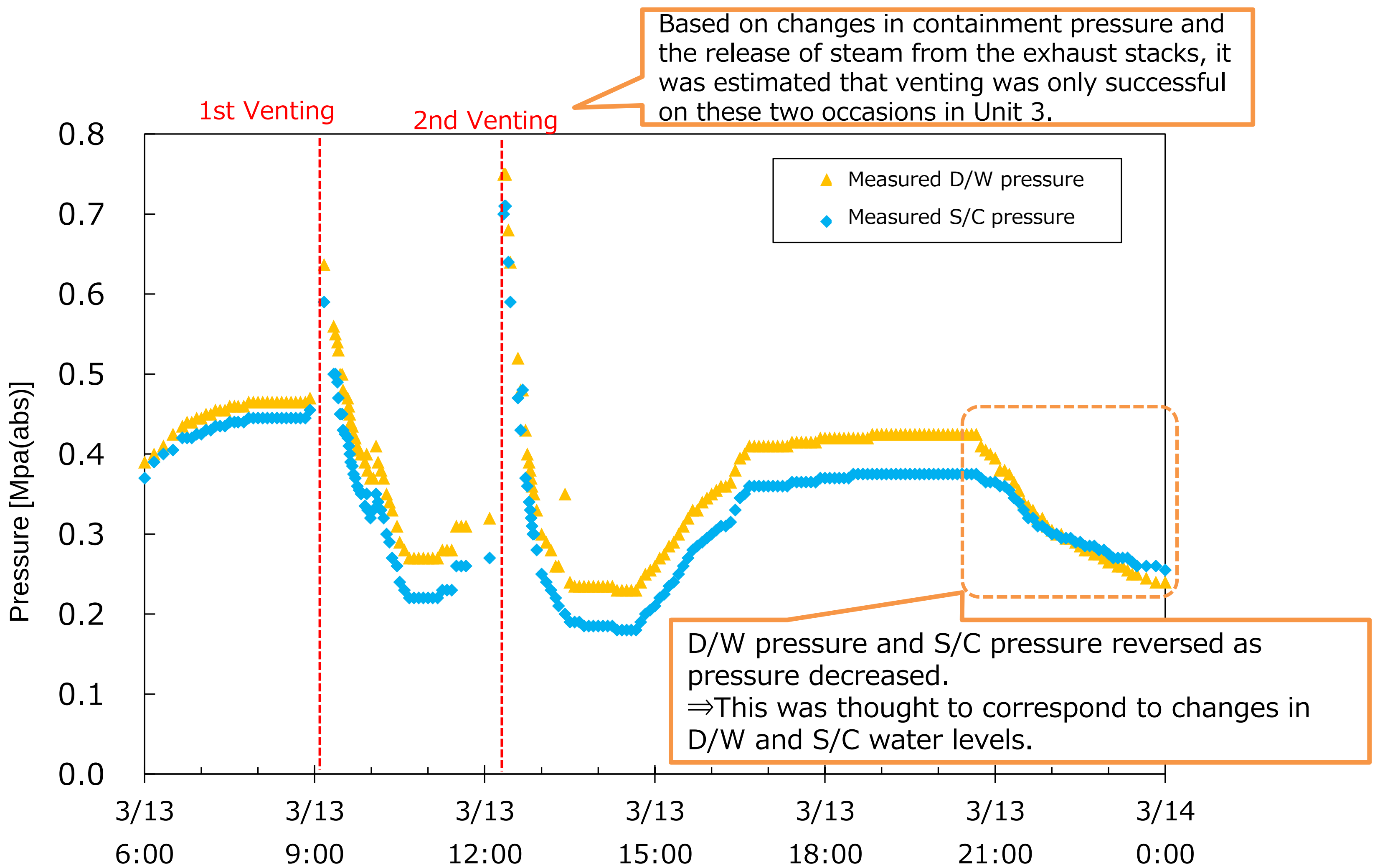


Fig. Actual measured values of containment pressure and S/C water level

Overview of accident progress and (2) Evaluation based on actual containment pressure measurements.

Focusing on the containment pressure depressurization behavior after 20:40 on the 13th, the water level at the time of the first vent was evaluated.



Estimation of S/C water level at the start of the first venting based on plant parameters (Accident progression scenario estimation using plant parameters)

As a preliminary step in the evaluation, the following three situations were assumed from the containment pressures before and after the S/C venting.
A: The D/W and/or S/C pressure gauges were misaligned, and the D/W-S/C pressure difference was overstated.
B: From the time of the first venting until 20:40 on the 13th, the water level in the vent tube was pushed down to the bottom of the downcomer. (Assumed to be due to S/C venting and gas phase leakage from RPV to D/W)
C: As of 20:40 on the 13th, the vacuum break valve was submerged.

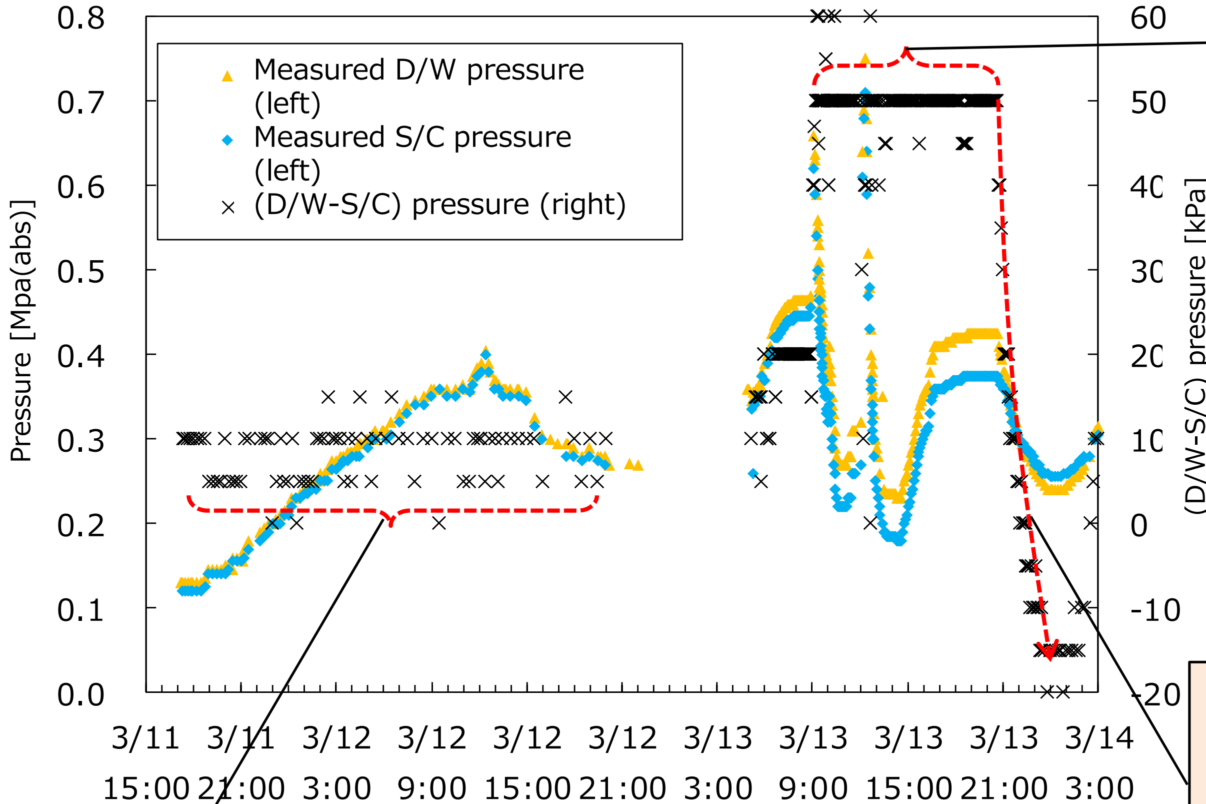


Fig. Actual measured pressure values of D/W and S/C, and pressure difference between them

<u>Situation B</u> Pressure difference between D/W and S/C increased to +50kPa and continued after S/C venting (below figure).

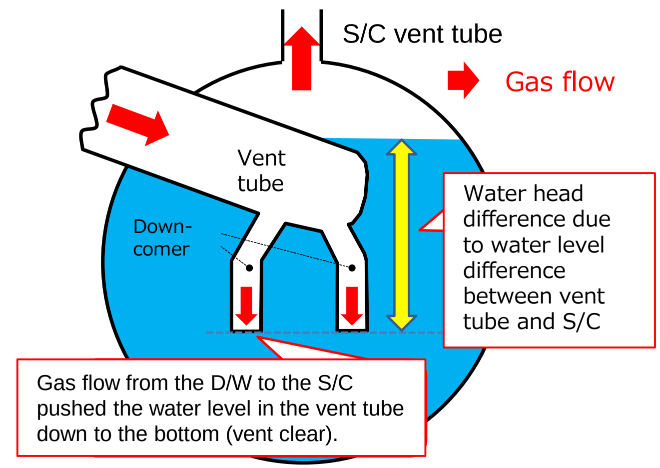


Fig. Water level status of vent tube and S/C after S/C venting.

<u>Situation A</u> PCV pressure rose faster than expected due to decay heat, presumably due to temperature stratification of S/C. D/W pressure > S/C pressure, when originally D/W pressure < S/C pressure.

<u>Situation C</u> During this period, the change in the D/W-S/C pressure differential (+50 kPa to -15 kPa) indicated that the water level in the vent pipe rose about 6.5 m relative to the S/C water level, which meant that the vacuum break valve was submerged (if the vacuum break valve was not submerged, D/W and S/C would be at equal pressure).

Estimation of S/C water level at the start of the first venting based on plant parameters
(1) Evaluation based on measured S/C water level

- Based on the measured S/C water level up to 20:00 on March 12, S/C pool water mass balance (items ① to ④ below) and energy balance (items ① to ⑤ below), the S/C water level behavior up to the time of the first venting was evaluated.
- Although there were uncertainties in items ②,③,⑤ and ⑥, it was confirmed that the influence of ② spray water injection rate was dominant.

⇒The S/C water level at the time of the first venting was evaluated in two cases, one in which the amount of spray water injection was high (high S/C water level case) and the other in which it was low (low S/C water level case), while satisfying the actual measured value of the S/C water level.

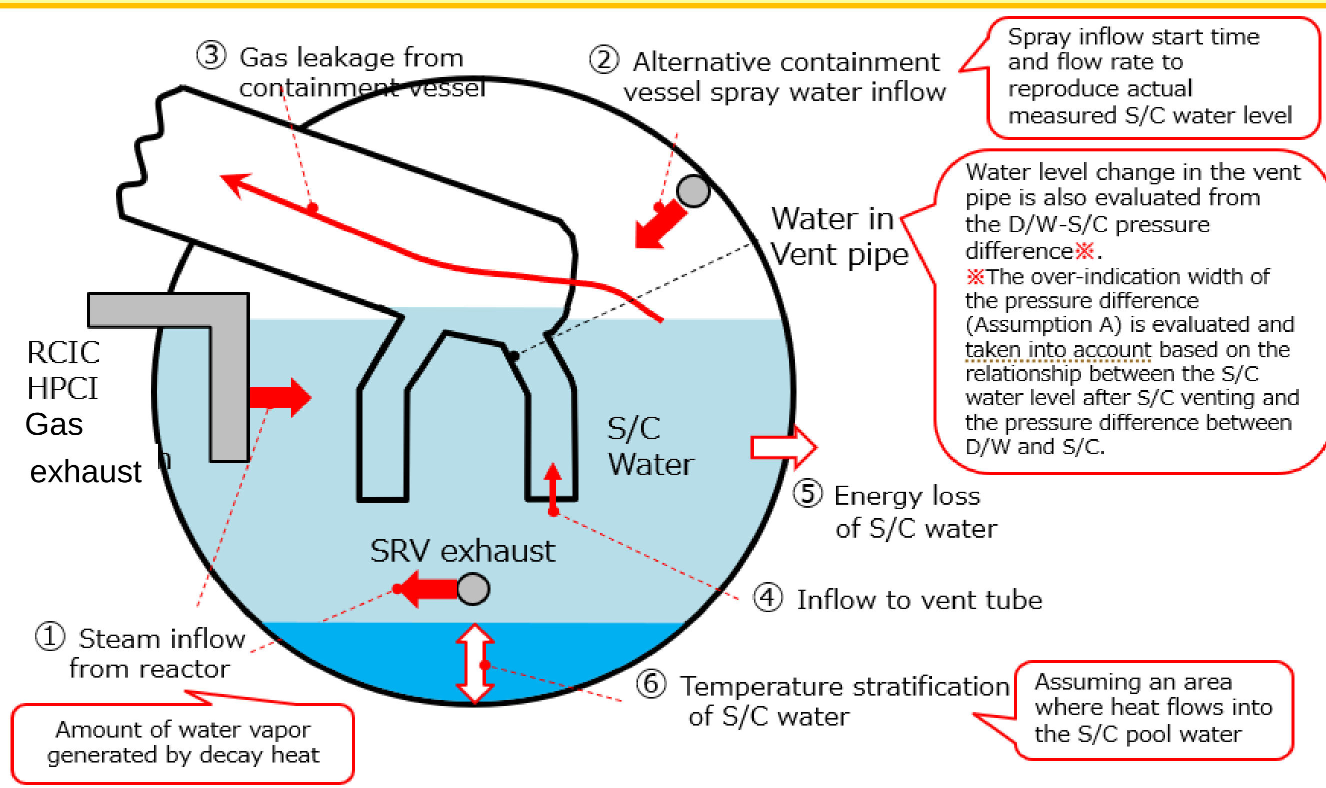
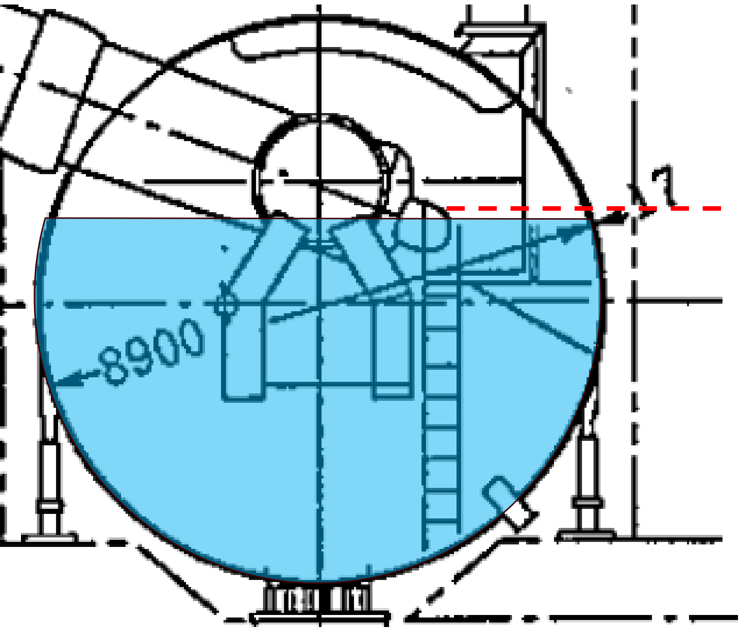
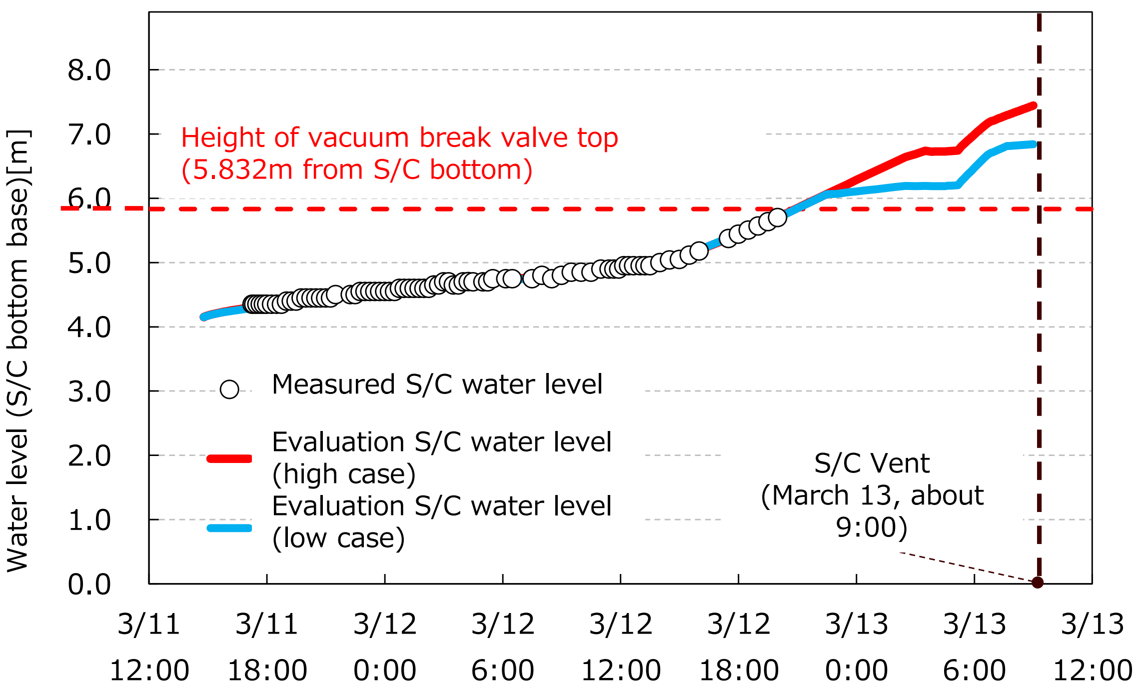


Fig. Schematic of the evaluation model

Estimation of S/C water level at the start of the first venting based on plant parameters
(1) Evaluation based on measured S/C water level

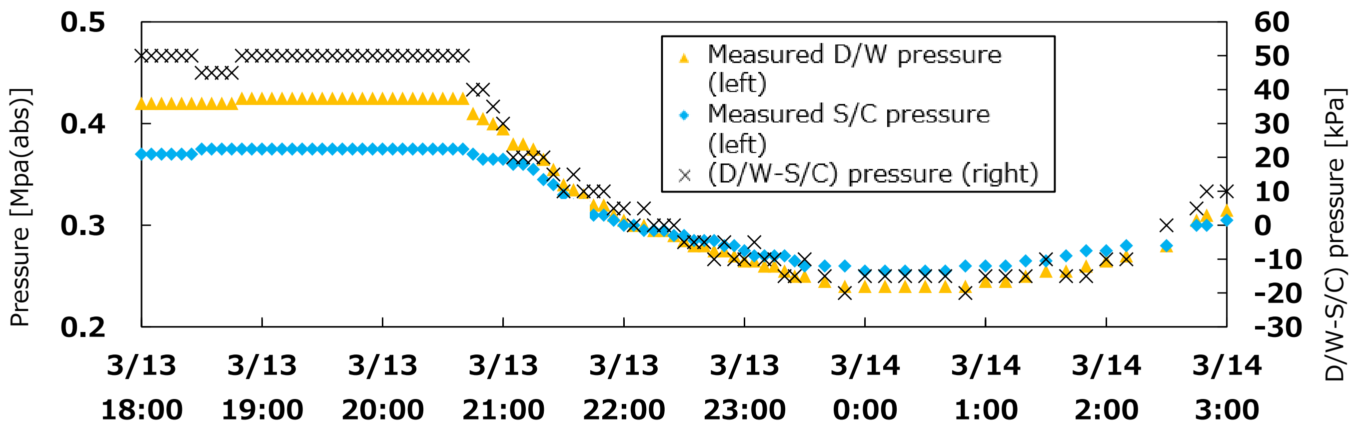
The S/C water level at the start of the first venting was 7.4m in the high water level case and 6.8m in the low water level case.





Estimation of S/C water level at the start of the first venting based on plant parameters
(2) Evaluation based on actual containment pressure measurements

After 20:40 on the 13th, S/C pressure became higher than D/W pressure amidst the decline.
It was presumed that this corresponded to the water level change as shown in the below figure.
Based on situations A, B, and C, the S/C water level at 20:40 on the 13th was evaluated.



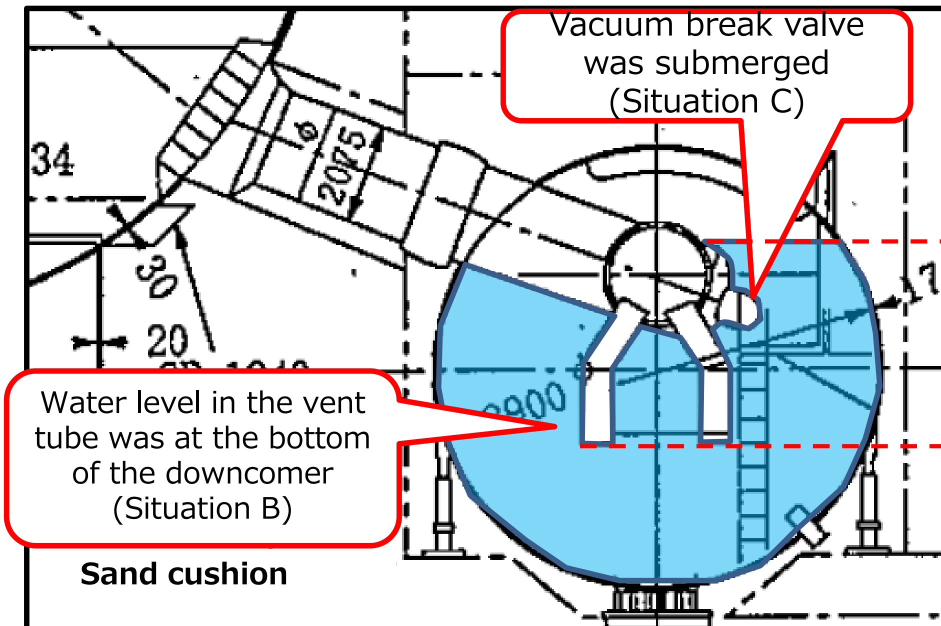


Fig. Image of water level before depressurization (at 20:40 on March 13)

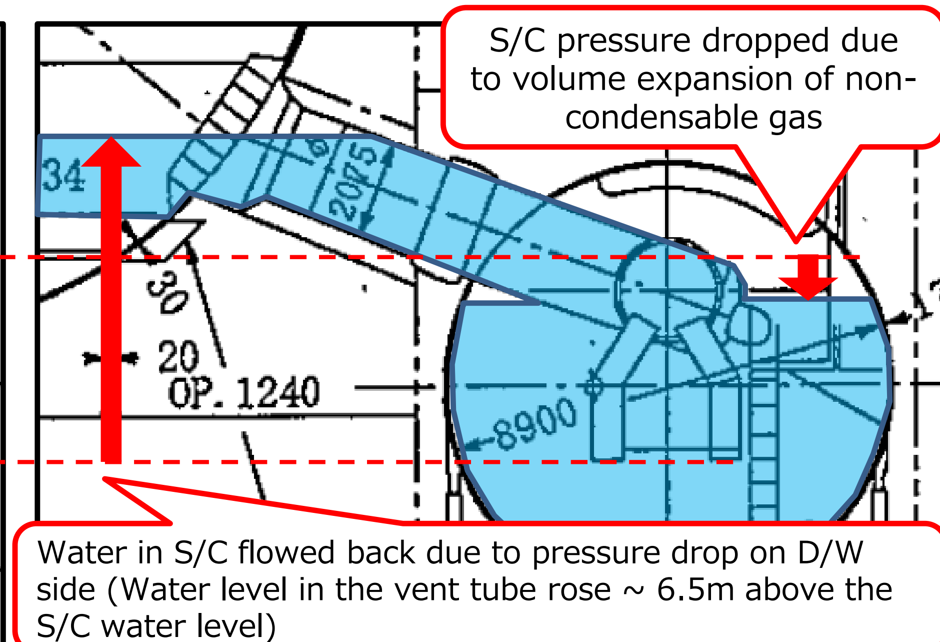


Fig. Image of water level after depressurization (at 0:00 on March 14)

Estimation of S/C water level at the start of the first venting based on plant parameters
(2) Evaluation based on actual containment pressure measurements

- Based on the D/W-S/C pressure difference of +50kPa before depressurization, the S/C water level at 20:40 on the 13th was estimated to be 6.8 to 8.3m from the bottom of the S/C (depending on the D/W-S/C pressure difference, below figure).
- Estimated maximum rise of S/C water level during approximately 12h from the start of the first venting to 20:40 on the 13th was 0.9m. (Estimated value of water pushed out of the vent pipe during venting + steam inflow from the reactor)

⇒Based on the evaluation result of the S/C water level at 20:40 on the 13th (6.8-8.3m), it was estimated that the S/C water level was at least 5.9m from the bottom of the S/C when the first venting started.

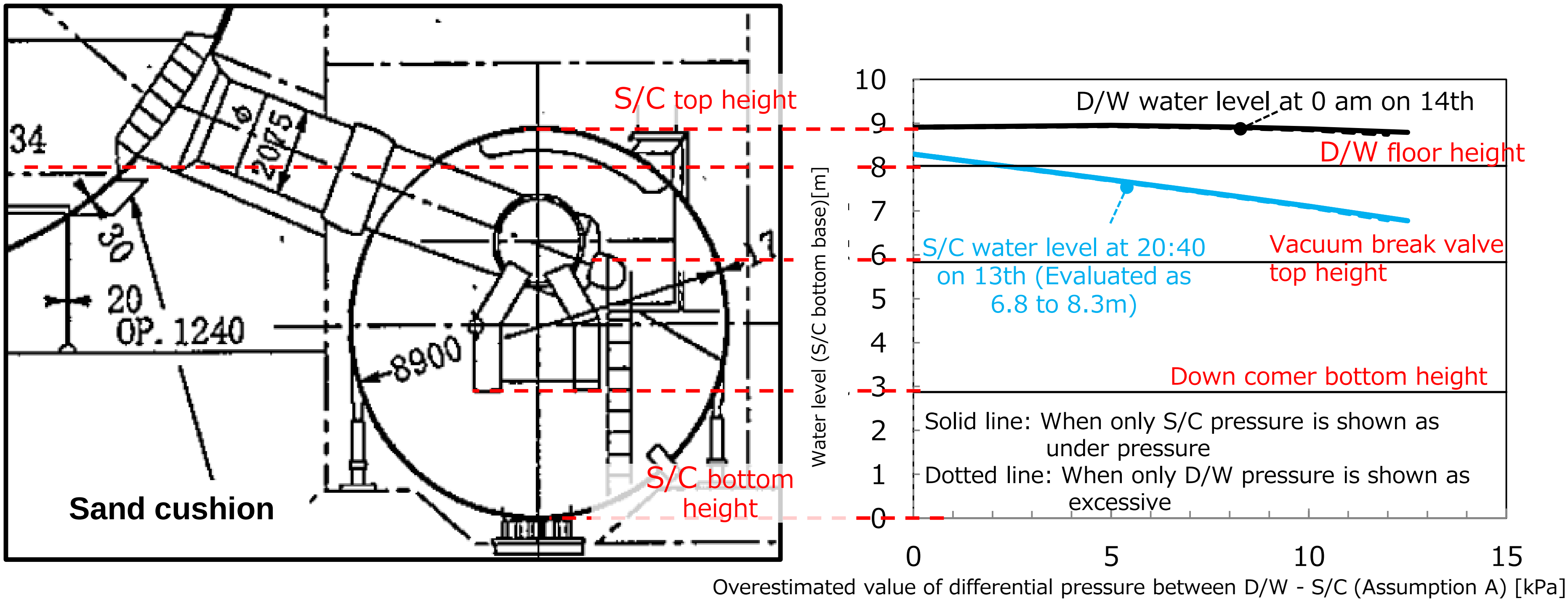
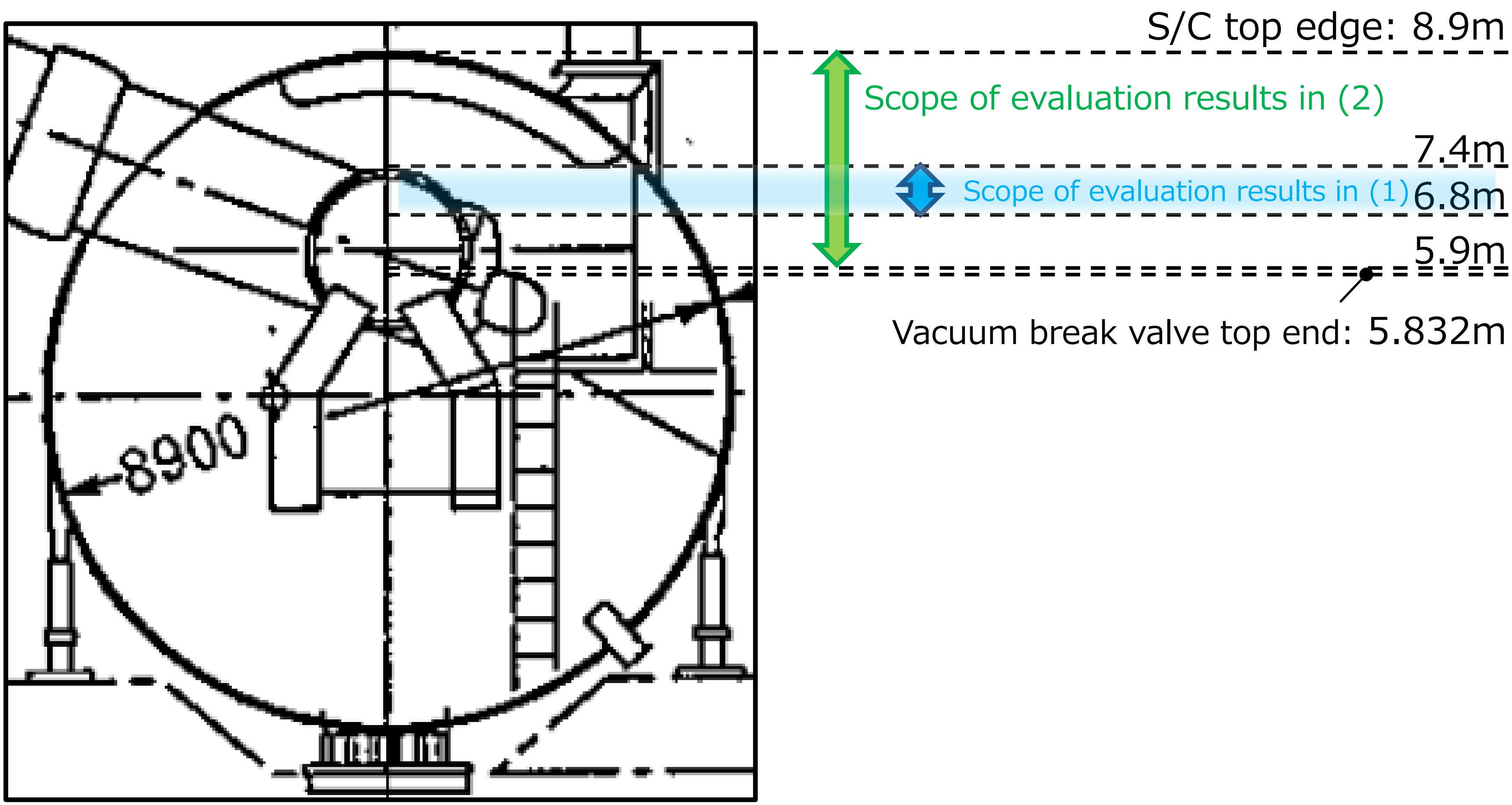


Fig. Results of S/C water level evaluation at 20:40 on March 13

Unified estimation of S/C water level at the start of the first venting based on plant parameters and evaluation results

Although there were uncertainties in the time of S/C spray and D/W spray and the amount of water injected, based on the evaluation using the S/C water level in (1) and the containment pressure in (2), the range of the S/C water level at the start of the first venting was estimated to be around 7m from the bottom of the S/C, which was higher than the vacuum break valve position.



Unified estimation of S/C water level at the start of the first venting based on plant parameters and evaluation results

- In Unit 3, the presence of water flowing back from the S/C in the pedestal might have inhibited spreading and MCCI during fuel debris fallout, and it might not have resulted in damage to the shell below the D/W.
- Leakage from the sand cushion drainpipe in Unit 1 suggested that the shell below the D/W was damaged.

⇒The high S/C water level estimated in this study was consistent with the observed fact that the current water level in the D/W of Unit 3 was higher than that of Unit 1, and that it could be treated as a possible accident progression scenario.

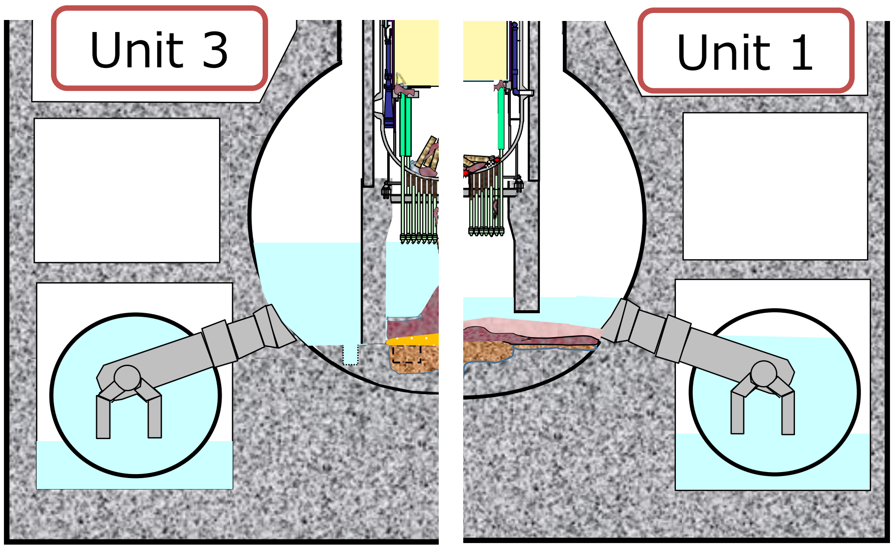


Fig. Current D/W water level (image)

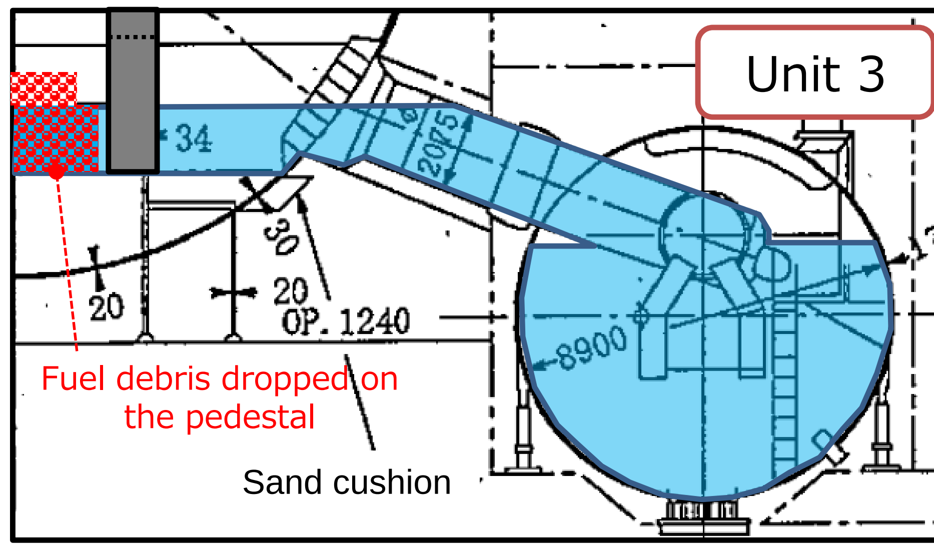


Fig. Image of fuel debris cooling by water in pedestal

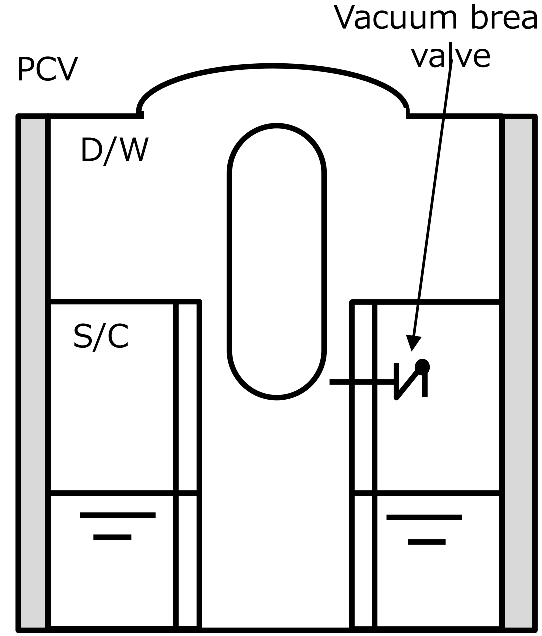
Safety measures in the Kashiwazaki-Kariwa NPS

Lesson learned: It is important to control the water level to avoid submerging the vacuum break valve.
(The vacuum break valve functions to prevent the PCV from becoming negatively pressurized, so it is important to maintain this function.)

- The decay heat generated in the PCV can be removed by circulating the PCV water through a residual heat removal system or a newly installed alternative circulation cooling system, while waste heat is transferred to seawater through a heat exchanger, in which case the water level in the PCV does not rise, and there is no danger of the vacuum break valve being submerged.

- If the above systems are not available, the water level of the PCV will rise due to continued water injection and spraying from outside the PCV to cool the PCV, but the procedure is to stop spraying before the vacuum break valve is submerged and to vent the PCV.

- Even if the vacuum break valve is submerged, the PCV can be prevented from being damaged by the negative pressure by stopping the PCV spraying before the PCV reaches negative pressure when the PCV spraying is conducted after the venting is stopped, etc., and by supplying nitrogen gas inside the PCV in the medium to long term.



PCV vacuum break valve

8. Accident progress after the Unit 3 reactor depressurization

Overview

- It is important to understand the containment pressure data in order to estimate the accident progression (containment vessel venting, gas phase leakage from the pressure vessel and containment vessel, hydrogen explosion, etc.) and the cooling status of the fuel debris after the reactor depressurization of Unit 3 at around 9:00 on March 13.
- Based on the estimation of the S/C water level at the time of containment venting of Unit 3 after 9:00 on March 13 and previous studies, we further examined accident progression scenarios for Unit 3 from 9:00 on March 13 to 0:00 on March 14.

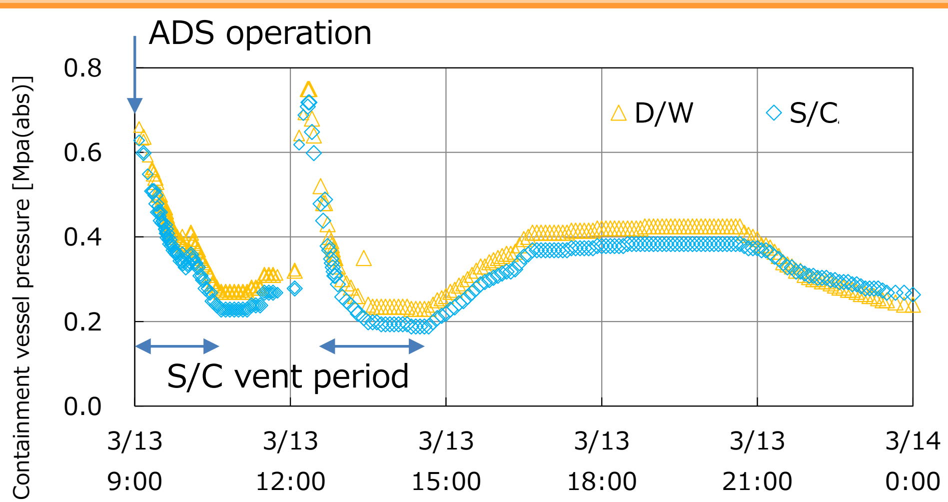


Fig. Containment pressure after ADS activation in Unit 3

<Examination approach>

- Accident progression scenarios were developed based on measured behavior and the results of previous studies.
- The developed accident progression scenarios were also examined from a quantitative perspective through analyses that reproduced the behavior of measured values.

Examination results: Estimation of accident progression scenarios that could quantitatively reproduce measured trends (main ones below)

- Possible gas leakage from the pressure vessel to the D/W occurred at about the same time as the ADS activation.
- Possibility that opening of 6 SRVs could not be maintained between ADS activation and about 12:00.
- Possible gas leakage from D/W occurred at about 16:40 on the 13th.
- Possibility that depletion of lower plenum water in RPV affected D/W depressurization from about 20:40 on 13th.

Related safety measures at Kashiwazaki-Kariwa NPS:
Reinforcement of depressurization maintenance function and containment leak prevention measures

Major past examinations

- RPV depressurization around 9:00 on the 13th was caused by ADS operation of SRVs (3rd progress report)
- Of the PCV vents conducted at Unit 3, only two were successful, after 9:00 and after 12:00 on March 13 (4th progress report)

* For RPV pressure charts, adjusted to match the time of scram on record (+7.5 min)
For RPV pressure (both charts and non-charts), consider the evaporation of water in the water level gauge pipe and its relationship with PCV pressure (+90 kPa after ADS).
For S/C pressure, consider the possibility that the pressure difference between D/W and S/C was overstated (+8.2 kPa), as mentioned in the "Study on the water level in the pressure suppression chamber of Unit 3".

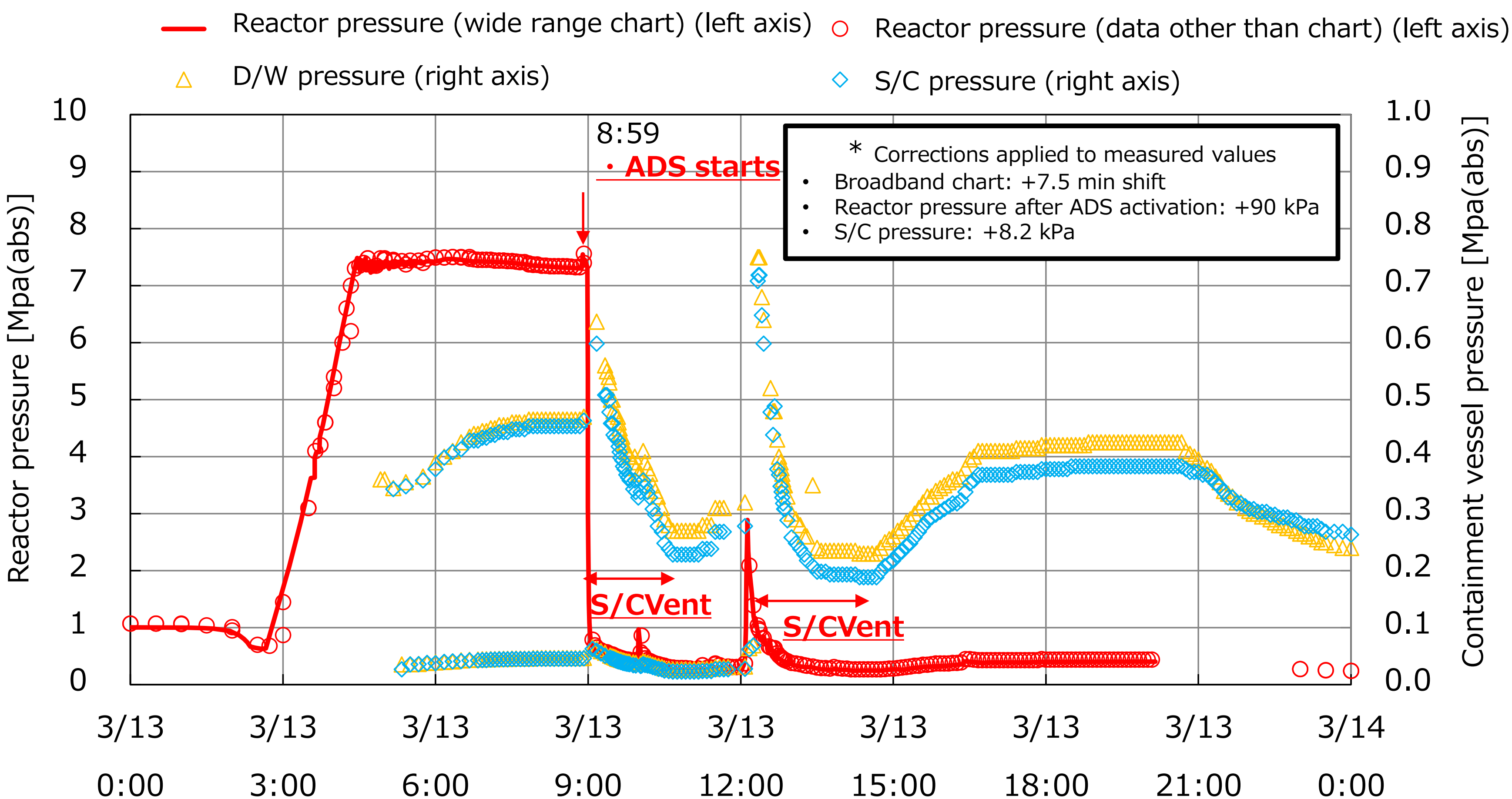


Fig. Overview of accident progression scenarios developed

Estimation of major accident developments addressed in this examination

① Possible gas leakage from the pressure vessel to the D/W occurred at about the same time as ADS activation.
② Possibility that opening of 6 SRVs could no longer be maintained between immediately after ADS activation and around 12:00.
③ Possible gas-phase leakage from D/W occurred at around 16:40 on the 13th.
④ Possibility that depletion of the lower plenum water in the RPV affected the D/W depressurization at around 20:40 on the 13th.

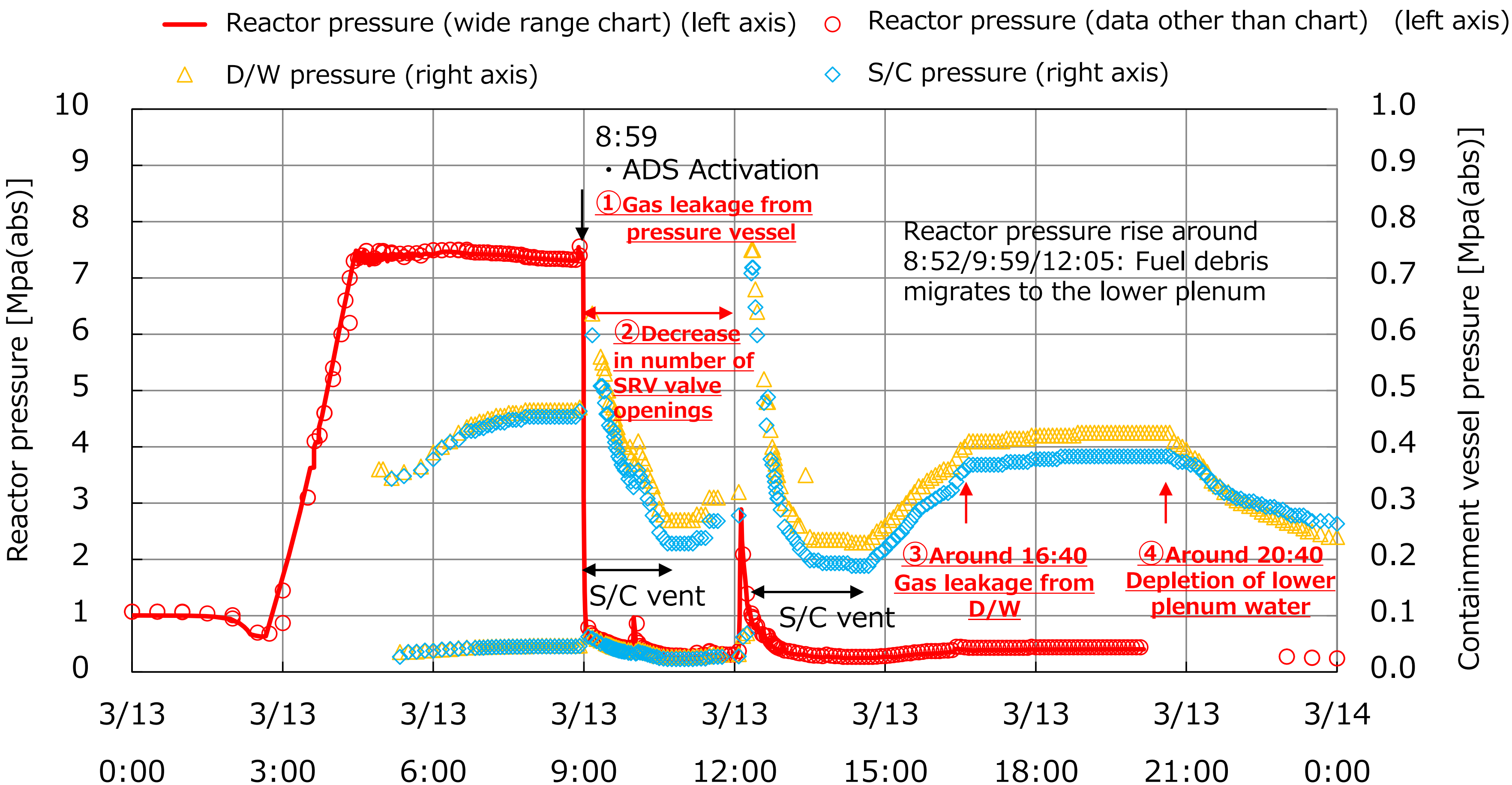
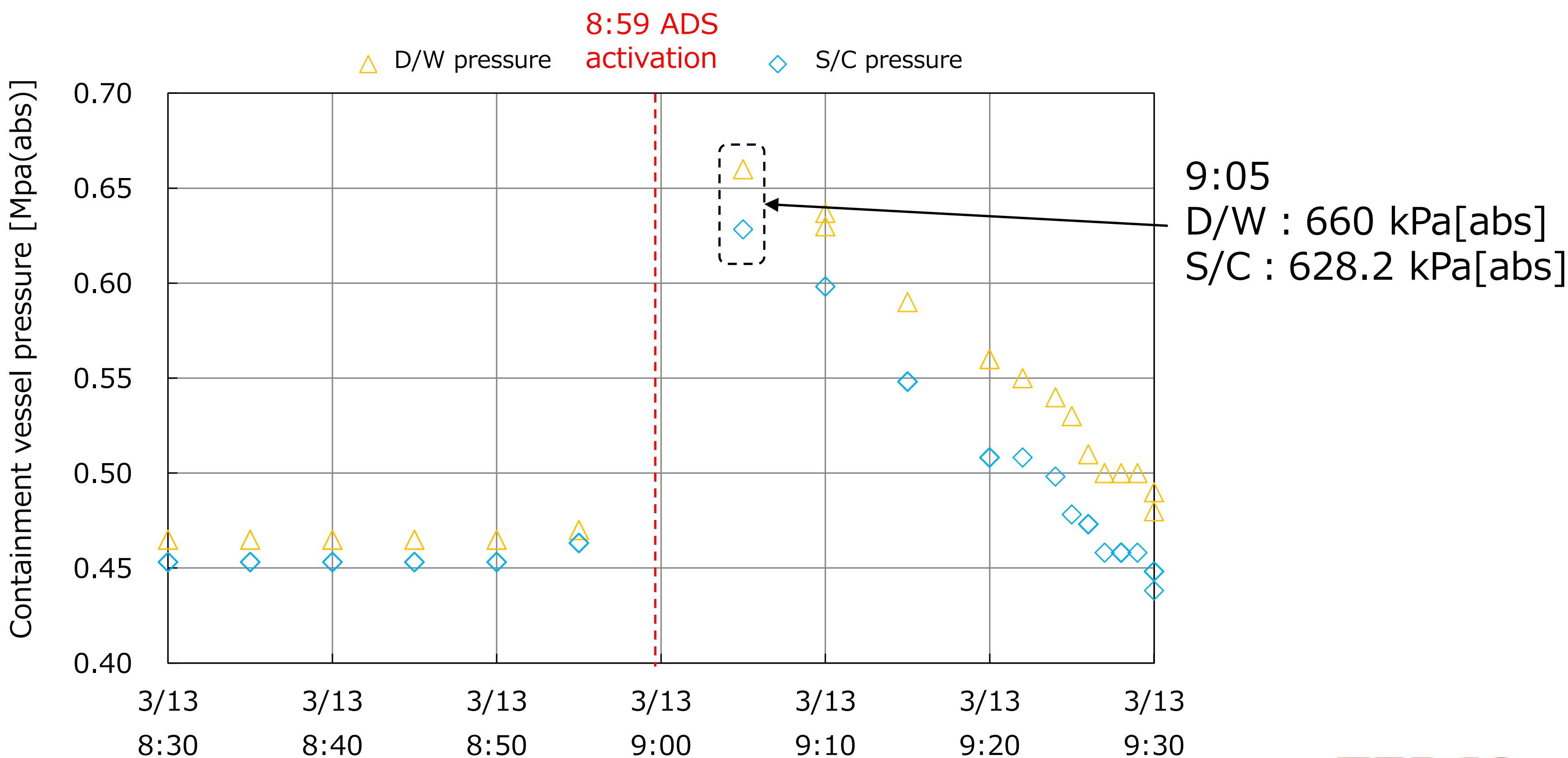


Fig. Overview of accident progression scenarios developed

①Gas phase leakage from pressure vessel

- Regarding the PCV pressure after ADS activation, the relationship was D/W > S/C for recorded data, where D/W < S/C was temporarily due to the large amount of gas flowing from the RPV into the S/C as a result of ADS activation. (As a result of S/C venting, the final result was D/W > S/C.)
- D/W pressure obtained at 9:05, immediately after ADS activation, was several tens of kPa[abs] higher than S/C pressure.

⇒ Possibility of leakage from RPV to D/W at the same time as ADS activation.



①Gas phase leakage from pressure vessel

Evaluate the leakage area (minimum leakage area required to reproduce RPV and PCV pressures) to estimate the presence or absence of leakage from RPV to D/W (using GOTHIC code).

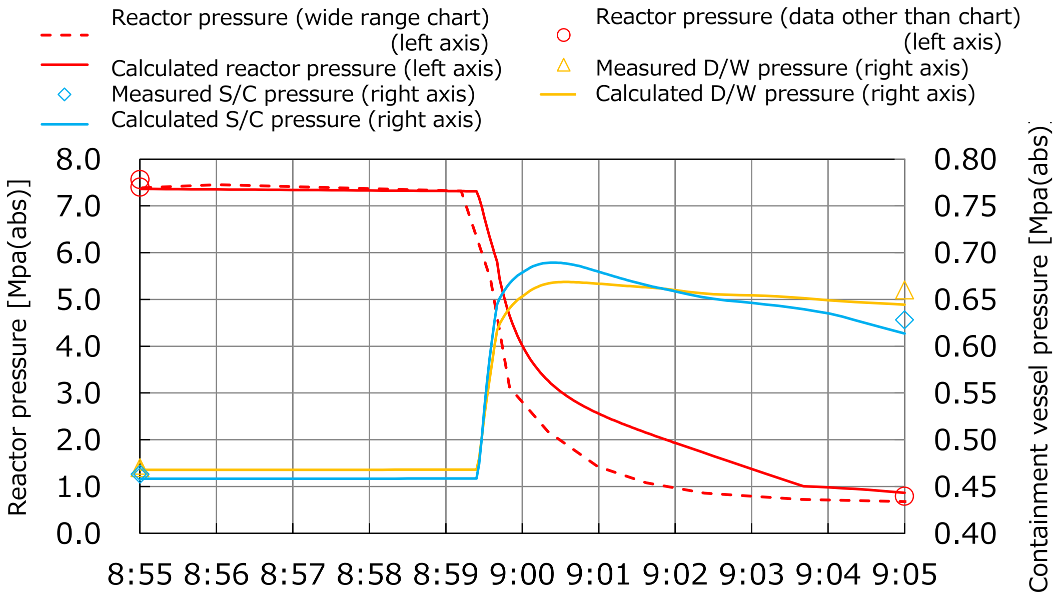


Fig. Evaluation results (gas-phase leakage area of pressure vessel: 30 cm^2)

＜Evaluation method＞

- Assumed conditions that facilitated reproduction of pressure behavior even with a small leakage area.
  - ✓ Assumed reduction in SRV area immediately after ADS activation (to reduce gas inflow to the S/C side in order to reproduce the relationship of D/W pressure > S/C pressure).
  - ✓ Assumed water vapor generation during depressurization process after ADS operation (to make the pressure difference between D/W and S/C easier to determine).
- The leakage area of RPV was varied to find the smallest value that could reproduce the pressure behavior.

- The minimum leakage area that could roughly reproduce the relationship between D/W and S/C pressures at 9:05 and the RPV pressure was found to be 30cm^2 (upper figure: there was a difference between measured and evaluated RPV pressure decompression rates, but if leakage area were further reduced, discrepancy from the measured RPV pressure value would further increase).

⇒ It was estimated that gas phase leakage from the RPV to the D/W occurred at about the same time as the ADS operation. A possible cause was the high temperatures in the RPV.

②Number of SRV valves opened

- Even after ADS activation of the SRVs, RPV pressure increase was observed, as seen around 10:00 and after 12:00.
- Regarding condition of SRVs, the main control room confirmed a situation in which both open and closed lights were lit only for 2 SRVs.
- ⇒Because the SRVs might have closed early after ADS activation, the period during which the SRVs were able to maintain 6 valves open was evaluated in the analysis (using GOTHIC code).

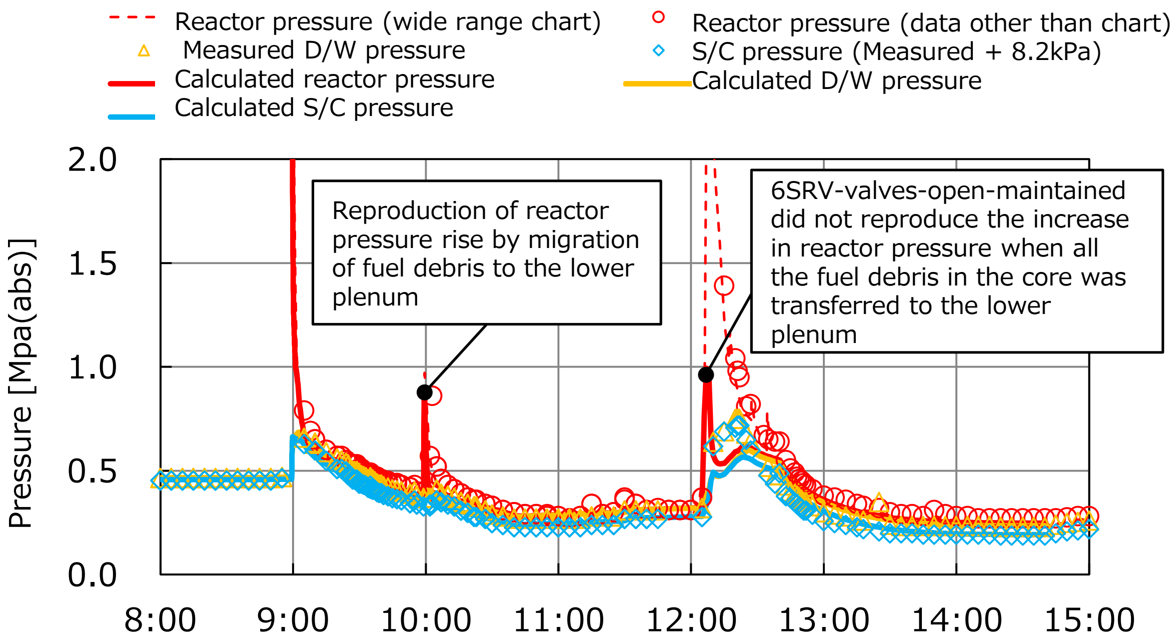


Fig. Evaluation results (6 SRV valves opening maintained case)

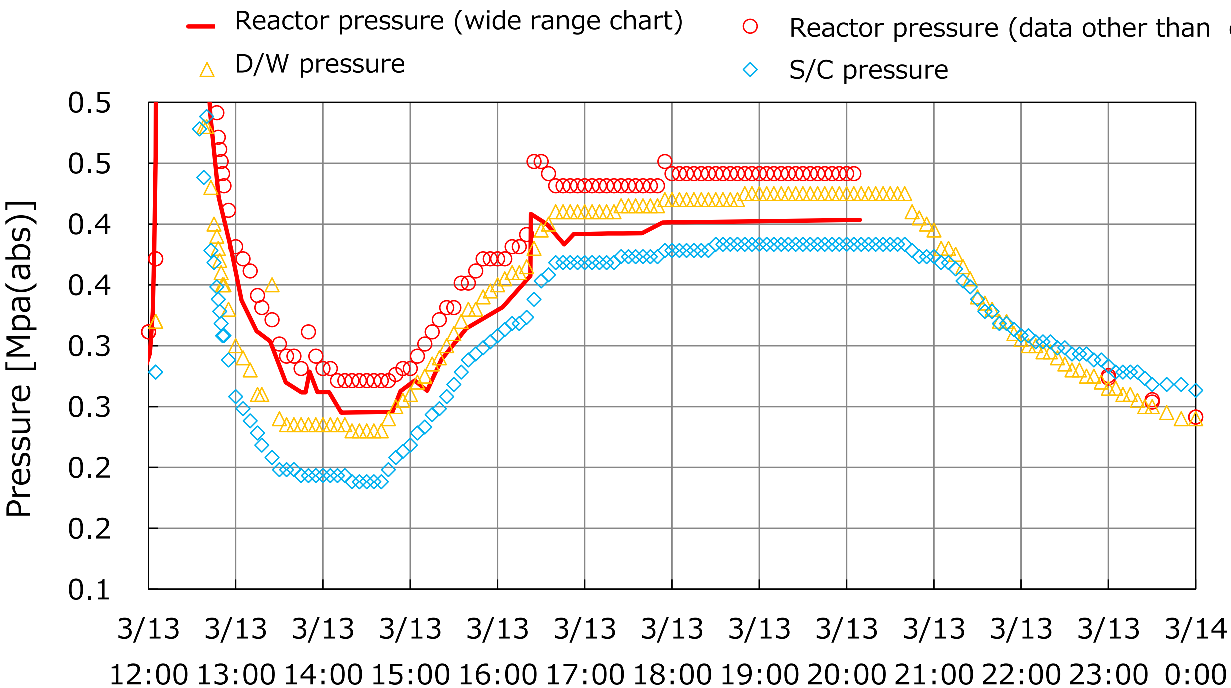
<Examination method>

- The analysis evaluated the longest and shortest time periods that 6 SRVs opening was maintained
- Case of 6 SRVs opening maintained
  - ✓ Analyze the point at which the increase in reactor pressure caused by the transfer of fuel debris into the lower plenum could be reproduced while the 6SRVs were kept open.
- Case of fully closed SRVs
  - ✓ The case in which SRVs were fully closed right after ADS activation (during depressurization) was also evaluated.

- In the 6-valves-open-maintained case, the increase in reactor pressure around 12:00 could not be reproduced even when all the fuel debris was transferred from the core to the lower plenum (above figure). But it was confirmed the pressure behavior could be generally reproduced, in the fully closed SRV case.
- It was highly probable that the 6 SRVs could no longer be kept open from just after the ADS activation to around 12:00. The cause of the inability to keep the SRV open could be lack of a power supply or deterioration of the SRV operating environment due to high temperatures in the PCV.

③D/W gas phase leakage

- By the time of the second S/C venting, no signs of gas phase leakage from the D/W could be read.
- On the other hand, during the period when the PCV pressure decreased after 20:40, the D/W pressure changed to below the S/C pressure, and it was estimated that leakage from the D/W occurred at this time (4th progress report).
- ⇒Timing of gas phase leakage from the D/W was qualitatively estimated from the plant data behavior.



Containment pressure after venting
about 14:40 & beyond: increased
about 16:40 & beyond: almost constant
about 20:40 & beyond: decreasing

⇒

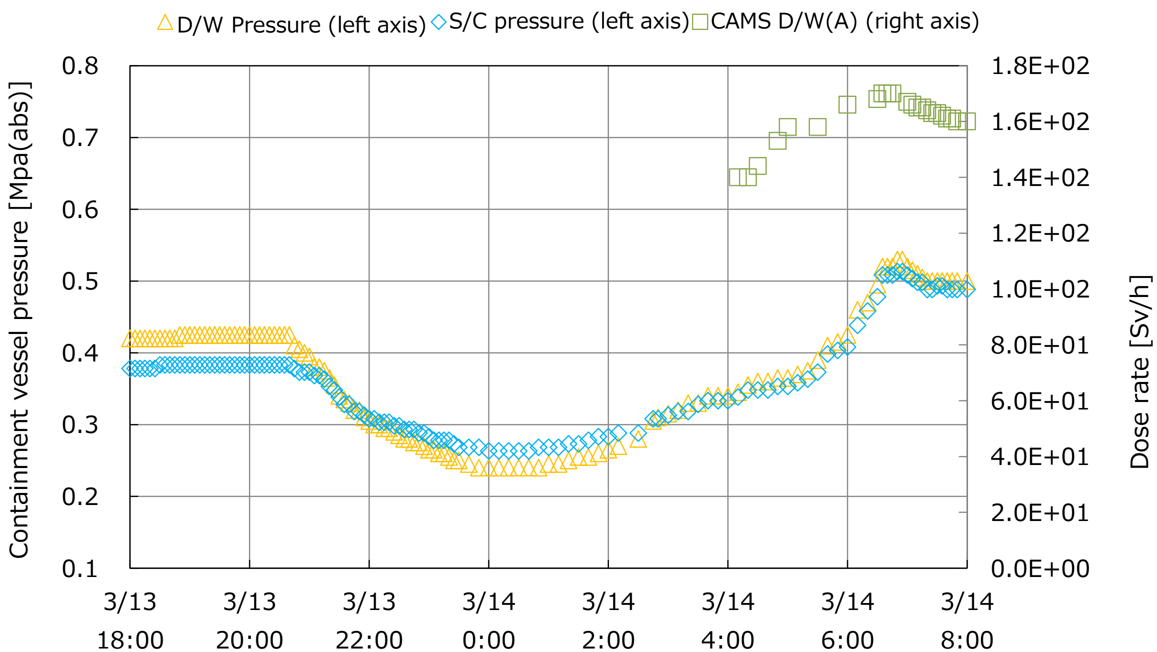
Factors of pressure change
- ✓ Gas phase leakage from D/W
- ✓ Decrease in gas generation (Decrease in vapor generation due to depletion of water in RPV)

- Corrected RPV pressure was higher than D/W pressure during constant PCV pressure; possible gas generation in the RPV, such as evaporation of lower plenum water.
- Presumed gas phase leakage from RPV after ADS operation.
- ⇒The PCV pressure did not increase under the condition of leakage from the RPV, so it was highly possible that there was gas phase leakage from the D/W after 16:40.

④D/W decompression around 20:40 on the 13th

- Two possible reasons for the decrease in PCV pressure were "expansion of gas phase leakage" and "decrease in gas generation".
- After around 0:00 on the 14th, the PCV pressure increased, and D/W CAMS(A) recorded a peak value (170Sv/h) at around 6:30 on the 14th; and other actual measurements were obtained that were suggestive of damage to the RPV lower head.

⇒ Qualitative estimation of D/W depressurization factors from around 20:40 on the 13th based on plant data behavior.



- PCV pressure began to increase after midnight on the 14th, and there was little possibility of expansion of the D/W gas phase leak.
- The lower head of the RPV might have been in an undamaged state shortly before around 20:40 on the 13th. In addition, there was a high possibility that the fire trucks were not injecting all the water into the reactor at that time, and the water level in the RPV might have dropped.

⇒The D/W depressurization from around 20:40 was thought to have been caused by gas phase leakage from the D/W and depletion of the lower plenum water which contributed to a decreased amount of water vapor generation in the RPV.

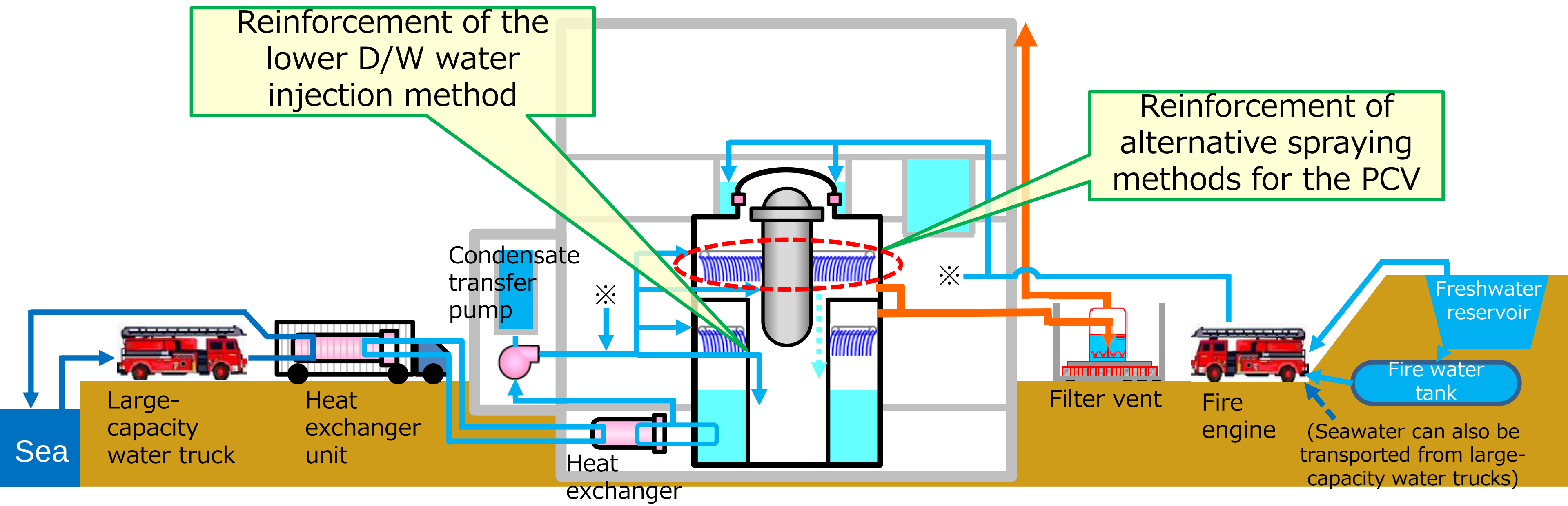
Safety measures in the Kashiwazaki-Kariwa NPS

Lesson learned:

- It was possible that the SRVs could not maintain the open state due to the high temperatures inside the PCV due to the gas phase leakage from the RPV, and that gas phase leakage could occur from the D/W. The importance of PCV cooling was once again suggested.
- Ways to supply nitrogen and a power supply need to be strengthened to keep the SRVs open.

At the Kashiwazaki-Kariwa NPS, the following measures have been implemented to control the temperature and pressure rise in the PCV and prevent PCV leakage.

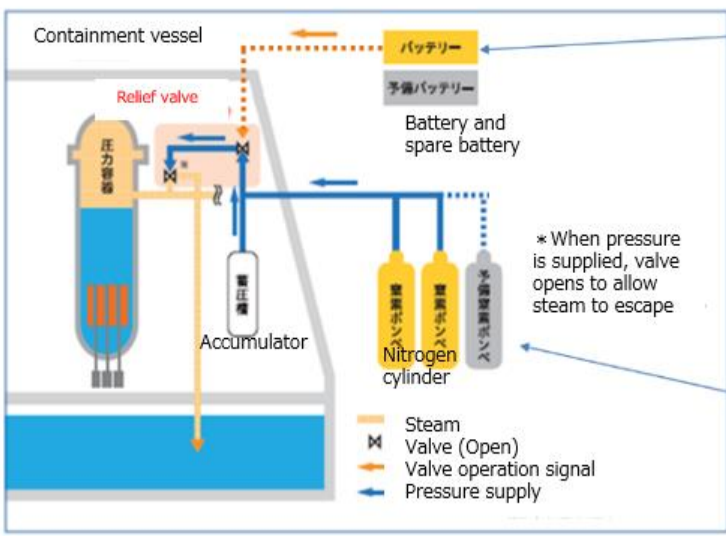
✓ Reinforcement of alternative spraying methods for the PCV
✓ Reinforcement of the lower D/W water injection method

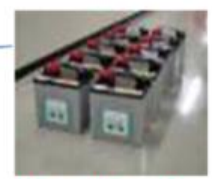


Safety measures in the Kashiwazaki-Kariwa NPS

■ The following measures ensure the ADS function of the SRVs, the manual forced depressurization function and the maintenance of openings.

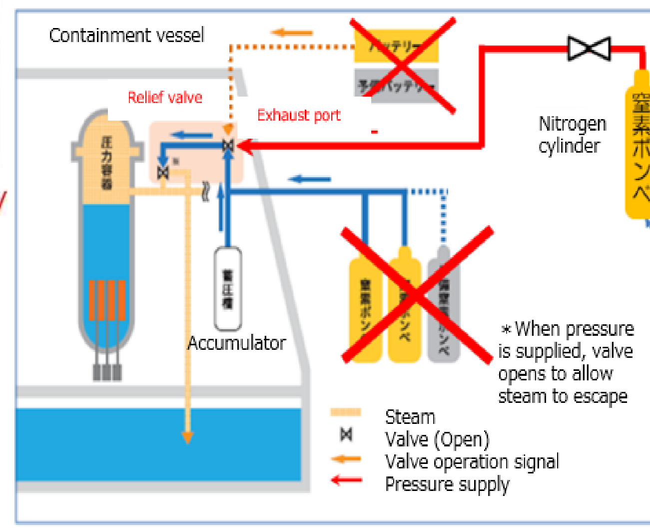
- A nitrogen supply method is secured by a cylinder in the high-pressure nitrogen gas supply system in case the accumulator loses nitrogen. In addition, a line independent of the high-pressure nitrogen gas supply system was added to allow SRV operation with only nitrogen supplied from the cylinder.
- The sealing material for the solenoid valve in the nitrogen supply line to the SRVs was changed to EPDM, which has excellent high-temperature resistance.
- An alternative spray procedure was added to mitigate thermal effects on the SRVs.
- To prepare for the loss of the permanent DC power supply, a supply method using storage batteries for the AM, portable DC power supply equipment (power supply vehicle), or portable storage batteries for SRVs has been added.





Portable battery

Portable gas cylinder



Portable gas cylinder

9. Examination of plant conditions during RCIC operation of Unit 3

Overview

- In the RCIC operation of Unit 3 after the arrival of the tsunami, the water source return line to the CST was utilized and the amount of water injected into the reactor was further adjusted to prevent it from tripping at the high reactor water level.
- The behavior of the reactor pressure during this period was recognized to be due to the complex situation in which the SRVs were opened and closed while RCIC was being operated in a special way.
- A study was conducted to confirm the validity of this qualitative explanation.

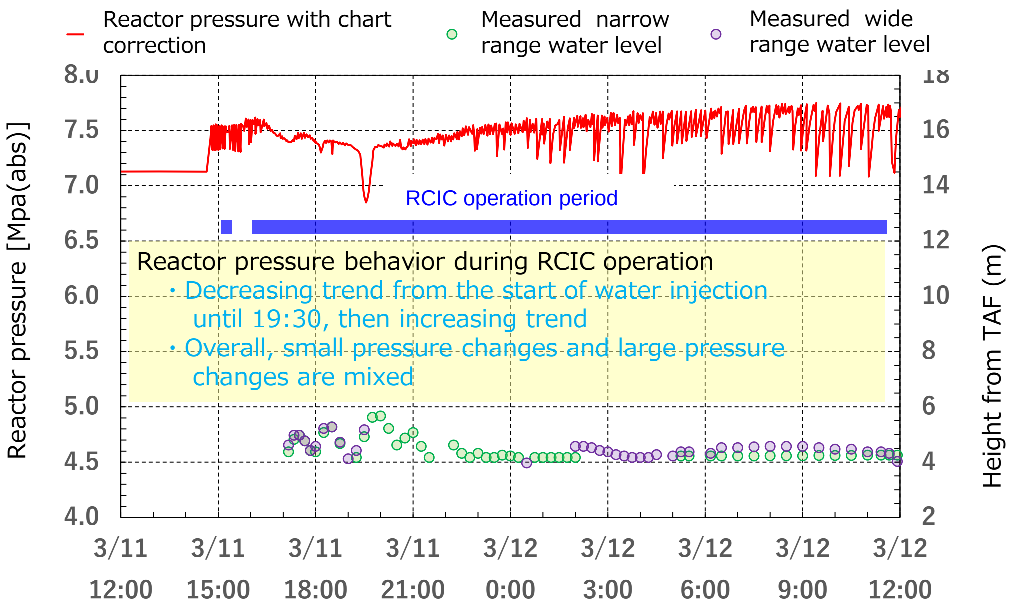


Fig. Reactor pressure and reactor water level during RCIC operation

＜Examination approach＞

- Confirmation and examination of the situation during the relevant period
  - ➢ RCIC operating results and operator operating procedures
  - ➢ Decay heat (energy balance)
  - ➢ Expected SRV opening modes
- Reproduction analysis of reactor pressure

- Through reproduction analysis that simulated water injection into the reactor by the RCIC and the opening and closing of the SRV, the validity of the understanding so far regarding the plant behavior during this period was confirmed, and the following results were obtained.
  - ➢ Decrease in reactor pressure due to water injection from RCIC to reactor.
  - ➢ Since the decay heat could not be consumed by the steam supplied to the RCIC turbine alone, steam was released via the SRV (it was thought the SRV was opened to some extent, but it was not fully opened).

Related safety measures at Kashiwazaki-Kariwa NPS: Enhanced decompression maintenance

RCIC operation after tsunami arrival (2nd operation)

Time series of RCIC operation

Time	RCIC related event	
3/11 15:05	RCIC manual start	
15:25	RCIC automatic stop (reactor water level high)	1st operation
15:27	Tsunami arrival (first wave)	
16:03	RCIC manual start	
16:16	RCIC water injection start	2nd operation
⟆		
3/12 11:36	RCIC automatic stop (Presumably due to high turbine exhaust pressure)	

Features of RCIC operation (2nd time)

- A line configuration in which water was passed through both the reactor water injection line and the test line, and a portion of the injected was returned to the CST in the test line.
- The flow rate was adjusted to prevent an automatic shutdown due to high reactor water level.

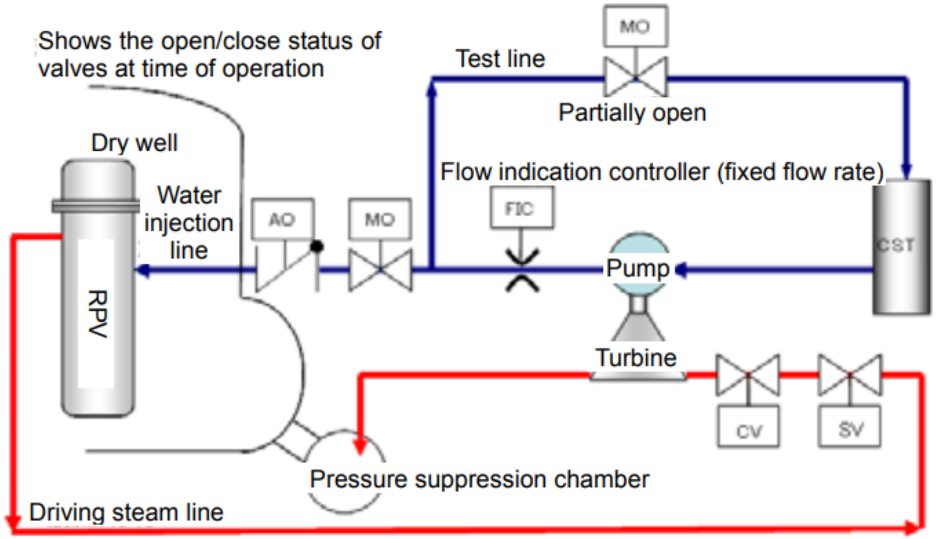


Fig. Reactor water injection line and test line

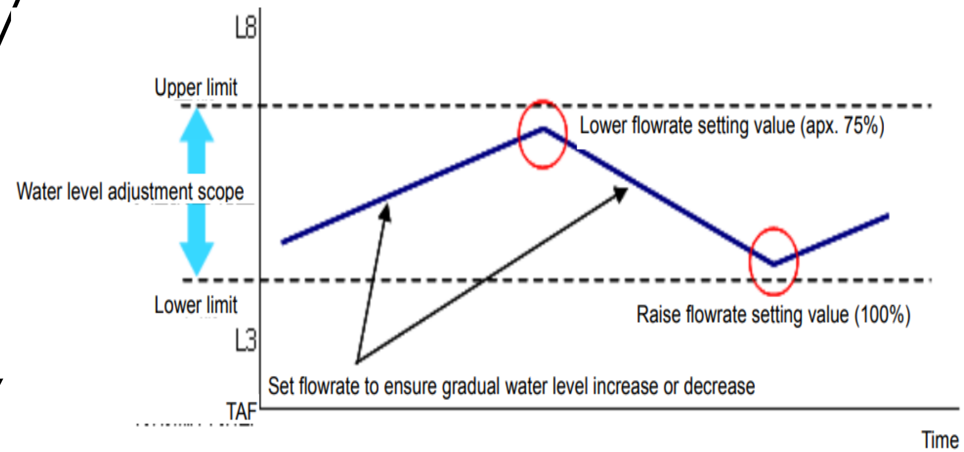
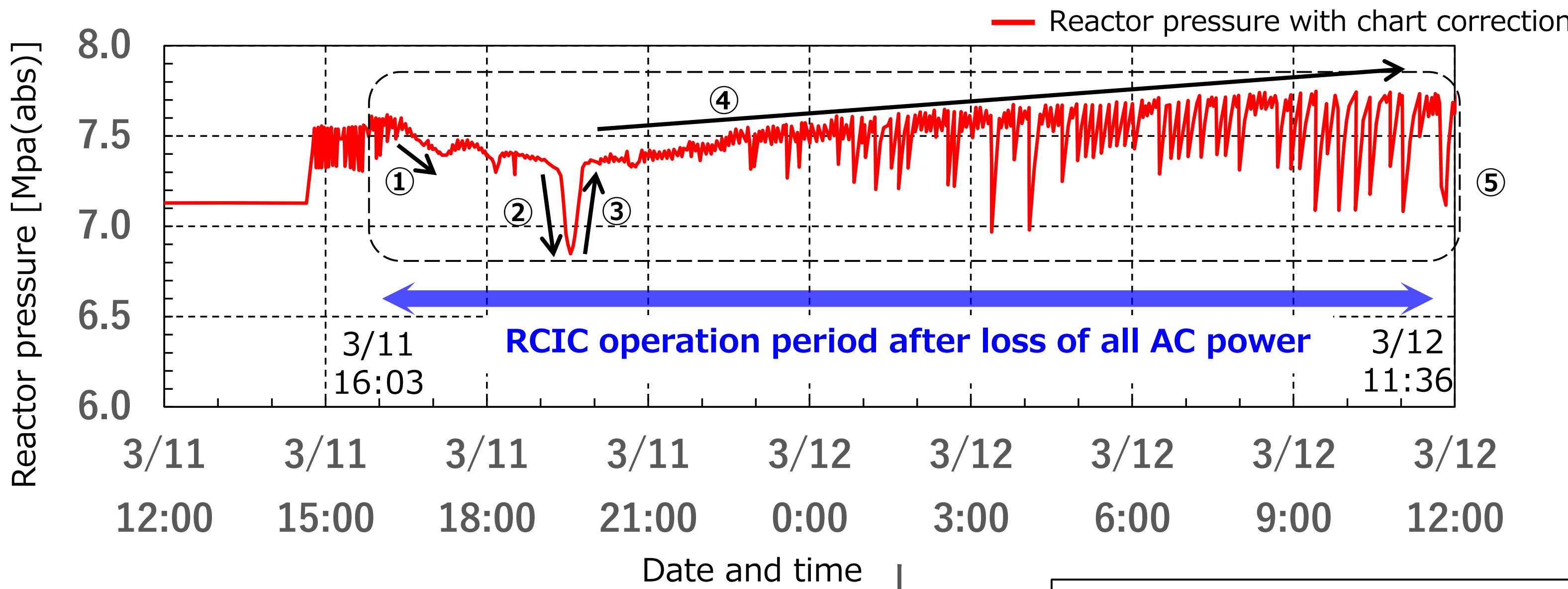


Fig. Schematic of the method to adjust reactor water level

Reactor pressure behavior during RCIC operation after loss of all AC power



Reactor pressure behavior during RCIC operation after loss of all AC power

①Pressure slowly decreased from the start of water injection (3/11 16:16).
②The pressure drop accelerated from 19:20, and dropped to about 6.85 MPa[abs] (around 19:30).
③The pressure began to rise from 19:30, reaching approximately 7.35 MPa[abs] (around 19:50).
④After that, a gradual upward trend continued until RCIC stopped.
⑤During this period, two behaviors were observed in the pressure change: a large pressure drop and rise, and a small pressure drop and rise.

Although the reactor pressure behavior during this period could not be explained by the normal opening and closing of SRVs, it was recognized as due to effects by water injection from RCIC to the reactor, extraction to the RCIC turbine, and unusual opening and closing of SRVs.

This examination will confirm the validity of the perception (qualitative explanation) during the (second) RCIC operation period through a reproducible analysis of the reactor pressure

Energy balance between decay heat and heat removal by RCIC operation

Decay heat at RCIC water injection start (16:16 on 11th) was ~ 27MW and less than the heat removal by normal RCIC operation of ~ 70MW

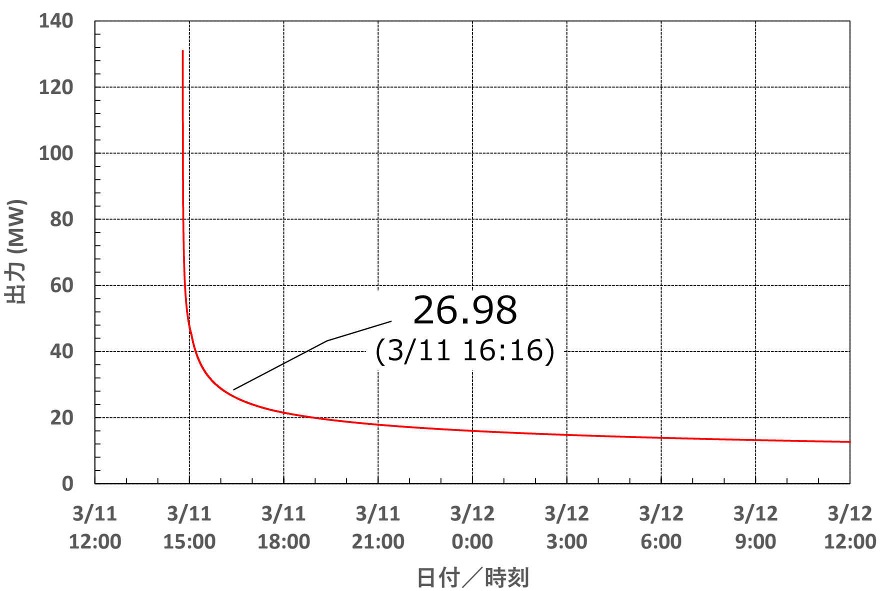


Fig. Time change of decay heat (Unit 3)

⇒ RCIC was operated during this period while the decay heat was reduced.

Water injection and steam release during RCIC operation in this period

- It was necessary to consider that part of the water injected by the RCIC was returned to the CST (not all of the water was injected into the reactor).
- It was necessary to consider the presence or absence of steam release* via SRV from the balance between changes in the reactor water level and the amount of water flowing in and out of the reactor.

* Steam release other than via SRV includes RCIC turbine extraction. During this period, water was also passed through the test line, so it was thought that the steam consumption at the RCIC turbine was relatively high.

SRV opening conditions in each operating mode

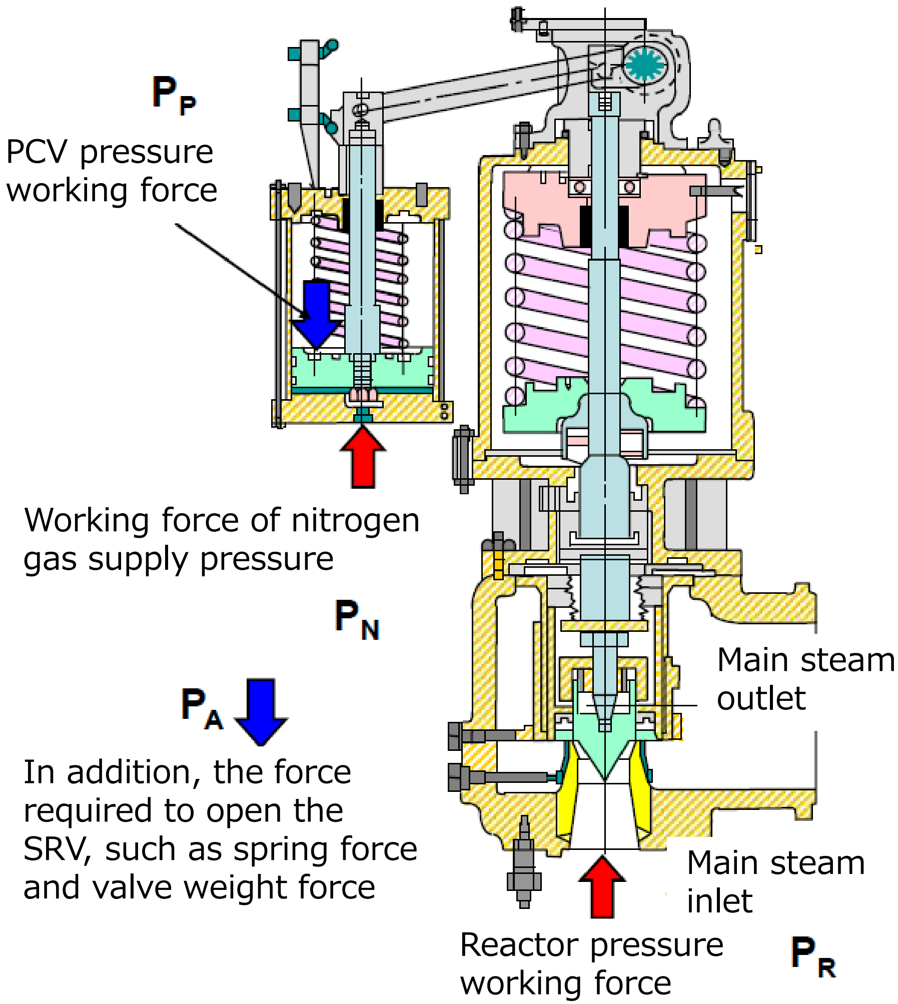


Fig. SRV cross-sectional view

【Relief valve mode】

Open at $\mathbf{P_P} + \mathbf{P_A} < \mathbf{P_N} + \mathbf{P_R}$

→Possibility of opening when reactor pressure (PR) rises if not depressurized to the return value

【Safety valve mode】

Open at $\mathbf{P_A} < \mathbf{P_R}$

→Possibility of opening at a lower pressure than the set value due to a decrease in Young's modulus caused by an increase in spring temperature.

Since this was the time when the amount of steam generation had decreased due to the decrease in decay heat, it was possible that the pressure would drop, and the valve would close immediately even if it opened in any of the modes.
(Possibility of partial opening)

Reproduction analysis of reactor pressure

■ **Reproduction analysis of reactor pressure (using RELAP5 code: below figure)**

⇒Although there is a difference* between the analysis results and the measured values, the behavior of the reactor pressure was approximately reproduced.

・The large pressure drop represented at around 19:30 on the 11th was reproduced (analysis was the effect of RCIC water injection)

・For the period from around 21:00 on the 11th, the behavior of the reactor pressure shown in the chart was reproduced by opening and closing the SRV (limited opening) and water injection from the RCIC.

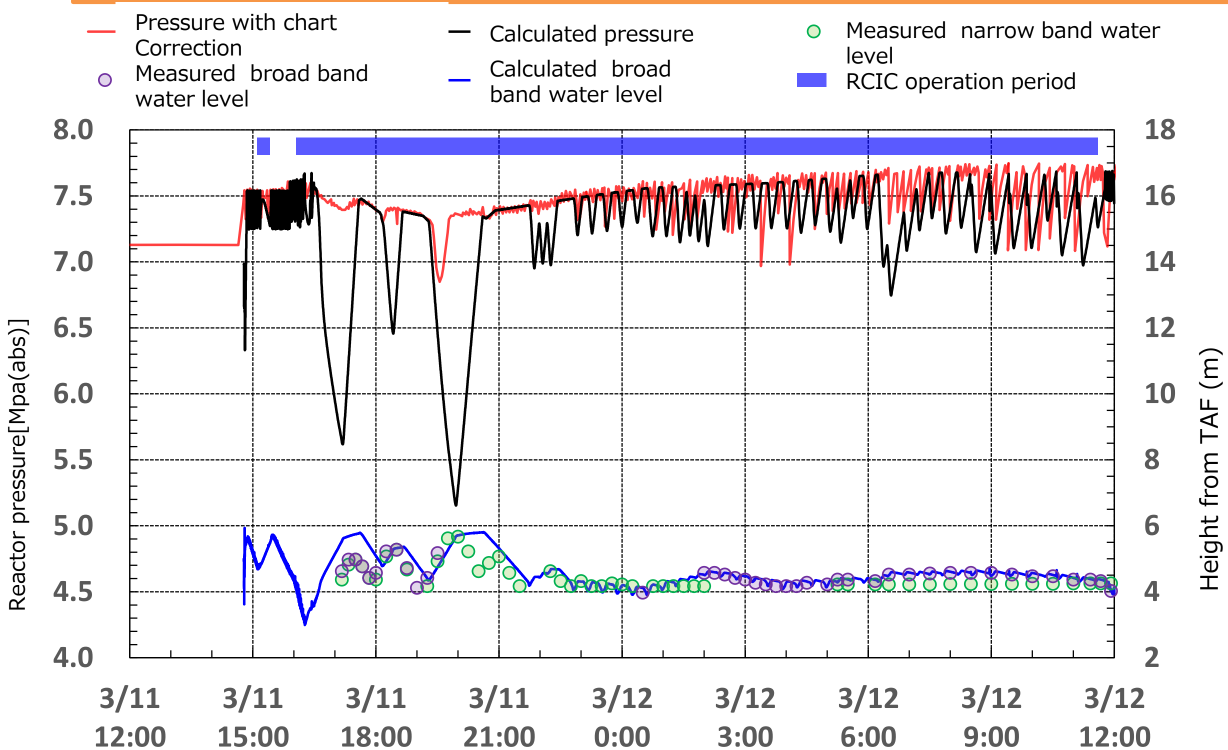


Fig. Reactor pressure and water level

【Main analysis settings】

①Adjust the amount of water injected into the reactor to satisfy the measured reactor water level.

⇒In the analysis, intermittent operation was performed at 80% of the rated flow rate in order to consider the return of water to the CST.

②Set SRV open/close to satisfy the measured reactor pressure.

⇒Analytically, a line was drawn that matched the indicated values in the reactor pressure chart (gradual decrease and increase during the period in question), and the SRVs were set to open when the pressure exceeded the line and to close when the pressure fell below the line (SRV opening was 10% of the fully open position).

※Main factor for difference

- In the analysis, water injected by RCIC tended to cause an excessive pressure drop due to its instantaneous mixing with water in the reactor.

Reproduction analysis of reactor pressure

SRV open/close: Design value
8 times open/close after loss of all AC power
(Consumption of accumulator pressure was considered)

Extraction to RCIC turbine continued during RCIC operation

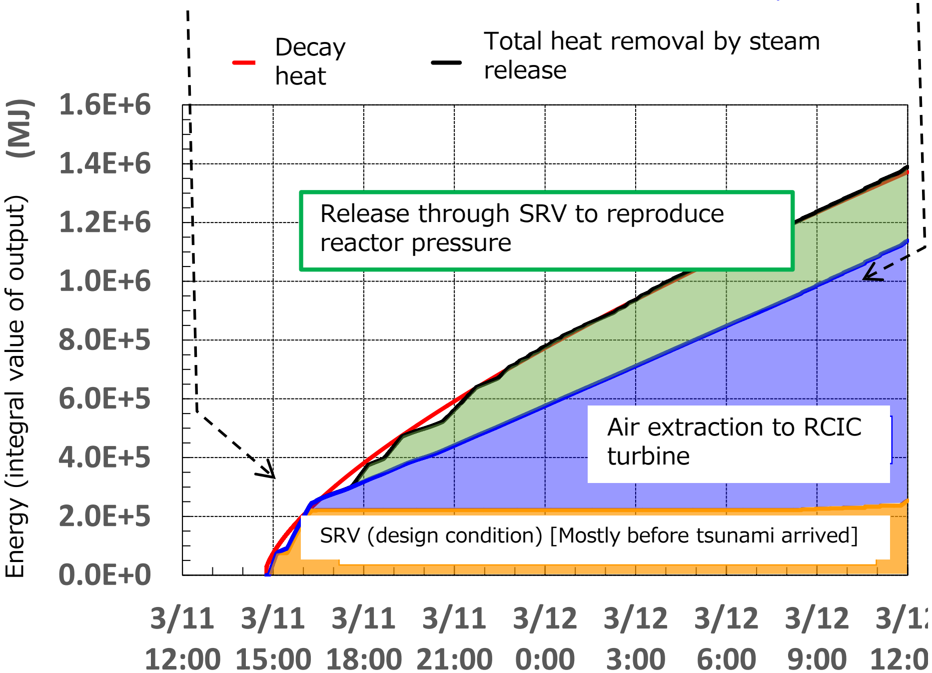


Fig. Relationship between decay heat and integral of heat removal by vapor release

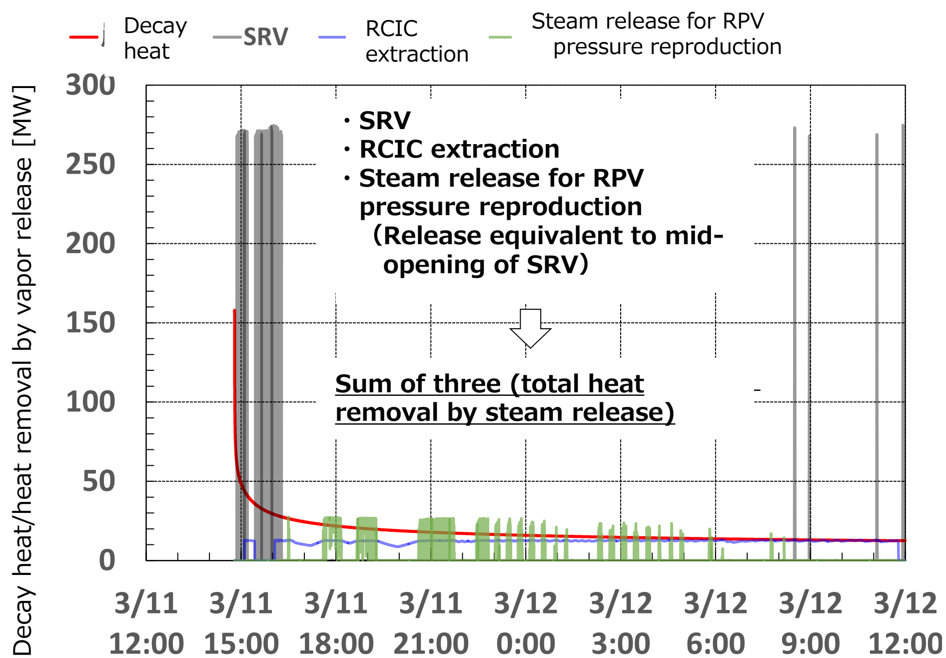


Fig. Heat removal by decay heat and vapor release

➢ The results showed that in addition to the steam supplied to the RCIC turbine, excess steam released via the SRV (left figure, green area) was required to remove decay heat during the RCIC operation period.

⇒It was highly likely that steam was released via SRVs.

Summary

- Reactor pressure behavior during RCIC operation of Unit 3 after the loss of all AC power was examined.

- The following conditions were confirmed through the analysis of the reproduction of the reactor pressure behavior during this period.
  - ➢ There was a decrease in reactor pressure due to RCIC water injection.
  - ➢ There was a high possibility of steam release via the SRVs, in addition to their opening and extraction of air to the RCIC turbine according to the design conditions.
    (Estimated flow rate was just below the point of opening (full opening).)

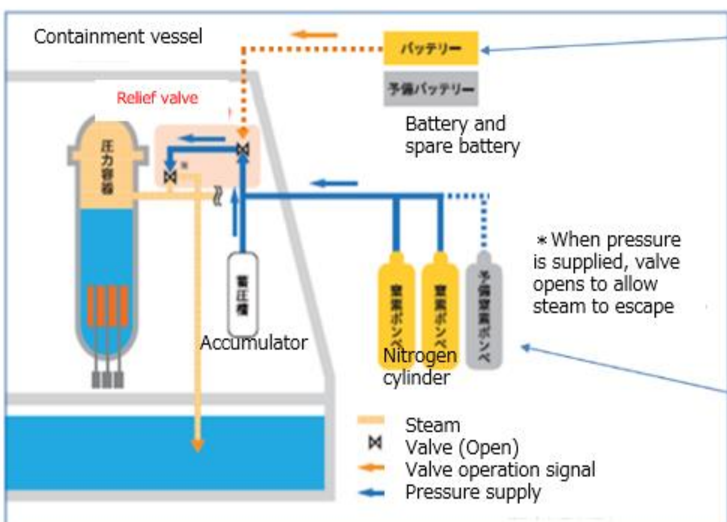
- The validity of the previous understanding was confirmed: the reactor pressure behavior during the RCIC operation period after the loss of all AC power was due to a combination of RCIC water injection into the reactor by special operations and intermittent steam release via SRVs in response to changes in reactor pressure.

Safety measures in the Kashiwazaki-Kariwa NPS

Lesson learned: Ways to supply nitrogen and a power supply need to be strengthened to maintain SRV function.

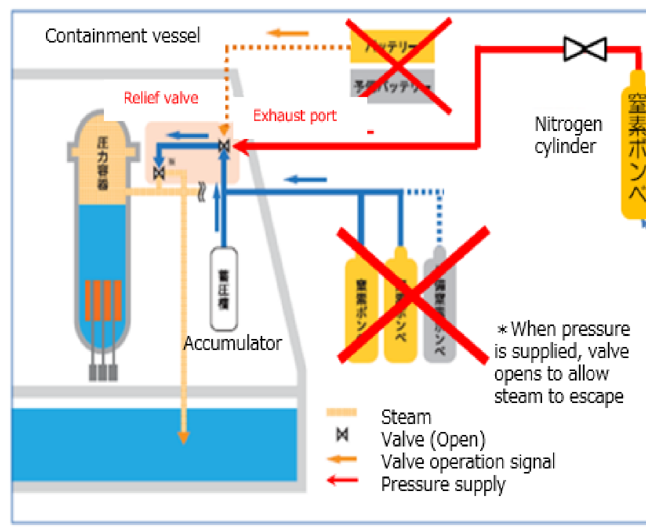
■ The ADS function and manual rapid depressurization function of the SRV are secured by the following measures. (From the viewpoint of ensuring reactor safety, the ADS function and manual rapid depressurization function to depressurize reactor and promote low-pressure water injection are more important than the relief valve function.)

- A nitrogen supply method is secured by a cylinder in the high-pressure nitrogen gas supply system in case the accumulator loses nitrogen. In addition, a line independent of the high-pressure nitrogen gas supply system was added to allow SRV operation with only nitrogen supplied from the cylinder.
- The sealing material for the solenoid valve in the nitrogen supply line to the SRVs was changed to EPDM, which has excellent high-temperature resistance.
- An alternative spray procedure was added to mitigate thermal effects on the SRVs.
- To prepare for the loss of the permanent DC power supply, a supply method using storage batteries for the AM, portable DC power supply equipment (power supply vehicle), or portable storage batteries for SRVs has been added.



Portable battery

Portable gas cylinder

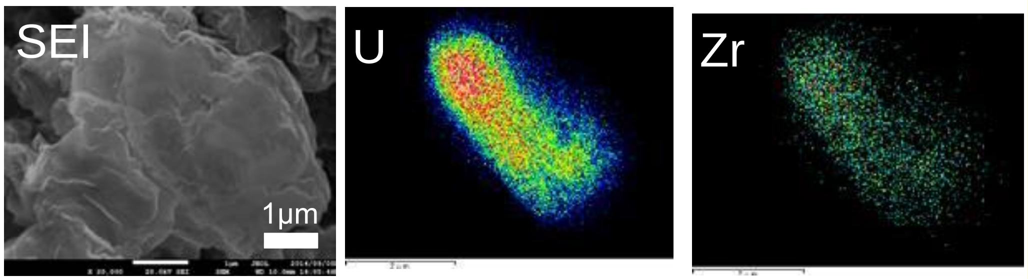


Portable gas cylinder

10. Sample analysis to determine accident progress

Overview

- Particles containing uranium (U) were detected from analysis samples collected inside and outside the PCVs of Units 1-3.
- Insoluble cesium (Cs) particles were detected in the environmental samples and their compositions were reported.
- These radioactive particles were thought to have originated from the high-temperature fuel at the time of the accident. If the formation process of these particles is known, information on the atmosphere (temperature change rate, hydrogen/steam ratio) inside the reactor pressure vessel (RPV) at the time of their formation, etc. can be obtained.
- Such knowledge will be used to understand the state of fuel debris and the progress of the accident.



U-containing particles on Unit 2 operating floor cover sheet
SEI (secondary electron image), element mapping (U, Zr) by SEM/WDS

<Examination Approach>

Examine the formation process of radioactive particles.

(1) Analysis focusing on U-containing particles

- Mixing state of fuel components was evaluated from the distribution of U isotopic ratios among samples.
- Formation process of U-containing particles was estimated by focusing on their composition and crystal structure and classified according to whether U underwent a melting and solidification process or an evaporation and condensation process.

(2) Examination of insoluble Cs particles

- Estimated formation process of spherical insoluble Cs particles。

<Findings on the condition of fuel debris>

- Most of the contamination sources in the stagnant water were present in particulate form and more than 90% could be removed by filtration. U is chemically stable in the form of cubic UO_2 and is unlikely to change over time.
- From the analysis results of the U isotope ratio (^{235}U/total U) in the sample, it was thought that the mixing of U isotopes progressed due to fuel melting.

<Findings on accident progress (from evaluation of the formation process of radioactive particulates)>

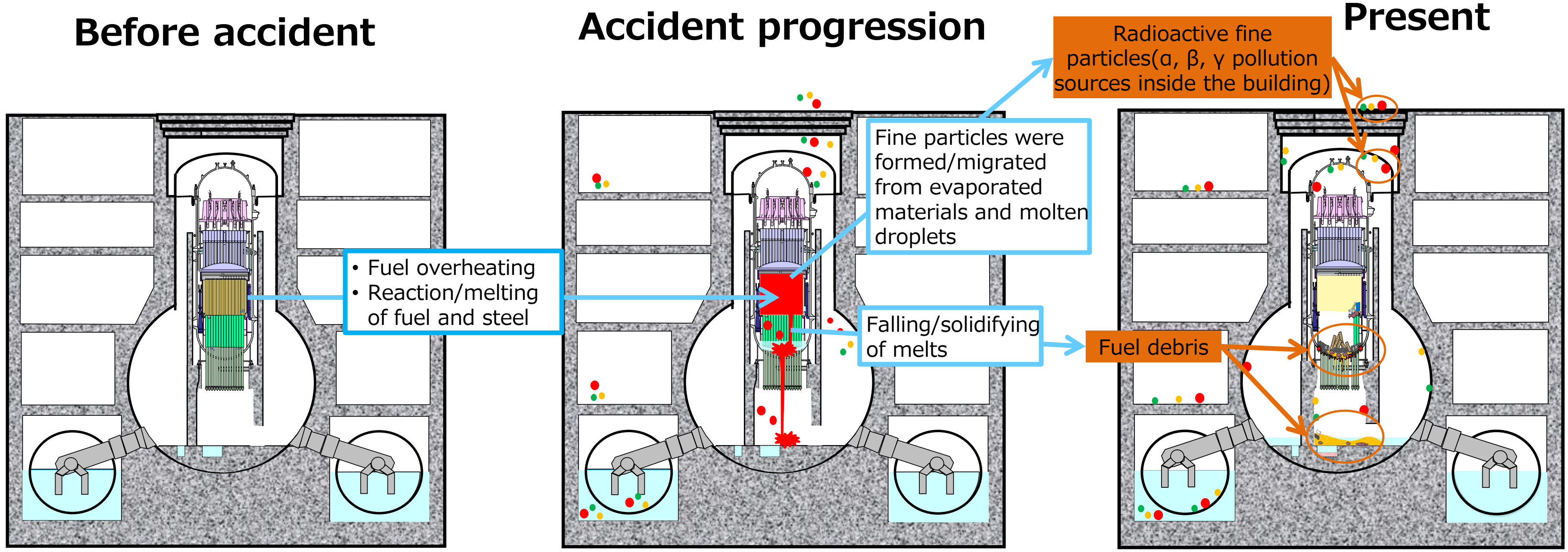
- Results suggested that the chemical environment (e.g., hydrogen/steam ratio) within the RPV and PCV changed with time and location.

- In Unit 1, particles thought to have been formed in a hydrogen-rich environment were confirmed, and these particles might be related to insufficient water injection into the reactor at the beginning of the accident.
- In Unit 2, particles thought to have been formed in an environment with a lot of water vapor and particles thought to have been formed in an environment with a lot of hydrogen were confirmed. The timing of formation of insoluble Cs particles was thought to be at the beginning of fuel temperature rise, which was considered to be a clue to the atmosphere inside the RPV at the time of formation.

Purpose of contaminant sample analysis and evaluation

10. Sample analysis to determine accident progress

Before accident **Accident progression** **Present**



The product changes depending on the atmosphere in the reactor at this time and cooling conditions of particle formation.

Sample collection and analysis from inside the building, PCV, etc.

- U-containing particles formed in the melting and solidification process
- U-containing particles formed during evaporation and condensation processes
- Particulate FPs

The composition and structure characteristics of U-containing particles and particulate FPs detected in the atmosphere are considered to include information on accident progression and information on fuel debris properties.

⇨ We believe that the results of the analysis of contaminated material samples will be useful in understanding the accident situation.

⇨ We believe that the knowledge and experience gained through the analysis and evaluation of contaminated material samples obtained will form the basis for the analysis and evaluation of fuel debris.

U-containing particles in building stagnant water

- Stagnant water was collected from the reactor building (R/B) torus rooms of Units 2 and 3 and filtered through 0.1μm filter paper; the total α activity concentration decreased by more than 90%. This indicated that most of the α contamination sources existed as particles.
- The possibility of aging was considered small because U-containing particles were detected when the filter paper was observed by SEM-EDS/WDS, and further observation by TEM-EDS-electron diffraction confirmed that U was chemically stable in the form of UO_2.

Raw water (before filtration)	
Sampling place	**Total α activity conc. (Bq/L)**
Unit 2 R/B	2.61E+05
Unit 3 R/B	1.50E+03

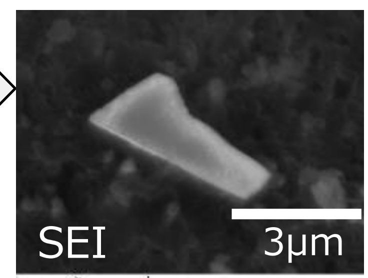
0.1μm filtering

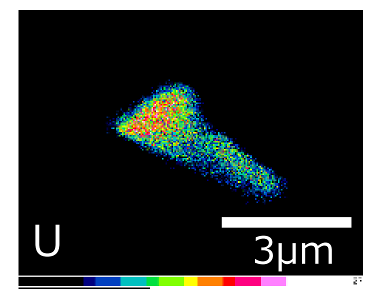
(Pass through filter)

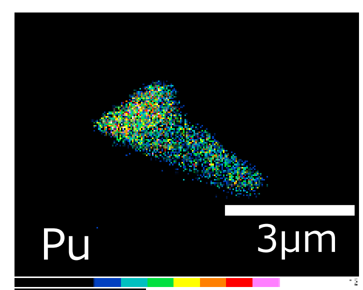
After filtration (0.1μm)	Removal ratio(%)
Total α activity conc. (Bq/L)	
9.54E+02	99.6
1.12E+02	92.5

(Remain on filter)

Residue

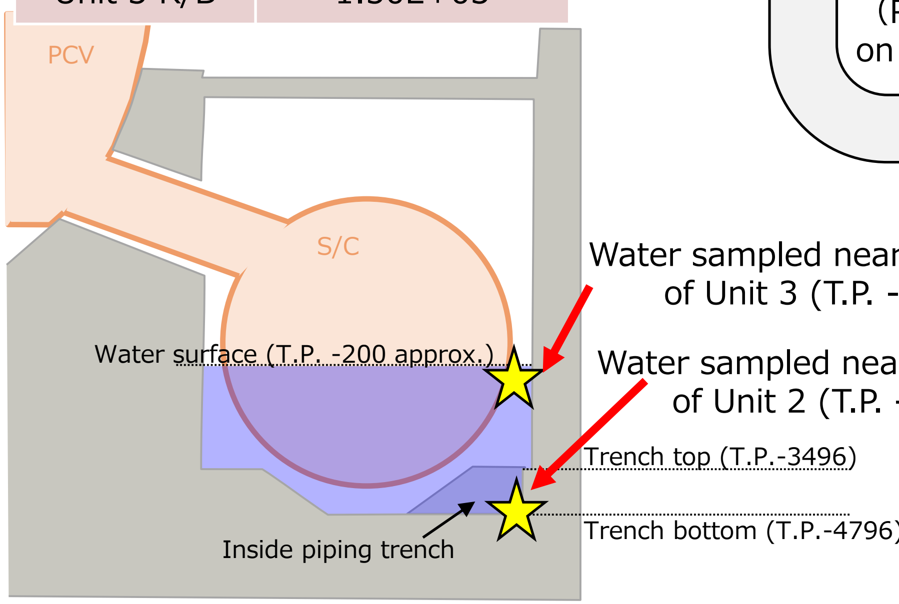






Element distribution obtained by SEM-WDS (Unit 2 residue)

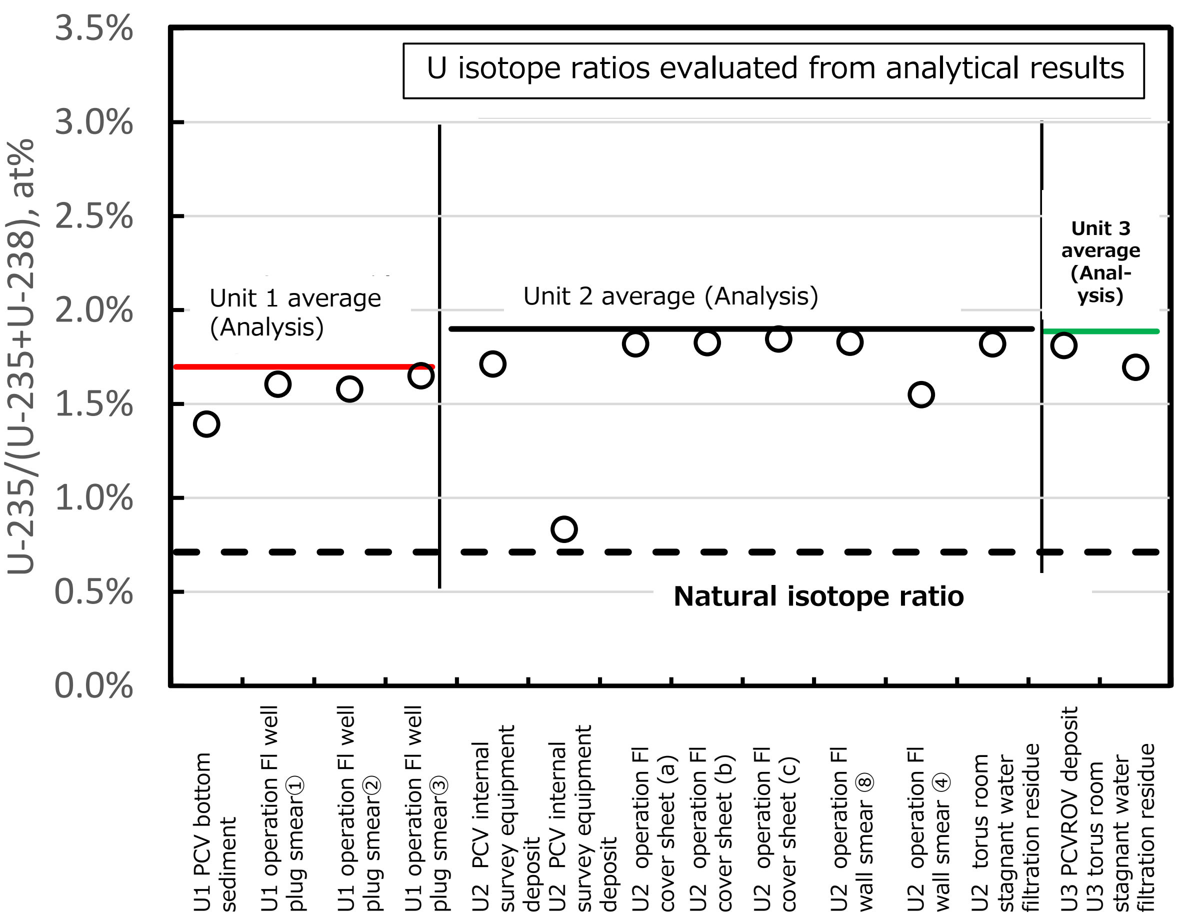
※Brightness of the characteristic X-ray image does not indicate the absolute value.



Cross section of R/B torus room (when sampled)

Evaluation of U isotope ratio distribution

- A distribution of U isotope ratios existed in the fuel prior to the accident as a result of the design enrichment (=U isotope ratio) distribution and the burnup during power operation.
- The U isotope ratio is a parameter important for criticality assessment and safe handling of fuel debris.
- Of the contaminated material samples collected from Units 1-3, those with high U concentrations found in SEM-EDS/WDS analysis were analyzed by ICP-MS to evaluate the U-235/total U ratio.



Many were found to be close to the core average calculated by the analysis code.

⇒It is considered that the U isotope ratios close to the core average calculated values are due to diffusion and melt mixing under high temperatures at the time of the accident. The reason for this is that the distribution range of U isotope ratios of the products is narrowed due to diffusion and melt mixing of materials under high temperatures during accidents.

(Some samples were found to have nearly natural isotope ratios.)

Evaluation of U-containing particle formation process

Aim to evaluate the formation process

- Obtaining knowledge on fuel debris properties and environmental information at the time of particle formation (H_2/H_2O ratio (molar ratio) in pressure vessel, temperature, etc.)

Classification by formation process of U-containing particles

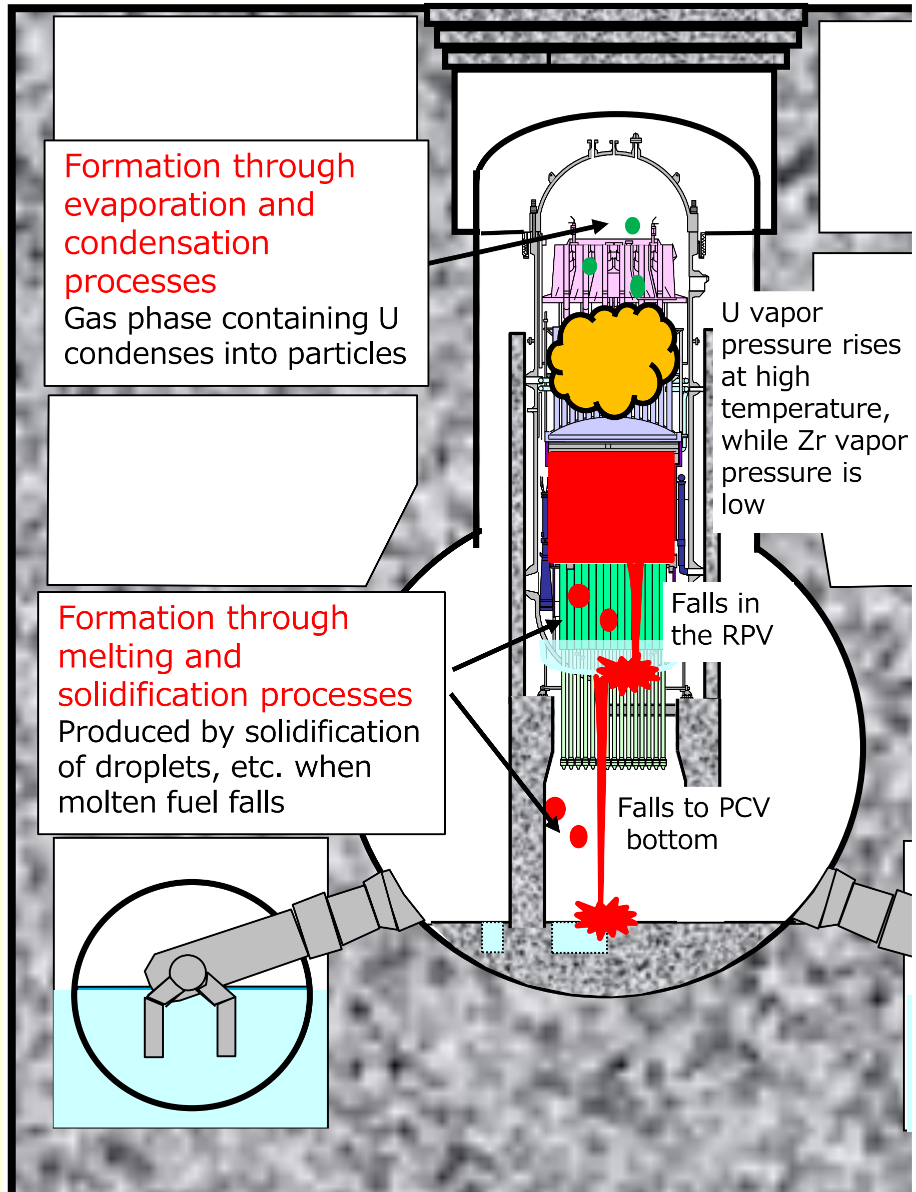
- When dissimilar materials react and melt, the composition of the constituent elements will be close to the average composition ratio of the material from which they originate, while when they evaporate and react in the gas phase, the vapor pressure difference is expected to result in a characteristic composition.
- Based on this, we classified particles into (1) and (2) below based on their composition (Zr content, etc.) and shape.

(1) Melting and solidification processes

- For particles containing Zr, we basically grouped them as particles formed by melting and solidification processes.
- The particles might have fracture surfaces associated with fracturing.

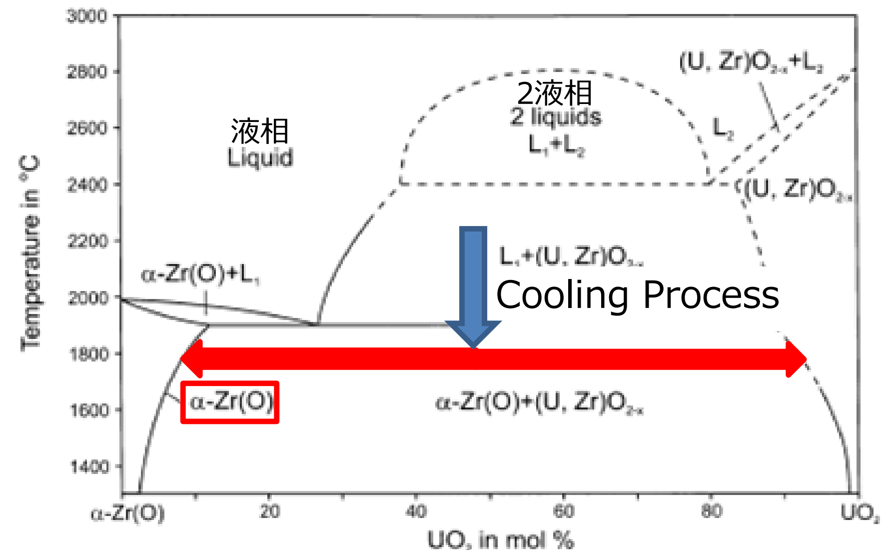
(2) Evaporation and condensation processes

- In the presence of sufficient water vapor, the vapor pressure of Zr oxides is low compared to values for other oxides.
- Microscopic particles might agglomerate and have a nearly spherical shape.
- Therefore, particles without Zr were classified as those formed by the evaporation and condensation processes.

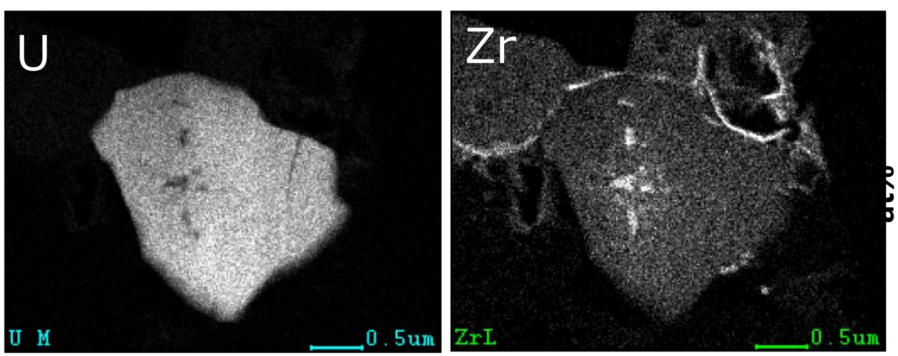


(1)-1 U-containing particles with α-Zr(O) phase (melting/solidification process)

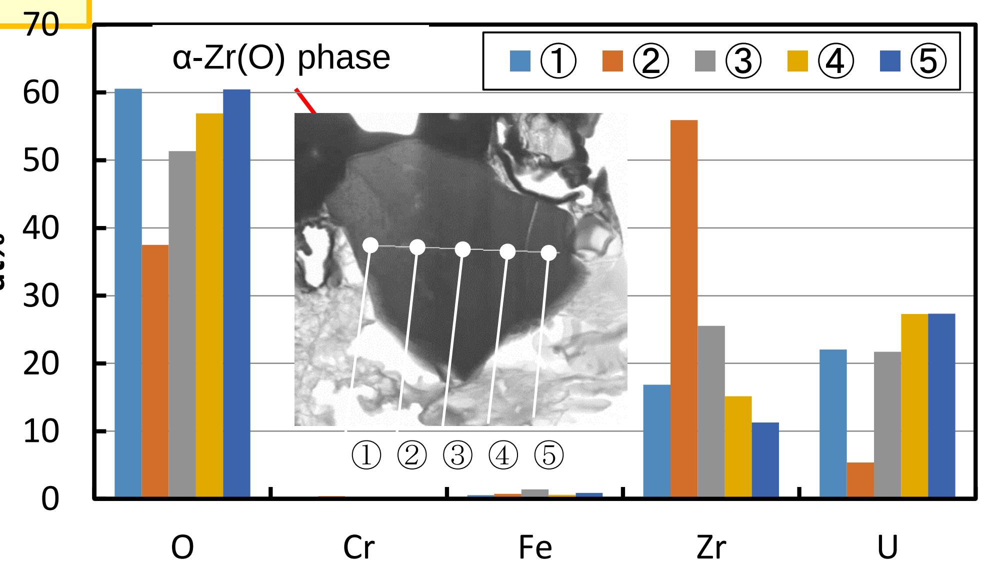
- One of the particles detected in the sediment at the Unit 1 PCV bottom was approximately 2μm in diameter (lower left figure).
- It contained Zr, and it was presumed to have been formed thorough the melting and solidification processes.
- A high-Zr region (analysis point ② in the lower right figure) was observed in the $(U,Zr)O_2$ matrix (analysis points ①, ③, ④, ⑤).
- Analysis point ② was thought to be α-Zr(O) phase separated from $(U,Zr)O_{2-x}$ during the cooling process (right figure).
- Lack of Zr oxidation suggested a reducing atmosphere (relatively hydrogen-rich situation) might have existed in the time before these particles solidified.



α-Zr(O)-UO_2 phase diagram
P. Hofmann (1999)



Element map obtained by TEM-EDS



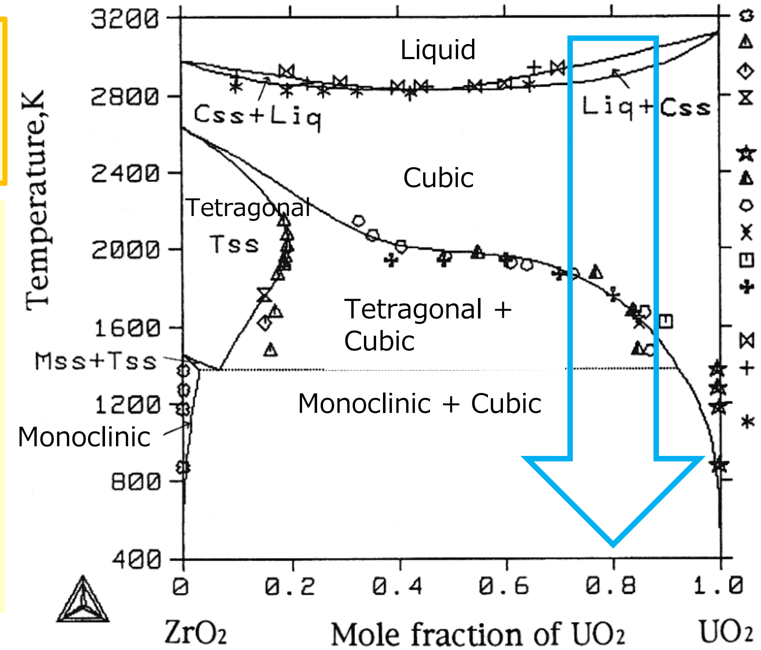
Element composition

(1)-2 Particles containing monoclinic ZrO_2 phase(melting and solidification process)

- In Unit 1, U-containing particles with monoclinic ZrO_2 phase were found.
- The matrix phase (① to ③) was cubic $(U,Fe,Cr)O_2$ and precipitate (④) monoclinic ZrO_2.

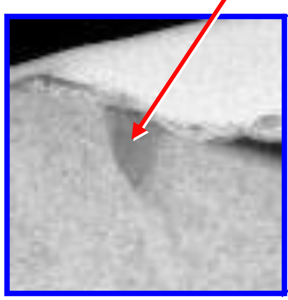
<Estimation of formation process>

- Particles contained Zr and were presumed to have been formed through the melting and solidification processes.
- The liquid phase of U-Zr-O was considered to have separated into cubic $(U,Zr)O_2$ and tetragonal $(Zr,U)O_2$ during cooling.
- Tetragonal crystal was transformed to monoclinic ZrO_2 (④).
- It was considered that the debris cooled slowly enough to cause separation, which might be related to the large enthalpy value of the fuel debris that fell in Unit 1 and the fact that water injection was not performed for a long time period.



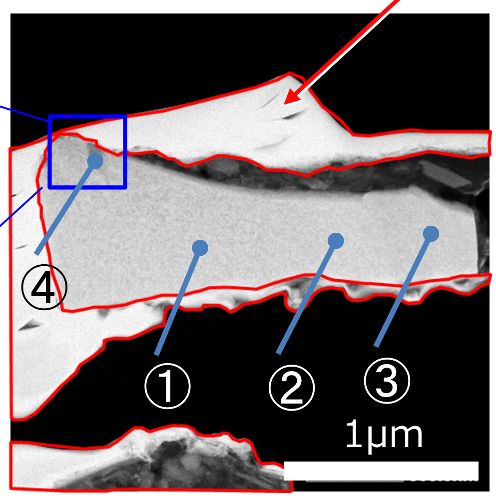
ZrO_2-UO_2 system phase diagram
M. Yamashita, et al. (1996)

Monoclinic ZrO_2



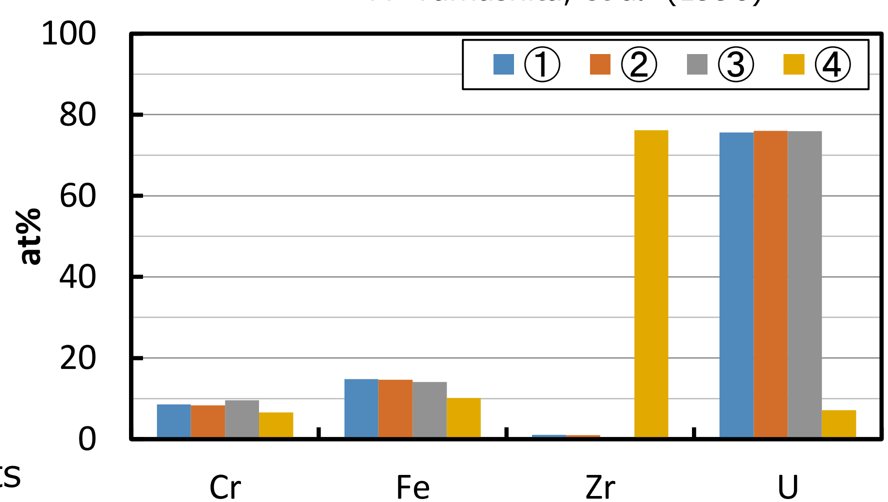
Magnified image of monoclinic ZrO_2 section

Protection film



TEM image and analysis points

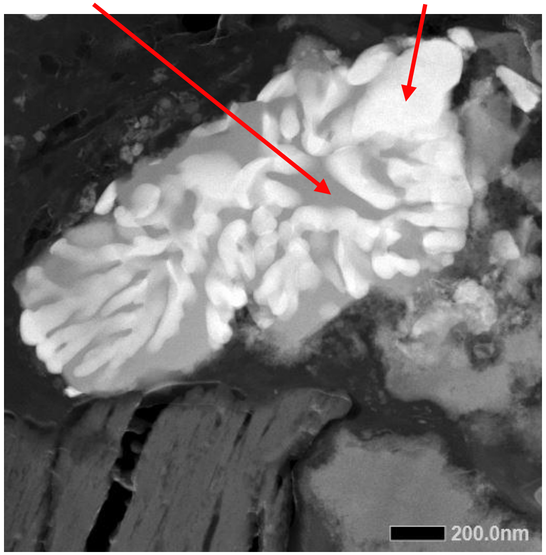
(Deposit at Unit 1 PCV penetration (X-2))



Element composition

Multi-constituent particles of fused fuel rod and steel constituents

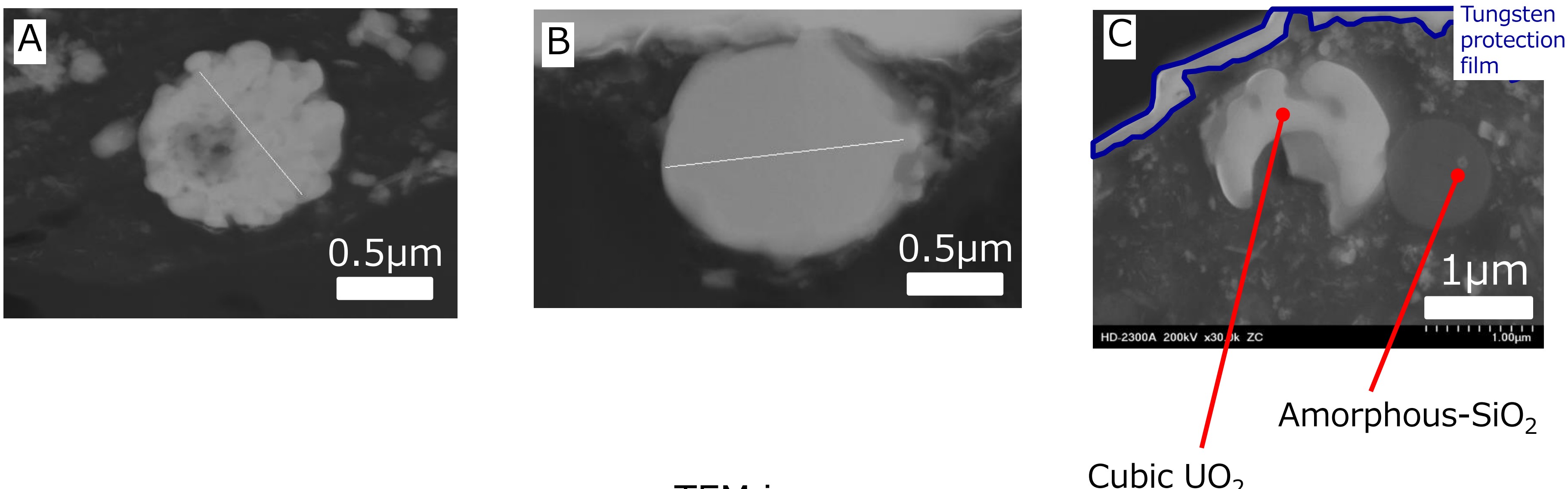
- On the Unit 2 operating floor, particles consisting of fused together steel constituents (Fe,Cr) and fuel rod constituents (U,Zr) were found; the presence of Zr suggested that they were formed through the melting and solidification processes.
- The particles consisted of a mixed phase of cubic $(U,Zr,Fe,Cr)O_2$ and $FeCr_2O_4$, and were considered as particles that had phase-separated during the cooling process of the U-Zr-Fe-Cr-O system melt.
- This was consistent with existing findings that the fuel reacted with steel to form debris.
- Since the precipitate size depends on the cooling rate, it may be useful for estimating the cooling rate.



TEM image
(Unit 2 operating floor cover sheet)

Particles formed by evaporation and condensation processes

- U-containing particles containing almost no Zr were detected in a sample taken at the Unit 2 operating floor. It was considered that they were formed by evaporation and condensation.
- It was considered that a part of the U component in the fuel evaporated and immediately became UO_2 and produced particles.
- Particle A looked like a secondary particle of agglomerated particles with a diameter of ~ 100nm; particle B had a dense spherical shape formed by crystal growth of agglomerated particles like A was.
- Particle C (cubic UO_2) was a particle considered to be in an intermediate state between A and B in terms of crystal growth, and its shape suggested that it might have been deposited on the surface of nearby spherical amorphous-SiO_2 (on the right in the TEM image), which was later separated.

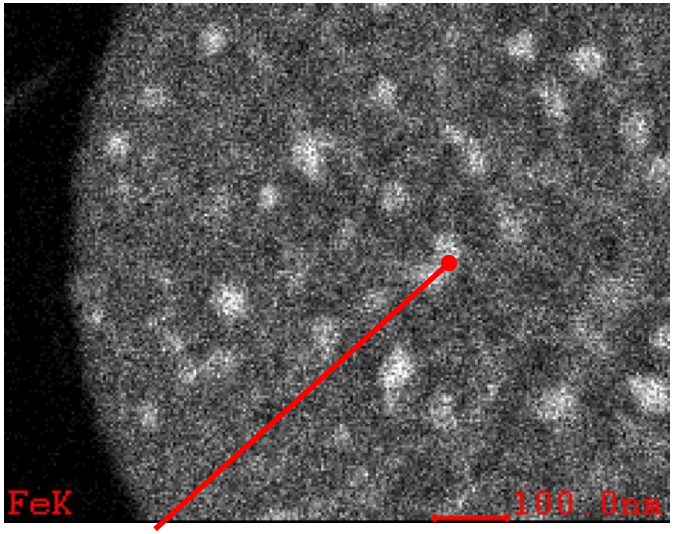


TEM image
(Unit 2 operating floor cover sheet)

Fine particles mainly composed of iron

- During the analysis of U-containing particles, one iron-based particle containing a small amount of U was found on the cover sheet of the Unit 2 operating floor.
- From the results of TEM-EDS and electron diffraction, pure Fe and Fe_3O_4 were found to be present together.
- Since the particle was spherical, it possibly formed from FeO in the liquid phase and separated into Fe and Fe_3O_4 during the cooling process.

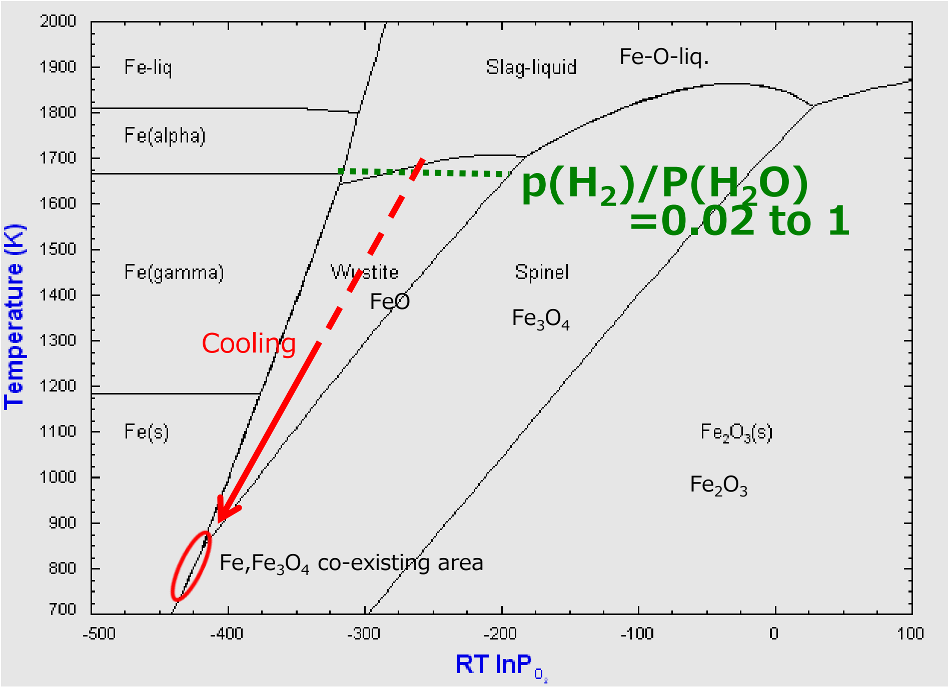
Crystal structure analysis revealed body-centered cubic (pure Fe) and spinel (Fe_3O_4)



Pure Fe fine particle

Fe distribution

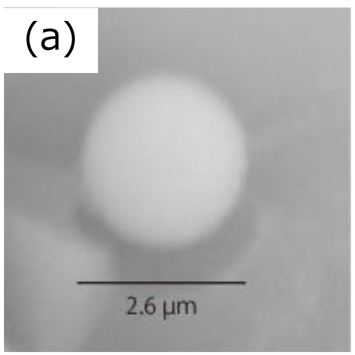
Coexistence of Fe and Fe_3O_4



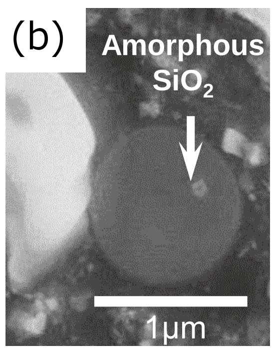
Oxygen potential (kJ/mol)

(2) Examination of insoluble Cs particles in the environment

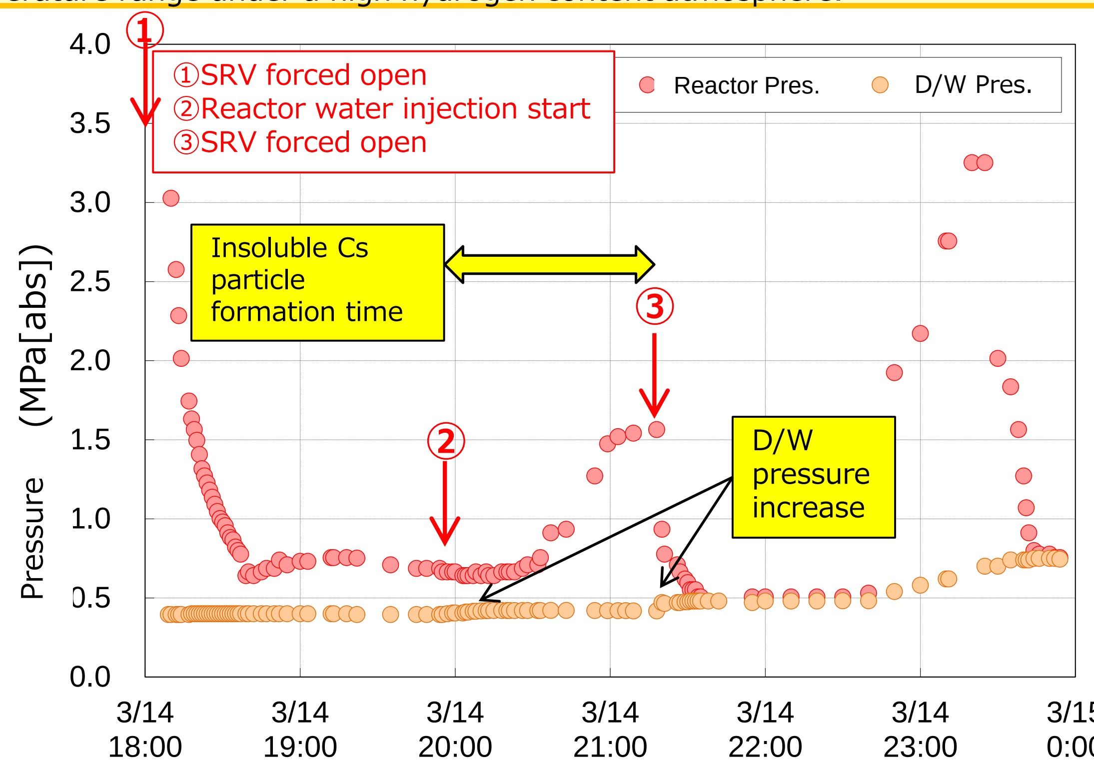
- Spherical insoluble Cs particles (Type A*) with a diameter of 1 to 10 μm containing radioactive Cs in an amorphous SiO_2 matrix were found in the environment, and we thought it would be useful to examine the accident progression by studying their formation process.
- They were presumed to have originated from Unit 2 based on the meteorological conditions at the time and the origin of the constituent elements.
- Based on the accident progression of Unit 2 (below right figure), it was considered that Si and Cs reacted in the RPV and formed particles, which migrated to the PCV when the PCV pressure was rising (marked by black arrows), and they were released into the environment through a rapid cooling process.
- Under high-temperature conditions by which Mo was released from the fuel, the formed SiO_2 particles would not contain Cs because of the high affinity between Cs and Mo.
- The fuel temperature at the time the particles migrated to the PCV might have been above the SiO_2 liquefaction temperature, above the Cs release temperature from the fuel, and below the Mo release temperature (depending on the atmosphere), i.e., in the 1500 to 2300°C temperature range under a high hydrogen content atmosphere.



Insoluble Cs particles (Type A)
Adachi et al. (2013)



Amorphous SiO_2 particles found on Unit 2 operating floor



Unit 2 Pressure indication

* Spherical insoluble Cs particles of 1 to 10 μm size with high specific activity were collected in the south and west areas of 1F

Summary

＜Findings on the state of fuel debris＞

- Most of the sources of α contamination in the stagnant water were present in particle form and more than 90% could be removed by filtration. U is chemically stable in the form of cubic UO_2 and has little possibility of aging.
- The results of the analysis of the U isotope ratio (U-235/total U) in the sample suggested that the fuel melting caused the U isotopes to mix.

＜Findings on accident progression (from evaluation of the formation process of radioactive particles)＞

- Results suggested that the chemical conditions (e.g., hydrogen/steam ratio) within the RPV and PCV changed with time and location.

- Particles that were thought to have been formed in a hydrogen-rich condition have been observed in Unit 1, and these particles may be related to insufficient water injection into the reactor in the early stages of the accident.
- In Unit 2, particles that were thought to have been formed under the steam-rich condition and particles that seemed to have formed under the hydrogen-rich condition were observed. Insoluble Cs particles were considered to have formed at the early stage of the fuel temperature rise, and this offers a clue to the atmosphere in the RPV at the time of formation.

Based on the knowledge obtained through sample analysis and evaluation, we will continue to estimate the state of the fuel debris and deepen our understanding of the accident progression.