Evaluation of the situation of cores and containment vessels of Fukushima Daiichi Nuclear Power Station Units-1 to 3 and examination into unsolved issues in the accident progression

— Progress Report No. 1 —

December 13, 2013
Tokyo Electric Power Co., Inc.

Preface

Based on surveys and analysis relating to the Fukushima Nuclear Power Station accident, TEPCO considers that many items pertaining to the causes and development of the accident are now clear.

At present, however, remaining records and on-site investigations are still limited, and there are still some aspects that remain unconfirmed or unexplained with regard to the locations, the extent and the causes of the damage to the Fukushima Daiichi Nuclear Power Station, arising from the development of the accidents that followed the Tohoku Region Pacific Coast Earthquake.

As the main party responsible for the Fukushima Nuclear Power Station accident, TEPCO will continue to conduct systematic on-site surveys and simulation analysis aimed at gaining a clear understanding of all aspects of the behavior of the nuclear reactors during the accident. We consider that this will prove useful in fulfilling our obligation as the operator of the Nuclear Power Station to improve safety, and to aid the work of decommissioning the plant.

As the first progress report, this report focuses on the unconfirmed and unexplained issues from immediately after the accident through to the end of March 2011.

Contents

Report on the survey and study results of unconfirmed and unexplained events of the Fukushima Nuclear Power Station accident

Overview

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1. Objective of survey and study of unconfirmed and unexplained events

Explaining what actually happened in the Fukushima Daiichi Nuclear Power Station accident will help improve the safety of power generating facilities in Japan and the rest of the world

As the operator of the nuclear power station and the main party responsible for the accident, we are fully committed to clarifying all aspects of the accident

Solving reactor decommissioning issues and accumulating information

Improvement in safety measures and heightened safety at Kashiwazaki-Kariwa Nuclear Power Station

2. Overview (approach to unconfirmed and unexplained events)

From a broad range of perspectives
Assigning unconfirmed and unexplained events to the two categories below

(Target period: To the end of March 2011)

Understanding the status of the reactor cores and containment vessels and the main flow of accident development	Accumulating the information needed for detailed understanding and assessment of the development of the accident
Example 1) Cause of loss of cooling system function in isolating the reactors Example 2) Details that observations during the accident cannot fully explain and issues we cannot explain	Example) Confirming the state of the residual heat removal system in unit 2 before and after the tsunami [Reason] We need to confirm the cooling status of this system, and this may also help to improve safety, such as by providing knowledge that will help to prevent accidents.

Note: Issues regarding the emission of radioactive material outside the plant have been detailed in the report, “Radioactive material released into the atmosphere in the Fukushima Daiichi Nuclear Power Station accident”, published in May 2012. This report will mainly focus on explaining the development of the accident.

2. Overview (organizing, extracting issues and approach to studies)

Information gleaned as far as possible from the field, records, interviews and information gathering will be used to establish the facts of the accident in conjunction with analytical methods

↓

Compiled and published in the TEPCO Fukushima nuclear accident investigation report (December 2011/June 2012)
Further surveys will be made

↓

52 instances of unconfirmed and unexplained events were extracted

↓

Events we have studied

- Re-investigation of events where our accident survey findings differ from those of others (2 instances)
- **10 instances that are almost completed** *

Content of this report

→ Continued study 2nd time —

Events whose study is not completed

- 10 issues with the highest priority for future consideration
- 34 issues that are not such a high priority

→ These issues will be studied with outside researchers in a flexible structure

→ Explanations will be found starting from high priority issues

We intend to reach conclusions regarding high-priority issues within 2 years

* 2 events that are fundamental to conclusions in this report

2. Overview (overview of first progress report)

Of the 10 issues for which studies were almost fully completed in the first survey results, 5 issues are considered to be key to understanding the accident. The 5 issues are summarized below.

● Was it not the earthquake that caused the loss of the "Cooling" function?

* Data recorded by wave height meter records and other instruments and photograph sequences of the incoming tsunami have made it clear that the loss of the seawater pump and the emergency generator functions were caused by the tsunami.

● Was it not the earthquake that caused water to leak from important equipment, resulting in the inflow of water into nuclear reactor building, unit 1?

* Drawing surveys, eyewitness accounts, plant data and other information have made it clear that water flowing into the 4th floor of the nuclear reactor building, unit 1 flowed in via a duct from the spent fuel pool.

● Why was it that water injected from fire trucks failed to sufficiently cool the reactor?

* Piping drawings allowed us to confirm that some of the water injected from the fire trucks into the reactor may have flowed into other systems; however, the actual amount of water injected and its impact on the development of the accident will be the subject of a future study.

● Has not the time of the manual stoppage of the High Pressure Coolant Injection (HPCI) System in nuclear reactor unit 3 been properly compared with internal reactor data?

* Confirmation of data trends has confirmed that the water injection system in nuclear reactor unit 3 may not have been able to provide enough water prior to the time it was manually stopped. We plan to reassess the development of reactor core damage.

● Was not the sudden loss of reactor pressure in nuclear reactor unit 3 caused by a hole in important equipment?

* Checking of the startup conditions confirmed that several safety relief valves may, in fact, have opened automatically. The opening of the valves may have caused the sudden decrease of pressure.

* Refer to the attached information for other study results and for details of the studies

Report on the survey and study results of unconfirmed and unexplained events of the Fukushima Nuclear Power Station accident

Main Report

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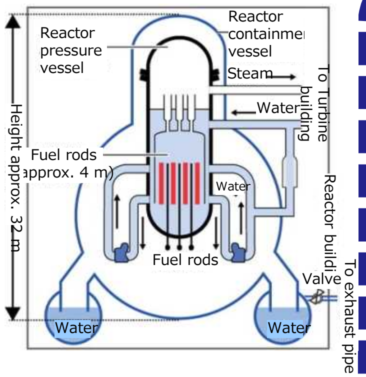
1. Approach to unconfirmed and unexplained events

(1) Extraction range for unconfirmed and unexplained events

The designated ranges are from ① to ⑤ below (primarily for actual correlations) and from ① to ⑨ (primarily for the development of the accident, the damage processes, etc.)

Time course

Pre-accident status

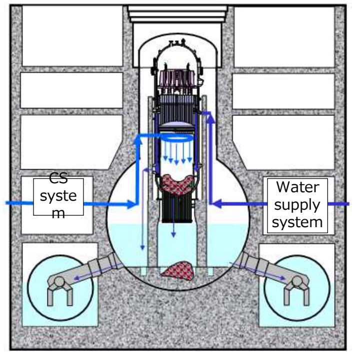


11 March 2011

① Earthquake
② Tsunami
③ Equipment damage
④ Control by operators
⑤ Equipment operation
⑥ Reactor behavior
⑦ Reactor damage
⑧ Release of radioactive materials
⑨ Containment vessel damage

End of March 2011

Post-accident status



Range ①	Because the views of TEPCO and those expressed in external accident investigation reports differ in part, surveys are ongoing.
Range ②	Studies will be conducted aimed at organizing, extracting and explaining unexplained issues, such as the detailed behavior of the equipment in steps ⑤ and ⑥ of the accident development, the reactor and containment vessel damage processes in steps ⑦ to ⑨, and their post-accident statuses.

1. Approach to unconfirmed and unexplained events

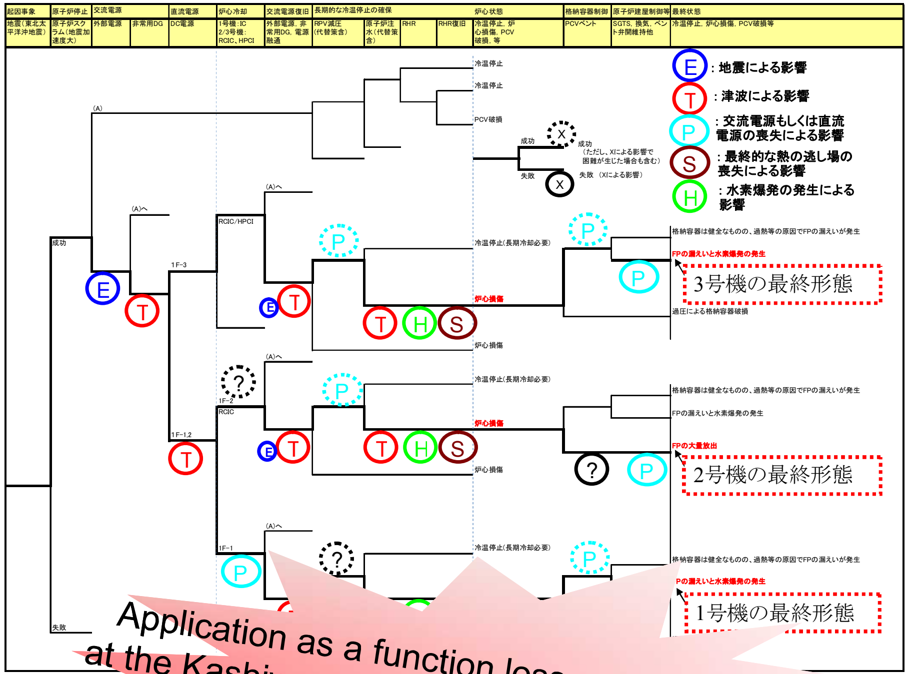
(2) Explained events

- **Analysis of the processes starting from just after the earthquake through to the most severe events using an "event tree". Organization of the characteristics of accident development from Unit 1 to Unit 3.**
- **Identifying the factors and causes leading to the loss of function in the safety equipment, and the application of measures to improve safety**

[Example of safety improvement measures using event tree analysis]

Understanding the causes of function loss	**Loss of DC power due to the tsunami** Equipment control/measurement became impossible
Function loss prevention measure	**Install watertight doors** in areas with DC power installed
Alternative security measure	Install backup batteries and dedicated chargers in **elevated locations**
Safety assurance	**Maintain DC power during accidents**

[Event tree image]



Application as a function loss countermeasure at the Kashiwazaki-Kariwa Nuclear Power Station

[What is event tree analysis?] A safety assessment method that analyzes the processes from the initial event that triggers the accident through to the final status by developing a branching structure (tree). The tree branches at each stage based on criteria such as whether safety equipment functions, allowing the characteristics of the accident to be organized.

1. Approach to unconfirmed and unexplained events

(3) Classifying and organizing the extraction of unconfirmed and unexplained events

The organization and extraction of unconfirmed and unexplained issues with the aim of fully clarifying the development of the accident, including events that are difficult to explain, events that unfolded over a long period, and events that require wide-ranging argument within the scientific community.

-- Classifying and organizing the extraction of unconfirmed and unexplained events --

- Points raised by experts, such as the report from the Investigation Commission appointed by Diet
- The mechanisms of the earthquake and tsunami, and their effects
- Equipment operating states and details of the reasons why they stopped
- Points that cannot be fully explained regarding the observed facts during the accident
- Core damage processes and the radioactive material release processes

Understanding the state inside the core and containment vessels

-- Application of the study findings --

Improved equipment reliability and operating procedures

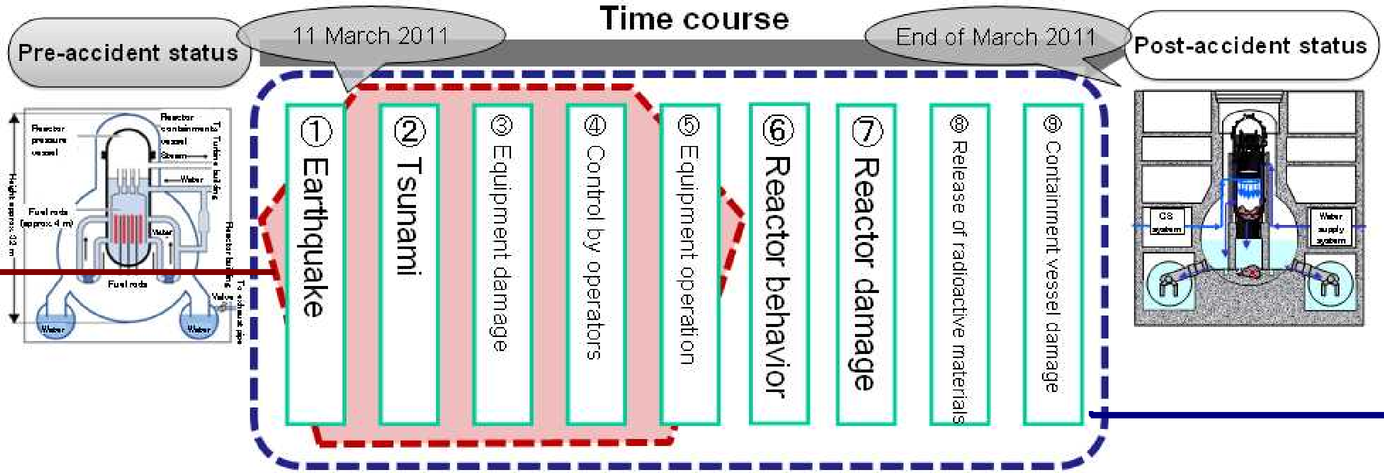
Useful in further improving safety and accurately assessing accident development behavior

More accurate estimation of the state inside the core and containment vessels

Useful in formulating efficient decommissioning policy

2. Unconfirmed and unexplained events for which studies have been completed

At present, of the 52 extracted events, explanations are almost completed for 10. Of those, 5 that are key to understanding the accident are outlined below.



(1) Primary causes for the loss of cooling function

(2) Causes of the flooding of the Unit 1 reactor building (4th floor)

(3) Reasons why the water from fire trucks failed to sufficiently cool the reactor

(4) The possibility that the water inflow to the reactor was already insufficient even before the Unit 3 high-pressure water injection system was manually stopped

(5) Reasons why the reactor pressure in Unit 3 suddenly decreased.

3. Typical details of the studies into unconfirmed and unexplained events

(1) Primary causes for the loss of cooling function

Verification of the high likelihood that the loss of emergency cooling and emergency generator function was caused by the tsunami rather than the earthquake

-- Understanding and verifying the facts regarding the arrival of the tsunami at the site -

Using wave height meter at the Fukushima Daiichi nuclear power plant
to understand the circumstances of the tsunami impact **1**

Using consecutive photography of the scene of the tsunami impact
to analyze the timeline of the tsunami arrival **2**

Analysis of the timing of the consecutive photography of the scene of the tsunami impact **3**

-- Key study points aimed at verification and explanation --

The time difference between the arrival of the tsunami at the site and the loss of emergency generation function (the events are simultaneous) **4**

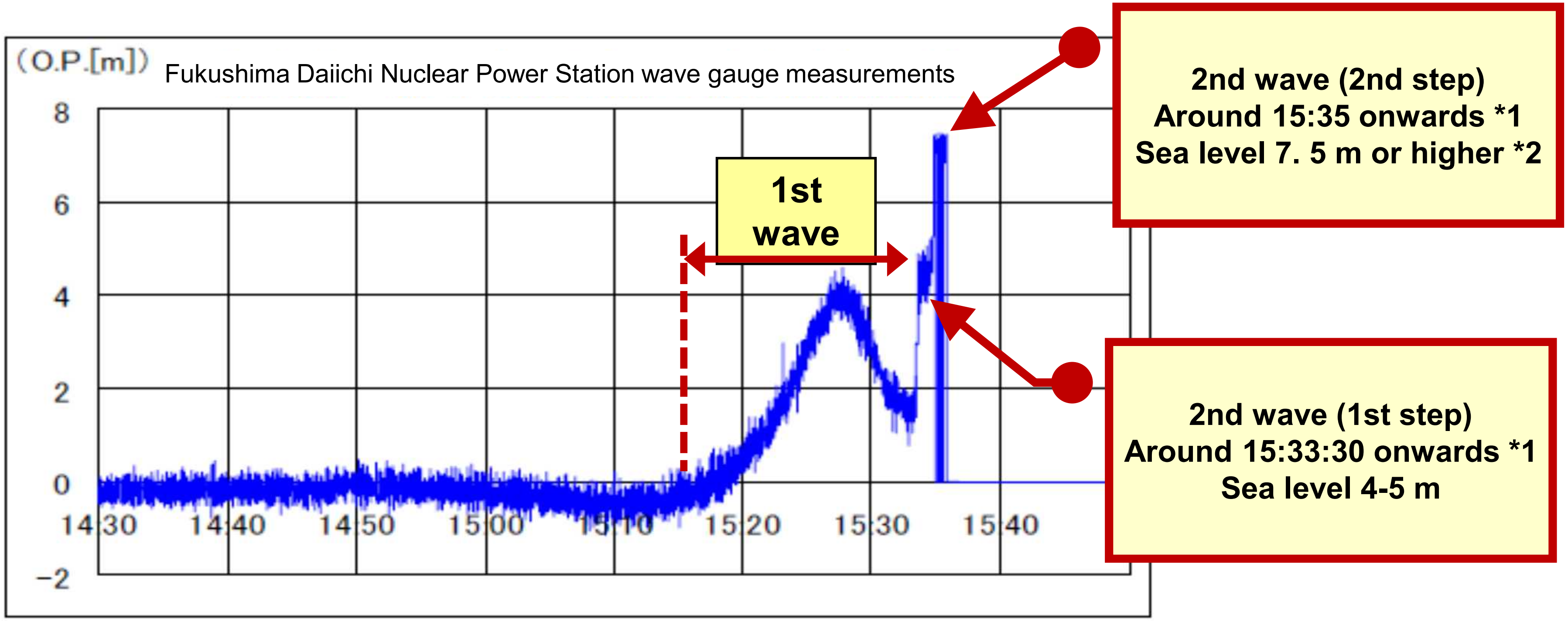
The sequence of function loss in equipment within the grounds of the Fukushima Daiichi Nuclear Power Station (function lost sequentially starting from the ocean side) **5**

(1) Primary causes for the loss of cooling function

1

Using wave height meter **to understand the circumstances of the tsunami impact**

From the recordings made using wave height meter of the tsunami that struck the Fukushima Daiichi Nuclear Power Station, **observations show that the biggest tsunami were made up of the 1st wave & the 2nd wave (1st step), and the 2nd wave (2nd step)**



*1: A problem occurred at around 15:36, resulting in a measured sea level of zero

*2: The observation range of the wave height meter is up to 7.5 m, so levels above 7.5 m are not recorded.

[What is a wave gauge?] An instrument that measures the height of waves close to the shore. At the Fukushima Daiichi Nuclear Power Station, an ultrasonic wave gauge is installed at a location roughly 1300m away from the site shoreline.

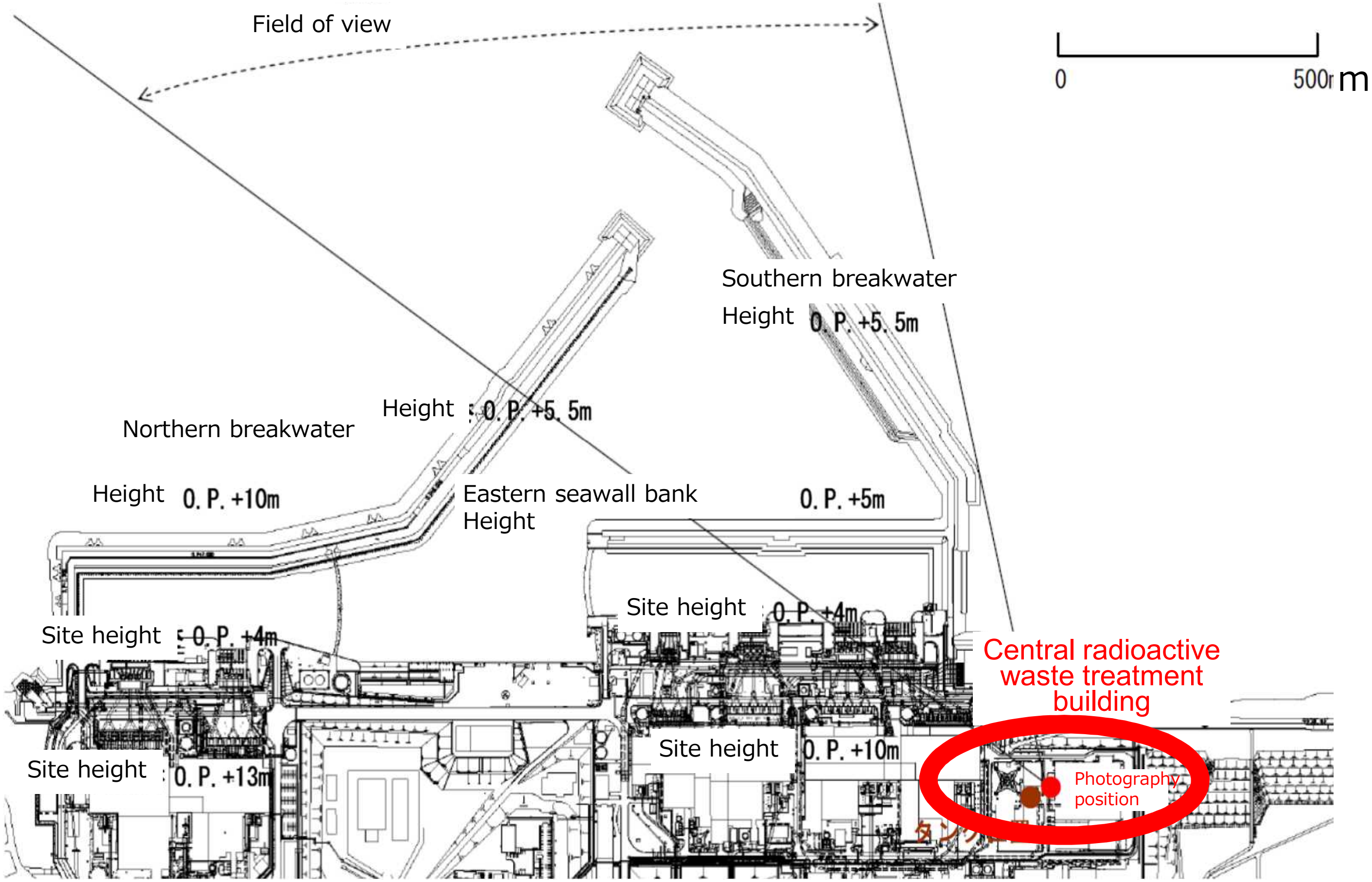
(1) Primary causes for the loss of cooling function

2

Using consecutive photography of the scene of the tsunami impact
to analyze the timeline of the tsunami arrival

Analysis of the first 19 of 44 photographs shot from the central waste treatment building

Wave 2, 1st step: At 15:35:40, the wave comes ashore in the site at 4 m above sea level (4 m level); Wave 2, 2nd step: From 15:36:30 to 15:37, the wave is estimated to be flowing into the site at 10 m above sea level (10 m level).



Around 15:34:56

Around 15:35:40

Around 15:36:46

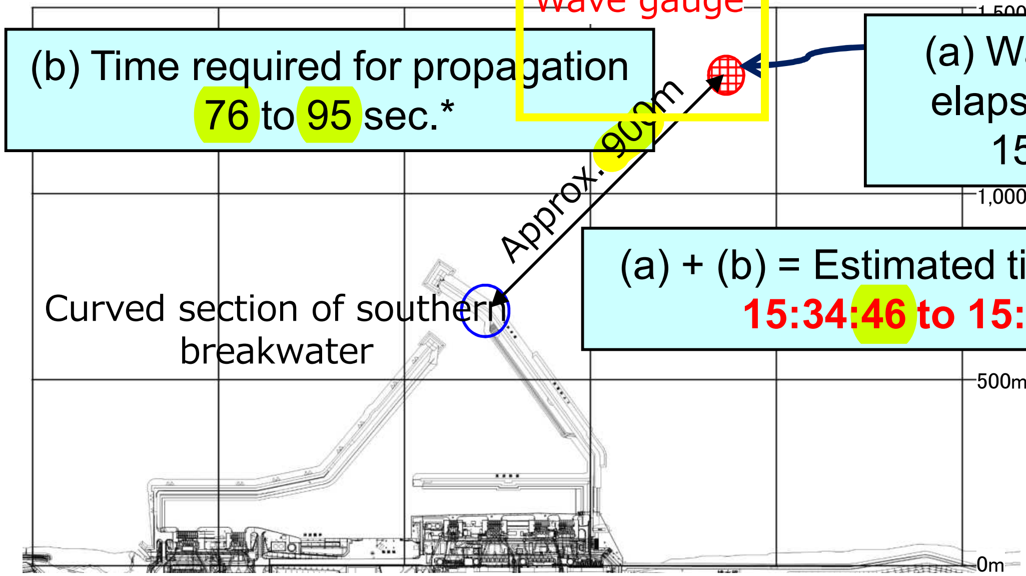
(1) Primary causes for the loss of cooling function

3

Analysis of the timing of the consecutive photography of the scene of the tsunami impact

We analyzed the time when tsunami wave 2 (1st step) struck and estimated that the camera's internal clock was fast by between 6 m 31 sec. to 6 m 50 sec. (Hereafter, photography times have been adjusted assuming a median value of 6 m 40 sec.)

The wave arrival times estimated here were obtained using an appropriate method and there are no significant discrepancies.



Wave gauge

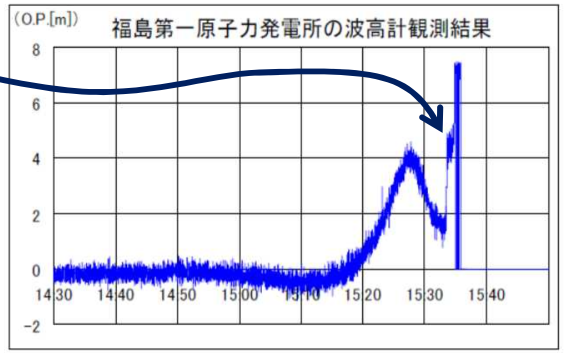
(b) Time required for propagation 76 to 95 sec.*

Approx. 900m

Curved section of southern breakwater

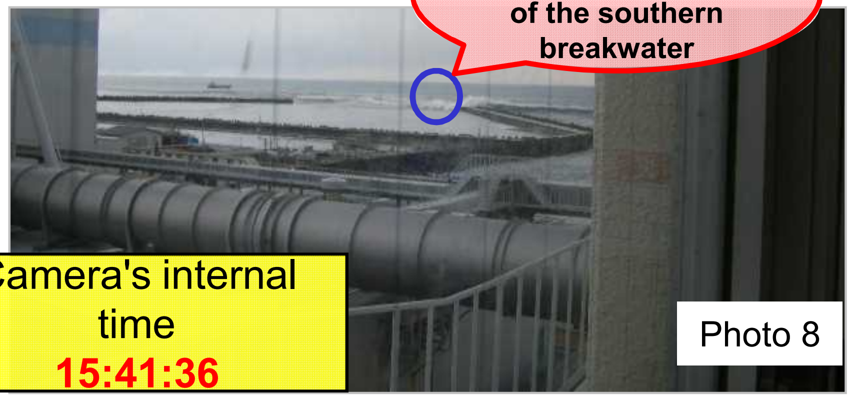
(a) Wave gauge elapsed time of 15:33:30

(a) + (b) = Estimated time of arrival **15:34:46 to 15:35:05**



Comparison

The wave arrives at the curved section of the southern breakwater



Camera's internal time **15:41:36**

Photo 8

*: Calculated using the estimated wave velocity based on total water depth and still water depth

We analyzed the time when the 2nd wave (1st step) of the tsunami reached the curved section of southern breakwater and compared it with the photography time (internal clock) for photo 8 (also shown on previous page).

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(1) Primary causes for the loss of cooling function

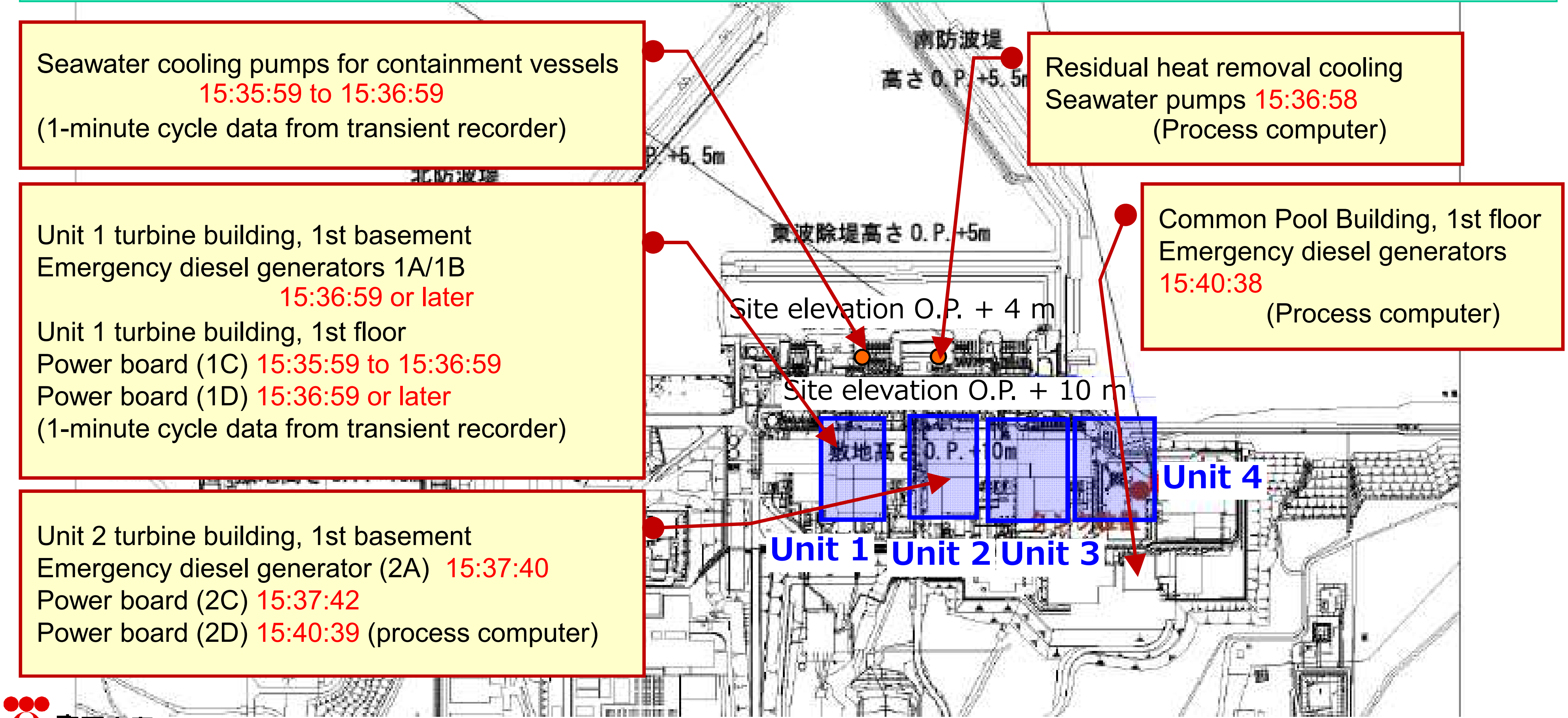
4

The time difference between the arrival of the tsunami at the site and the loss of emergency generation function (the events are simultaneous)

Using computer records to estimate the time when the emergency generator, pump and power board functions were lost

Measured data shows that function of the ocean-side pumps ① was lost due to the arrival of the tsunami around 15:36.

Function loss in the power boards ③ and emergency diesel generators ②, which were in a more elevated location, is estimated as being after ① .

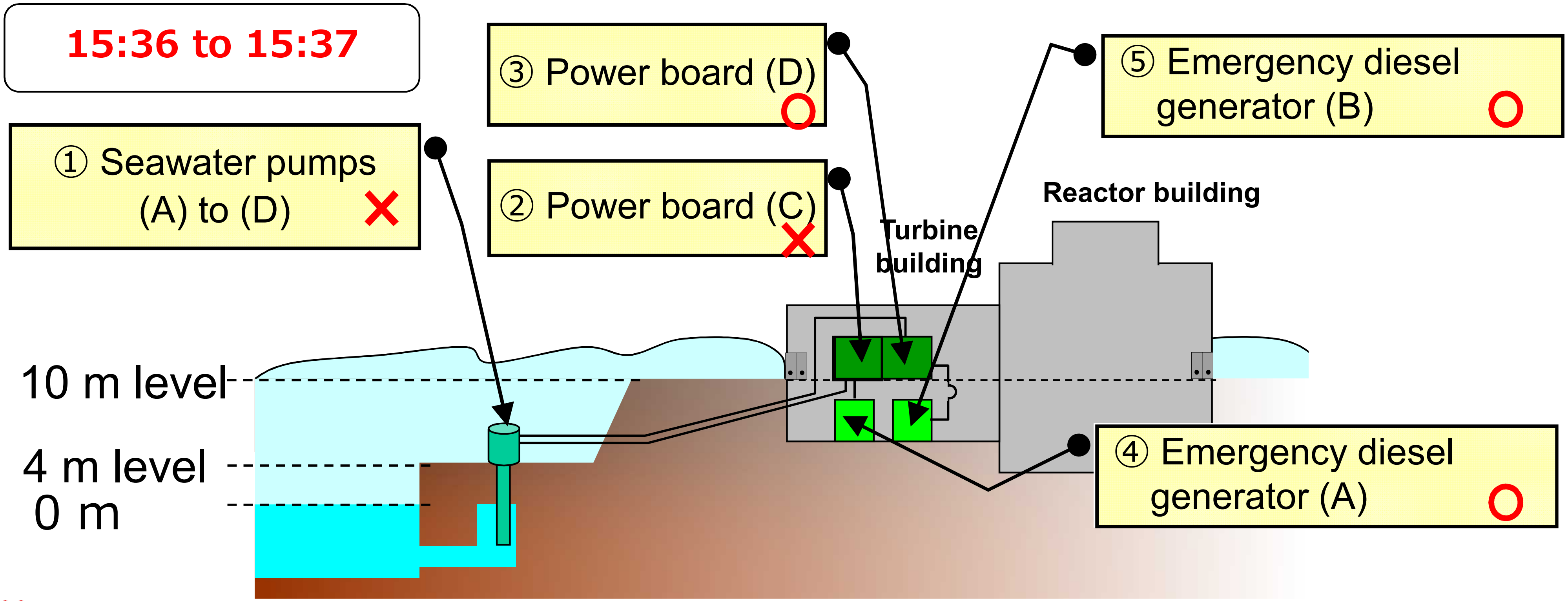


(1) Primary causes for the loss of cooling function

5

The sequence of function loss in equipment within the site
(function lost sequentially starting from the ocean side)

From data gleaned from process computers and transient recorders, it is estimated that **the effects of the tsunami progressed sequentially starting from the ocean side, as follows:**
[15:36] Seawater pump function lost → Bus voltage (C) function lost →
[Subsequently] function lost in bus voltage (D) and emergency diesel generators (A) and (B).



(2) The possibility that the flooding in the Unit 1 reactor building might be water leakage from important equipment due to the earthquake

The Investigation Commission appointed by Diet indicated the possibility of water leakage from inside the reactor, but...

with regard to the causes of flooding on the 4th floor of the Unit 1 reactor building when the earthquake struck, **it is highly likely that water in the spent fuel pool entered the air-conditioning ducts due to agitation during the earthquake, and the flooding was from the overflow prevention chamber**

-- Key study points aimed at verification and explanation --

Eye-witness testimony from site workers (2)
・ Estimation of flooding locations based on images

1

By studying drawings,
understanding which equipment/facilities may have been flooded

2

From the site survey conducted on 30 November 2012,
confirmation of the equipment/facilities that may have been flooded

3

By checking water levels in the isolation condenser (IC) heat exchanger,
confirming that there is no damage consistent with water discharge from inside

4

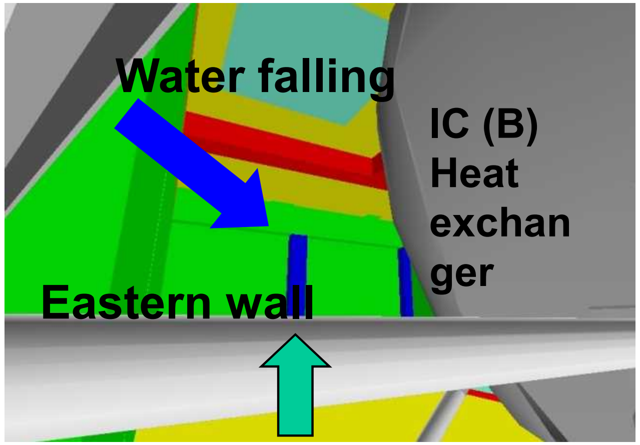
(2) The possibility that the flooding in the Unit 1 reactor building might be water leakage from important equipment due to the earthquake

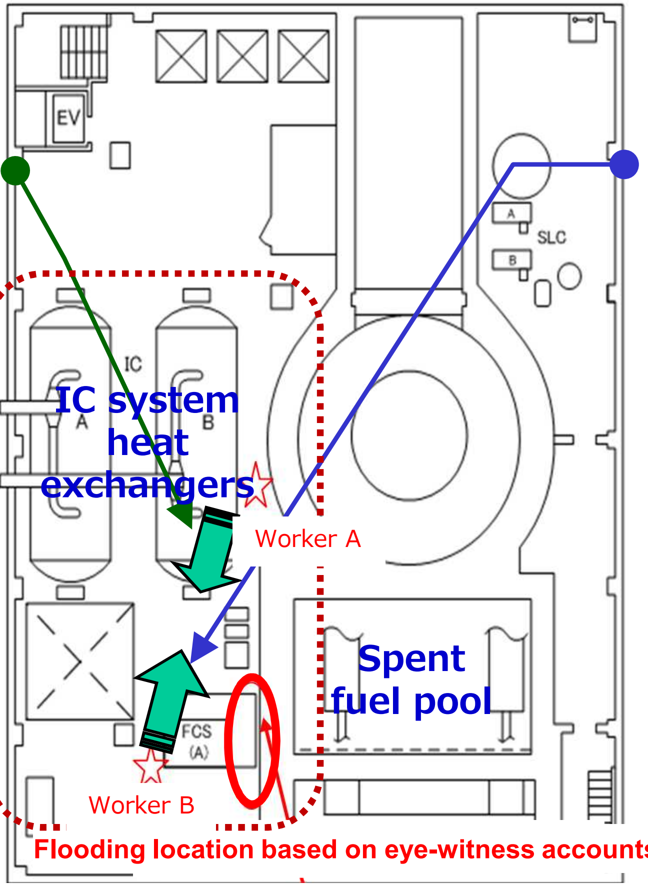
Eye-witness testimony from site workers (2) ・ Estimation of flooding locations based on images

1

Worker A

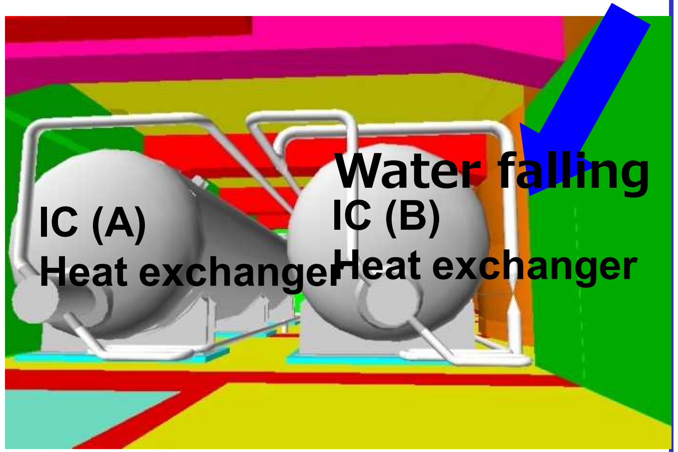
The water came out in a sudden rush from above at an angle of about 45°.





Worker B

Flooding from above on the right just shooting down like a solid wall



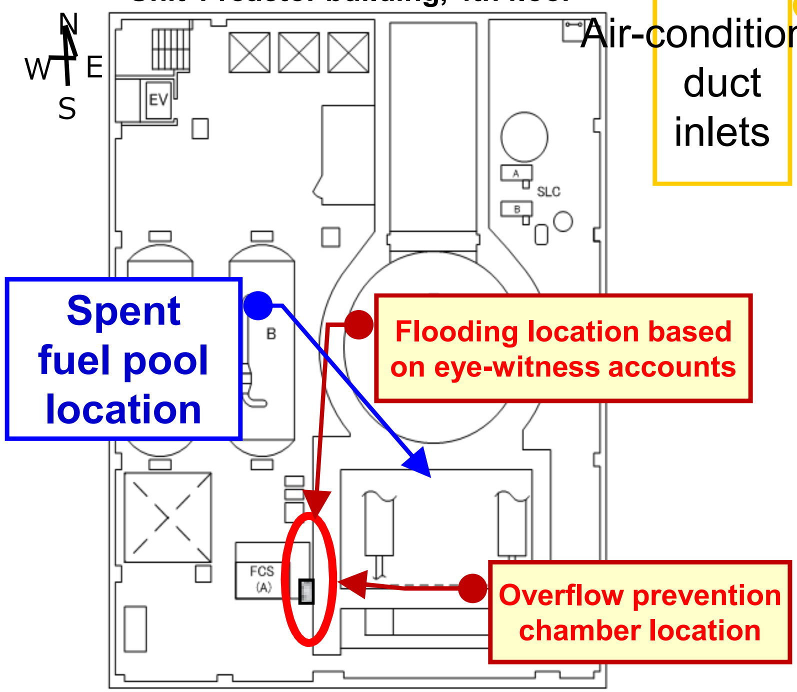
By matching the eye-witness accounts from both workers, we can estimate that the flooding locations were in the "overflow prevention chamber" shown above.

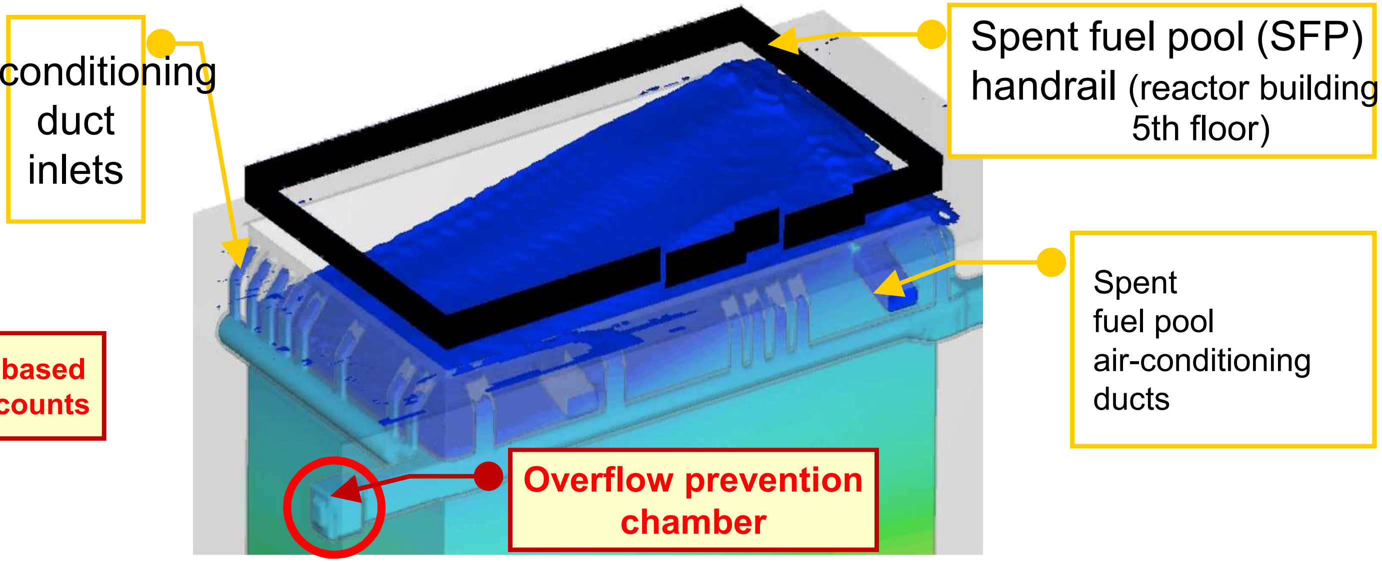
(2) The possibility that the flooding in the Unit 1 reactor building might be water leakage from important equipment due to the earthquake

2

By studying drawings,
understanding which equipment/facilities may have been flooded

Unit 1 reactor building, 4th floor





What is the "overflow prevention chamber"?
The overflow prevention chamber is installed to prevent water overflowing from the spent fuel pool (SFP) from getting through the air-conditioning ducts and leaking out of the radiation controlled area by holding and then draining the overflow. To ensure that there is no risk at all of leakage into the air-conditioning ducts, the chamber was closed by a diaphragm to isolate the overflow prevention chamber from the duct downstream.

- **Studying the drawings made it clear that, apart from this overflow prevention chamber, there is no other equipment near the flooding location that could cause an overflow that accords with the eye-witness accounts (see previous page).**
- **The "Committee on Accident Analysis" of the Nuclear Regulation Authority (NRA) also studied the possibility of flooding from this chamber and concluded that it was highly likely that the flooding was from that location.**

(2) The possibility that the flooding in the Unit 1 reactor building might be water leakage from important equipment due to the earthquake

3

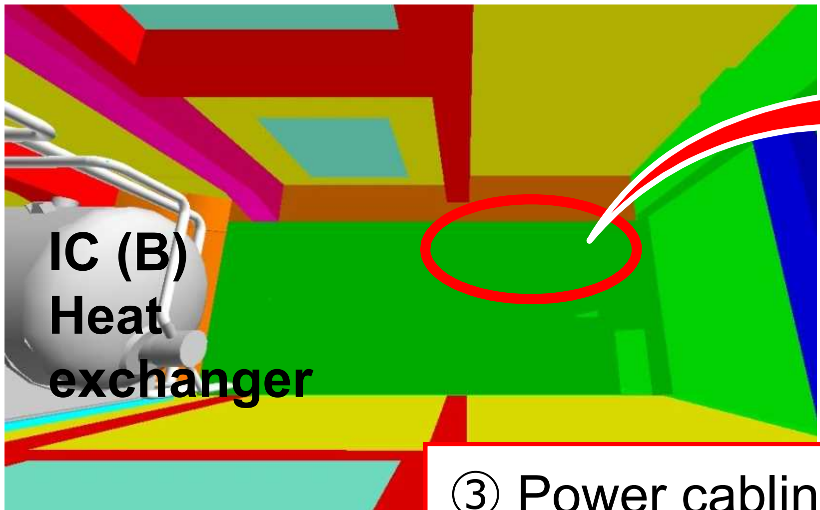
From the site survey conducted on 30 November 2012,

confirmation of the equipment/facilities that may have been flooded

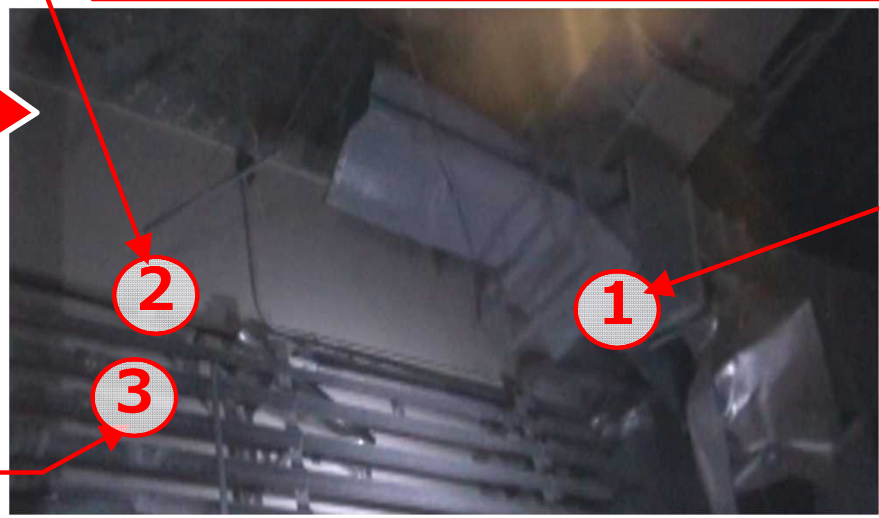
- **Site surveys confirmed that, of the equipment and tubing in the vicinity, water could only have leaked from the overflow prevention chamber.**
- **Similarly, site surveys included visual confirmation of deformation of the chamber itself and deformation or openings in the shutoff plates.**

② Isolation condenser vent lines
(lines that return steam to the main steam piping from the primary isolation condenser piping)
: Small-diameter (3-4 inch) pipes that carry high-temperature steam which, even if damaged in any way, would be extremely unlikely to cause flooding like a wall of water

① Air-conditioning ducts and overflow prevention chamber
: Connected to the surface of the spent fuel pool wall
Pool water can flow into the chamber during an earthquake



③ Power cabling



Note: Photograph shown with enhanced brightness and contrast.

(2) The possibility that the flooding in the Unit 1 reactor building might be water leakage from important equipment due to the earthquake

4

By checking water levels in the isolation condenser (IC) heat exchanger,
confirming that there is no damage consistent with water discharge from inside important equipment near the site

- Because there is no confirmation of steam at the site and given that the amount of water remaining in the IC tank can be confirmed (65% in A and 85% in B), **it is considered unlikely that any damage occurred that would cause an outflow of water inside important equipment** near the site.

Hypothesis ①: The primary piping in the IC system or the vapor phase side of the IC tank was damaged

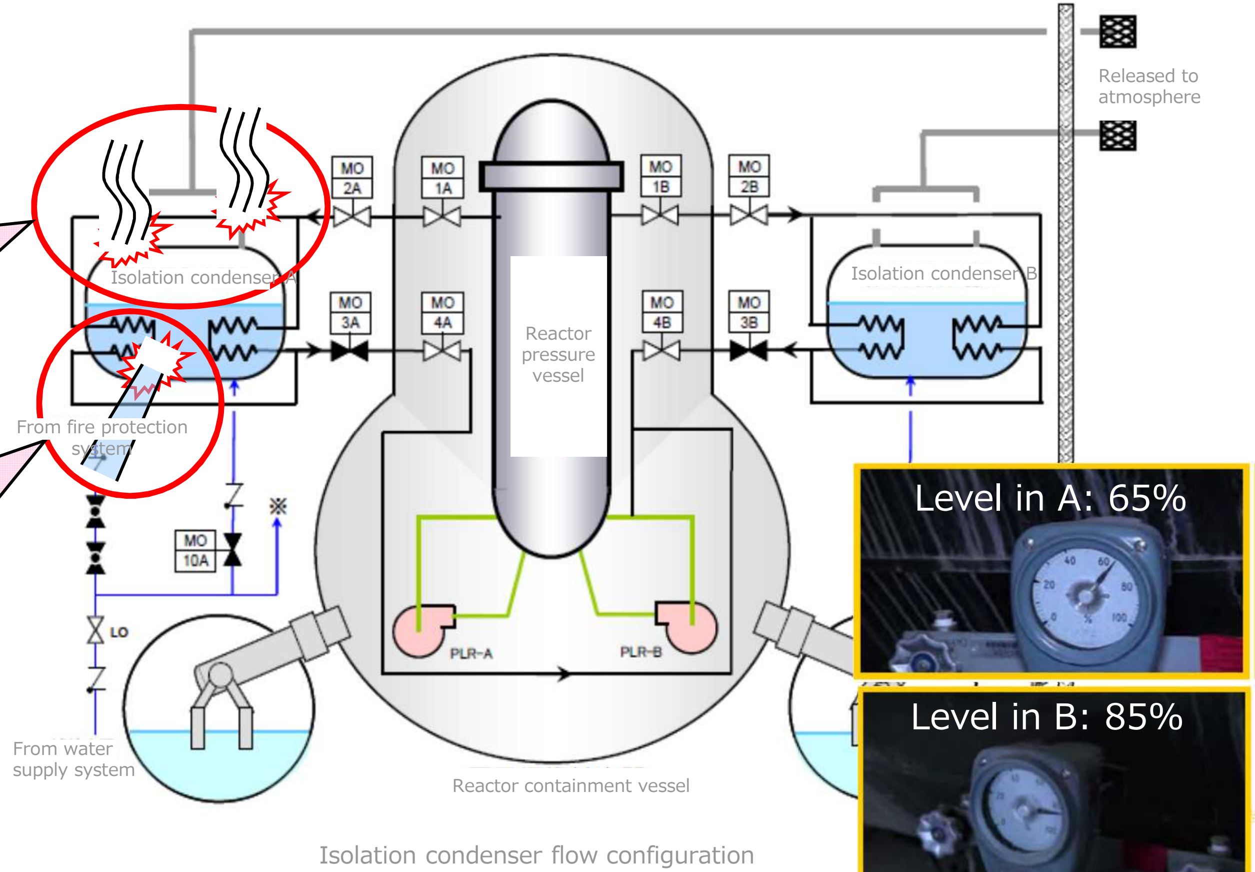
"Steam, not water, is generated" but
There was no confirmation of steam at the site.

~~Steam, not water, is generated~~

Hypothesis ②: The vapor phase side of the IC tank was damaged

"Large amounts of retained water are released" but
The site water level was checked on 18 October 2011
A: 65%, B: 85%
(A was used after the earthquake, so the level is considered to have dropped.)

~~Large amounts of retained water were released~~



Isolation condenser flow configuration

3. Typical details of the studies into unconfirmed and unexplained events

(3) Reasons why the reactor was not sufficiently cooled, despite water being injected in from fire trucks

If all the water injected in from fire trucks had reached the reactor, it should have adequately cooled the reactor.

Some of the water may have flowed into other systems

-- Key study points aimed at verification and explanation --

1

Information extremely important to assessing the accident development behavior is:

1-1 **The correlation between the amount of water injected into the building by fire trucks and the amount of water necessary for cooling**

and

1-2 **Identifying locations other than the reactor into which the injected water may have flowed**

-- Utilizing the study results in the Kashiwazaki-Kariwa Nuclear Power Station --	-- Future study points --
Review locations where inflow is possible **Use alternative water injection means to achieve effective water injection to the reactor** 2	**Assessment of the actual amount of water injected into the reactor and continued study into its effect on the development of the accident**

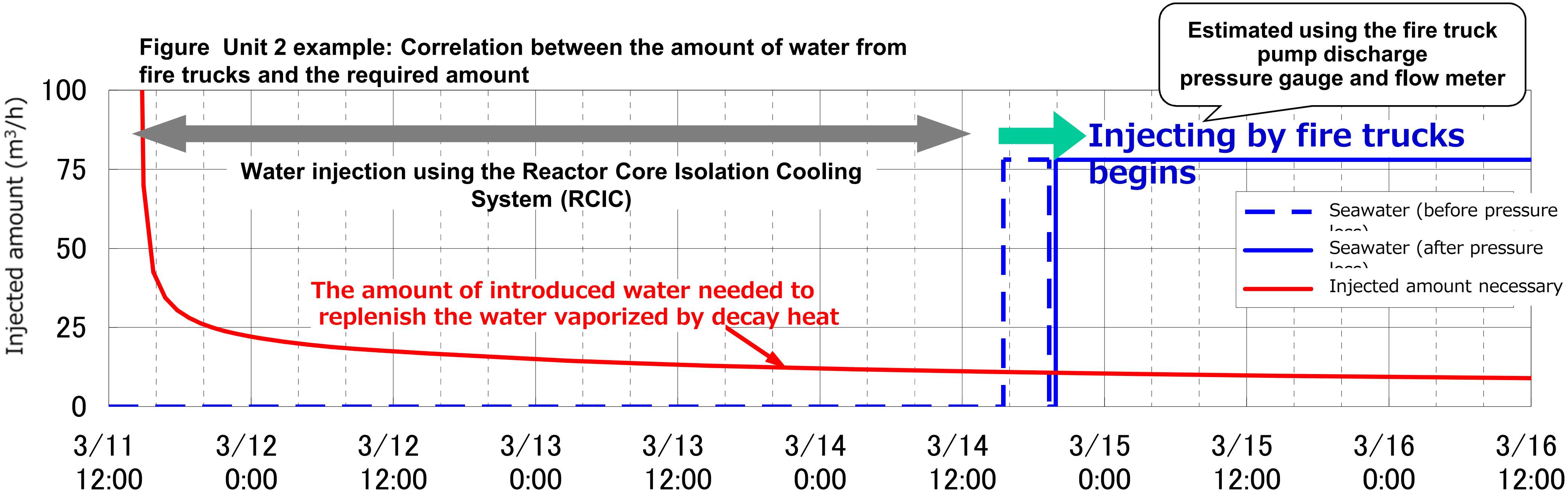
(3) Reasons why the reactor was not sufficiently cooled, despite water being injected in from fire trucks

1-1

Ascertaining the amount of water injected into the reactor by fire trucks and the amount of water necessary for cooling

In Units 1-3, the amount of water injected into the reactors by fire trucks was sufficient to replenish the water vaporized by the decay heat

Figure Unit 2 example: Correlation between the amount of water from fire trucks and the required amount



-- Challenges requiring continued study in the future --

Assessment of the actual amount of water injected into the reactor and its effect on the development of the accident

(3) Reasons why the reactor was not sufficiently cooled, despite water being injected in from fire trucks

1-2

Information extremely important to assessing the accident development behavior is:
Ascertaining locations other than the reactor into which water from fire trucks might flow

Confirming that there are paths that generate bypass flows to the main condenser and condensate storage tank

-- Background to the identification of possible locations --

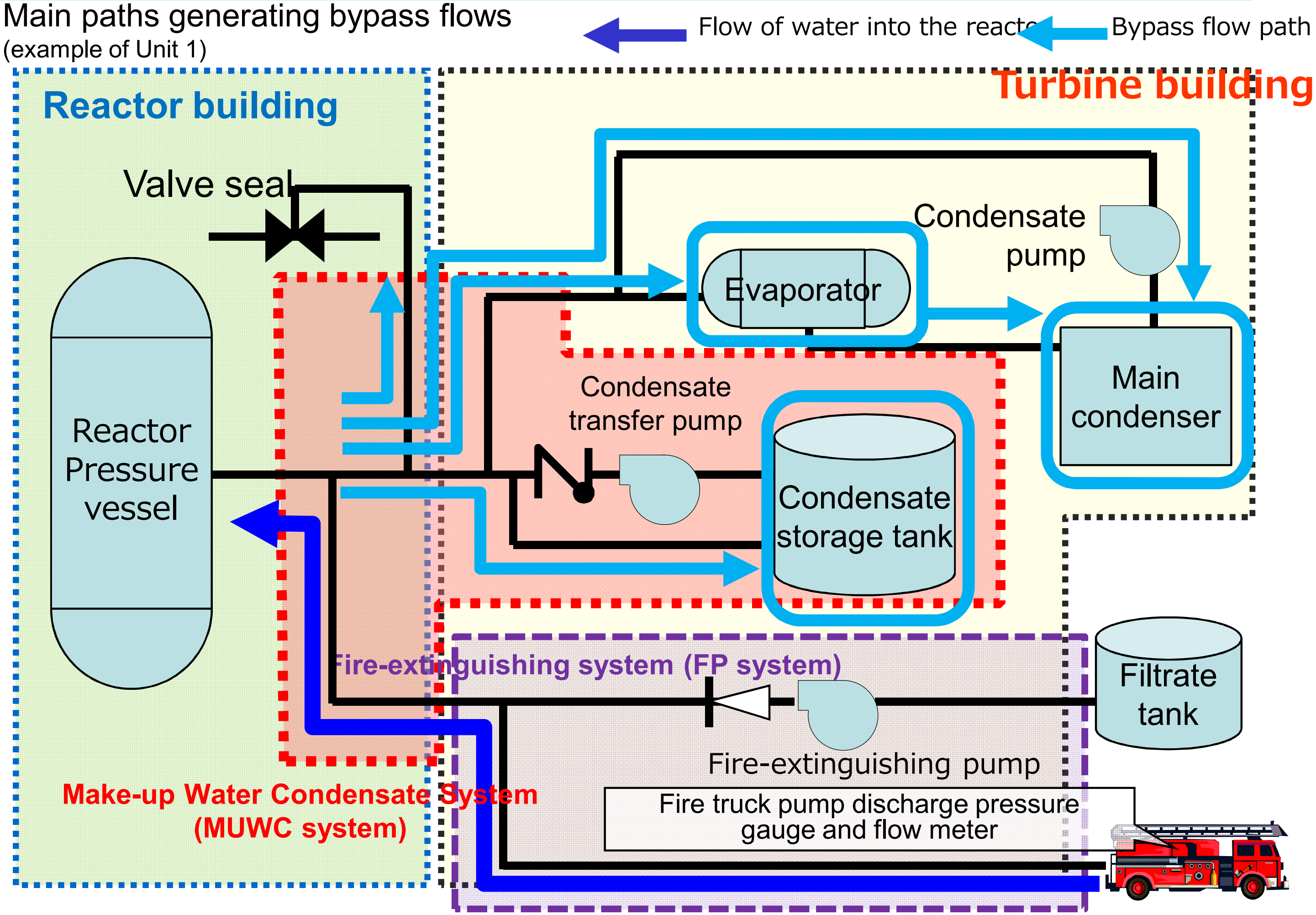
In Unit 2, from the time water injecting by fire trucks began until fuel rods were exposed and damaged

Late March 2011
Confirmation of water accumulated in the main condenser

Suggested possibility of bypass flows for water from fire trucks

Confirmation from piping drawings that there were paths that would generate bypass flows to the main condenser and condensate storage tank during the accident

Main paths generating bypass flows
(example of Unit 1)



(3) Reasons why the reactor was not sufficiently cooled, despite water being injected in from fire trucks

2

Reviewing locations where bypass flows are possible
Implementing measures to effectively inject water into the reactor using alternate water injection methods

Countermeasures implemented in the Kashiwazaki-Kariwa Nuclear Power Station

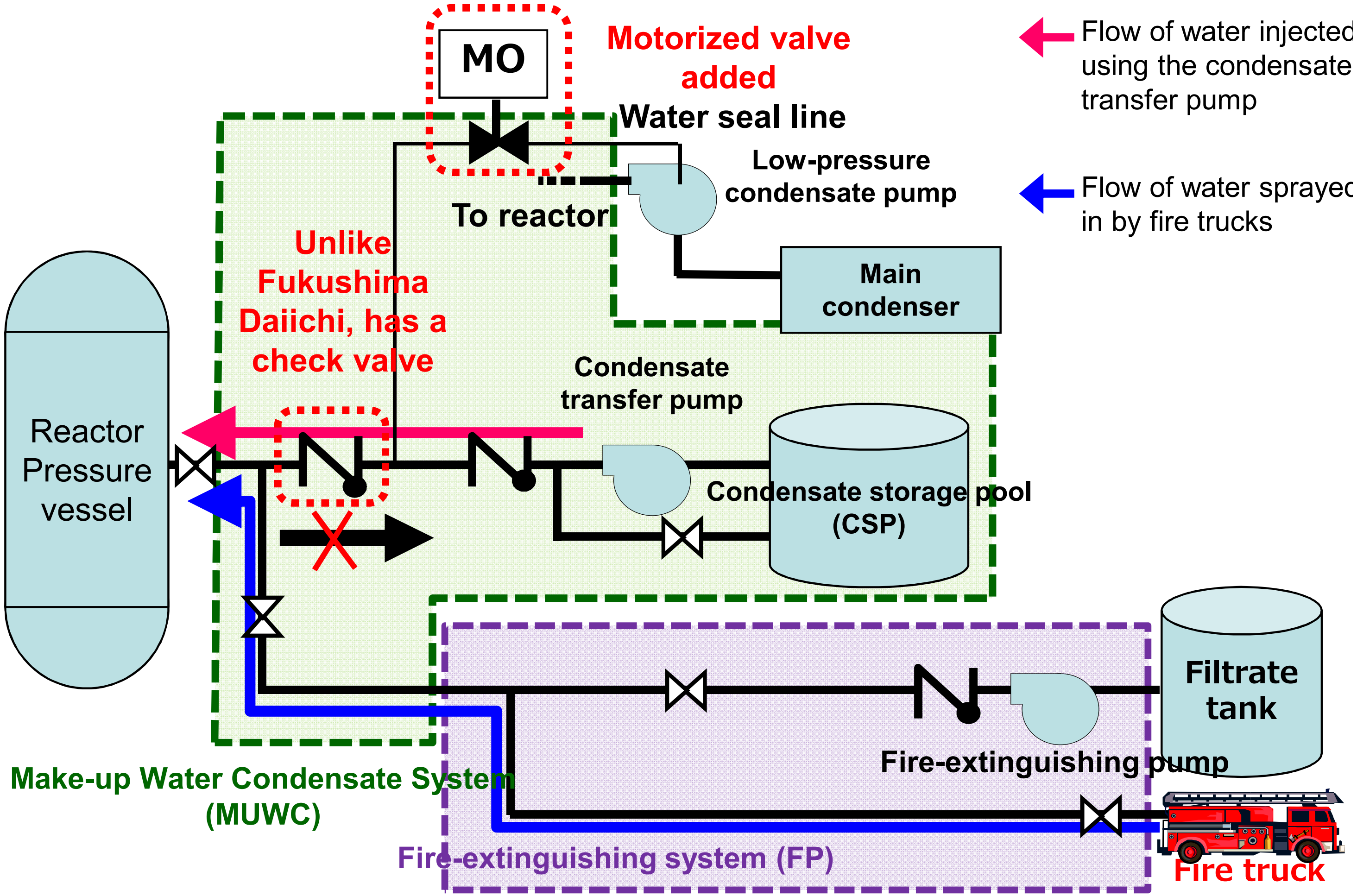
-- Utilization or application of study findings --

To prevent bypass flows, valves requiring a closing operation and valves where closure is to be confirmed clarified in the instructions

Training in operation & checking conducted

Addition of a motorized valve to prevent bypass flows when the Make-up Water Condensate System is used as an alternate water injection method

Provision of digital recorders and dedicated monitoring batteries for parameters such as the reactor water level and injection flow rate into the reactor



(4) Reason for the discrepancy between the time of manual stoppage of the high-pressure coolant injection system in unit 3 and internal reactor data

**Water injection may have been insufficient before the time of manual stoppage.
The process by which the lowering of the water level led to fuel exposure and damage has not been logically explained.**

Assessment of the operational status of the high-pressure coolant injection system in unit 3

Confirmation of the operational status of
the high-pressure coolant injection system (HPCI) in unit 3 **1**

Interpretation of internal reactor data and analytical data **2**

Investigation toward verification/explanation

Examination of the discrepancy between the time of manual stoppage of the HPCI system in unit 3 and internal reactor data **3**

Future investigation

Continued re-assessment of the advancement of core damage in unit 3

(4) Reason for the discrepancy between the time of manual stoppage of the high-pressure coolant injection system in unit 3 and internal reactor data

Confirmation of the operational status of the high-pressure coolant injection system (HPCI) in unit 3 **1**

Thus far, manual stoppage of the HPCI system at 2:42 on March 13 has been interpreted as the stoppage of coolant injection to the reactor. As the water level was not measured before the manual stoppage, the amount of water injected is unknown.

Chronology of major events

①3/12 11:36 : Automatic stoppage of the RCIC system

Low water level signal

②3/12 12:35 : Automatic start-up of the HPCI system in unit

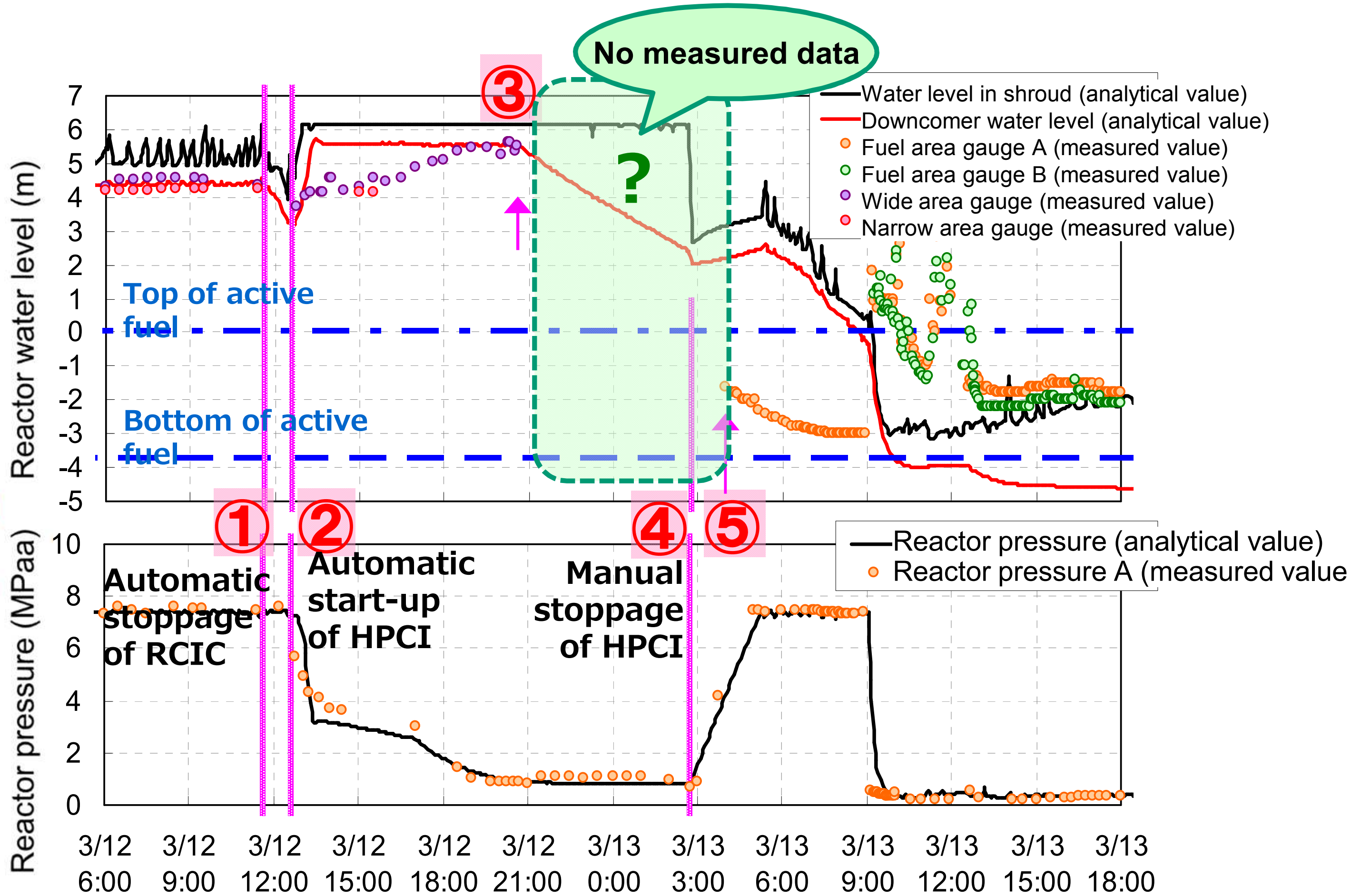
③3/12 20:36 : Reactor water gauge measurement suspended due to depletion

④3/13 02:42 : **Manual stoppage of the HPCI system in unit**

Reactor pressure buildup

⑤3/13 Around 04:00 : Reactor water gauge restored by battery

Water level measurement already below effective fuel top at this point



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(4) Reason for the discrepancy between the time of manual stoppage of the high-pressure coolant injection system in unit 3 and internal reactor data

Interpretation of internal reactor data and analytical data

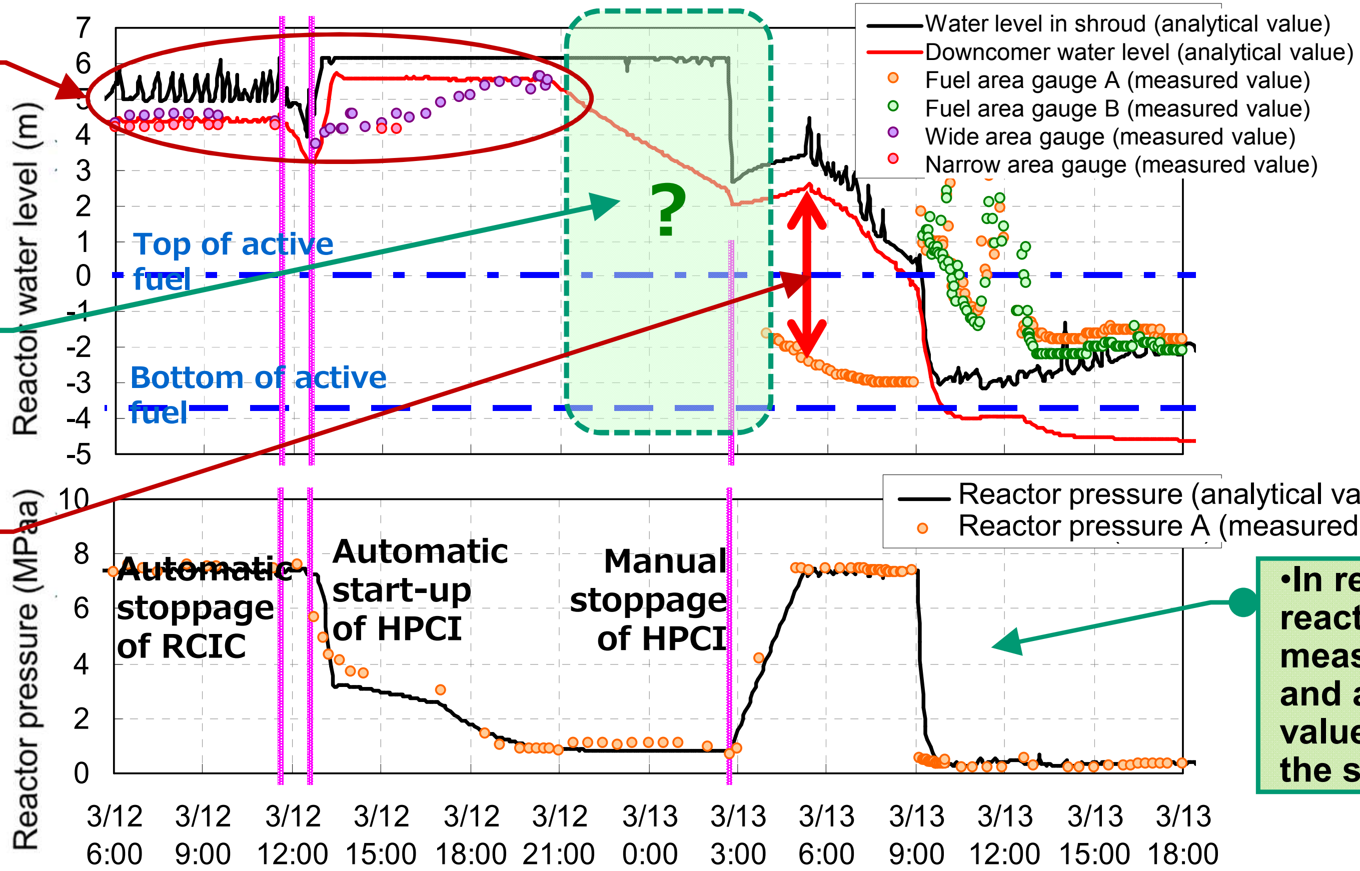
2

The HPCI system in unit 3 was manually stopped at 2:42 on March 13.
The reactor water level several hours before and after the manual stoppage shows discrepancies between the analytical results and measured values.

•Measured values and analytical values are roughly the same up to around 21:00

Data is missing during this period

•Measured values and analytical values differ greatly



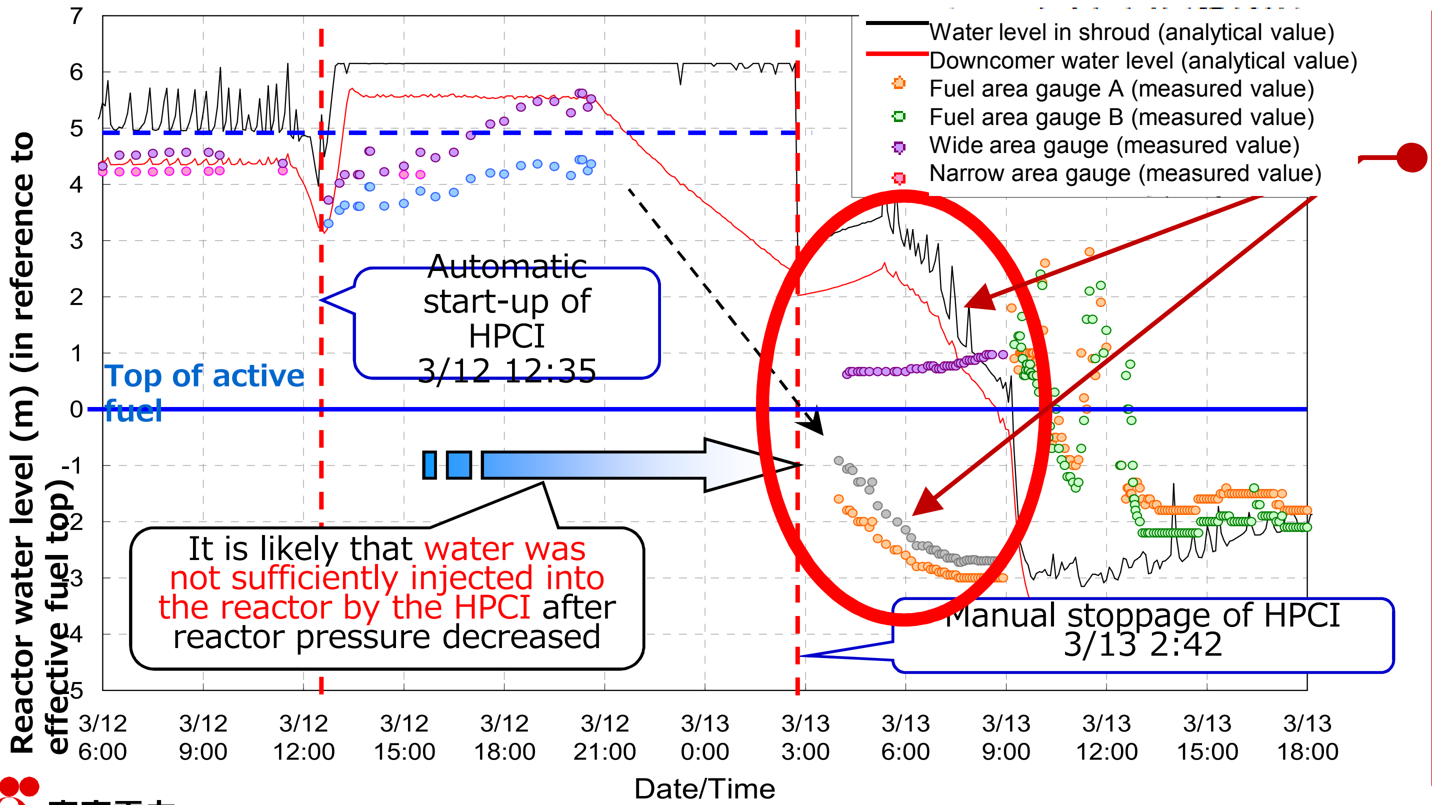
•In regard to reactor pressure, measured values and analytical values are roughly the same

(4) Reason for the discrepancy between the time of manual stoppage of the high-pressure coolant injection system in unit 3 and internal reactor data

Examination of the discrepancy between the time of manual stoppage of the HPCI system in unit 3 and internal reactor data

3

The process through which fuel was exposed and damaged has not been logically explained due to uncertainties in the operational status (actual amount of water injected) of the HPCI system, but the advancement of core damage in unit 3 will be re-assessed in the future based on the estimated result that water injection to the reactor by the HPCI system was insufficient.



Analytical water level (black line)

According to MAAP analysis, the water level was higher than the effective fuel top until 9:00 on March 13

Measured water level (gray circles)

The fuel area gauge indicated a water level lower than the effective fuel top at around 4:00 on March 13

The discrepancy between the analytical results and measured values cannot be reasonably explained at present

3. Detailed examination of representative unconfirmed and unexplained events

(5) Cause of the sudden decrease in reactor pressure in unit 3 (whether there was a hole in the reactor or other important equipment)

Understanding prior to investigation

The sudden decrease in reactor pressure in unit 3 at around 9:00 on March 13 was thought to be a result of an operator opening the safety relief valves (SRV).

Results of investigation

An examination revealed that reactor pressure had already been dropping while the operator was preparing for manual depressuring. It is likely that operating conditions were satisfied for the automatic depressuring system (ADS) to function and reduce the pressure.

Assessment of the cause of the sudden decrease in pressure

Circumstances of the sudden decrease in reactor pressure in unit 3 1

Confirmation of ADS operating conditions that allowed rapid depressuring 2

Investigation toward verification/explanation

Examination of the possibility that operating conditions were right for the ADS to function 3

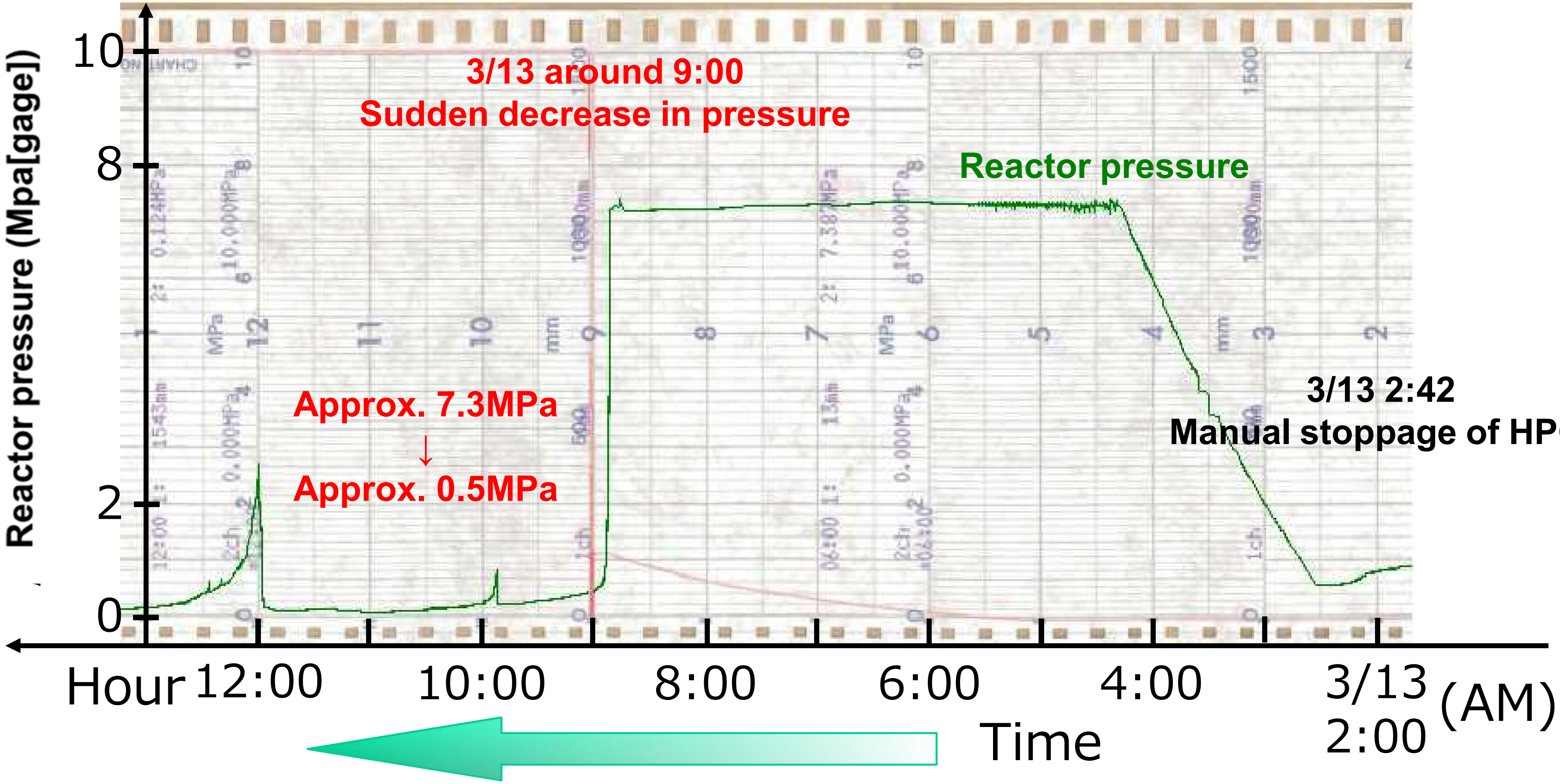
Investigation of the decrease in reactor pressure by comparing measured data and analytical data 4

(5) Cause of the sudden decrease in reactor pressure in unit 3 (whether there was a hole in the reactor or other important equipment)

Circumstances of the sudden decrease in reactor pressure in unit 3

1

At around 9:00 on March 13, reactor pressure dropped while an operator was preparing for manual depressuring. Depressuring occurred rapidly within 2–3 minutes, while this normally takes around 20 minutes after manually opening the safety relief valves (SRV).



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(5) Cause of the sudden decrease in reactor pressure in unit 3 (whether there was a hole in the reactor or other important equipment)

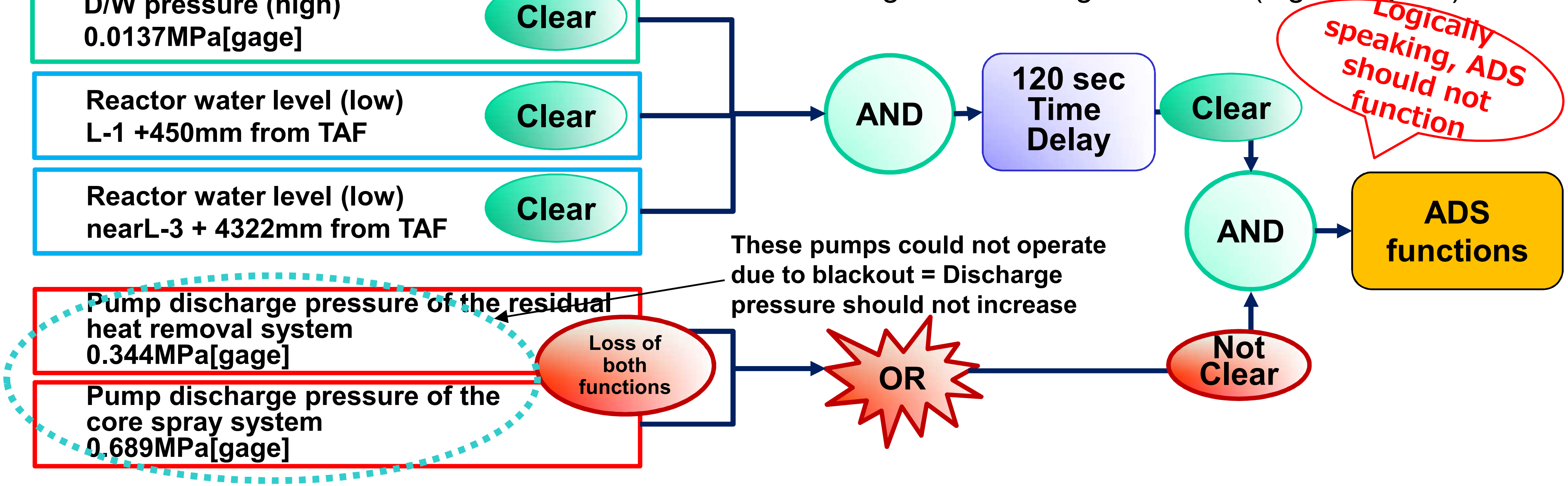
Confirmation of ADS operating conditions that allowed rapid depressuring

2

It was understood that rapid depressuring is possible if the ADS functions, but it was thought that conditions were not right for the ADS in unit 3 to function.

Note) When pressure in the reactor is high and reactor water level cannot be maintained, ADS (automatic depressrization system) opens the safety relief valves to lower the reactor pressure and allow water injection via a low-pressure water injection method. As one of the conditions for activating ADS, preparation of the low-pressure water injection system must be completed (pump discharge pressure must be established).

Fig. ADS starting conditions (logic diagram)



(*Logically, ADS should not function and perform rapid depressuring, but given the sudden decrease in pressure that actually occurred, the possibility of the ADS having functioned will be examined.)

(5) Cause of the sudden decrease in reactor pressure in unit 3 (whether there was a hole in the reactor or other important equipment)

Examination of the possibility that operating conditions were right for the ADS to function

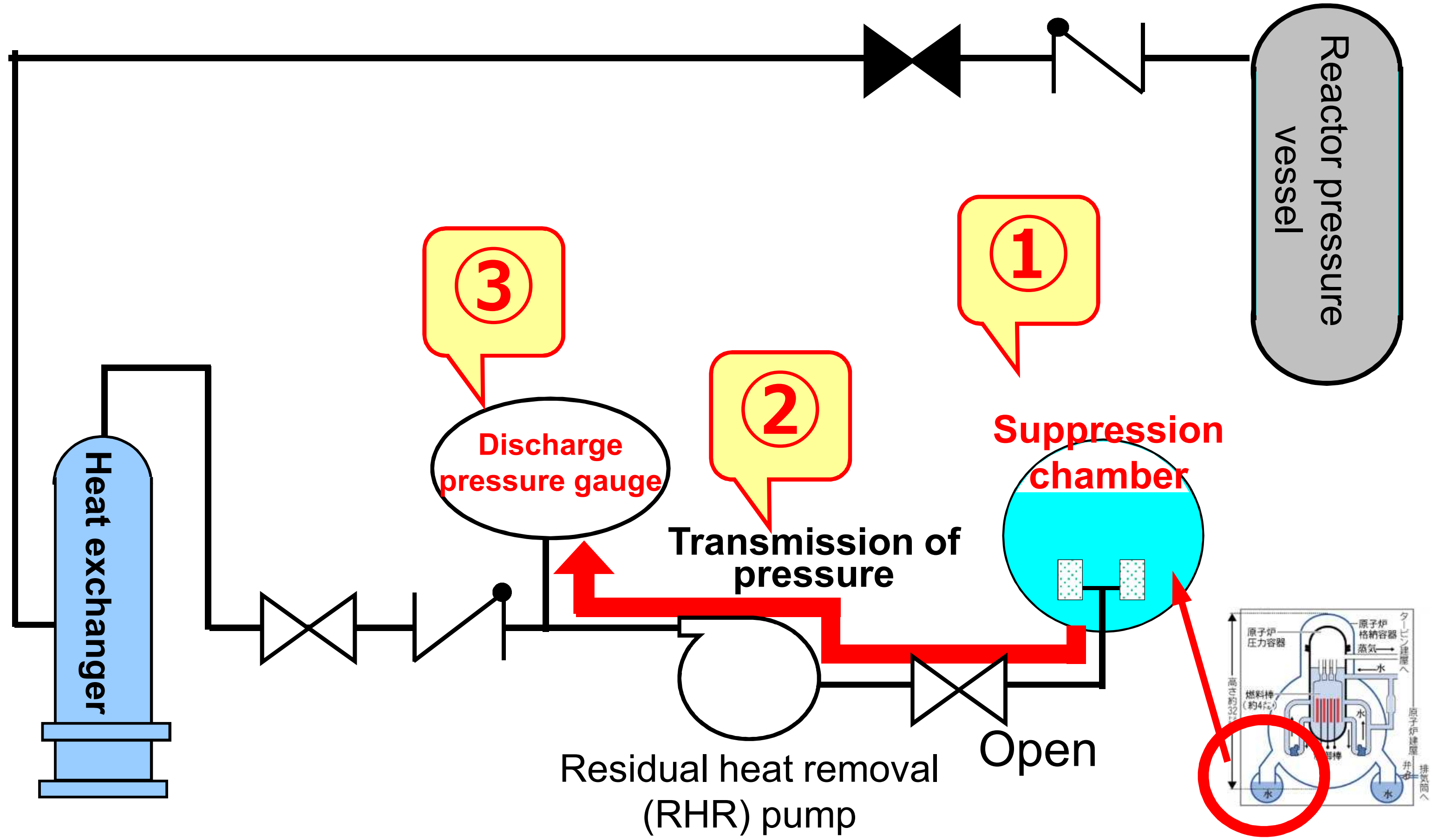
3

Due to an increase in pressure in the suppression chamber (S/C), the discharge pressure gauge measured the prescribed value even though the pump in the residual heat removal system (RHR) had not functioned.
This may have set the conditions for depressuring by the ADS.

① Increase in pressure in the S/C to 0.455MPa[abs]

② Transmission of pressure

③ The pressure gauge detected a value exceeding the 0.344MPa[gage] condition for ADS operation

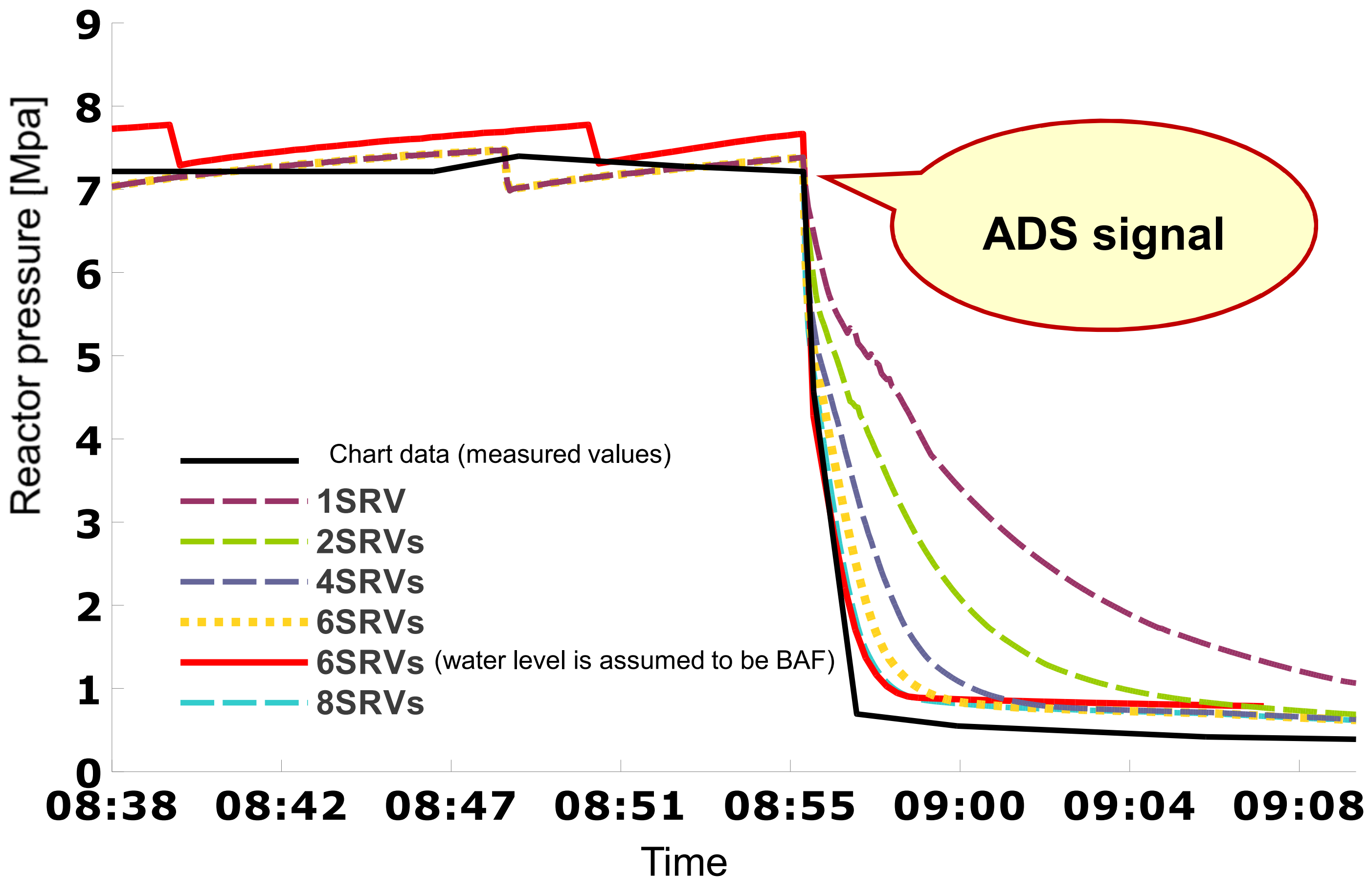


(5) Cause of the sudden decrease in reactor pressure in unit 3 (whether there was a hole in the reactor or other important equipment)

Investigation of the decrease in reactor pressure by comparing measured data and analytical data

4

When conditions for the functioning of ADS were analyzed with 6 safety relief valves (SRV) open and the water level near the fuel bottom as indicated by the measured value, the actual decrease in reactor pressure could be reproduced for the most part, so the possibility that operating conditions were right for the ADS to function has been verified.



Safety relief valve (SRV): When reactor pressure increases abnormally, the safety relief valve releases steam to the suppression chamber automatically, or manually via the central control room, to protect the pressure vessel (the released steam is cooled by water in the suppression chamber and condensed). It also functions as an automatic depressuring system (ADS) for the emergency core cooling system (ECCS).

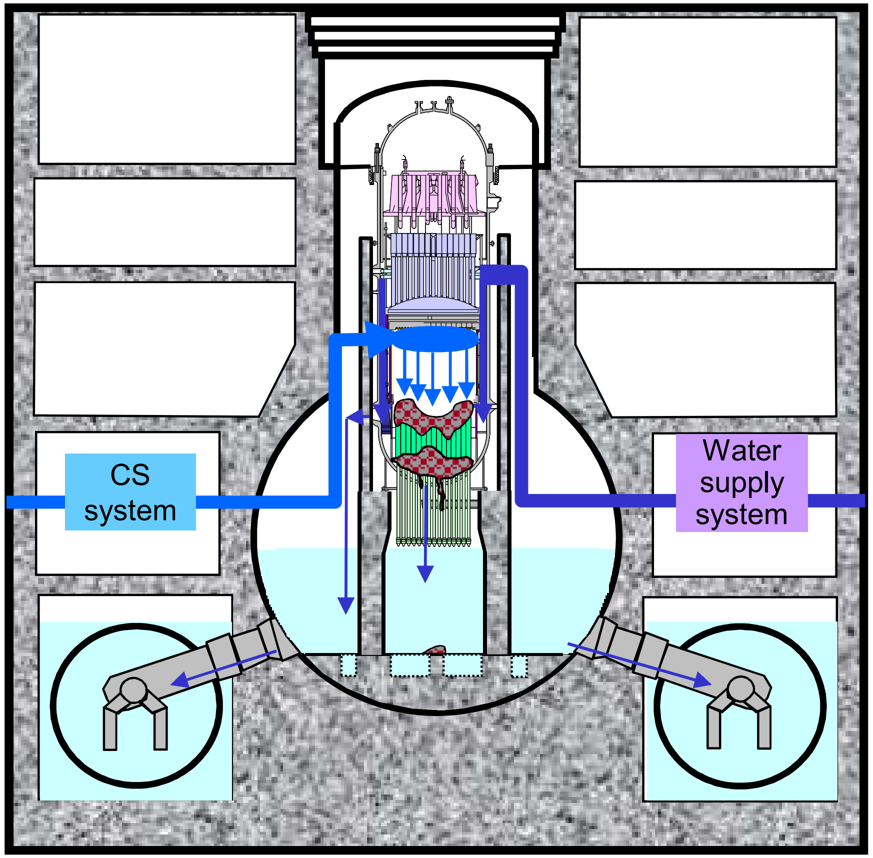
4. Estimated state of the reactor and containment vessel (unit 3)

State of the reactor containment vessel in unit 3 estimated from recent confirmation results

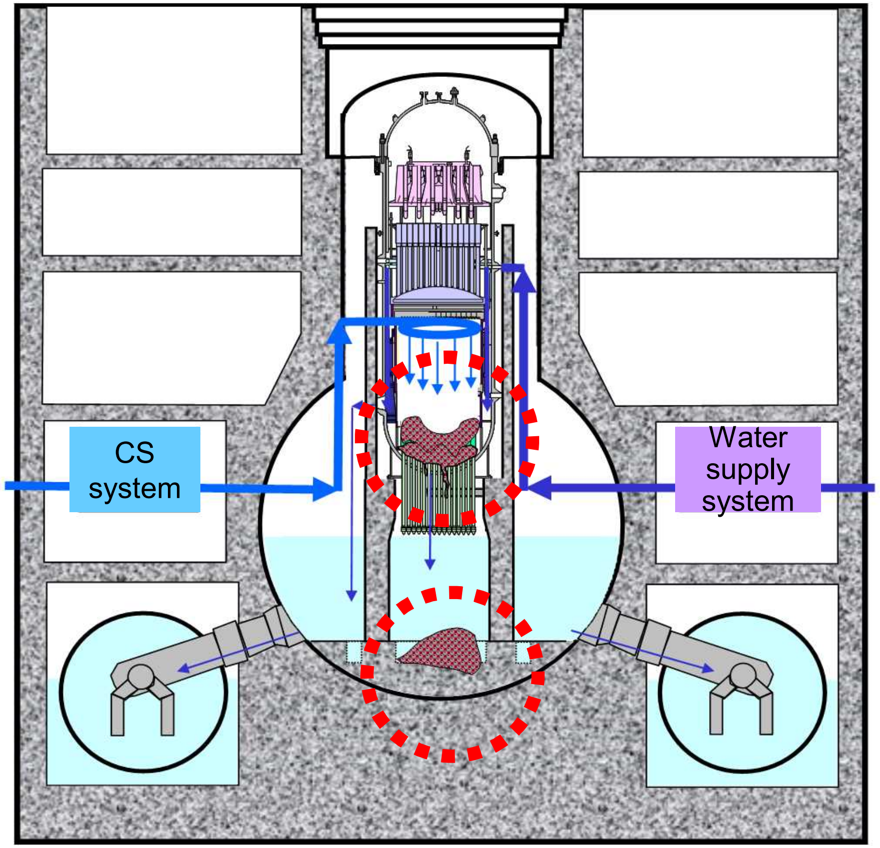
Revise the extent of core damage and core meltdown in consideration of the lack of water injection by the high-pressure coolant injection system (estimate that a larger amount of fuel than initially evaluated has fallen inside the containment vessel).

***The diagram is an illustration, and does not quantitatively express the size of the fuel debris, etc.**

<Previous evaluation*>



<Present estimation*>



<Previous evaluation>
Cited from and additions made to "Reactor core conditions of unit 1-3 of Fukushima Daiichi Nuclear Power Station" (Nov. 30, 2011)

5. Major issues for future examination

Of the 52 instances of unconfirmed and unexplained events extracted in this study, the following 10 issues will be given the highest priority for early clarification.

- **Examination of the operation of the safety relief valves after core damage**
- **Discharge status of radioactive material since March 20**
- **Enhancement of the accuracy of the amount of water injected into the reactor by fire trucks / Examination based on the conclusion of issue (3)**
- **Evaluation of the operational status of the high-pressure coolant injection system (HPCI) in unit 3 and effect on the accident / Examination based on the conclusion of issue (4)**
- **Dropping mechanism of the lower plenum of the melted core**
- **Identification of the cause of the high-dose contamination of the Reactor Building Closed Cooling Water System (RCW) in unit 1**
- **Increase in reactor pressure after forced depressuring in unit 2**
- **Operation of the rupture disk in unit 2**
- **Cause of the stoppage of the reactor core isolation cooling system (RCIC) in unit 3**
- **Thermal stratification of the suppression pool in unit 3**