Research plan regarding improvement of simulation code for understanding the status of fuel debris in the reactor

International Experts' Symposium on the Decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Plant Unit 1-4

March 14, 2012
Tokyo, Japan

Tokyo Electric Power Company, Inc.

TOKYO ELECTRIC POWER COMPANY

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O. Present Approach

① After MAAP analysis in May 2011, reanalyzing with following input
- Detail information of operation and plant parameter
- Operating condition of components assumed from design basis under SBO condition after the earthquake

From the above, possible to explain plant behavior from the earthquake to the core melt
- Still high uncertainty for debris location after the core melt

② Estimating the in-core status from temperature change before and after the CS water injection
- Unit 1: below 100℃ without CS water injection,
  Units 2 & 3: below 100℃ after CS water injection

Therefore, some fuel debris are existing in the core at units 2 & 3. (high uncertainty)

③ For MCCI, implementing individual parameter studies

④ Confirming no continuance of MCCI due to PCV gas measurement

Necessary to focus on identifying debris location including MCCI

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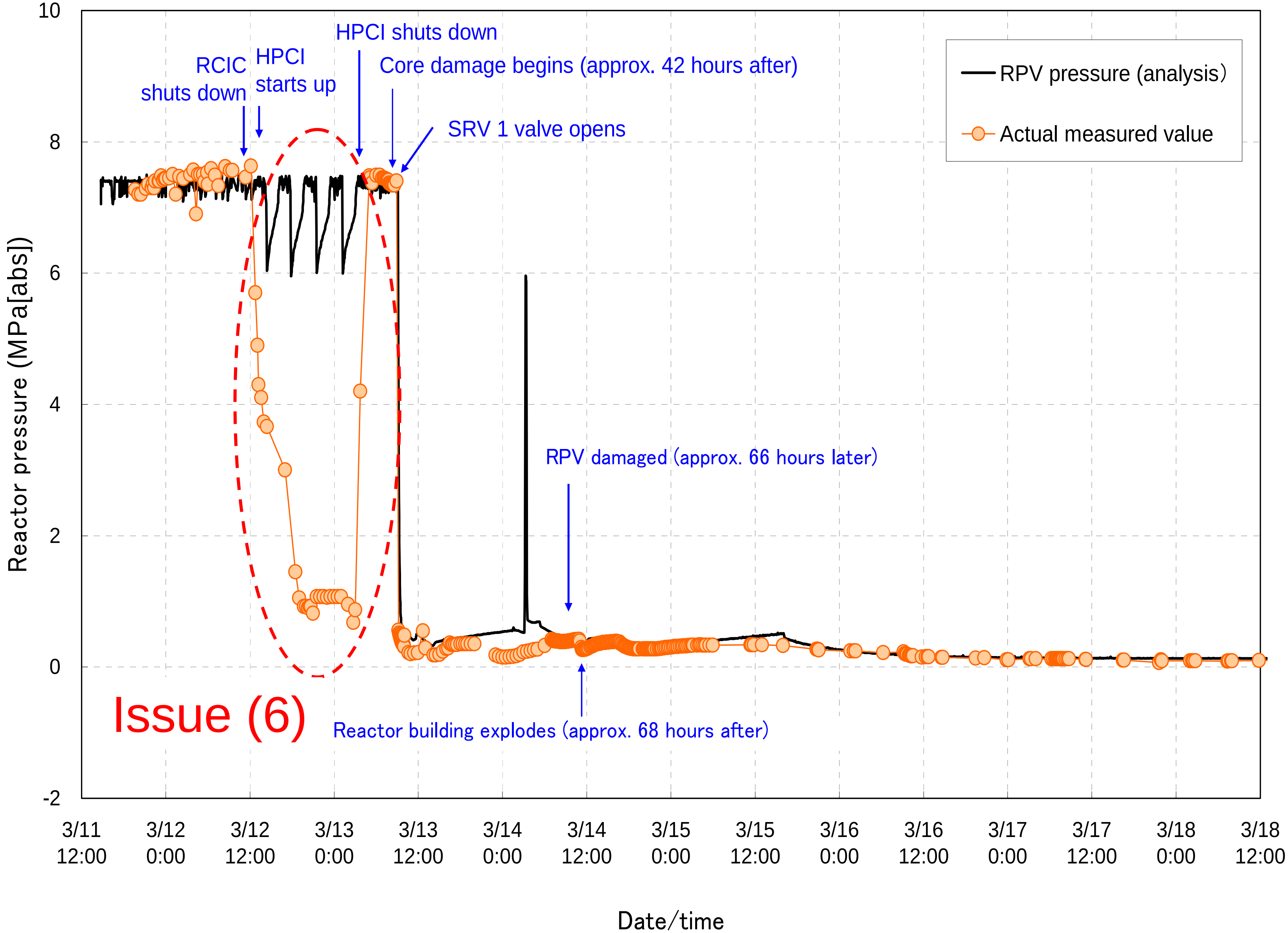
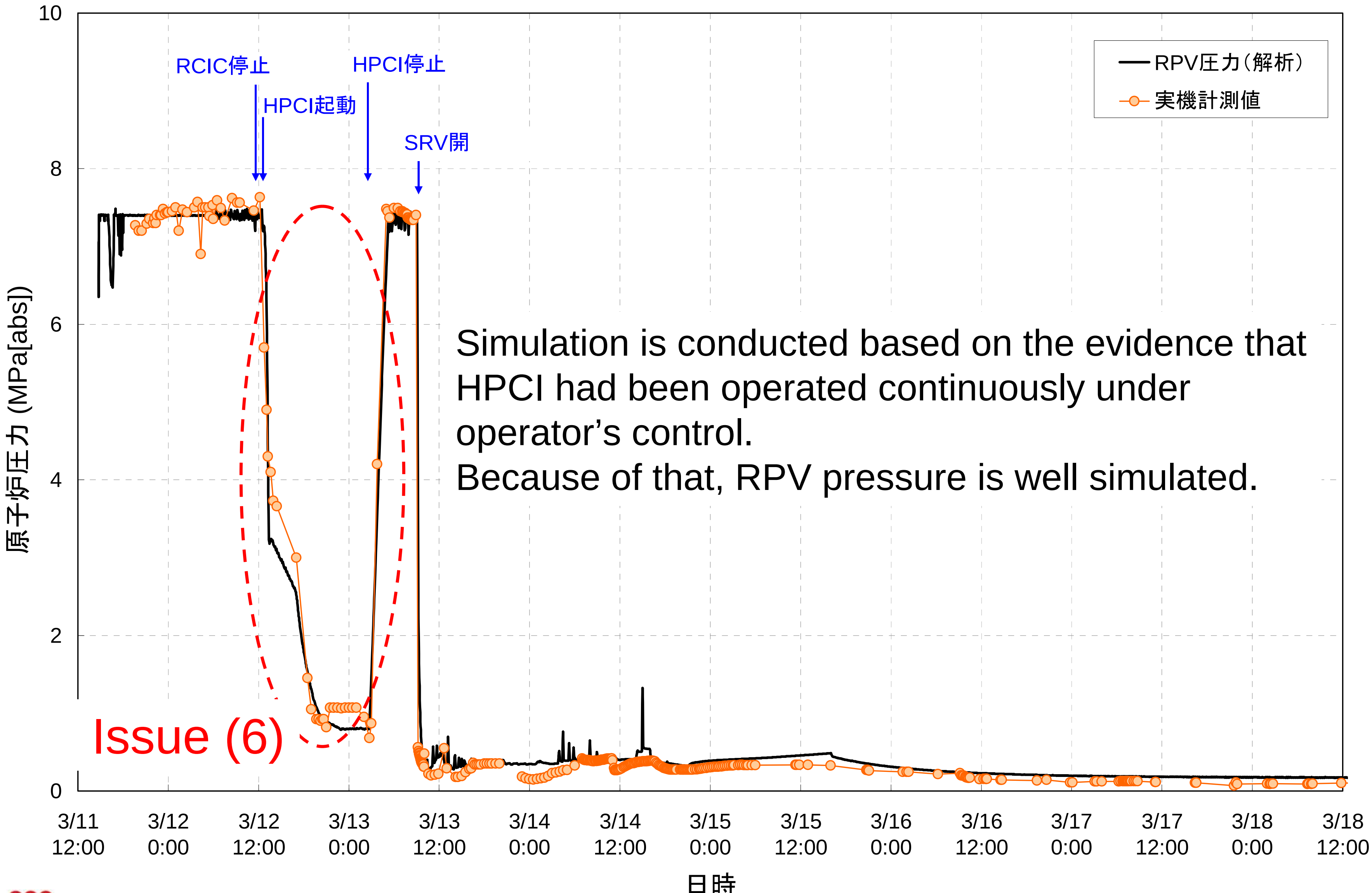
Unit 3 Analysis Results (Reactor Pressure) ：May 23rd



Unit 3 Analysis Results (Reactor Pressure) : latest ver.



Simulation is conducted based on the evidence that HPCI had been operated continuously under operator's control.
Because of that, RPV pressure is well simulated.

I. Introduction

- Purpose: Identifying the fuel debris status in the reactors
- Challenge: High uncertainty in the present analysis code（difficult to obtain precise information at present）
  - Inadequacy of simulating core internals
  - Many phenomenon not fully understanding of physical mechanism

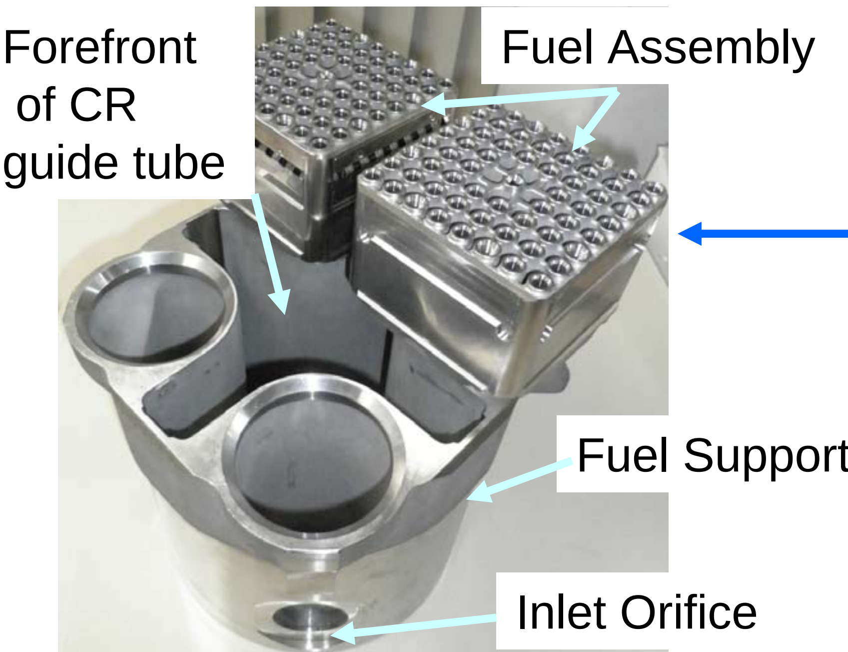


Fig. 1 BWR lower core structure

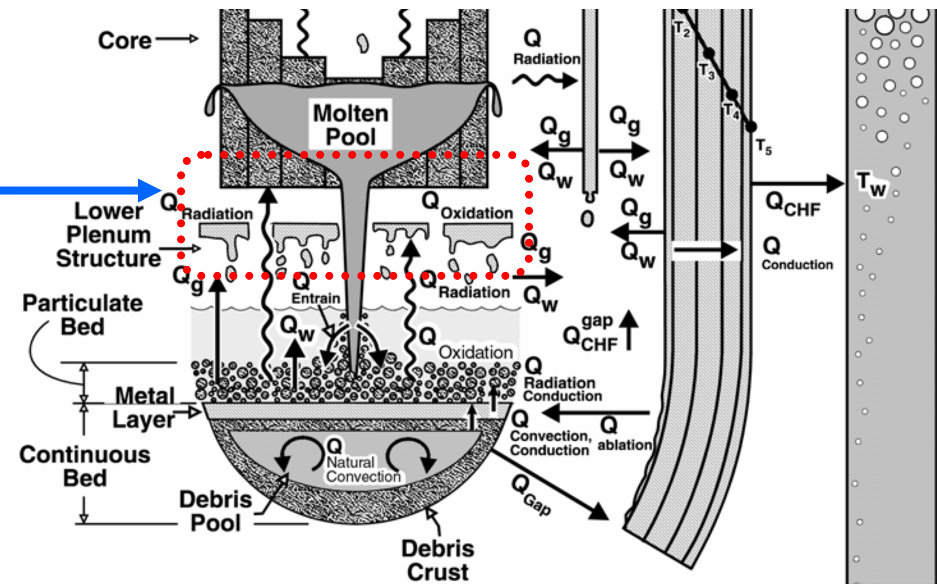


Fig. 2 MAAP model for BWR lower core structure

I. Introduction

- Referring PIRT (Phenomena Identification and Ranking Table), and defining priority of development item (PIRT of the Fukushima accident is in process of creation by AESJ)
  - Applying previous study such as PIRT by EURSAFE
  - Reflecting priority as indication for review of our development plan

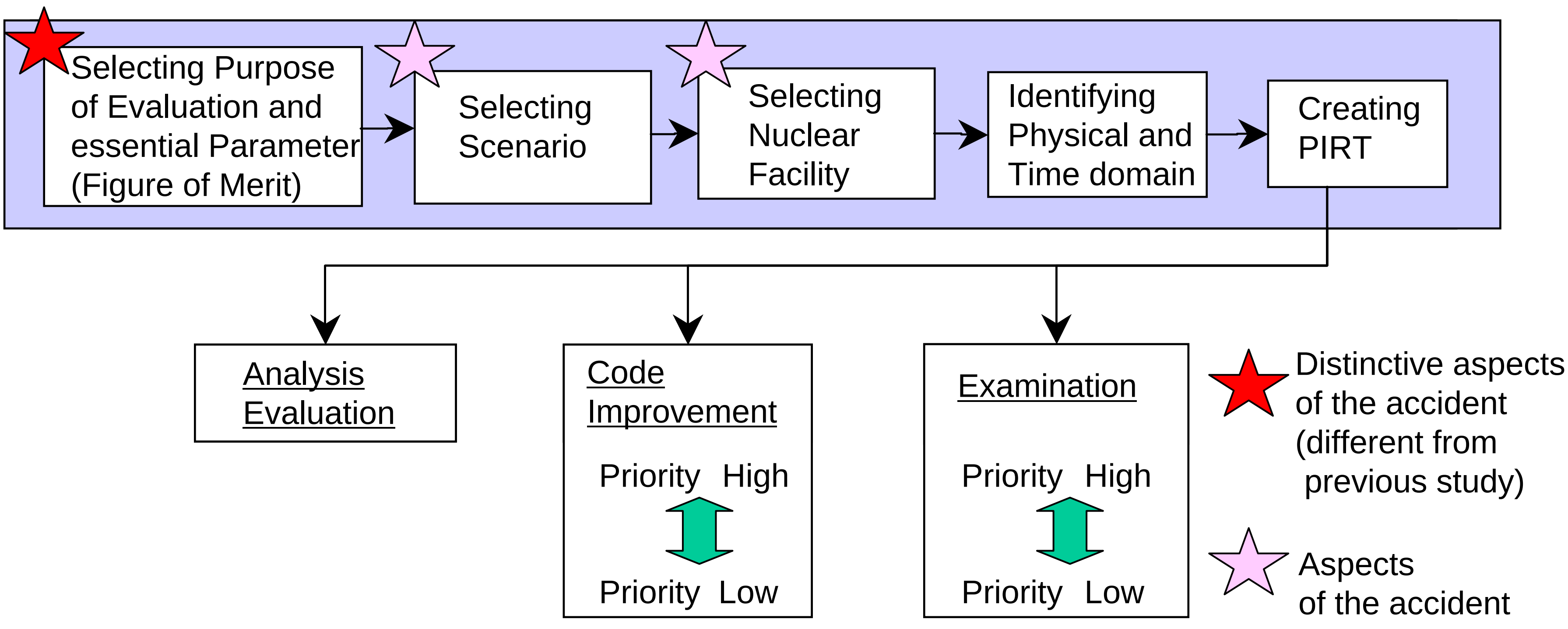


Fig. 1 Relationship between PIRT creation and analysis evaluation/code development/examination

II. Concerning Item

1. Analyzing and evaluating of plant behavior at the accident

- Analyzing and evaluating plant behavior based on operating information and actual measurement data, and clarifying essential information for accident progress analysis
- Referring to the result evaluated by various codes and simplifying calculations for identifying the in-core status

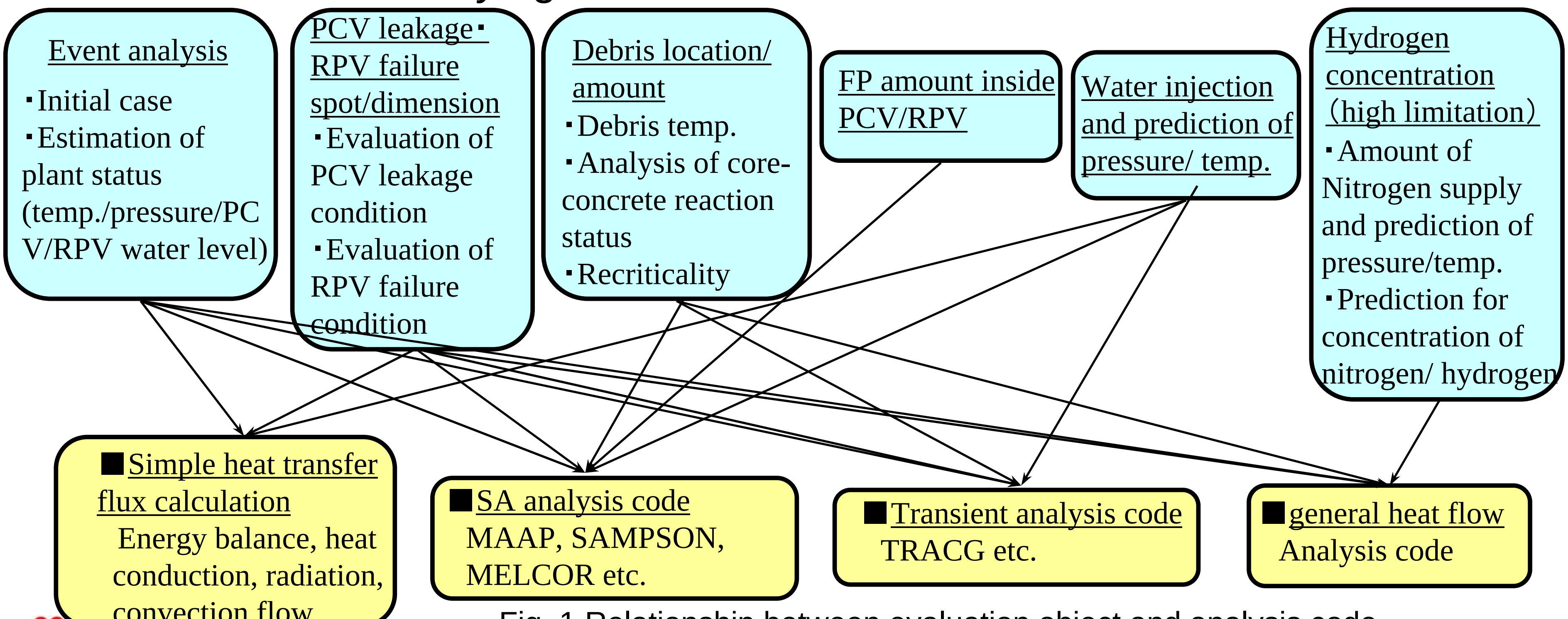


Fig. 1 Relationship between evaluation object and analysis code

II. Concerning Item

2. Upgrading Severe Accident Analysis Code

- Clarifying aspects for existing severe accident (SA) analysis code, evaluating applicability to understanding in-core status

＊：Object Code ①MAAP（Modular Accident Analysis Program）
②SAMPSON:（Severe Accident Analysis Code with Mechanistic, Parallelized Simulations Oriented towards Nuclear Field）

- Upgrading the code in accord with evaluation result of the accident behavior and in-core investigation
  – Adding debris transition model (reflecting core bottom structure) etc.

3. Conducting mock examination to contribute the detail analysis of the accident progression

- Performing survey regarding SA-related concerns ever conducted
- Commencing studies for mock examination from now on

III. Upgrading Severe Accident Analysis Code

- Conducting the accident progression analysis project by public offering of Agency for Natural Resources and Energy

"1. Analyzing of plant behavior at the accident"

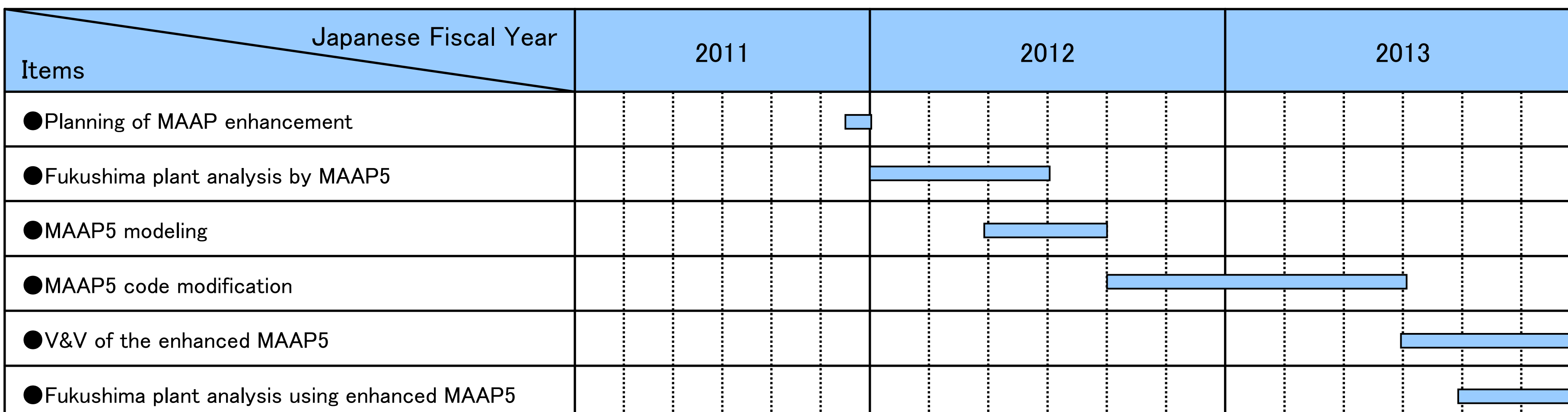
"2. Upgrading severe accident analysis code"

- Evaluating from broad standpoint in two approaches for the implementation of the projects
- Selecting models to be upgraded in accord with PIRT
- ①「User tuning application model」(2/15～)
  - Toshiba(applying proven "MAAP")
- ②「Mechanistic model」(2/15～)
  - The Institute of Applied Energy (applying "SAMPSON" installing detail model)

Enhancement of MAAP Code on User Tuning Application Mode Project

- **Project program**
  - Based on the analysis for progress in Fukushima event by severe accident analysis code MAAP, together with the clarifying of analysis model related to event sequence of inside of the vessel, prepare enhanced specification by extracting its subjects.
  - Based on enhanced MAAP specification, modify code and verify by comparison with experimental results.
  - Analyze progress in Fukushima event by enhanced MAAP code and contribute to the understanding the condition of reactor.

Schedule



• Problem of existing MAAP code

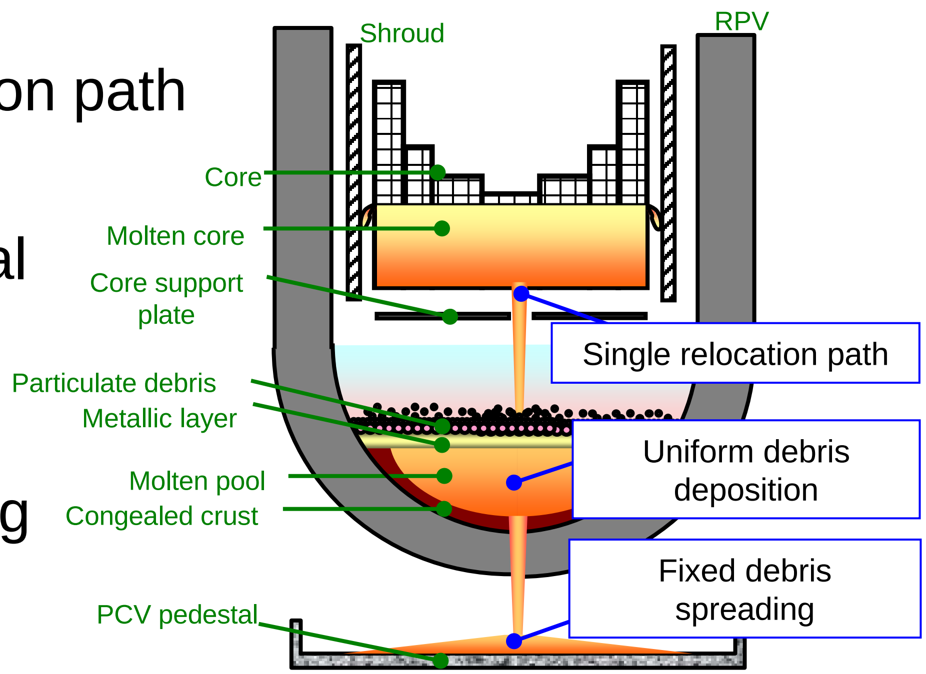
- – Limited core degradation progression during SA, because of simplified core relocation path.
- – Modify to model possible core degradation progression.

（Example）

✓Modeling of various relocation path of molten core

✓Modeling of non-symmetrical debris deposition complying with transition path

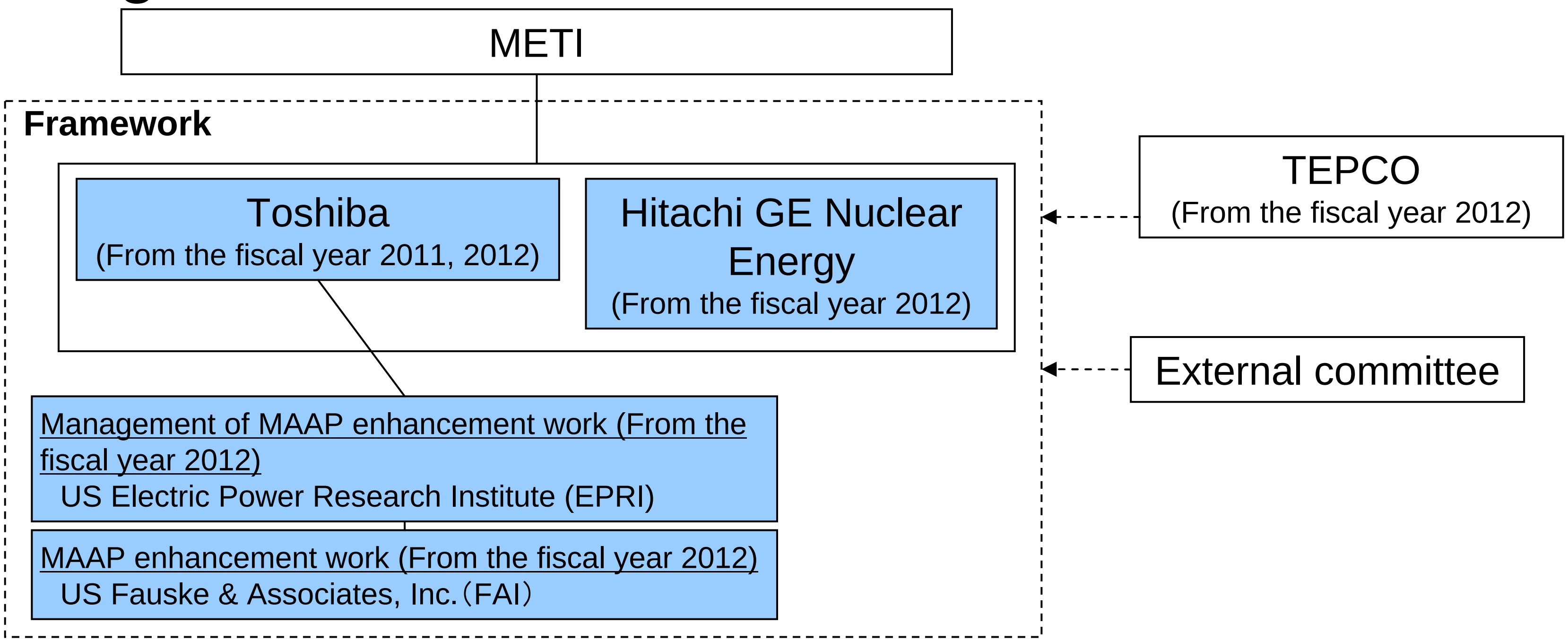
✓Modeling of debris spreading based on the property of molten debris



Existing core damage progress model of MAAP

Enhancement of MAAP Code on User Tuning Application Mode Project

- **Organizational chart**



【Role-sharing】

- Toshiba： Coordinating, accident progression analysis of the Unit 2 and 3 reactor of the Fukushima Daiichi nuclear plant, review of enhancement result
- Hitachi GE Nuclear Energy： accident progression analysis of the Unit 1 reactor of the Fukushima Daiichi nuclear plant, review of enhancement result
- EPRI： Management of MAAP enhancement work
- FAI： MAAP enhancement work
- External committee： Evaluation of the applicability of enhanced code and the achievement of objectives
- TEPCO： Provision of information about accident progression of Fukushima Daiichi nuclear plant

Upgrading SAMPSON Code on Mechanistic Model Project

- Major features of SAMPSON (advantage and weakness)
  - Maximum use of mechanistic models and theoretical-base equations
  - Minimum use of user tuning parameters
  - Evaluation of locations, amounts, and compositions of distributed debris
  - Explainable analysis results as physical phenomena
  - Long computational time (about 30 times of real time for initial rapid transients)
  - Large effort to prepare multi-dimensional input data (about 28,000 lines)
  - Insufficient I/O interface

- Major unknowns for analysis (points to be improved in SAMPSON)
  - Part-load operation of cooling systems such as IC, RCIC, HPCI
  - Possible deformation of core internals, especially core shroud
  - Damage behavior of RPV bottom having many pipes such as CRD and ICM
  - RPV depressurization behavior (possible steam leakage from RPV or pipes)
  - Leakage from PCV
  - Improvement of I/O interface
  - Reduction of CPU time (for parametric analysis)

Upgrading SAMPSON Code on Mechanistic Model Project

• Overall SAMPSON code improvement plan

Items	Phase 1		Phase 2		Phase 3	
	2011	2012	2013	2014	2015	2016
New models	• Part-load operation of cooling systems, • Deformation of core internals, • Heat loss from RPV, CV					
Model improvements	Thermal-hydraulics (T/H) in RPV			Thermal-hydraulics (T/H) in PCV		
Reduction of CPU time	• Convergent calculation improvement • Mesh division optimization 30% reduction		• Algorism optimization • Parallel efficiency improvement Real time simulation		Model simplification Faster than real time	
User interface	Introduction of GUI					
Fukushima Analysis	With current SAMPSON		Analysis with improves code at each phase			
Major analysis results	Location/amount/ composition of debris with RPV/CV leakage and core internal deformation models		Same as on the left, but more detailed and realistic with improved T/H models in RPV and considering part-load operation.		Same as on the left, but more detailed and realistic with improved T/H models in PCV. Identification of detailed accident progression.	

Upgrading SAMPSON Code on Mechanistic Model Project

- Implementation structure of the project

Atomic Energy Society of Japan
Severe Accident Research Committee
(Voluntary study by AESJ members)

Ministry of Economy, Trade and Industry

Implementation structure

Evaluation committee
- Objective evaluation of implements, results, and applications by specialists

The Institute of Applied Energy

(Application of expertise)

Fukushima Analysis Project Team

Survey Team
- Current technology survey
- Input data preparation

Code Improvement Team
- New model development
- Current model improvement

Analysis Team
- Analysis of Fukushima accident progression

Subcontractors (Soft-houses): for coding

(Provision of plant data) TEPCO, Plant makers, Fuel company

IV. Mock Examination

- Conducting mock examination to contribute to upgrading SA analysis code in accord with PIRT (to be put into practice from now on)

[Reference]：Examination plan byJAEA

- – High temperature material characteristics examination and structural response analysis as RPV soundness evaluation
- – Loss-of-coolant mock examination and fuel melting examination as fuel melting progression evaluation
- – Debris-concrete reaction basic examination
- – Source term examination, etc.

V. Knowledge Base･International Corporation

- As input data for fuel debris removal,
  analysis results including following items are required.

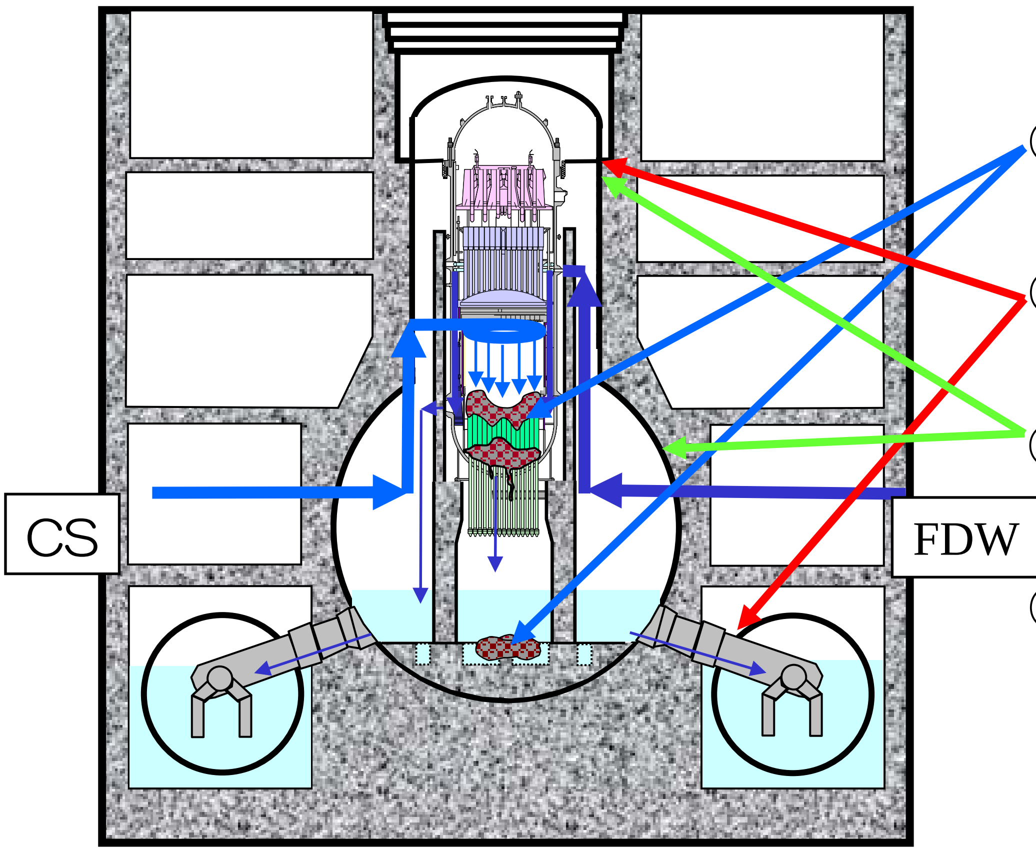


Fig. 1 Efficient information toward fuel debris removal

V. Knowledge Base･International Corporation

- Necessity of precise information about the accident to conduct high-quality analysis
  - Compiling database of previous published information and analysis evaluation for analysts' convenience

Actual measurement data of plant status	Actual performance of operation at the accident	Material/measurement data of each plants
･RPV pressure ･PCV pressure ･RPV water level ･CAMS gamma dose ･Temperature ･etc.	･IC、RCIC operation ･HPCI operation ･Venting operation ･Loss-of-power time ･Water injection actual performance	･Fuel and control rod ･Core internals ･RPV and PCV ･Pump performance

Fig. 1 Possible necessary information

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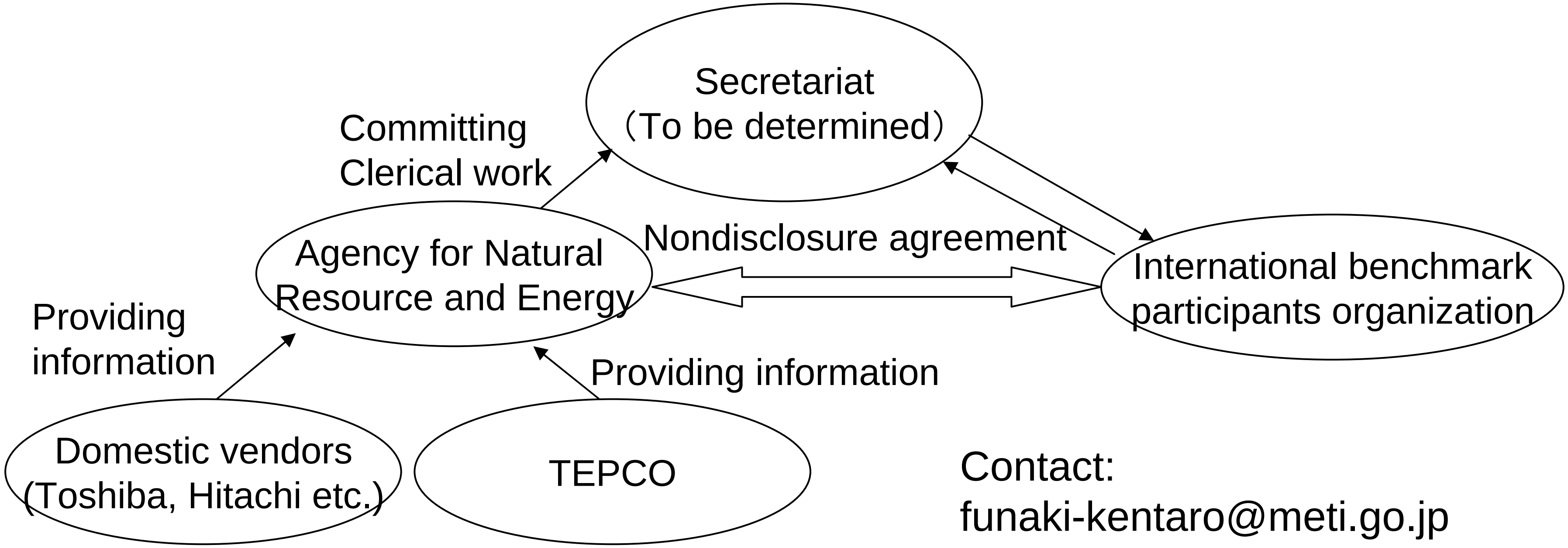
V. Knowledge Base･International Corporation

- Effective if develop and process framework for gathering a wide range of information such as selection of international benchmark
  - Considering the framework for conducting the analysis efficiently from now on
  - Collecting feedback and suggestion from the WS participants



Contact:
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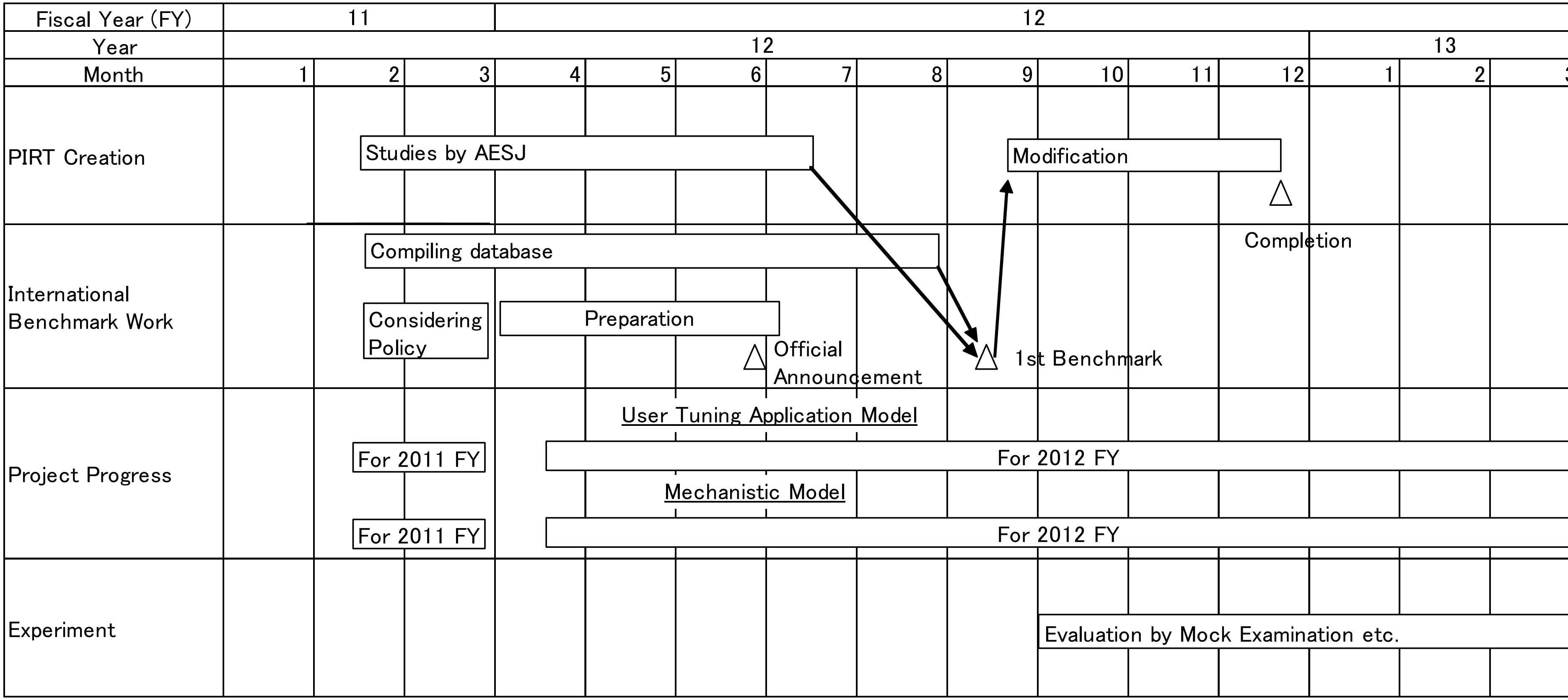
yamanaka.yasunori@tepco.co.jp

Fig. 1 Proposed framework of international benchmark implementation

VI. Schedule



1st phase

2011～2013

(Mainly understanding Plant behavior)

2nd phase

2014～2020

(SA evaluation in accord with the result at the 1st phase)

VII. Summary -challenge and future action-

- Upgrading the present SA code
  - –Clarifying ability and limitation of the code
  - –Conducting studies on not only SA analysis code but various codes
- Applying the PIRT to advance development efficiently
  - –PIRT specifying for the Fukushima accident
  - –Making effective use of previous knowledge for PIRT construction
- Understanding the status inside the Fukushima reactor with concentrating the world's knowledge & experience
  - –Collecting the analysis evaluation to contribute to understanding the debris location
  - –Complying necessary database for the analysis
- Commencing studies toward conducting international benchmark analysis
  - –determining regarding implementation framework such as secretariat.

Focus point for advice or proposal from domestic and international experts

1. Methodology for identifying the status of fuel debris in the reactors applying analysis code

– Plan to perform studies of the most precise possible accident progression with revising analysis code to understand the fuel debris position in the reactor. What is advise from this point?

– Plan to conduct studies of proceeding code revision by PIRT methodology with applying previous study of EURSAFE. What is advise from this point?

– Plan to obtain information regarding actual fuel debris with performing analysis in not only the SA analysis but the various resolution. How can we consider the relations among the analyses?

2. Upgrading SA analysis code

3. SA Mock Examination

Focus point for advice or proposal from domestic and international experts

1. Methodology for understanding the status of fuel debris in the reactor applying analysis code

2. Upgrading SA analysis code

- What do you think that we plan to proceed with the both analysis codes （MAAP, SAMPSON based on mechanistic）?
- Which one is suitable to understand precise location and aspect of fuel debris among MELCOR, ASTEC or other SA analysis codes?
- Is it possible to apply the suitable aspect of the SA analysis codes described above to put effort into revising the MAAP, SAMPSON?
- What do you think that we plan to organize international benchmark problem in Fukushima accident?

3. SA Mock Examination

Focus point for advice or proposal from domestic and international experts

1. Methodology for understanding the status of fuel debris in the reactor applying analysis code

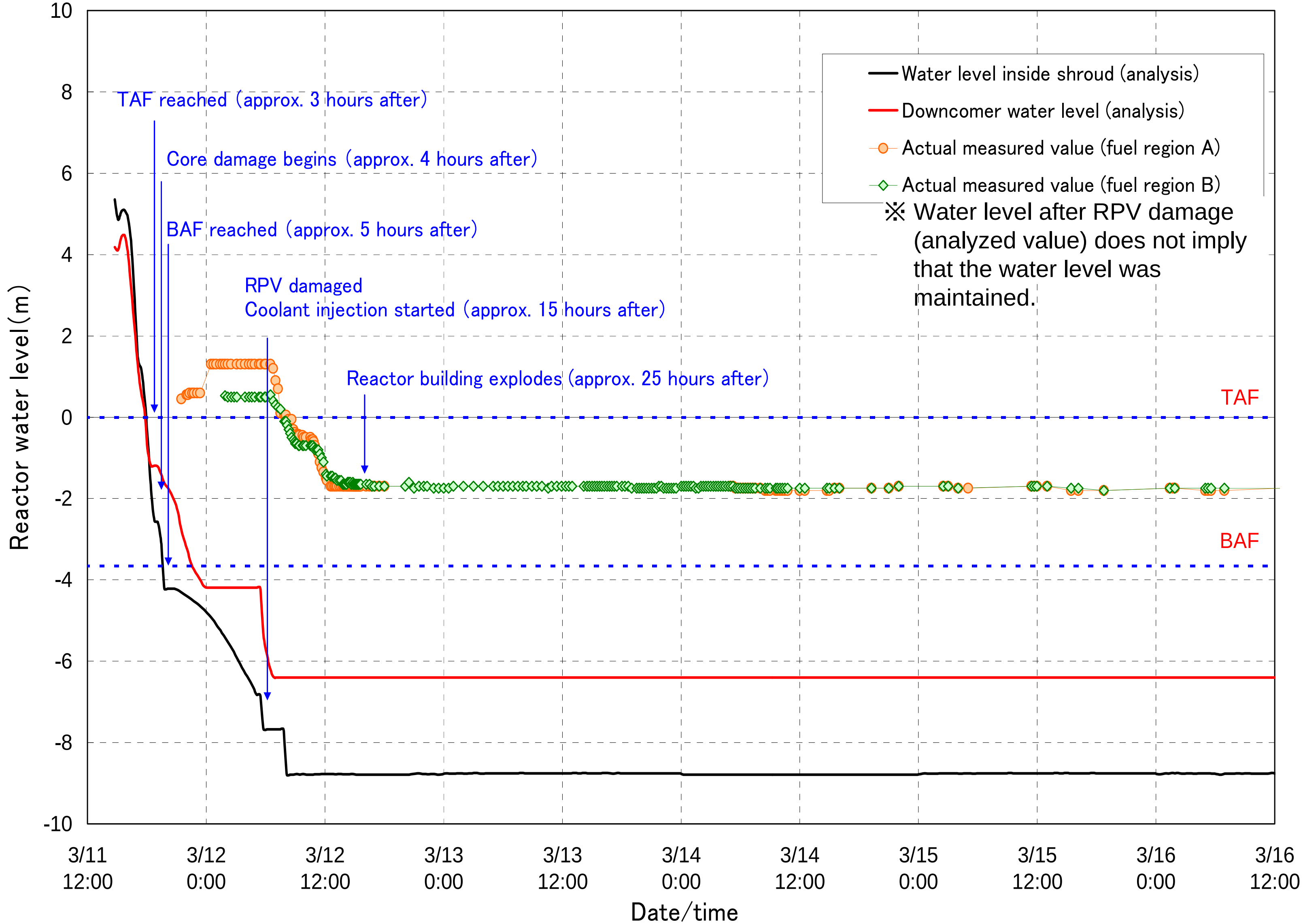
2. Upgrading SA analysis code

3. SA Mock Examination

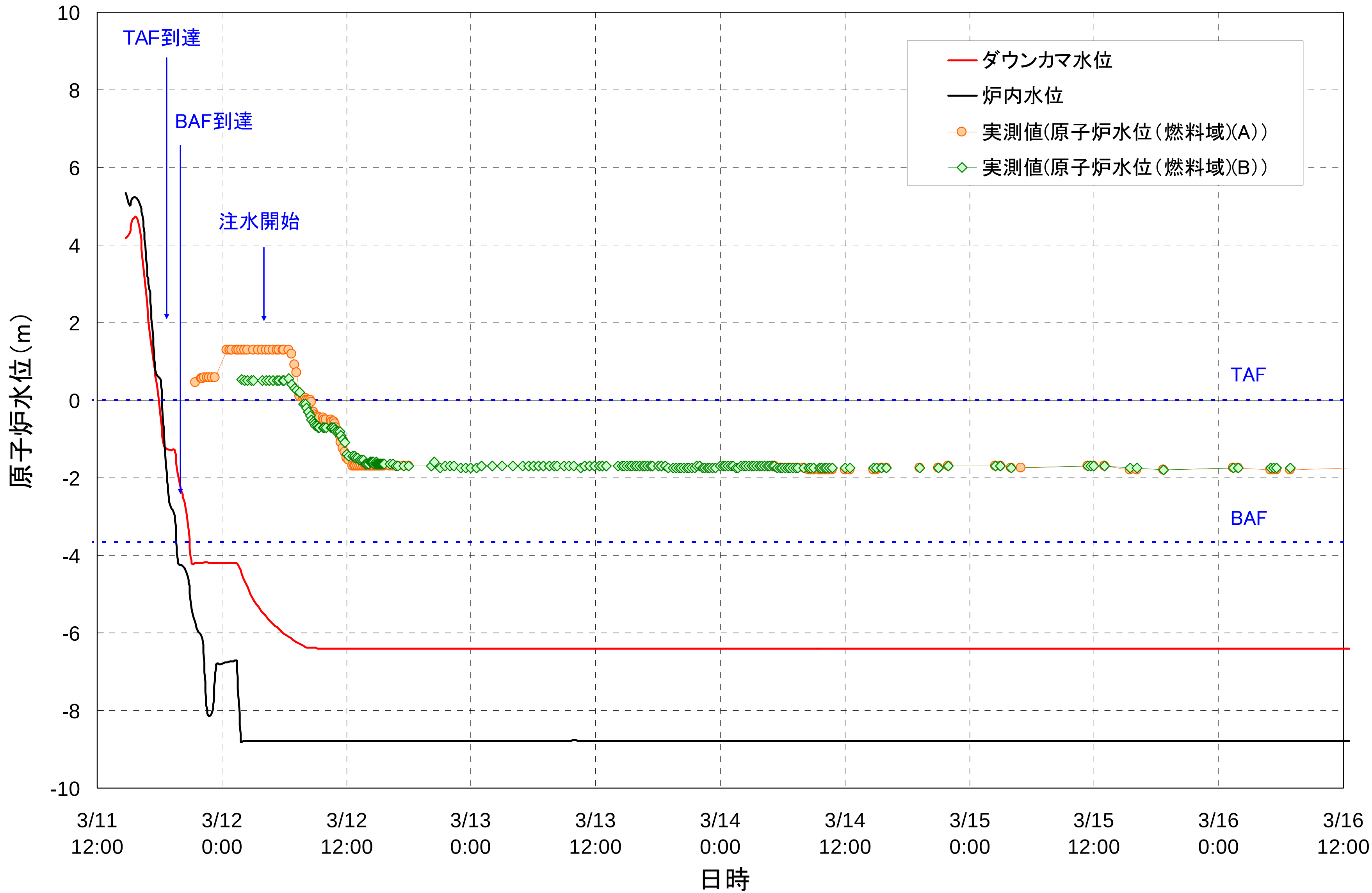
- How much SA mock examination is required for revising the above analysis code?
- What is to especially refer among the SA mock examination previously implemented in the US and Europe?
- Are there many research institutes which are interested in proceeding the SA mock experiment as international corporation project?

Appendix

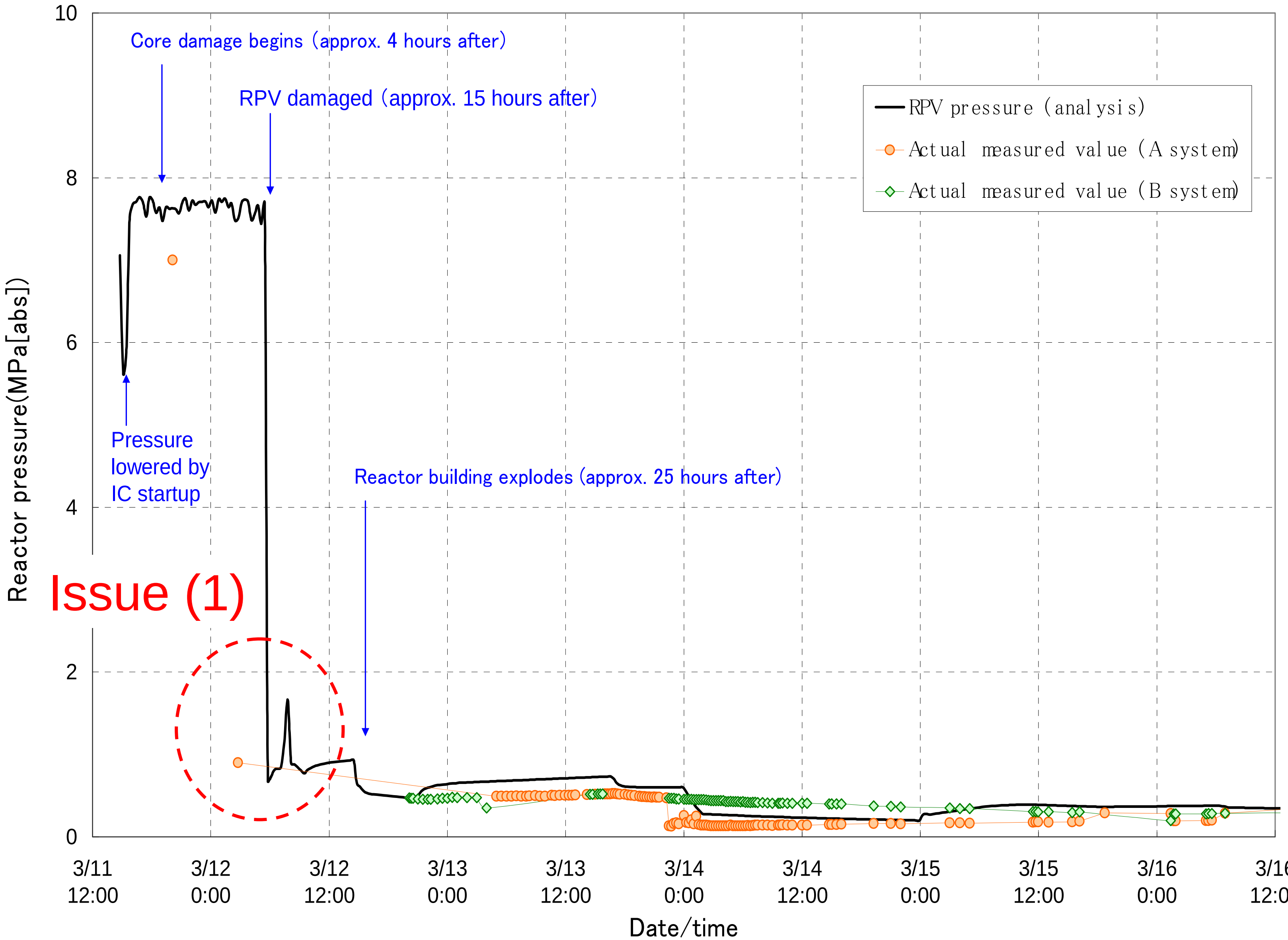
Unit 1 Analysis Results (Reactor Water Level)：March 23rd



Unit 1 Analysis Results (Reactor Water Level)：latest ver.



Unit 1 Analysis Results (Reactor Pressure) ：March 23rd



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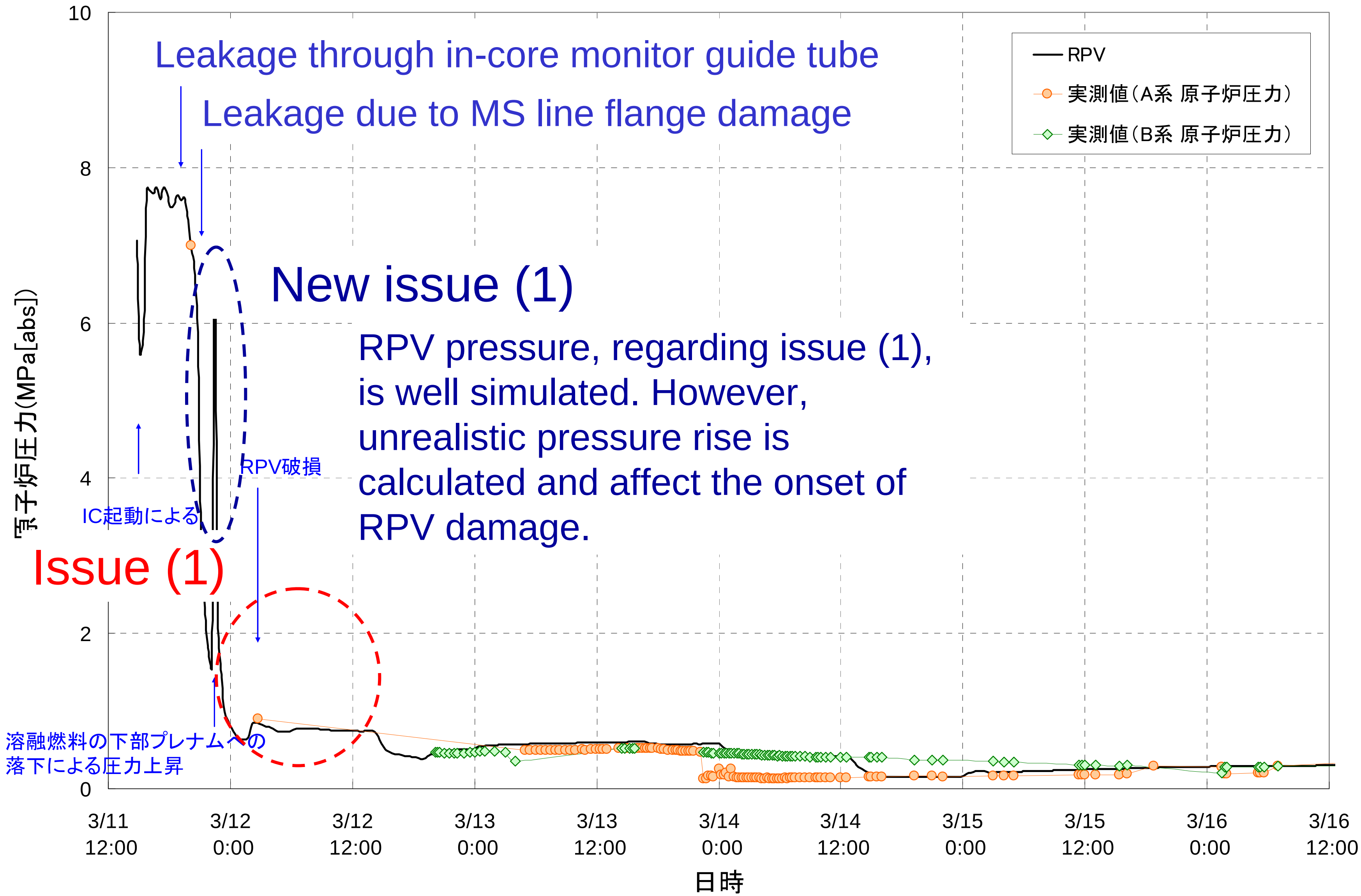
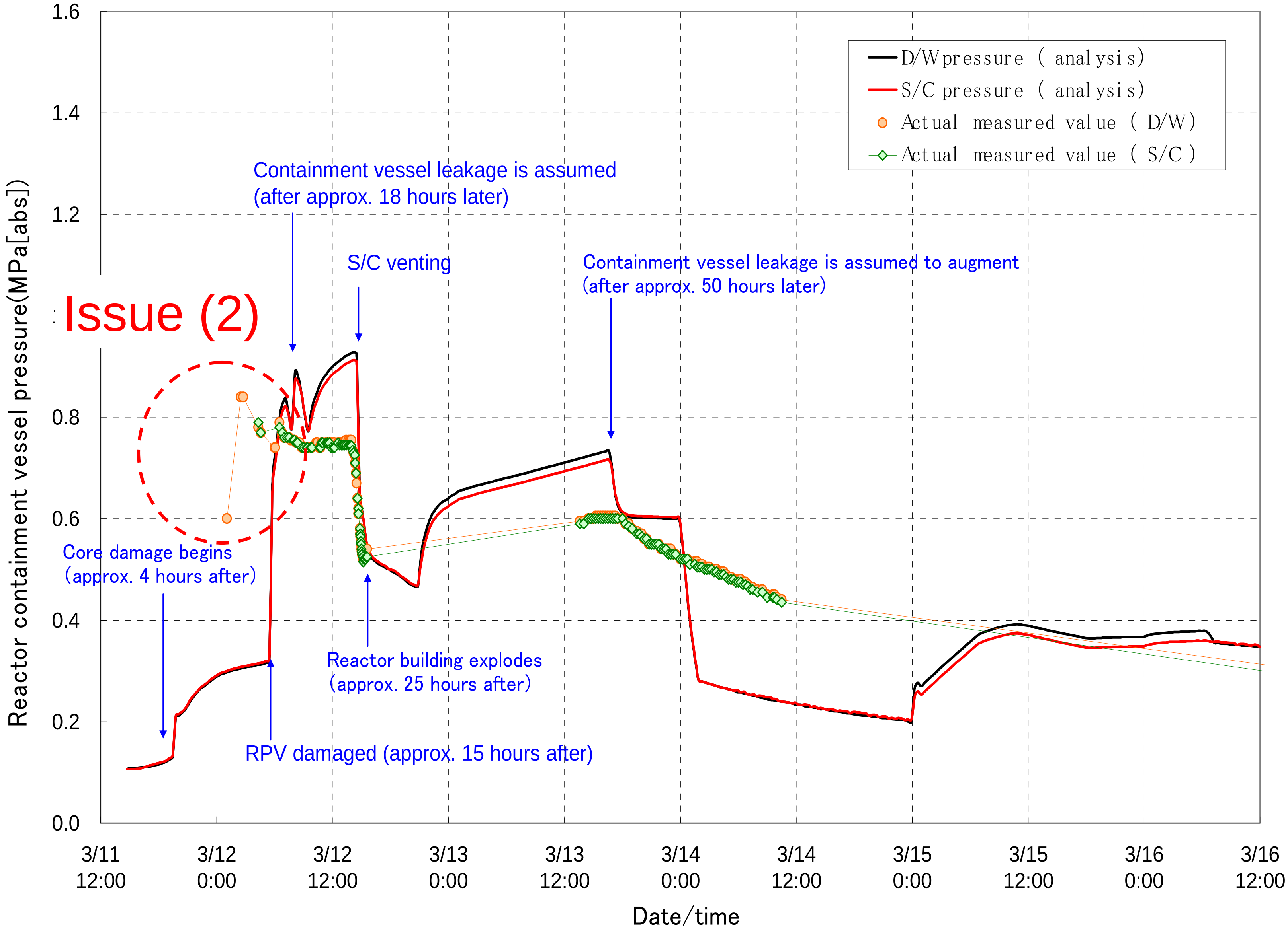
Unit 1 Analysis Results (Reactor Pressure) ： latest ver.



Unit 1 Analysis Results (Containment Vessel Pressure) : March 23rd



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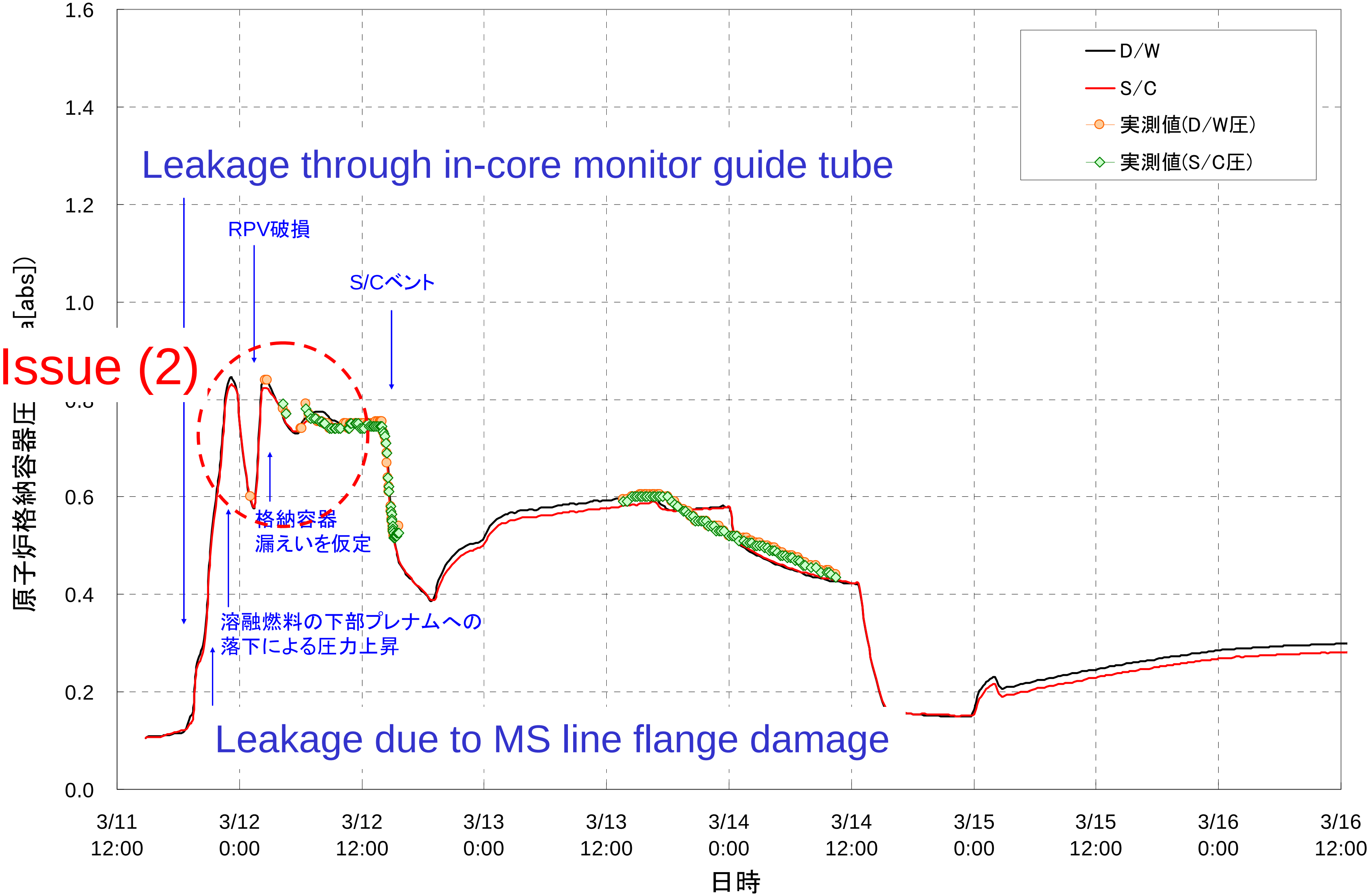
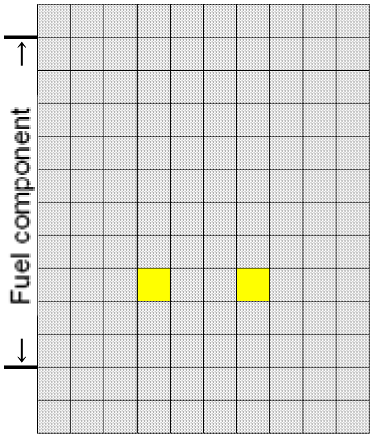
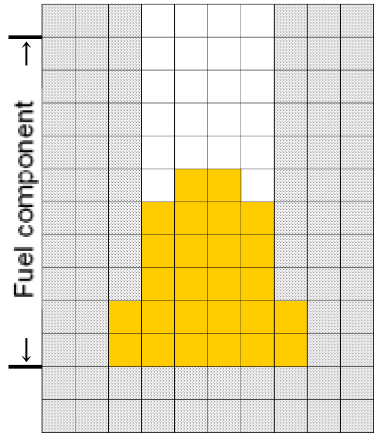
Unit 1 Analysis Results (Containment Vessel Pressure) : latest ver.



Unit 1 Analysis Results – Core Conditions ：March 23rd

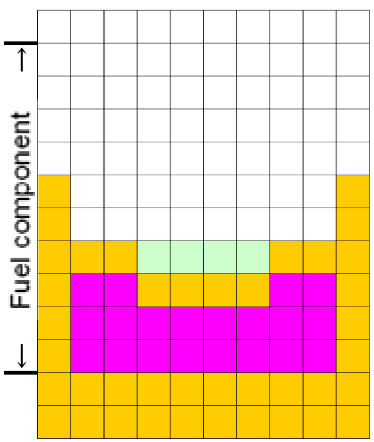


Approx. 4.7 hours after scram

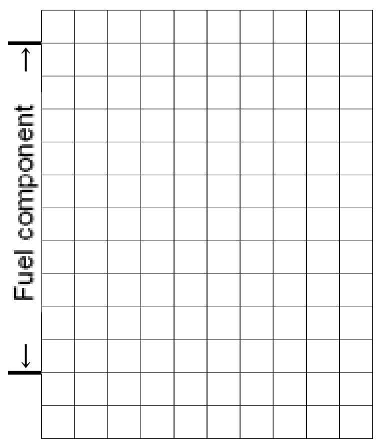


Approx. 5.3 hours after scram

Model of damage states
- : No fuel (collapse)
- : Normal fuel
- : Damaged fuel accumulates (fuel rod form maintained)
- : Melted fuel flows down the cladding surface and cools and solidifies on the control rod surface, increasing the diameter of the control rods
- : Fuel rod diameter further increases and flow channel is blocked by fuel
- : Melt pool formed

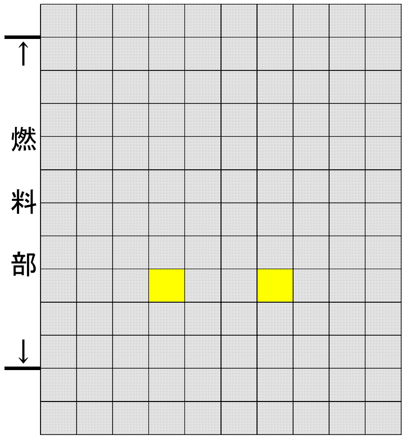


Approx. 14.3 hours after scram (just prior to reactor pressure vessel damage)

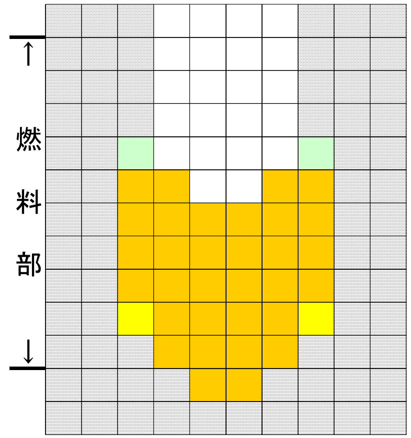


Approx. 15 hours after scram (reactor pressure vessel damaged)

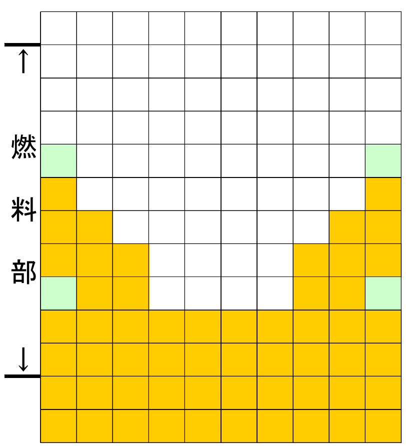
Unit 1 Analysis Results – Core Conditions : latest ver.



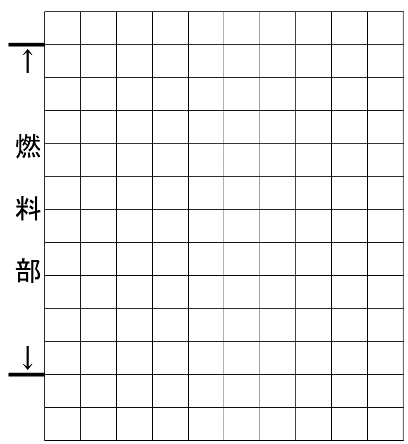
Approx. 4.8 hours after scram



Approx. 5.5 hours after scram



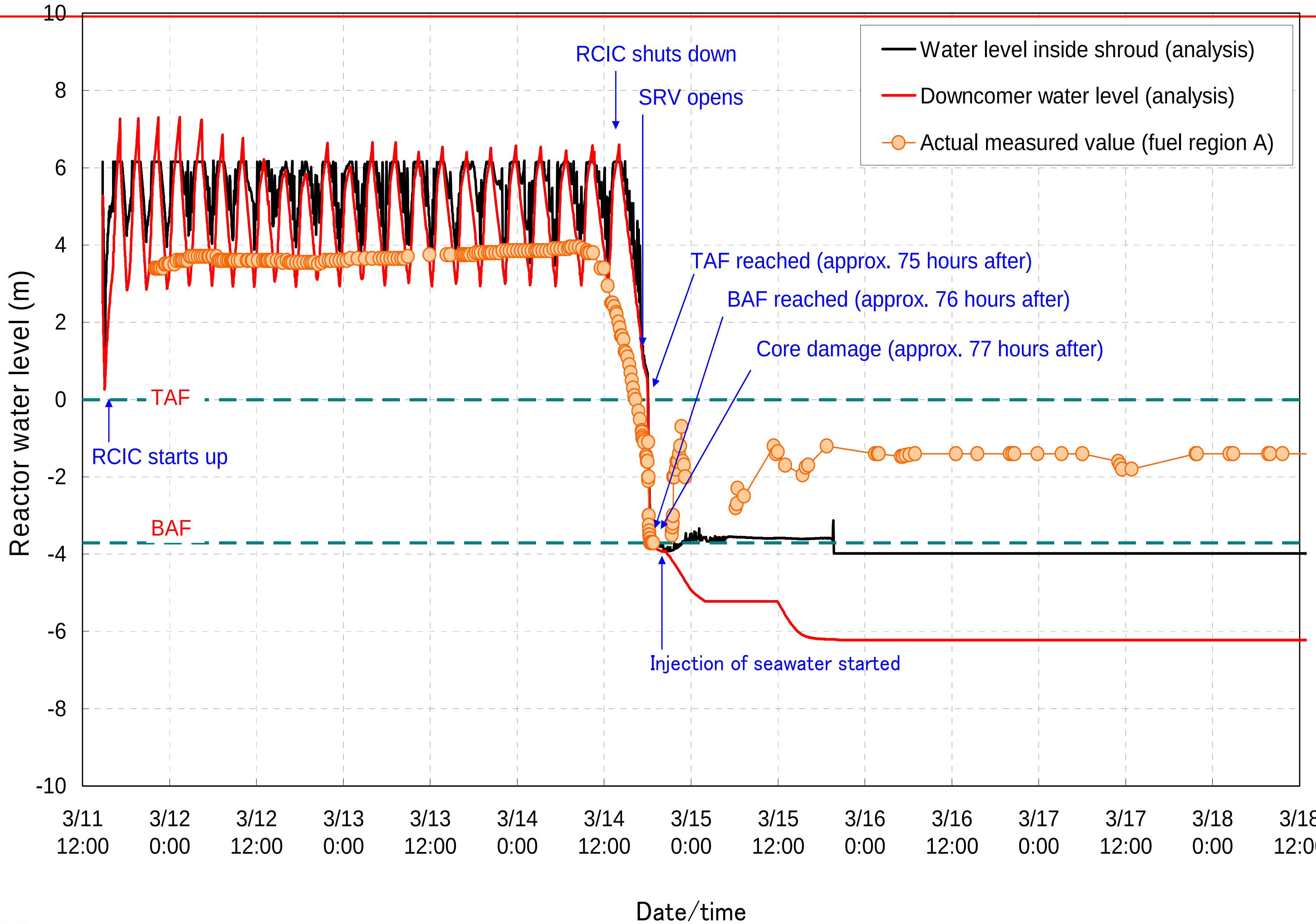
Approx. 7.6 hours after scram



Approx. 8.6 hours after scram

Model of damage states
- : No fuel (collapse)
- : Normal fuel
- : Damaged fuel accumulates (fuel rod form maintained)
- : Melted fuel flows down the cladding surface and cools and solidifies on the control rod surface, increasing the diameter of the control rods
- : Fuel rod diameter further increases and flow channel is blocked by fuel
- : Melt pool formed

Unit 2 Analysis Results (Reactor Water Level) :March 23rd



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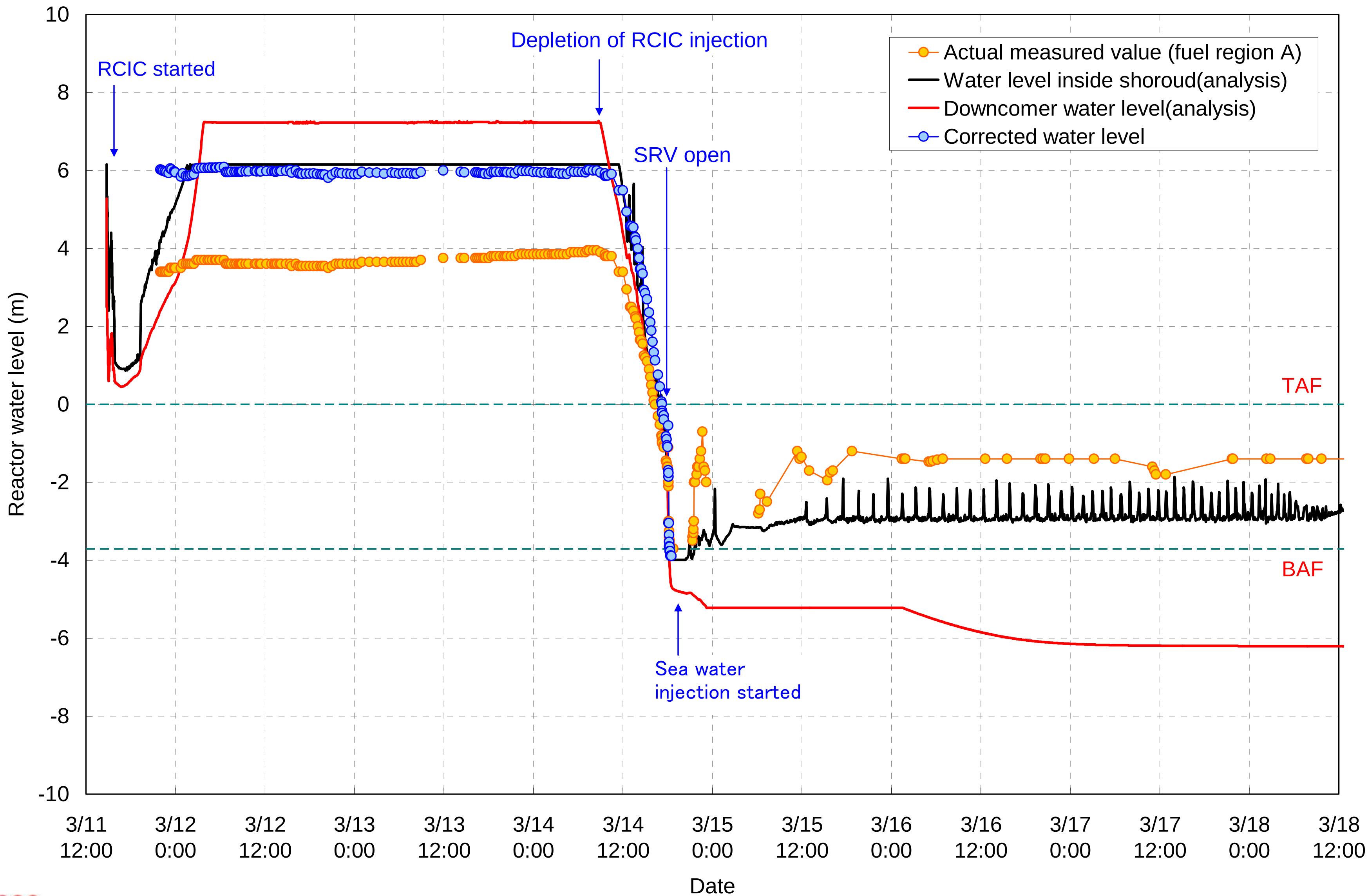
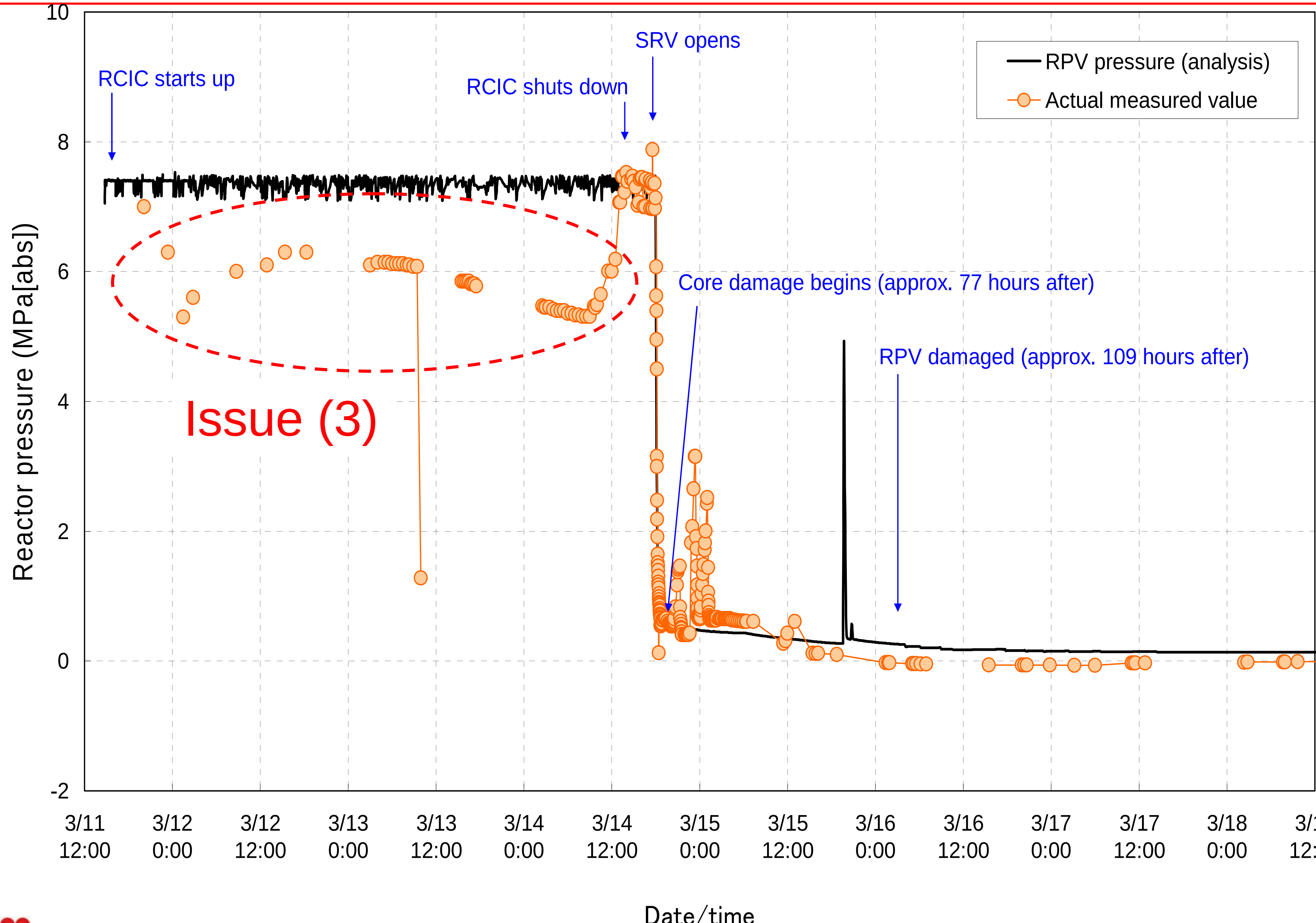
Unit 2 Analysis Results (Reactor Water Level) ： latest ver.



Unit 2 Analysis Results (Reactor Pressure) : March 23rd



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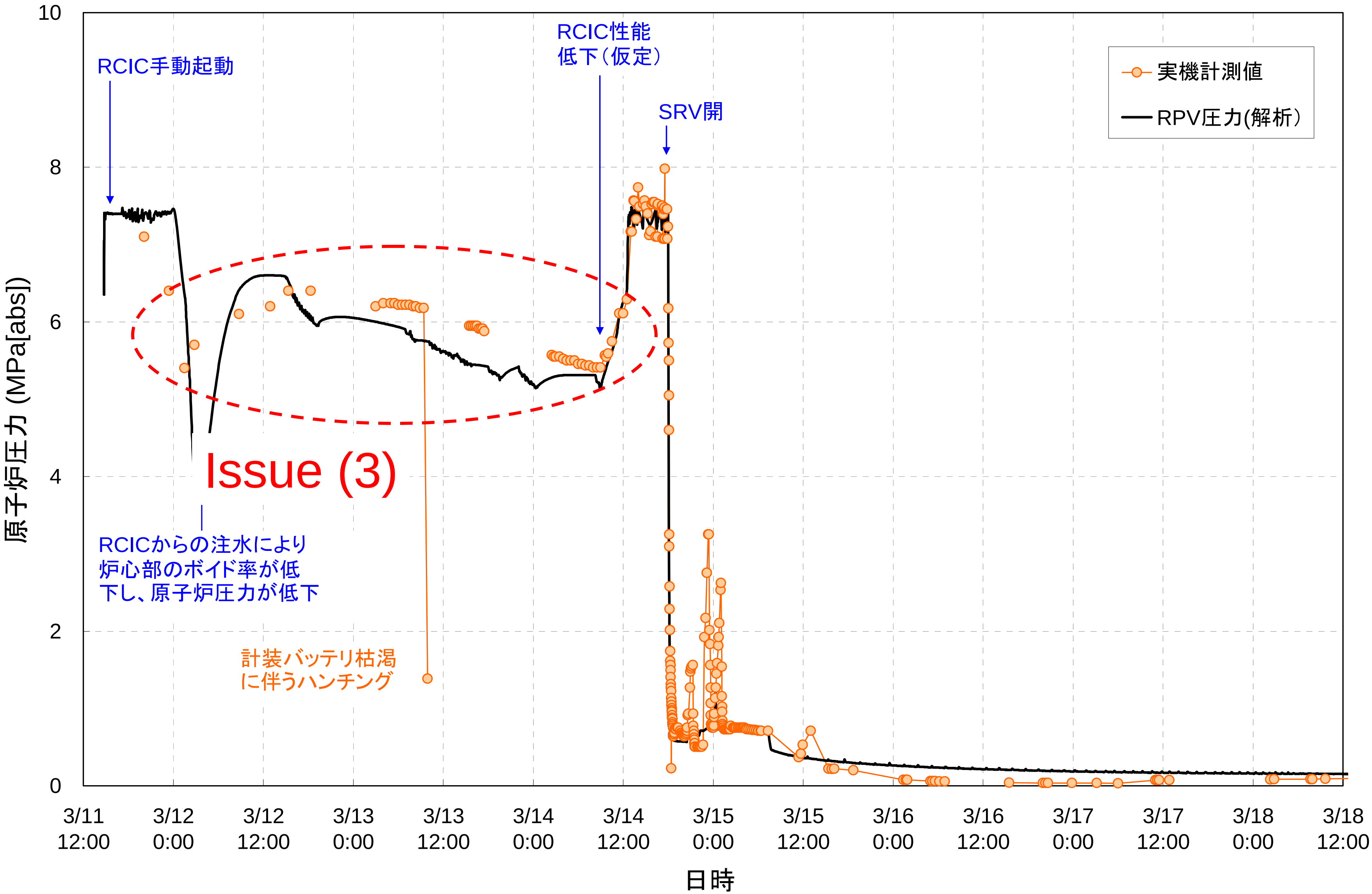
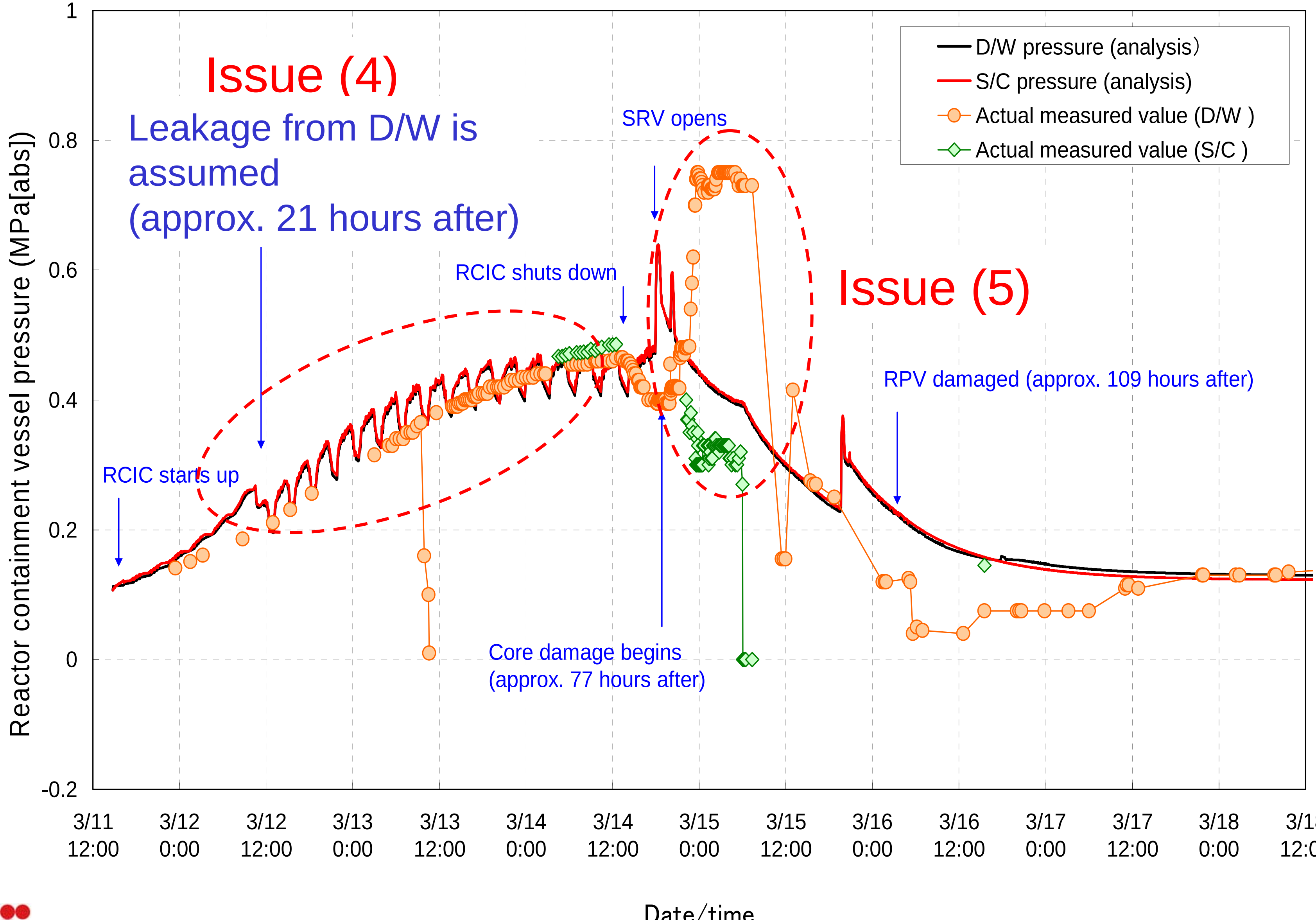
Unit 2 Analysis Results (Reactor Pressure) : latest ver.



Unit 2 Analysis Results (Containment Vessel Pressure) : March 23rd



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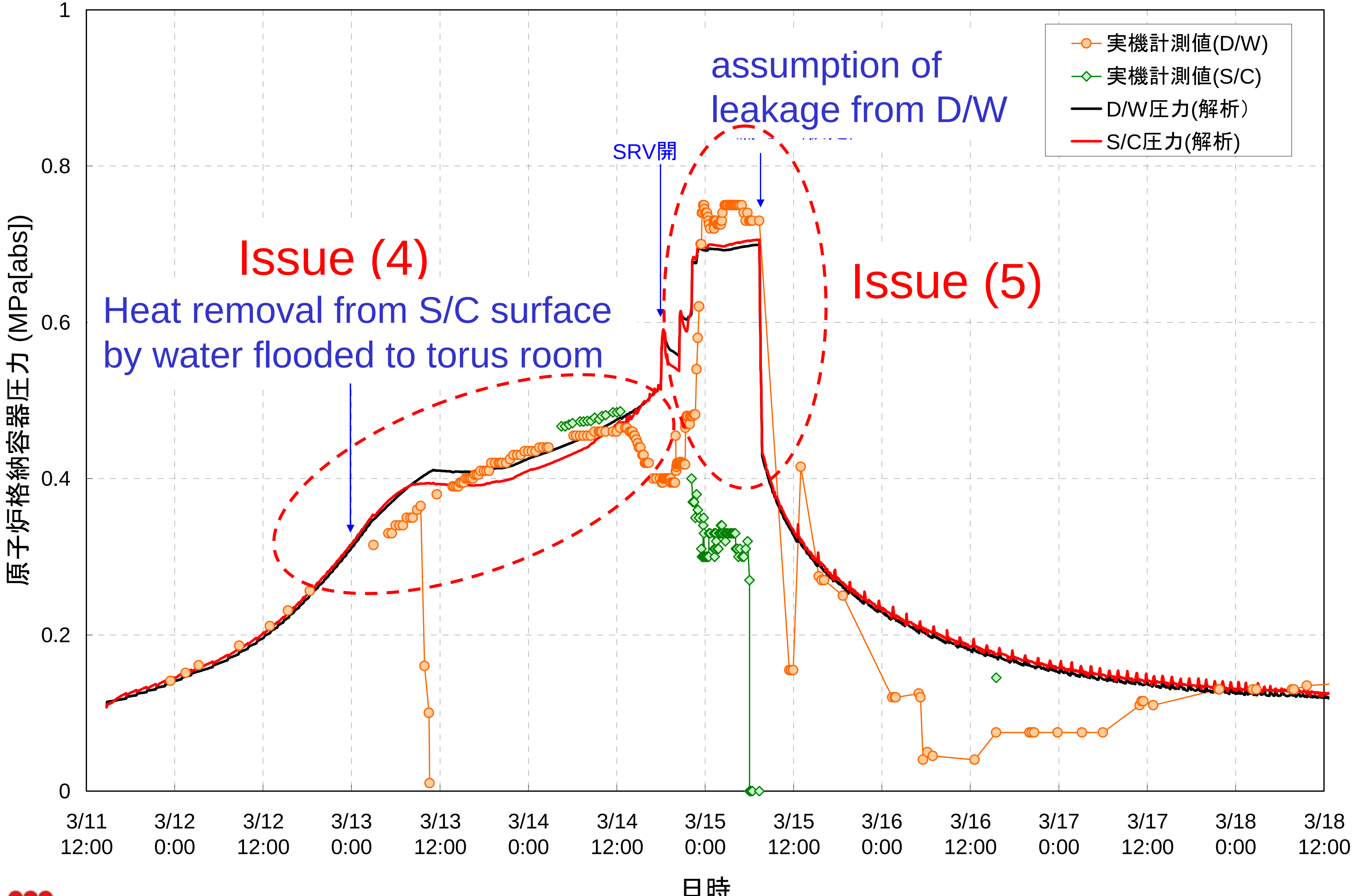
Unit 2 Analysis Results (Containment Vessel Pressure)：latest ver.



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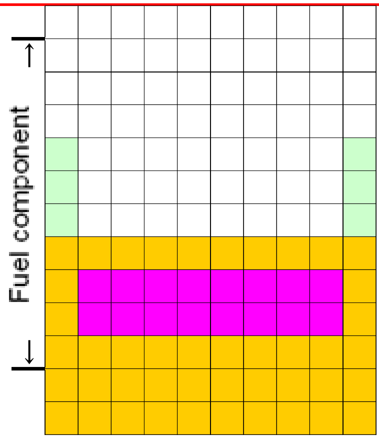
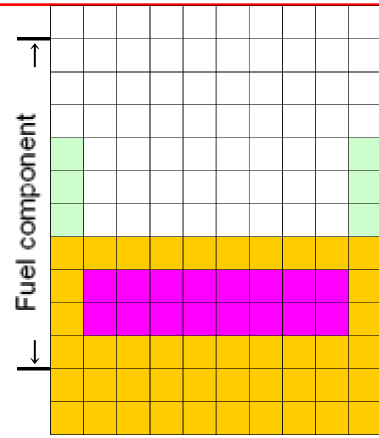
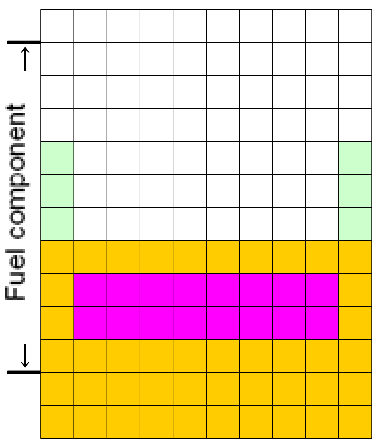
Unit 2 Analysis Results – Core Conditions：March 23rd



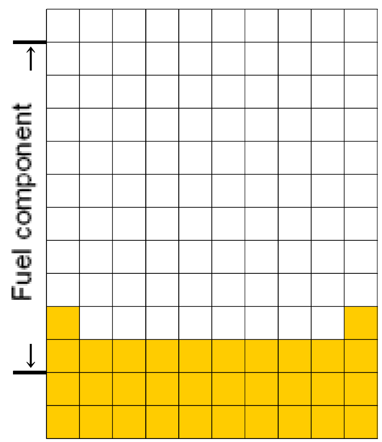
Approx. 87 hours after scram



Approx. 96 hours after scram



Approx. 100 hours after scram (just prior to reactor pressure vessel damage)



Approx. 109 hours after scram (reactor pressure vessel damaged)

Model of damage states

- : No fuel (collapse)
- : Normal fuel
- : Damaged fuel accumulates (fuel rod form maintained)
- : Melted fuel flows down the cladding surface and cools and solidifies on the control rod surface, increasing the diameter of the control rods
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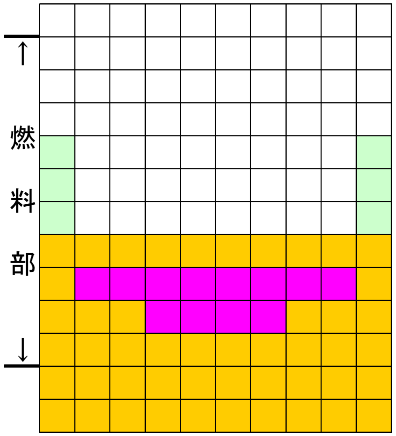
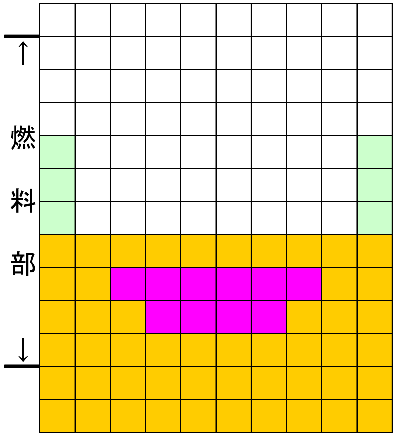
- : Melt pool formed

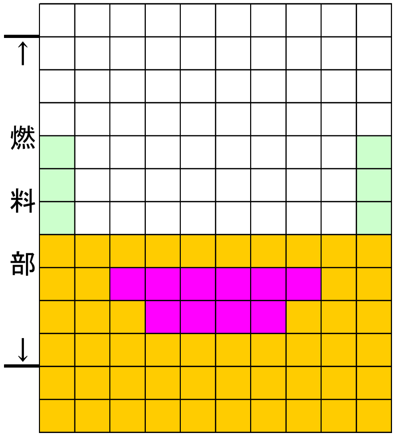
Unit 2 Analysis Results – Core Conditions : latest ver.



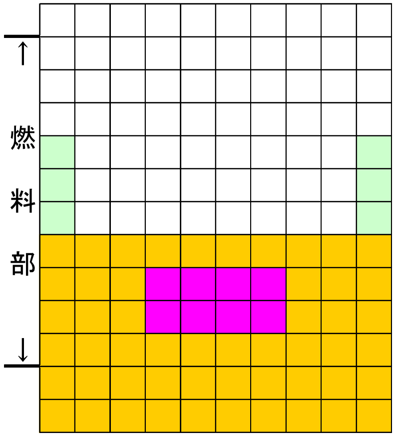
Approx. 96 hours after scram



Approx. 120 hours after scram



Approx. 144 hours after scram



Approx. 168 hours after scram

Model of damage states

- : No fuel (collapse)
- : Normal fuel
- : Damaged fuel accumulates (fuel rod form maintained)
- : Melted fuel flows down the cladding surface and cools and solidifies on the control rod surface, increasing the diameter of the control rods
- : Fuel rod diameter further increases and flow channel is blocked by fuel
- : Melt pool formed

New issue (2)

Plant behavior before core melt is well simulated. However, calculation results show no damage on RPV.

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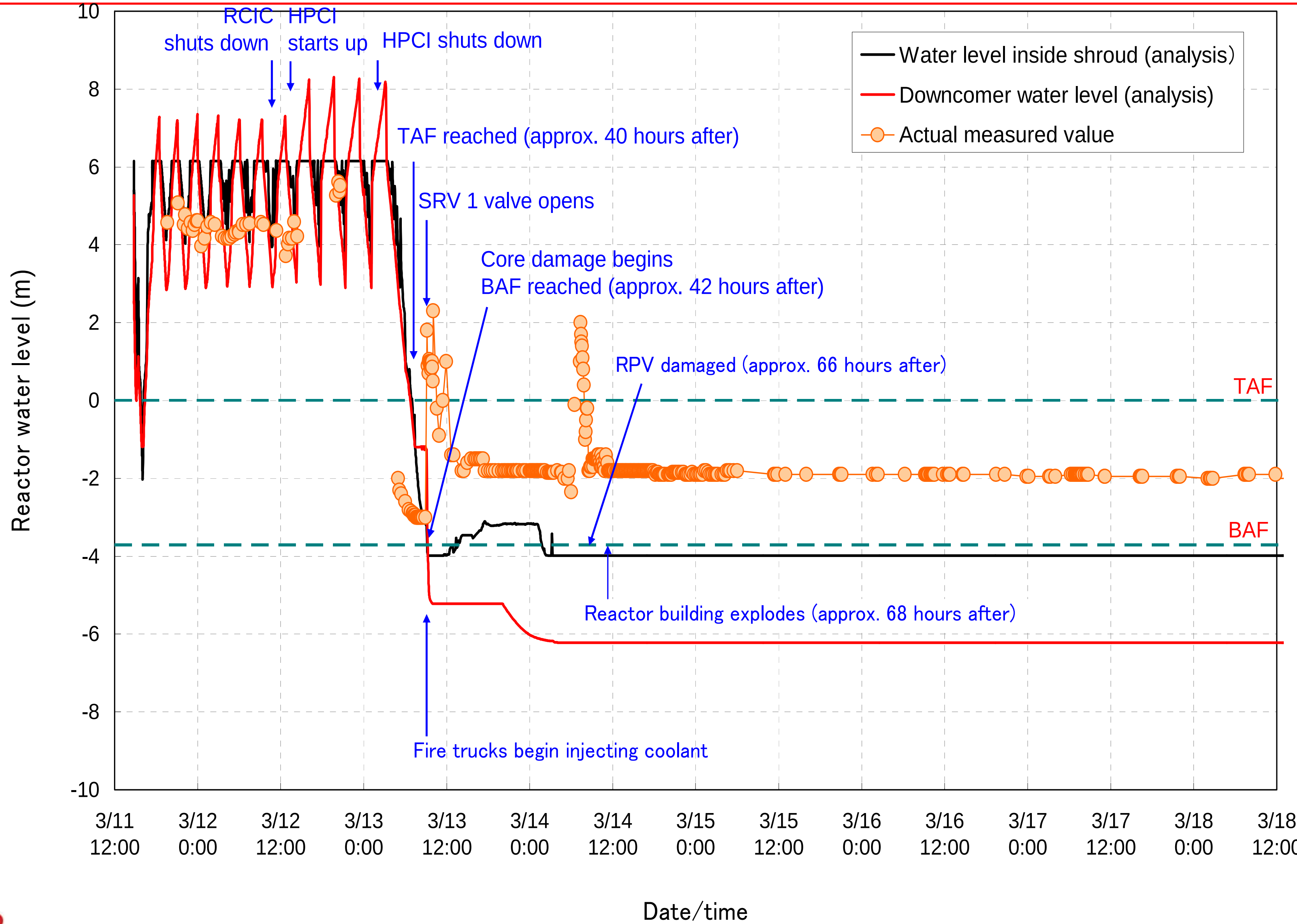
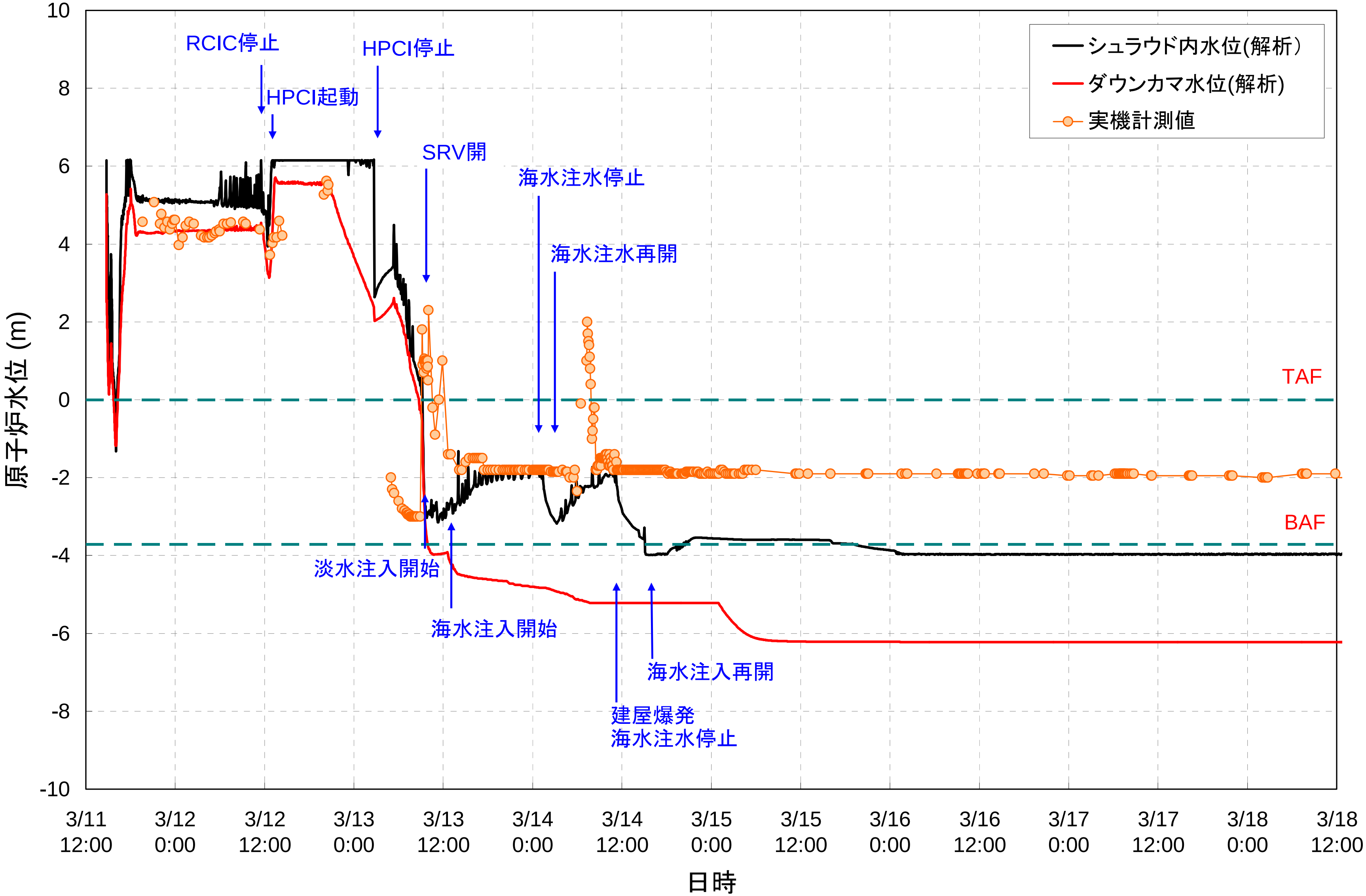
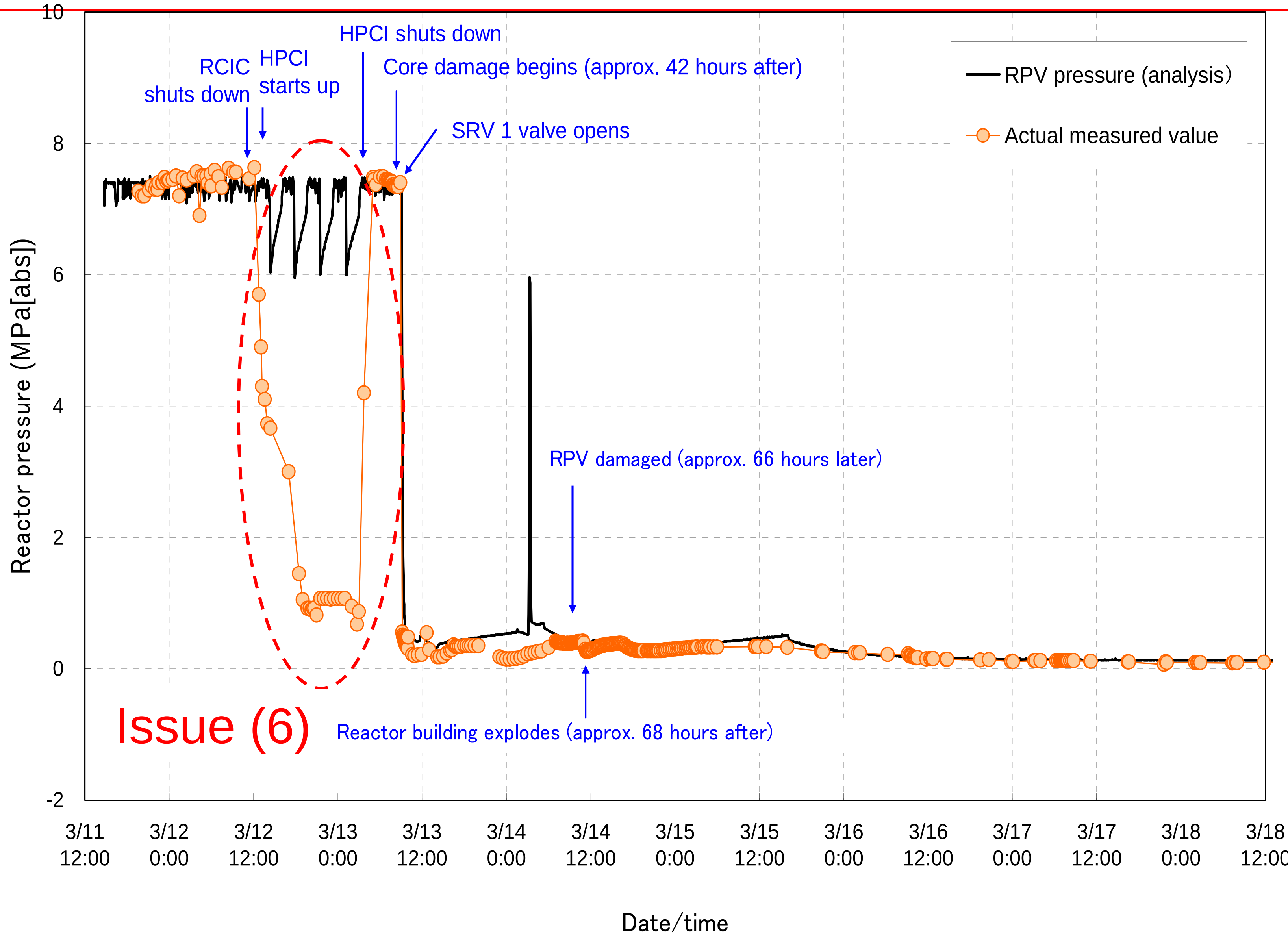
Unit 3 Analysis Results (Reactor Water Level) : March 23rd



Unit 3 Analysis Results (Reactor Water Level) : latest ver.



Unit 3 Analysis Results (Reactor Pressure) : March 23rd



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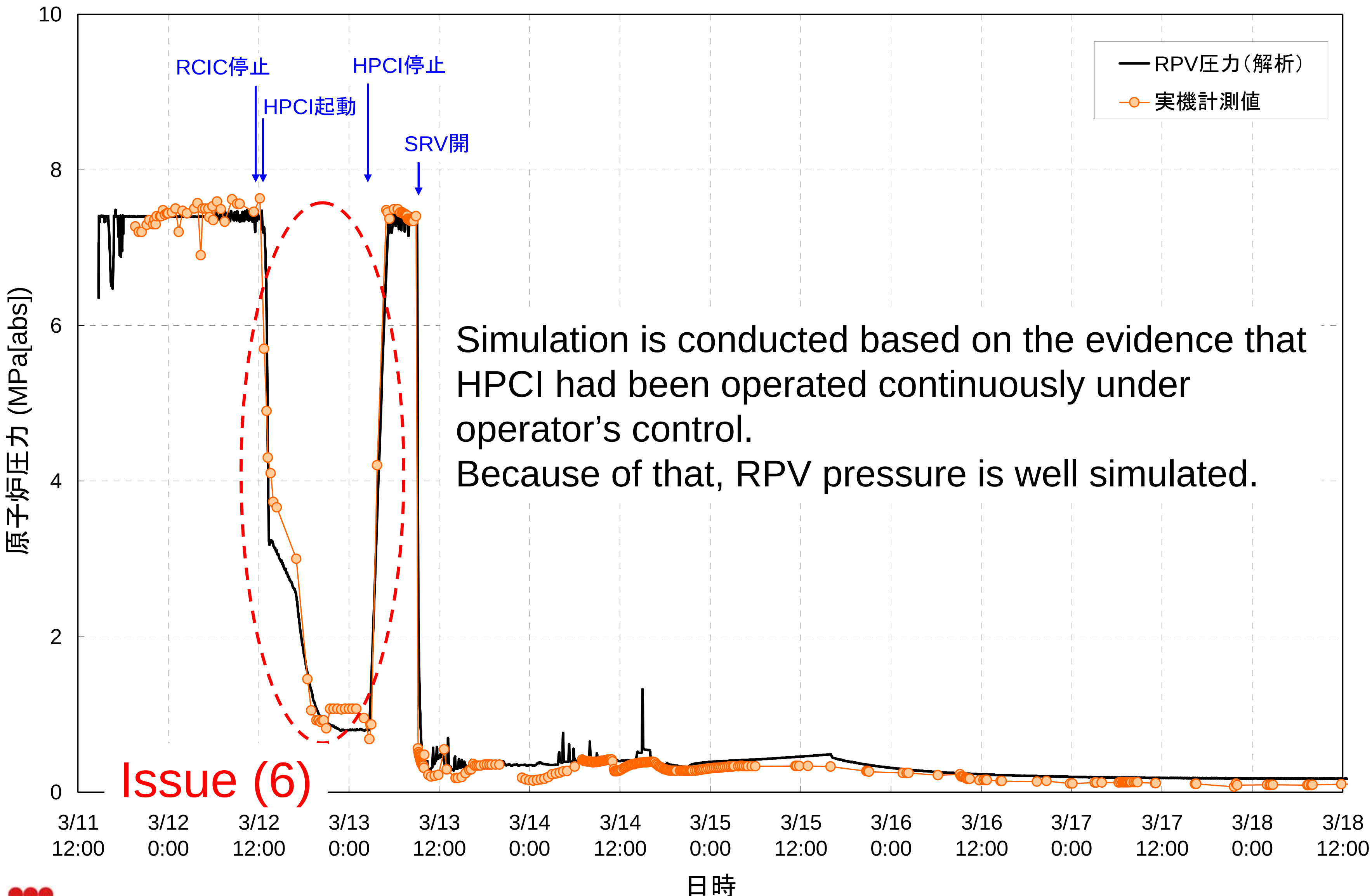
Unit 3 Analysis Results (Reactor Pressure) : latest ver.



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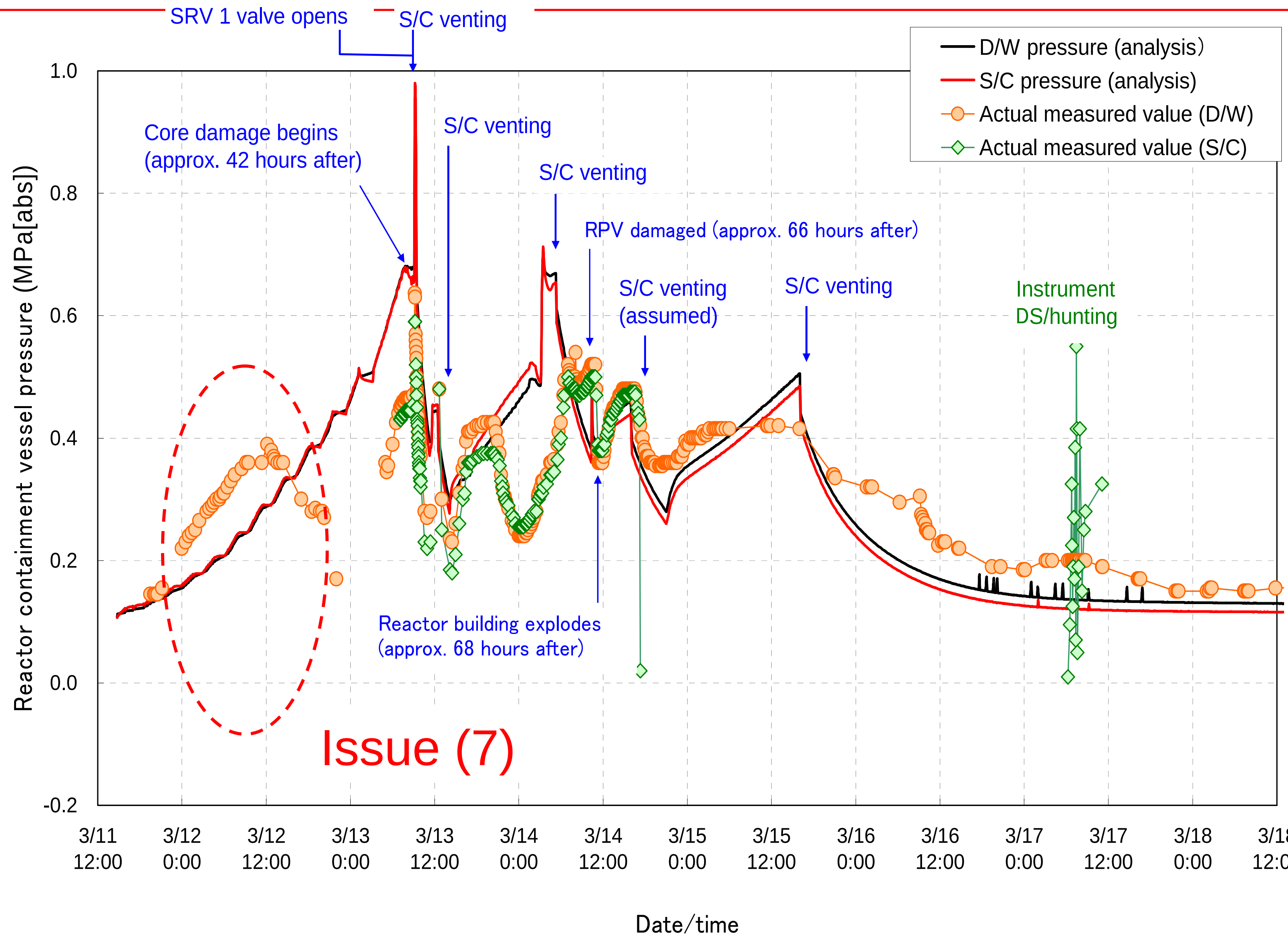
Unit 3 Analysis Results (Containment Vessel Pressure) : March 23rd



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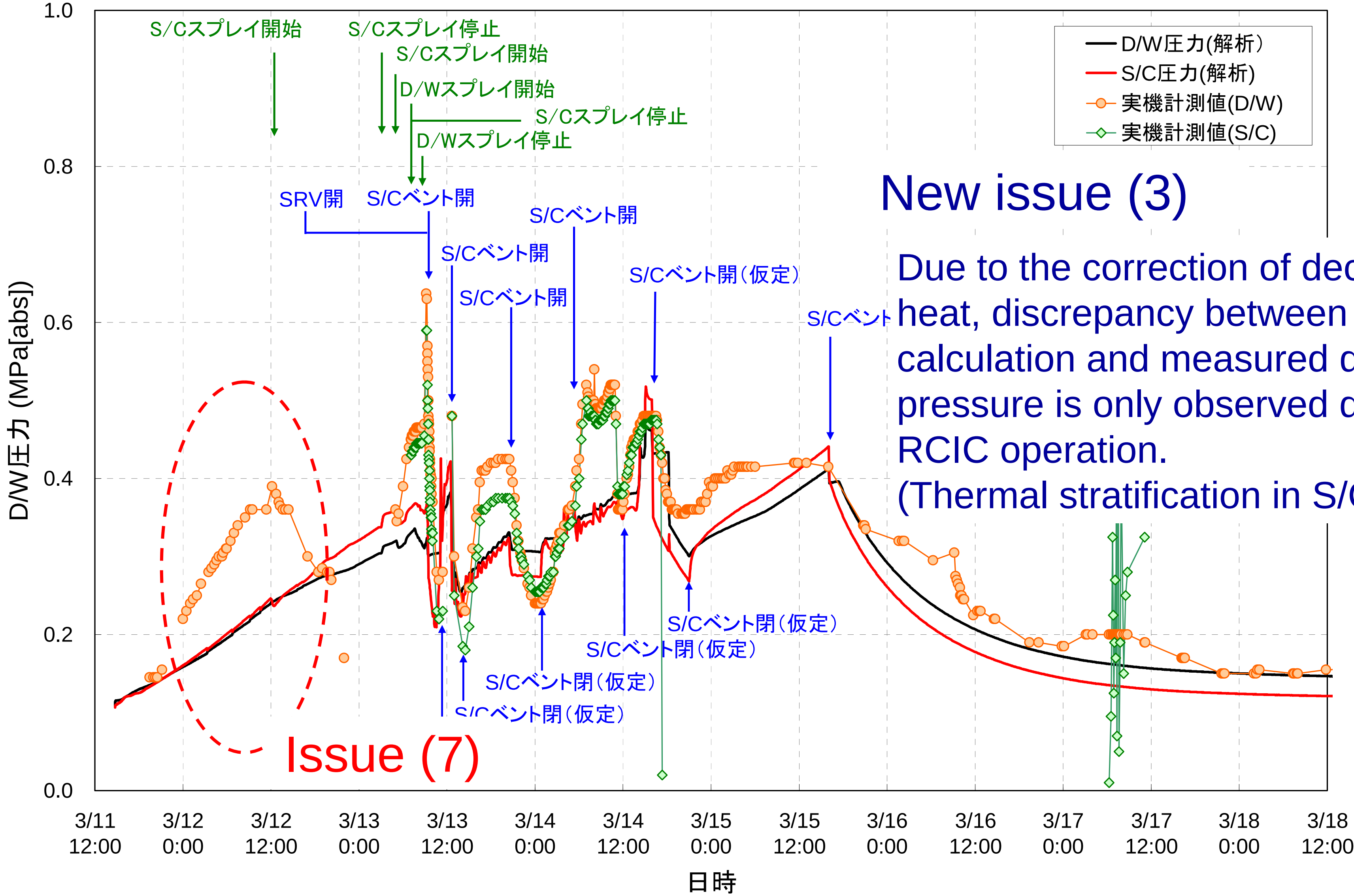
Unit 3 Analysis Results (Containment Vessel Pressure) : latest ver.



New issue (3)

Due to the correction of decay heat, discrepancy between calculation and measured drywell pressure is only observed during RCIC operation.
(Thermal stratification in S/C?)

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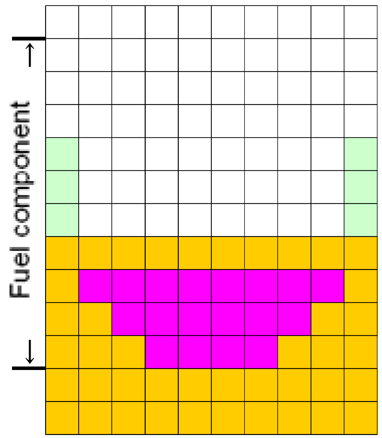
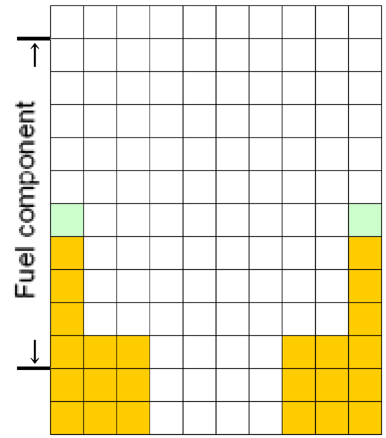
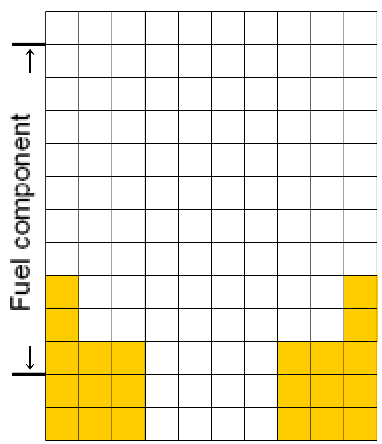
Unit 3 Analysis Results – Core Conditions ：March 23rd



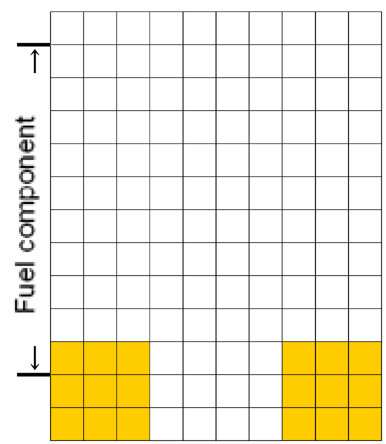
Approx. 58 hours after scram



Approx. 62 hours after scram
(just prior to reactor pressure vessel damage)



Approx. 66 hours after scram
(reactor pressure vessel damaged)



Approx. 96 hours after scram

Model of damage states

: No fuel (collapse)

: Normal fuel

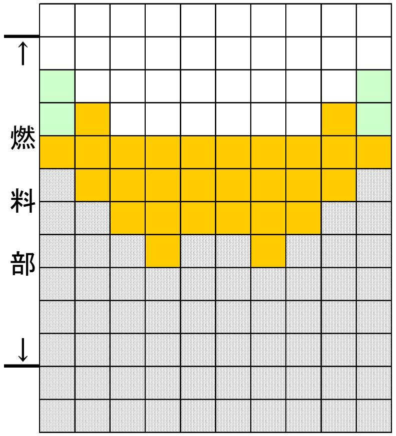
: Damaged fuel accumulates (fuel rod form maintained)

: Melted fuel flows down the cladding surface and cools and solidifies on the control rod surface, increasing the diameter of the control rods

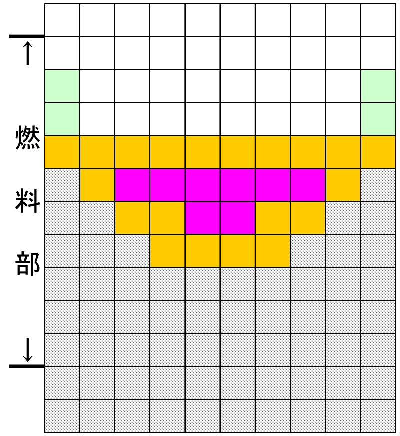
: Fuel rod diameter further increases and flow channel is blocked by fuel

: Melt pool formed

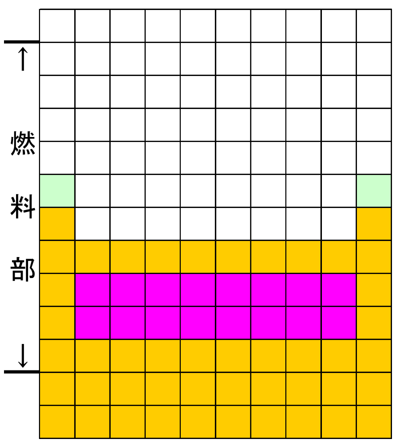
Unit 3 Analysis Results – Core Conditions : latest ver.



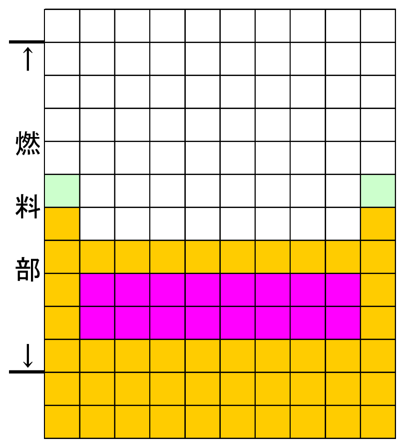
Approx. 48 hours after scram



Approx. 72 hours after scram



Approx. 96 hours after scram



Approx. 168 hours after scram

Model of damage states

: No fuel (collapse)

: Normal fuel

: Damaged fuel accumulates (fuel rod form maintained)

: Melted fuel flows down the cladding surface and cools and solidifies on the control rod surface, increasing the diameter of the control rods

: Fuel rod diameter further increases and flow channel is blocked by fuel

: Melt pool formed

New issue (4)

Plant behavior before core melt is well simulated. However, calculation results show no damage on RPV.

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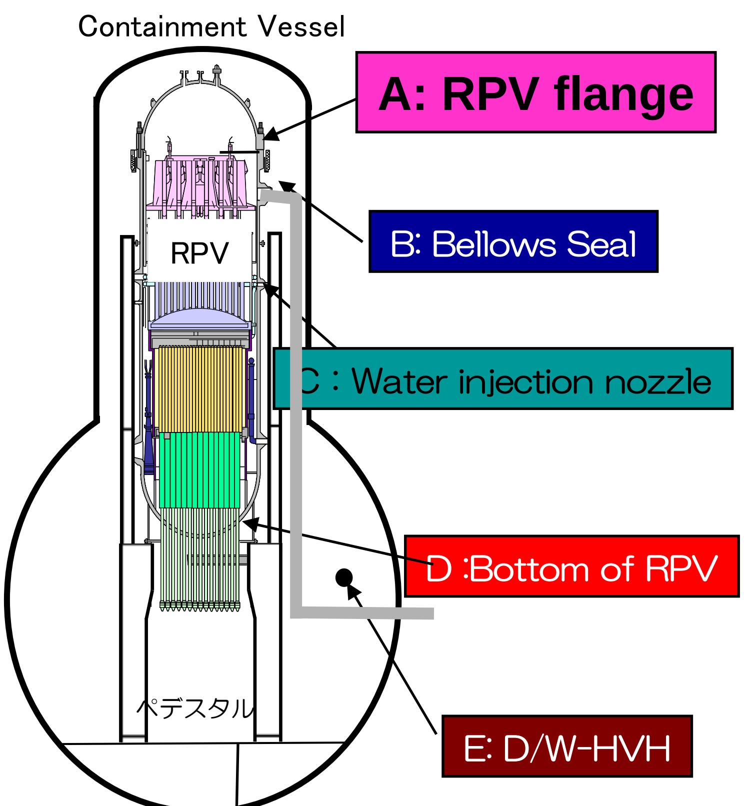
Behavior of temperatures and pressures at Unit 1 (Whole behavior : Mar. to Nov.)



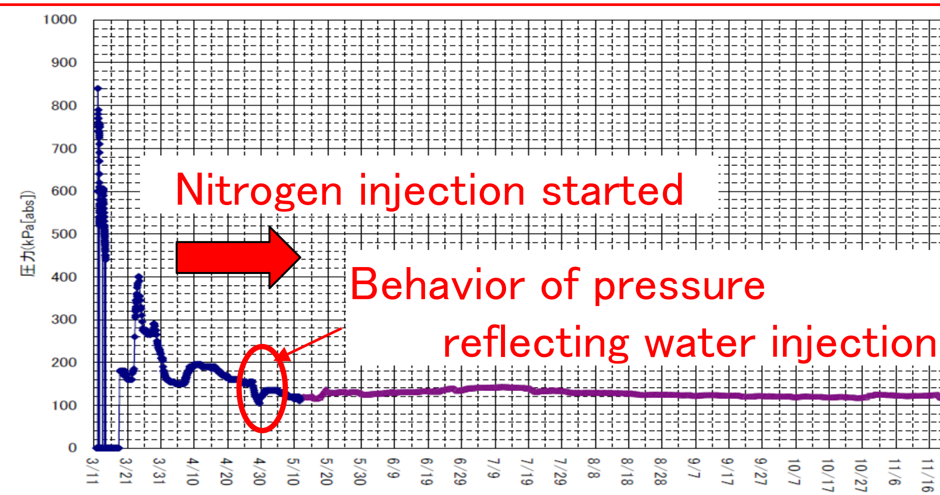


Fig.1 Behavior of D/W pressures after the accident

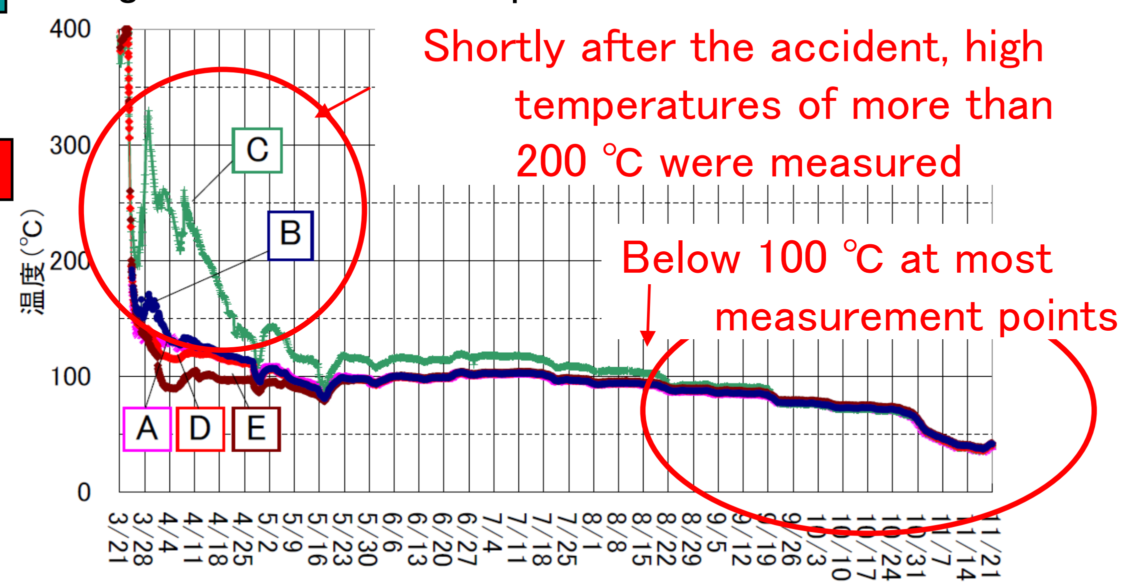
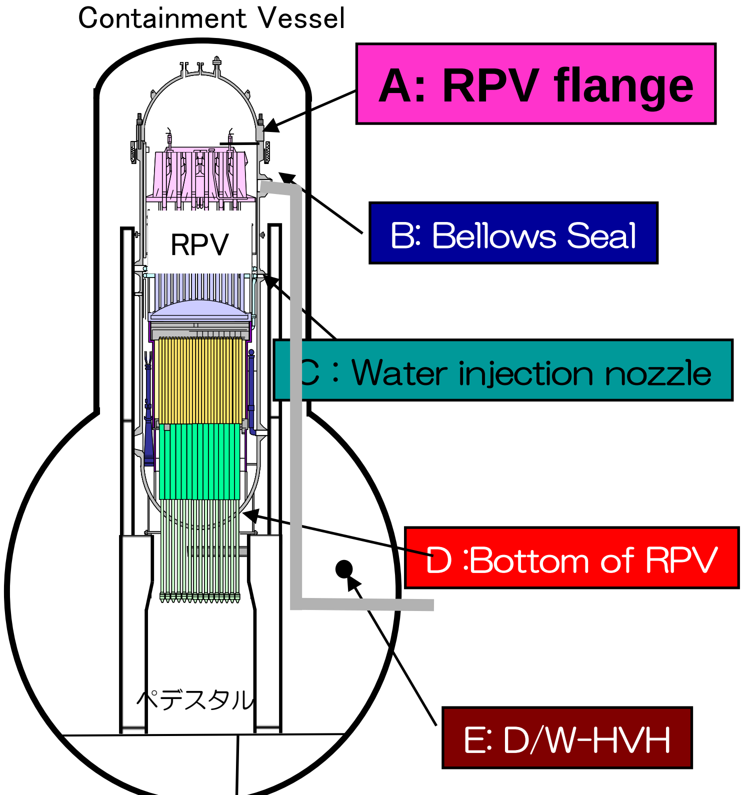


Fig.2 Behavior of temperatures after the accident

By FDW injection without direct injection to core, temperature became lower than 100 ℃

Behavior of temperatures and pressures at Unit 2 （Whole behavior : Mar. to Nov.）



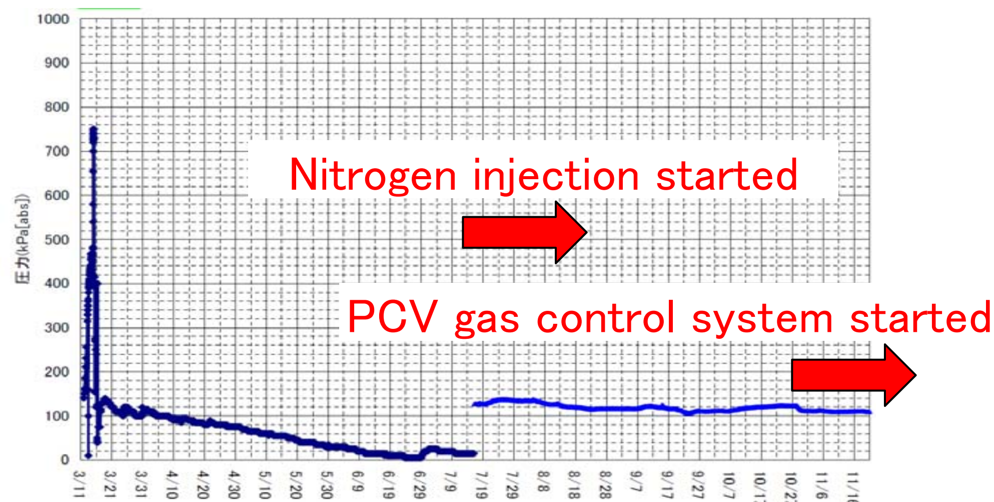


Fig.1 Behavior of D/W pressures after the accident

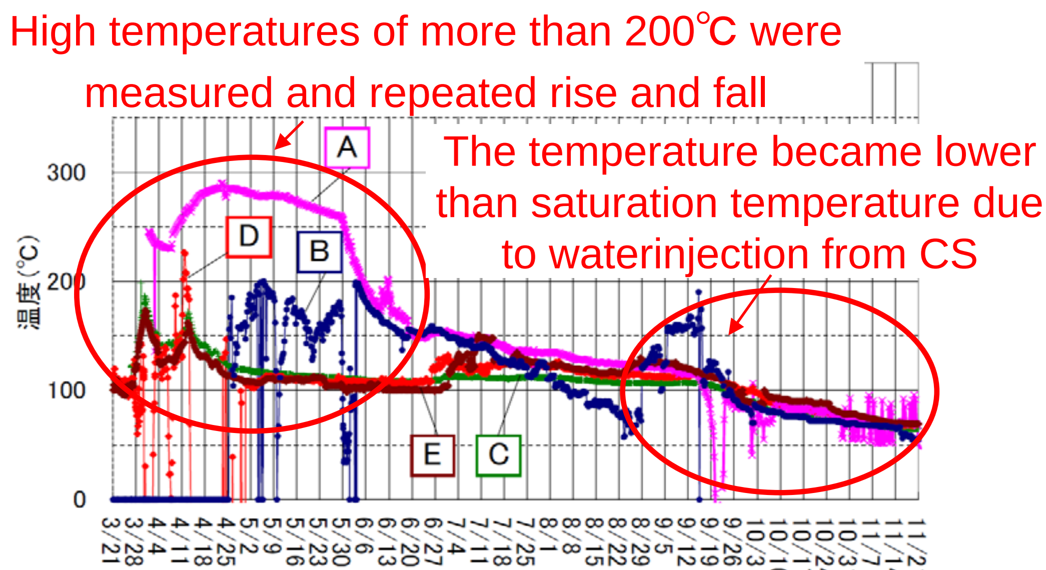
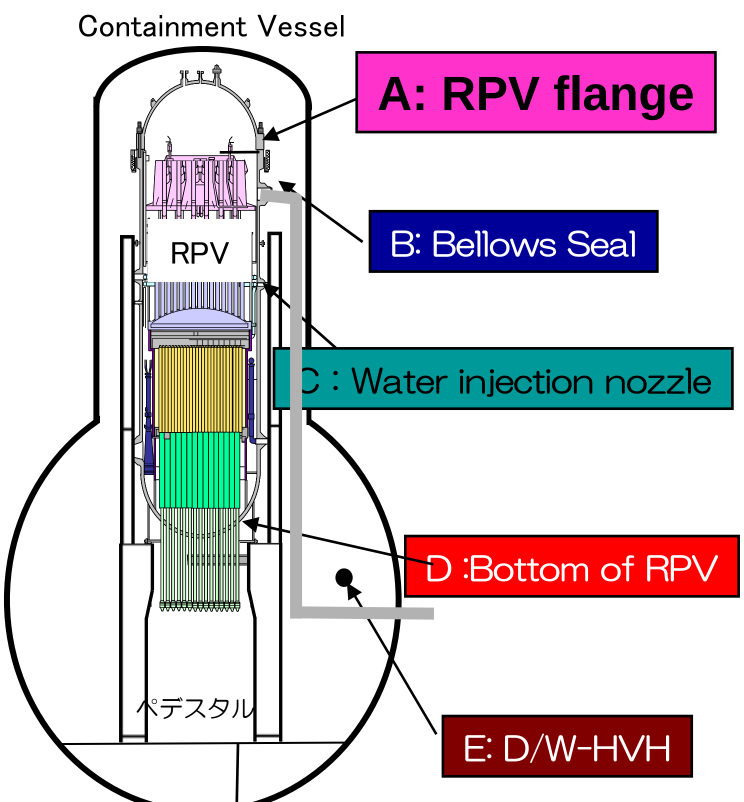


Fig.2 Behavior of temperatures after the accident

After direct injection to core from CS, temperature became lower than 100℃

Behavior of temperatures and pressures at Unit 3 （Whole behavior : Mar. to Nov.）



After direct injection to core from CS, temperature became lower than 100℃

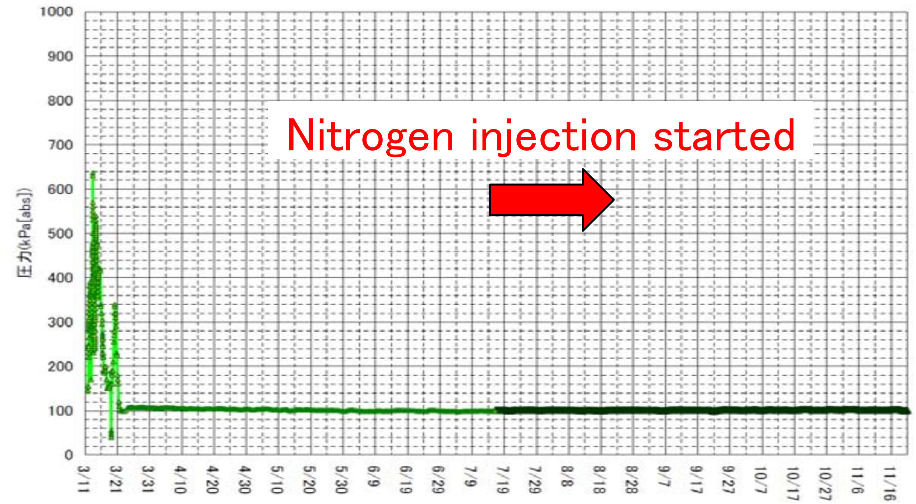


Fig.1 Behavior of D/W pressures after the accident

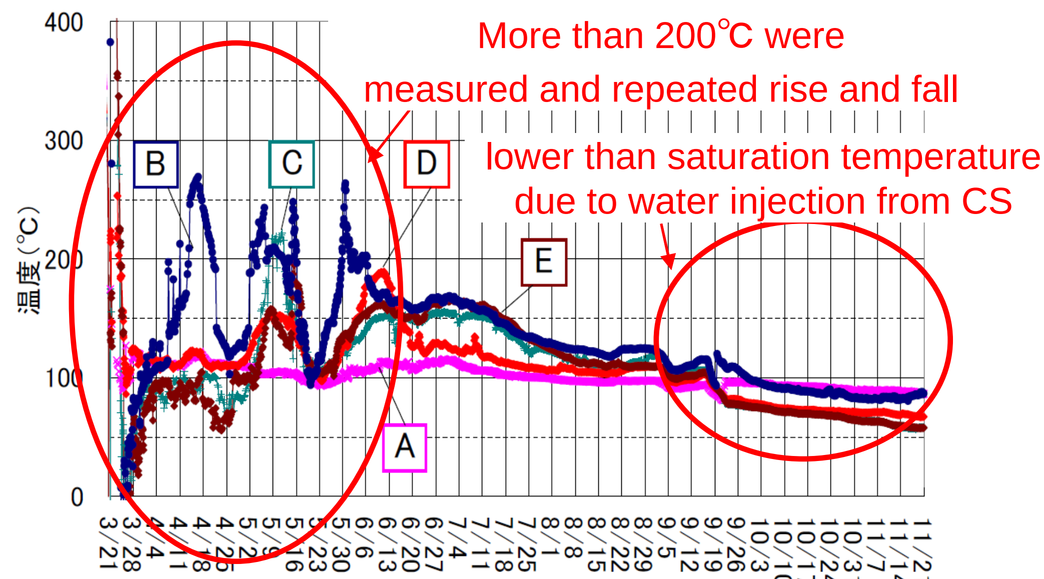
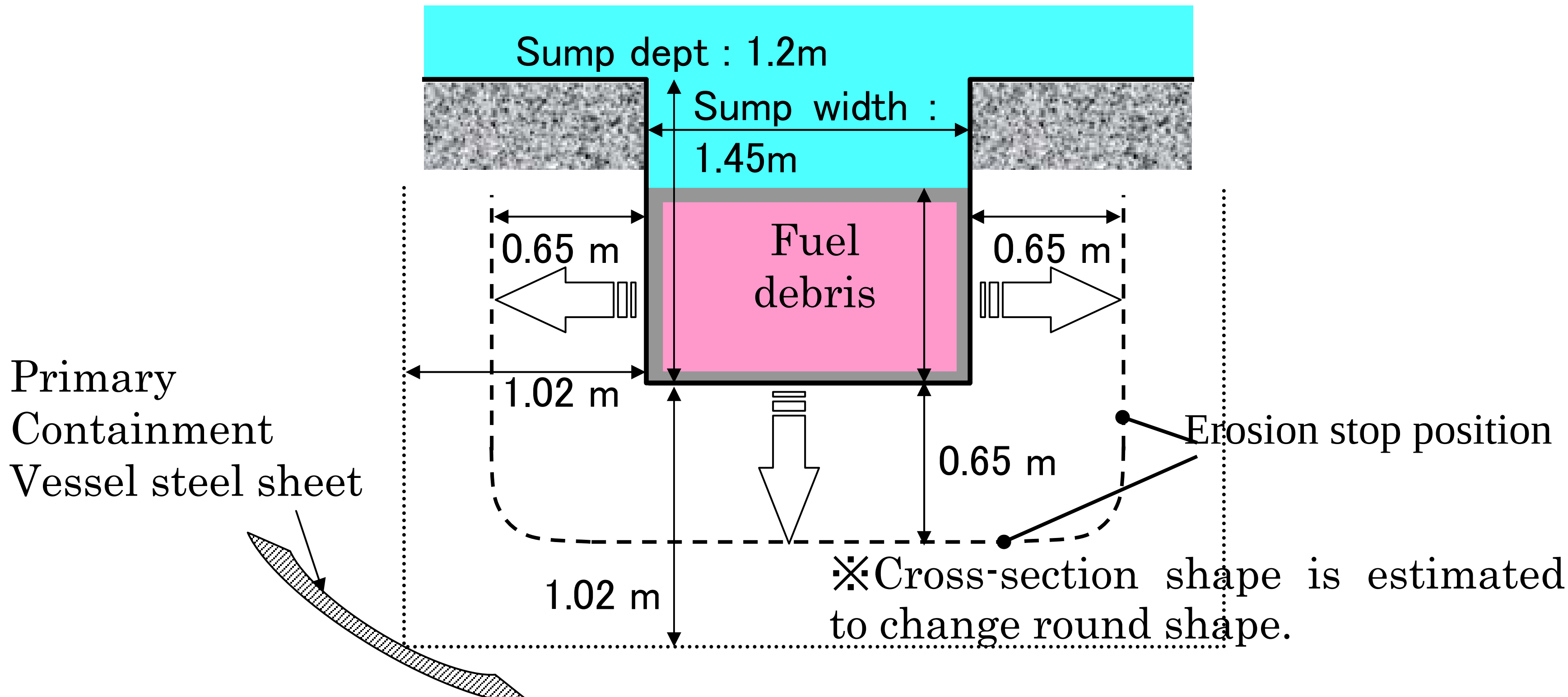


Fig.2 Behavior of temperatures after the accident

MCCI Analysis Result, Unit 1



Ratio of falling core	100%
Deposition thickness of fuel debris	0.81m
Erosion depth	0.65m

Analysis of Gas in PCV (Units 1 and 2)

We conducted a gas-chromatographic analysis of the same sample as the nuclide analysis was conducted on.

By measuring the concentration of hydrogen, carbon monoxide, and carbon dioxide we evaluated the possibility of the core-concrete reaction progress.

(Estimating the amount of the gas generated in the past is difficult since it is diluted by the vapor and nitrogen.)

Table) Analysis Result of Gas in PCV of Unit 1 (equivalent to the concentration in PCV)

Unit[%]

Samples	H	CO	CO2
Unit 1(September) ①	0.154	<0.01	0.118
Unit 1(September) ②	0.101	<0.01	0.201
Unit 1(September) ③	0.079	<0.01	0.129
Unit 2(August) ①	0.558	0.014	0.152
Unit 2(August) ②	1.062	0.016	0.150
Unit 2(August) ③	<0.001	<0.01	0.152

CO_2 concentration is significantly high, however, seeing that the ratio of H_2, CO, and CO_2 is different from the ratio of the gas generated by the core-concrete reaction, it is likely that CO_2 dissolved in the water injected to the reactor (fee carbon dioxide) contributes to it.

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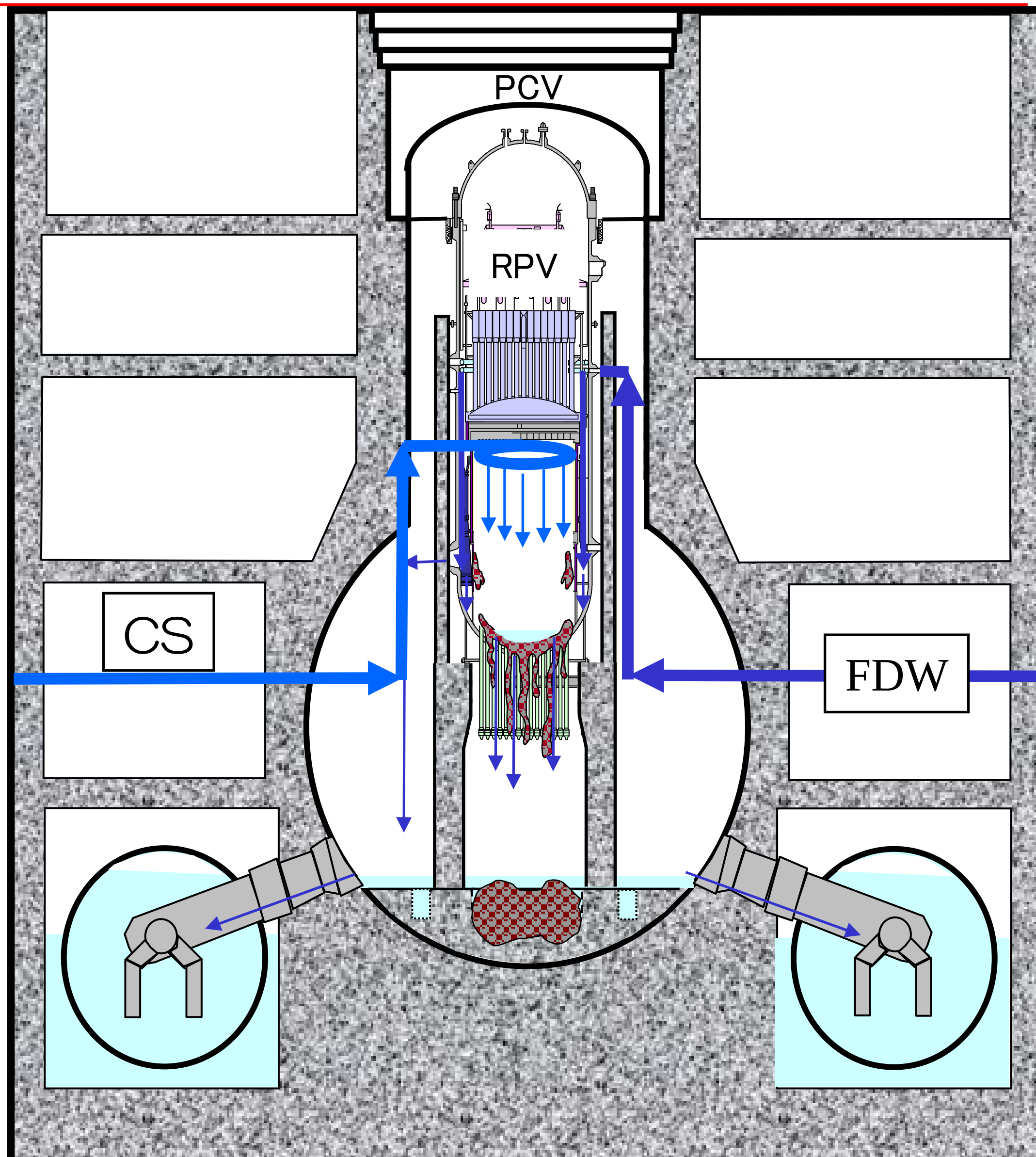
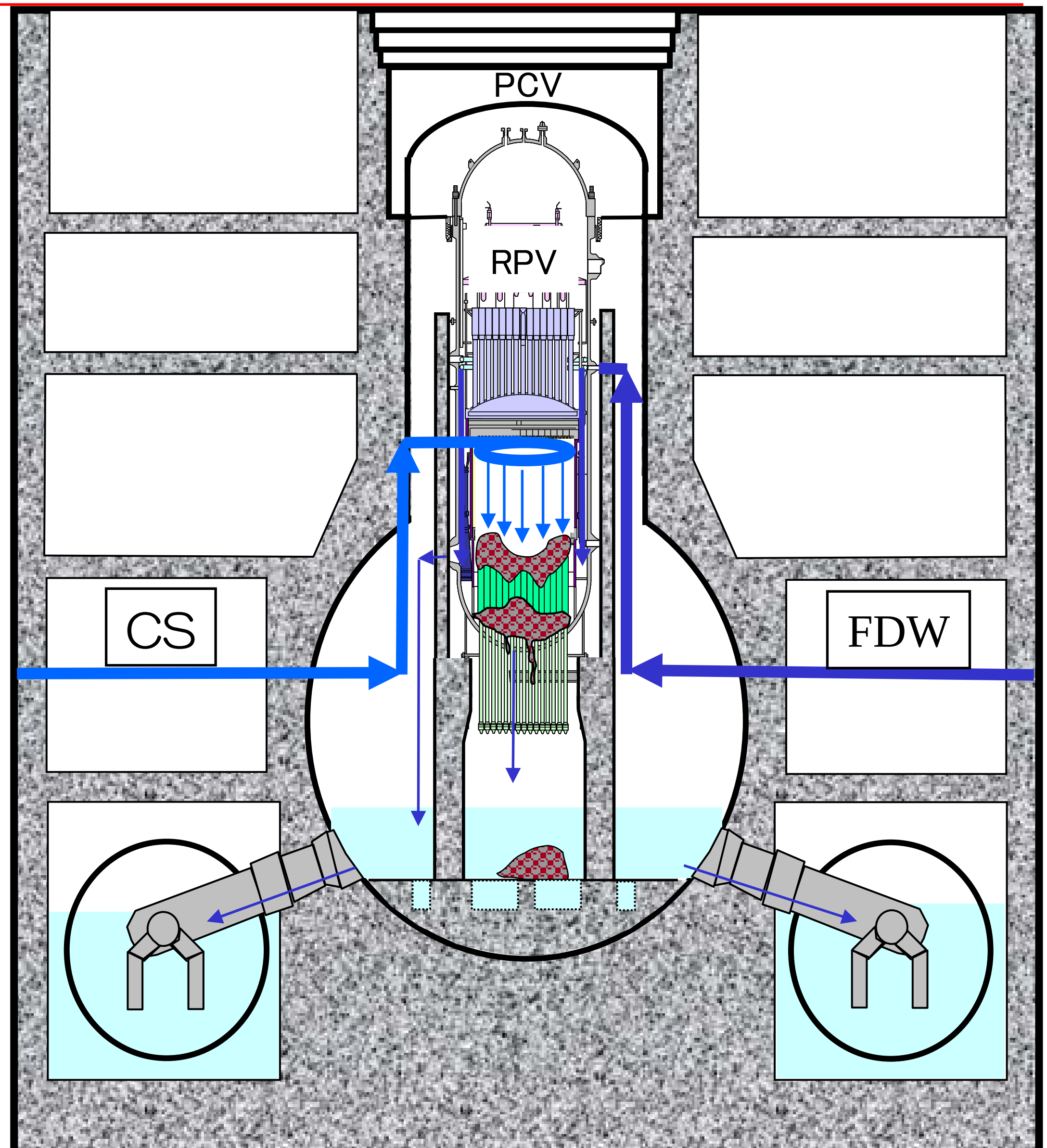
Presumption of reactor core statement (Unit 1)

- Almost no fuel was left at the original position, and completely moved downward after it melted.
- The moved fuel likely damaged PCV and assumed that most of it had dropped to the bottom. (Details for dropped fuel is unknown)
- Dropped fuel is assumed to have caused core concrete reaction.
- Therefore, it is evaluated that all the moved fuel is expected to be cooled directly by water injection. It is also evaluated that the core concrete reaction has been stopped.



Presumption of reactor core statement (Unit 2 and 3)

- There is a range in the evaluation result from "damaged fuel dropped to part of the bottom of PCV" to "Almost all the fuel is left inside RPV".
- If the part of damaged fuel were to have dropped to the bottom of PCV, it can be assumed that core concrete reaction was caused.
- Therefore, it is evaluated that all the moved fuel is expected to be cooled directly by water injection. It is also evaluated that the core concrete reaction has been stopped.



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