Reactor Core Conditions of Units 1 to 3 of Fukushima Daiichi Nuclear Power Station

* This part is a preliminary translation of a part of a report submitted by Tokyo Electric Power Company.

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1. Introduction

Due to the Tohoku-Chihou-Taiheiyou-Oki Earthquake centered off the coast of Sanriku, which occurred on March 11, 2011, Units 1 to 3 of Fukushima Daiichi Nuclear Power Station experienced accidents that not only went significantly beyond design basis, but also exceeded the extent of multiple breakdowns assumed in the preparation for accident management measures, such as the malfunction or loss of all Emergency Core Cooling Systems (ECCS) combined with the extended loss of all AC power sources. In order to stabilize and restore the accident in the future, it is important to understand the event progression of the plants since the earthquake and the current condition of the plants.

On April 25, 2011 TEPCO received an instruction from Nuclear and Industrial Safety Agency (NISA) titled "Collection of Report on Section 1 of the Article 67 of Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors" (April 24, 2011 NISA No.1). On May 16, 2011, based on the instruction, we collected as many plant data as possible at the time of the earthquake, sorted them and reported to NISA. Based on the above mentioned information on equipment status and operations at the early period when the earthquake occurred, we have evaluated plant status and sorted the information by means of Modular Accident Analysis Program (MAAP).

As this analysis was based on limited amount of information obtained at the time of preparing this report and adopted various estimates and assumptions for conditions necessary to run the analysis, there is extremely great uncertainty in the results. Therefore, as we proceed with investigation of causes, we may find the conclusion to be widely different from the analysis results.

2. Summary of Evaluation Results

As a result of an analysis by means of MAAP code, we have obtained an analysis result showing that after an isolation condenser system (IC) of Unit 1 of Fukushima Daiichi Nuclear Power Station (Unit 1) was assumed to stop, reactor core damage began at a relatively early stage, followed by a breach of the reactor pressure vessel.

At Unit 2 of Fukushima Daiichi Nuclear Power Station (Unit 2) and Unit 3 of Fukushima Daiichi Nuclear Power Station (Unit 3), though reactor core damages began as water levels of reactors decreased due to decline in a function of Reactor Core Isolation and Cooling System (RCIC) and High Pressure Coolant Injection System (HPCI), analysis results show that reactor cores in the reactor pressure vessels are maintained. However, in case the actual water levels were lower than the measured values and were below the bottom of active fuel, core damages will further propagate, leading to a breach of reactor pressure vessels.

According to current plant parameters of Units 1 to 3, such as temperatures of their reactor pressure vessels, as those behavior indicate that most of the heat source (fuels) is located in the reactor pressure vessels, even if reactor pressure vessels were to be breached, it is assumed not to be as significant as shown by the analysis. Therefore, we consider the analysis result to be more severe than in reality.

Hence, based on these analysis results together with consideration of plant parameters, we have evaluated that for all plants, significant amount of pellets has melted and the geometry and location of reactor cores have changed substantially.

According to current temperatures of reactor pressure vessels periphery, as reactor cores are being sufficiently cooled, by continuing the current injection of water, such events as large release of fission products (FP) will not occur in the future.

3. Analysis and evaluation regarding condition of reactor core

3.1 Unit 1 of Fukushima Daiichi Nuclear Power Station

3.1.1 Analysis condition

Principal conditions of analysis regarding Unit 1 of Fukushima Daiichi Nuclear Power Station are shown in Tables 3.1.1 and 3.1.2.

We implemented the analysis with the following assumptions regarding leakage from the primary containment vessel (PCV) and operation of the IC.

① Assumption of the gas-phase leakage from Primary Containment Vessel

In the analysis, it was presumed that a leak (about ϕ 3 cm) from gas phase of PCV (Dry Well (D/W)) occurred at about 18 hours after the earthquake, in order to match the actually measured pressure of PCV, to a certain extent. A leak expansion (about ϕ 7 cm) was assumed at about 50 hours after the earthquake.

However, these assumptions were made for the sole purpose of analysis, and it is currently unclear whether a leak from D/W actually occurred or it is a mismatch of measured figure and analyzed figure caused by the problem of the instrument.

② Observation of the operating condition of IC

As the operating condition of IC after Station Black Out (SBO) is still unclear, we did not presume the operation of IC after SBO in the analysis. In addition, we also analyzed the case that IC temporary operated after SBO as a sensitivity analysis.

We presumed that one side of the system of IC has operated intermittently before SBO, as reactor pressure had been fluctuating below the set pressure level (about 7.4MPa[abs]) for operation of Safety Relief Valve (SRV).

Table 3.1.1 Plant Conditions

Items	Conditions
Initial reactor output	1380 MWt (rated power output)
Initial reactor pressure	7.03MPa[abs] (normal operation pressure)
Initial reactor water level	Normal level
Open space volume of PCV	D/W open space : 3410m^3 S/C open space : 2620m^3
Suppression pool water volume	1750m^3

Table 3.1.2 Events

Explanatory notes ○ : Records available △ : Estimates based on records □ : Assumption used on analysis

Analysis Condition				Classification	Notes	In case of ○ : Referred part of the records In case of △ or □ : Estimated, presumed reasons etc.
No	Time and Date		Analyzed Events			
1	May 11th	2:46 pm	Earthquake occurred	○	—	
2		2:46 pm	Reactor scram occurred	○	Report on May 16th, 4. Operation daily, Handover diary of shift supervisor	
3		2:47 pm	MSIV closed	○	Report on May 16th, 4. Operation daily, Handover diary of shift supervisor	
4		2:52 pm	IC(A) (B) activated automatically	○	Report on May 16th, 3. Data of records on alarm generation, Alarm timer	
5		Around 3:03 pm	IC(A) stopped	△	It is assumed that IC stopped based on 6.The record of transient recorder described in the report on May 16th.	
6		Around 3:03 pm	IC(B) stopped	△	Same as above	
7		3:17 pm	IC(A) restarted	△	Estimated behavior of IC based on the transition of reactor pressure (described in 2.Record of the chart in the report on May 16th) ※1	
8		3:19 pm	IC(A) stopped	△	Same as above	
9		3:24 pm	IC(A) restarted	△	Same as above	

10		3:26 pm	IC(A) stopped	△	Same as above
11		3:32 pm	IC(A) restarted	△	Same as above
12		3:34 pm	IC(A) stopped	△	Same as above
13		3:37 pm	SBO occurred	○	Report on May 16th, 4. Operation diary, Handover diary of shift supervisor
14		6:10 pm	IC(A) system 2A, 3A valve opened, and steam generation was confirmed	□	It is described in 7.operation record in the report on May16, however, we presumed in the analysis that the function of IC had been lost since SBO ※2
15		6:25 pm	IC(A) system 3A valve was closed	□	Same as above
16		9:19 pm	The lineup from Diesel Driving Fire Protection Pump (D/D-FP) was implemented.	□	Same as above
17		9:30 pm	IC 3A valve was opened	□	Same as above
18		9:35 pm	Regarding IC, water was supplied from D/D-FP	□	Same as above
19	May 12th	1:48 am	Regarding IC, D/D-FP stopped supplying water by pump trouble, not by fuel run-out	□	Same as above
20		5:46 am	Injection of fresh water by the fire pump started	○	Report on May16th, 7. Operation records ※3
21		2:30 pm	Regarding the containment	△	Report on May16th, 7. Operation records

			vessel vent, operation of AO valve of suppression chamber side was implemented at 10:17am, and a pressure decrease was confirmed at 2:30pm.		We presumed that the success of vent is at 2:30pm, time when a pressure decrease was confirmed.
22		2:49 pm	Vent valve of PCV was closed	△	We presumed the event based on the pressure increase of PCV
23		2:53 pm	Injection of fresh water terminated	○	Report on May16th, 7. Operation records
24		3:36 pm	Explosion of reactor building of Unit 1 occurred	○	Report on May16th, 7. Operation records
25		8:20 pm	Injection of sea water started	○	Report on May16th, 7. Operation records ※3

※1 There are unclear points on the behavior of IC before SBO, however, as reactor pressure fluctuated between about 6.2 to 7.2MPa[abs] based on the 2.Records of charts in the report on May 16th, we presumed that the one of the system of IC had operated intermittently, as set pressure of SRV no.1 relief valve is 7.4 MPa[abs], and reseating pressure is 6.9MPa[abs].

※2 There are also unclear points on the operation of IC after SBO, however, we presumed that the function of IC was lost as we only have the insufficient amount of record which show IC is functioning.

※3 We set up the injection amount of water and the time we changed the amount based on the daily injection amount described in 7.Operation records of the report on March 16th, in order not to exceed the daily average injection amount and total injection amount.

3.1.2 Analysis Result

Table 3.1.3 shows the result of analysis based on the condition shown in 3.1.1. Fig. 3.1.1 to Fig 3.1.12 show the result of analysis on the trend of reactor water level, etc.

Table 3.1.3 Summary of Analysis Result on Unit 1

Item	Analysis Result
Start of reactor core exposure	Approx. 3 hours after earthquake
Start of reactor core damages	Approx. 4 hours after earthquake
Start of reactor pressure vessel damages	Approx. 15 hours after earthquake

The detail of analysis result is as follows.

The reactor water level reaches the Top of Active Fuel (TAF) approximately 2 hours after assumed timing of IC stop, followed by a reactor core damage (see Fig. 3.1.1).

After the earthquake occurred, the actual data of reactor water level was changing within the fuel range. It differs dramatically from the analysis result; the analysis results in the breach of the reactor pressure vessel, and the reactor water level cannot be maintained in the reactor pressure vessel. Regarding this matter, there is a possibility that the correct level is not shown because the water in the level gauge boiled away as the temperature of the PCV rose. As for Unit 1, after calibration of level gauge, it was confirmed that the water level is below the fuel range.

After assumed timing of IC stop, the reactor pressure increases, but it remains around 8MPa because of the SRV. After the timing of reactor core damage, the melted pellets move to the lower plenum, and then after 15 hours from the earthquake, the reactor pressure vessel breaches and the reactor pressure decreases rapidly (see Fig. 3.1.2).

The pressure of PCV temporarily increases because of the steam released from the reactor pressure vessel and hydrogen gas formed by the reaction of water and metal in the reactor, but after that the pressure shows a decreasing trend because of the leakage from the PCV assumed in the analysis, and then it decreases rapidly by the vent operation on Mar. 12 (see Fig. 3.1.3).

At the beginning of the incident the measured pressure of PCV is higher than the analysis result; this might be due to, for example, a release of steam from the reactor pressure vessel caused by a damage in the instrumentation pipe in the reactor at the initial stage of core damage, or a loss of sealing capability of a gasket used in the main steam system because of high temperature. However, currently the cause is unknown including whether it is a problem of instrumentation or not.

Regarding the assumption about the leakage of the PCV, at the time after 18 hours of the earthquake when the leakage is assumed, the PCV temperature is over 300 degree

C, which well exceeds the designed temperature of the vessel (138 degree C). Based on a Joint Utilities Studies conducted in the past, it is know that there is a possibility for gaskets to be damaged in such an over-temperature condition; therefore, if the leakage from the PCV is real, the damage of gasket by over-temperature can be one of the reasons. Also regarding the assumption of leakage from the PCV 50 hours after the earthquake, as the analysis shows a transition of PCV temperature in a high-temperature range (see Fig. 3.1.5), the gradual increase in leakage points may be attributed as one of the reasons for the increase in leakage.

Although water injection to the reactor starts at 14 hours after the assumed timing of IC stop, by then fuels melt by the decay heat and relocate to the lower plenum, leading to a breach in the reactor pressure vessel at about 15 hours after the earthquake (see Fig 3.1.4 and Fig 3.1.9).

Regarding the Fission Products (FP) released due to the reactor core damage, almost all noble gases are released to the atmosphere by the vent operation. According to the analysis, approximately 1% of cesium iodide and less than 1% of other nuclides are released (see Fig 3.1.7 and Fig 3.1.8). Plutonium belongs to UO_2 group as PuO_2, and the analysis shows that the release rate is under 10^{-7}.

Hydrogen is generated at the same time when the reactor core damage began, and it is possible that the cause of the explosion on Mar. 12 was the hydrogen generated during this period (see Fig. 3.1.6).

Operation of IC after the Tsunami arrival is unclear, however, we ran a case under the assumption that IC was temporary operable (assumed operation of one train of IC from approx. 6:00 pm, Mar. 11 to approx. 2 am, Mar. 12). The behavior of the reactor water level is similar although the absolute value is different (see Fig. 3.1.10). However, based on this assumption, the PCV pressure shows a totally different behavior from measured value (see Fig.3.1.11), such that it is impossible to clarify by this analysis about the operation of IC after SBO. Because the water level also cannot be kept in the fuel range in this sensitivity analysis of IC, the reactor core becomes damaged (see Fig.3.1.12).

This evaluation is implemented based on the analysis using MAAP code; it has some uncertainty on the determination of analysis condition and on the analysis model, so that it is necessary to take notice that the progresses of phenomenon as the result also have some uncertainty. Especially, the amount of released FP is greatly dependent on these uncertainties, so that the data should be used only as a reference.

3.1.3 Evaluation Result

As previously mentioned, we have had an analysis result that, after failure of all AC power (arrival of the tsunami), reactor core damages began at a relatively early period, leading to the breach of the reactor pressure vessel. However, we consider the analysis result to be stricter than in reality, when we refer to plant status estimated from temperatures in each part etc. in the following pages.

When we regained temperature measurement in various locations, those of reactor pressure vessels were above 400 degrees Celsius in multiple measuring points. At that time, though the condition of insufficient cool-down to the reactor core had continued, they have then sufficiently cooled down since we injected water through the feed-water line and, due to the secured water injection to the reactor, temperatures in each part rapidly decreased.

In addition, as a result of calibration of the water-level gauge, we have found that the water level in reactor pressure vessel is not in the fuel range.

On the other hand, most of the fuel is believed to be cooled in the reactor pressure vessel, because we can still measure temperatures in CRD housing, etc. at the lower reactor pressure vessel (if the reactor pressure vessel is breached, we may not be able to measure temperatures) and temperatures of the steel of the reactor pressure vessel currently change around between 100 degrees Celsius and 120 degrees Celsius and correlate changes in amount of water injection in multiple measuring points and temperatures in multiple points of the upper reactor pressure vessel are higher and the heat source is estimated to come from the inside of the reactor pressure vessel.

Hence, according to the analysis and plant parameters (temperatures at the periphery of the reactor pressure vessel), the reactor core has been significantly damaged, relocated below or slumped to the lower plenum from the fixed position of fuel loading, however, most of it is being stably cooled down around there.

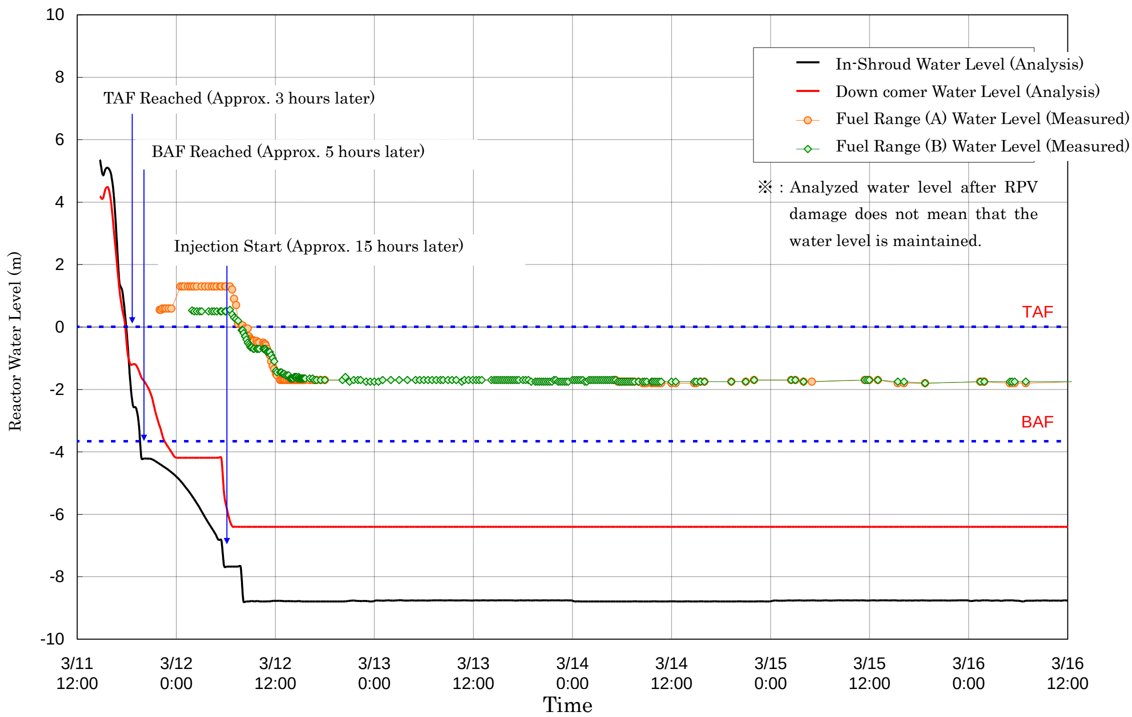


Figure3.1.1 Unit1 Reactor Water Level

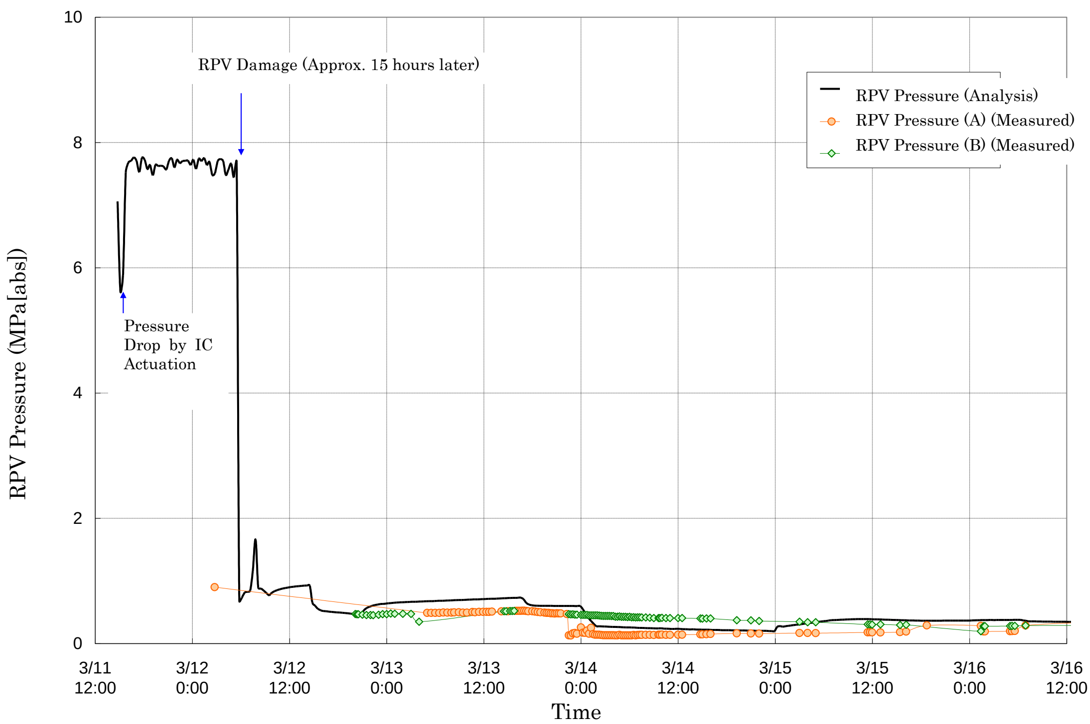


Figure3.1.2 Unit1 RPV Pressure

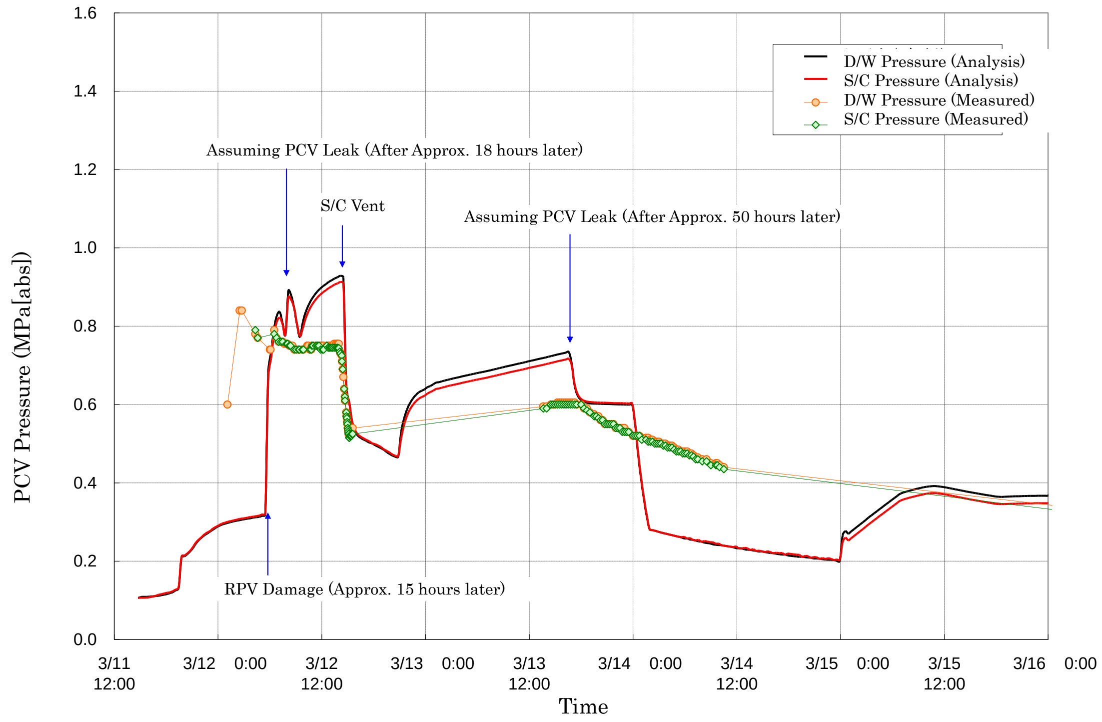


Figure3.1.3 Unit1 PCV Pressure

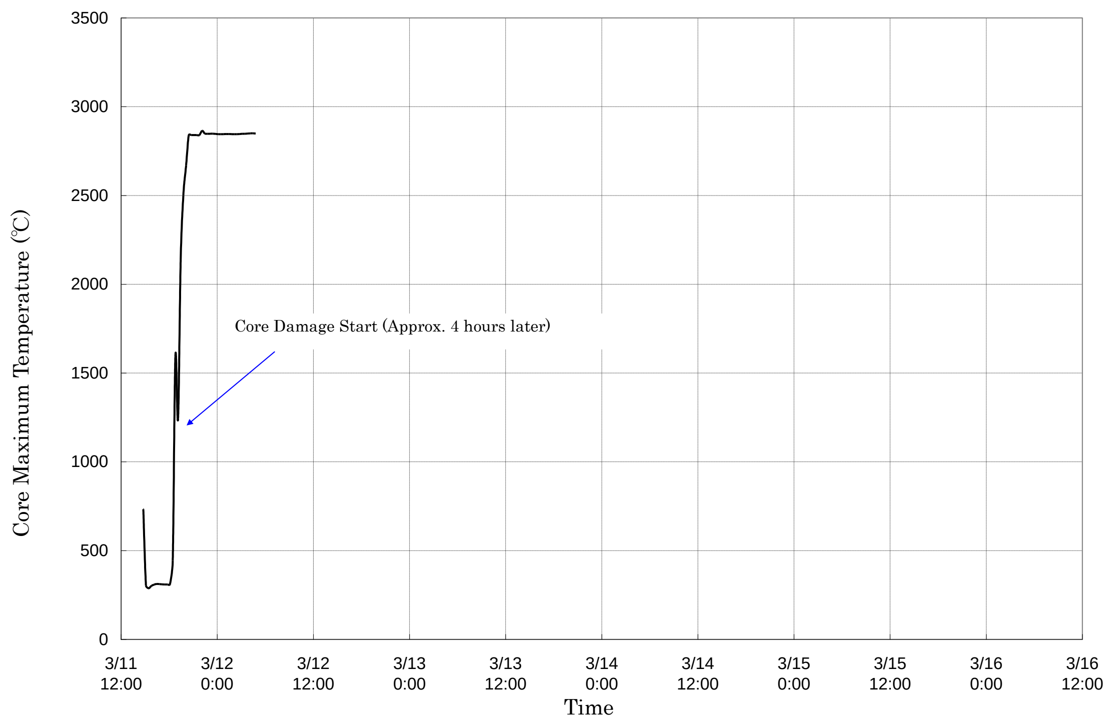


Figure3.1.4 Unit1 Core Temperature

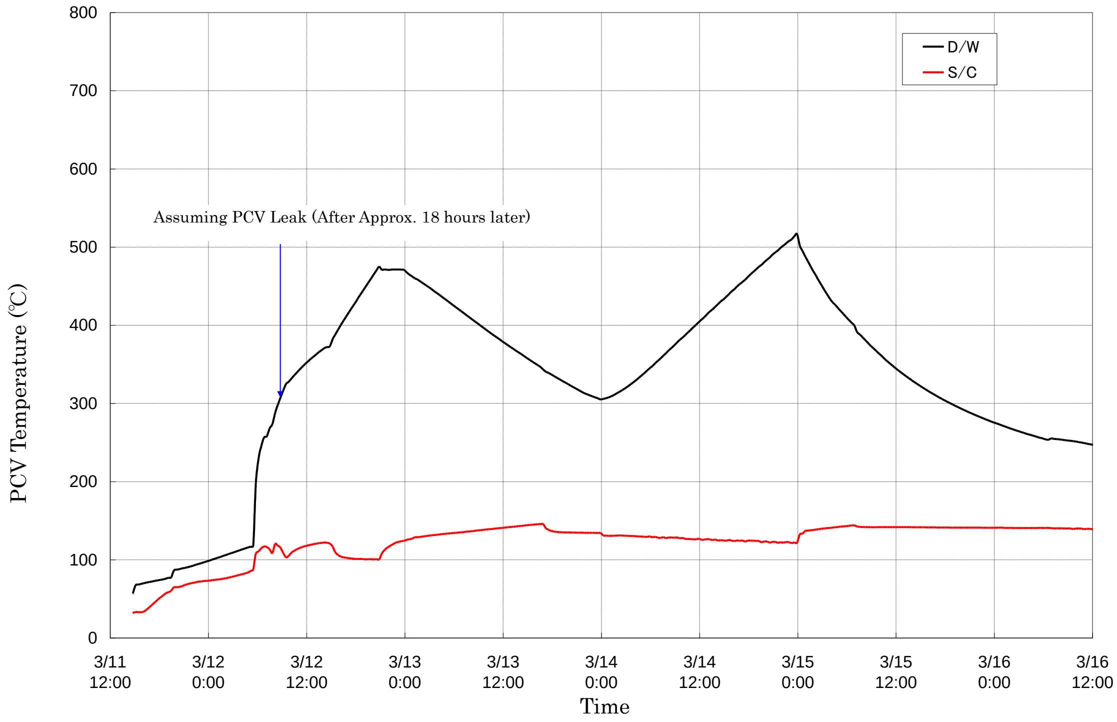


Figure3.1.5 Unit1 PCV Temperature

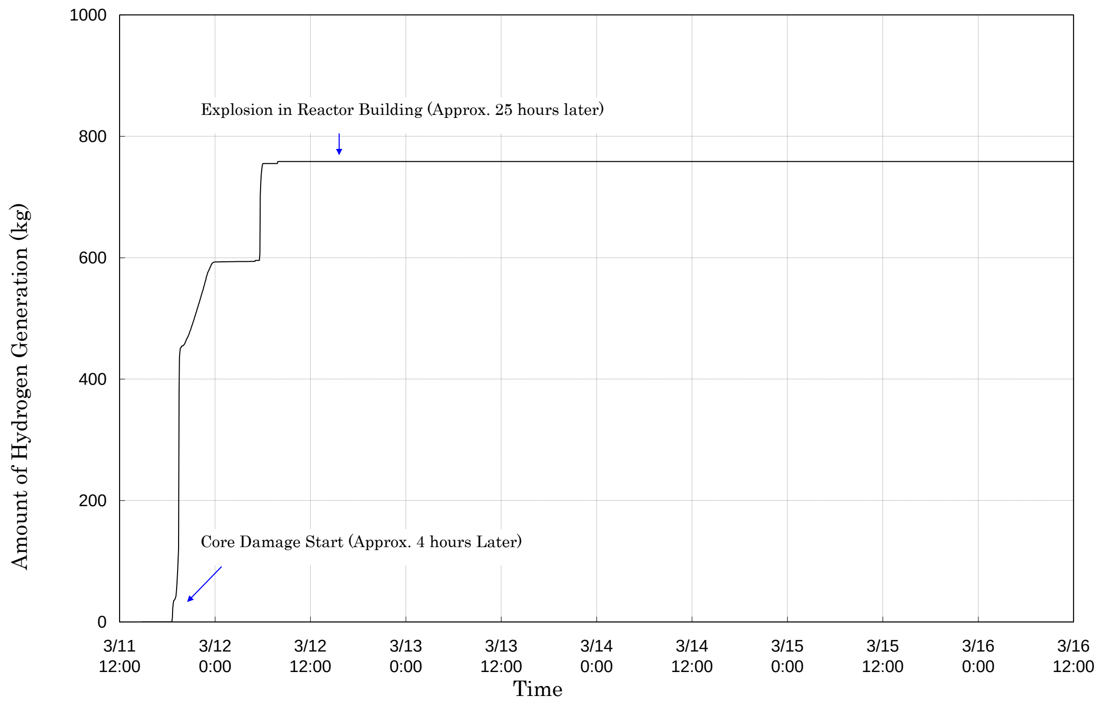


Figure3.1.6 Unit1 Amount of Hydrogen Generation

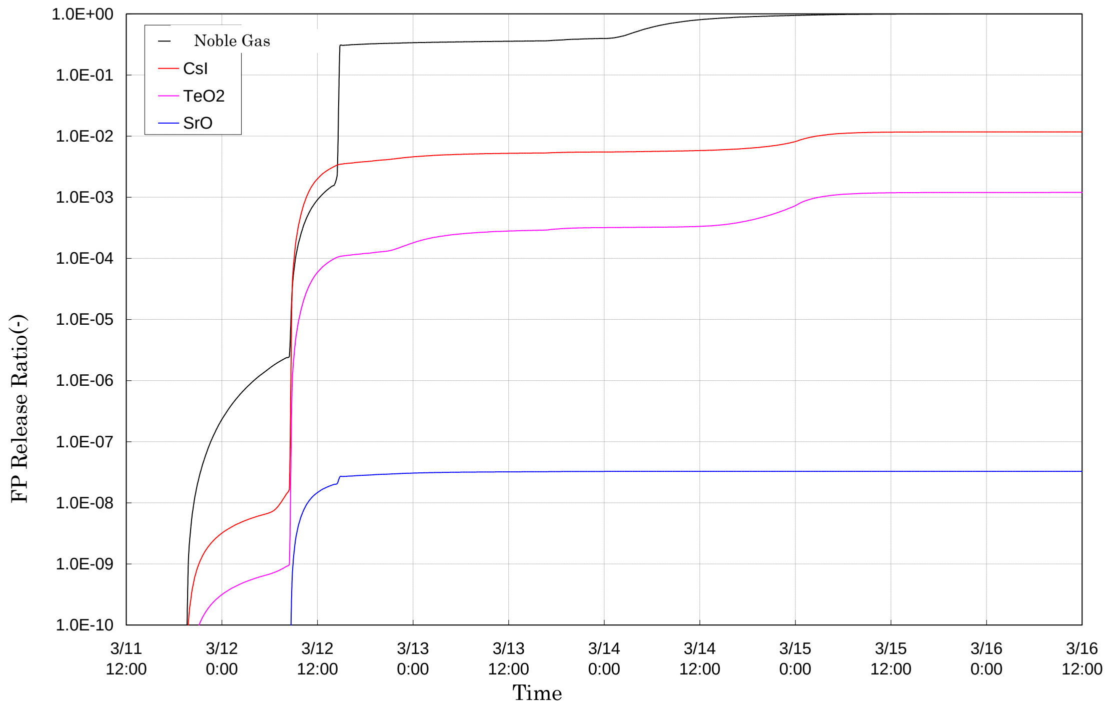


Figure3.1.7　Unit1 FP Release Ratio（１／３）

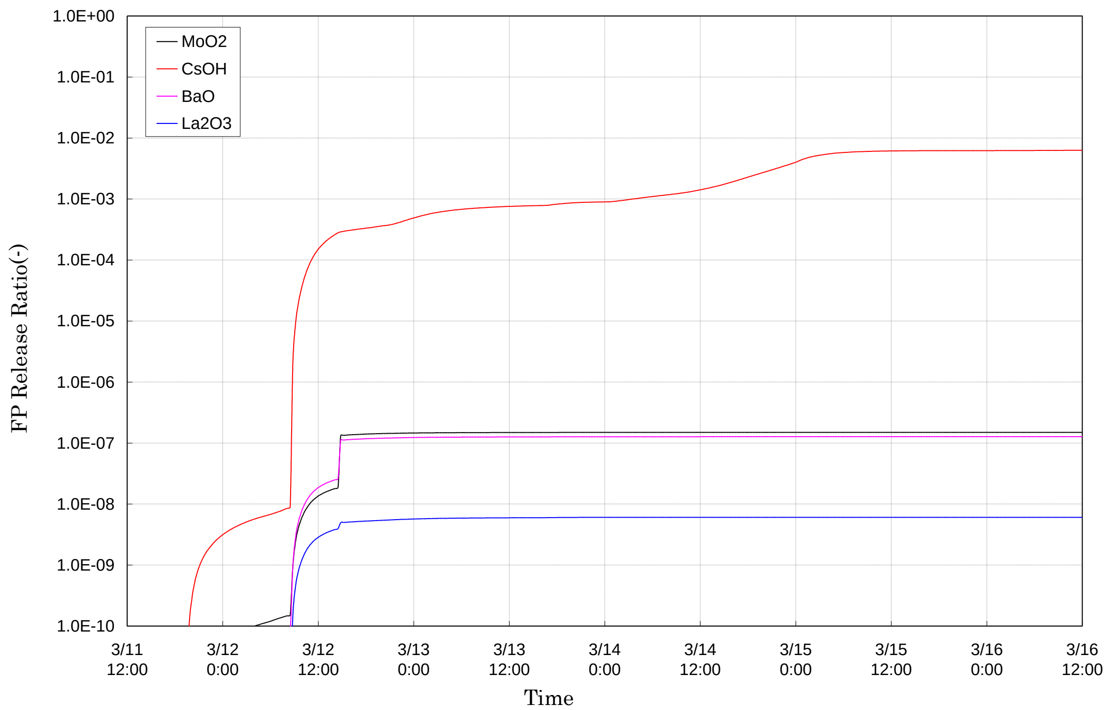


Figure3.1.7　Unit1　FP Release Ratio（２／３）

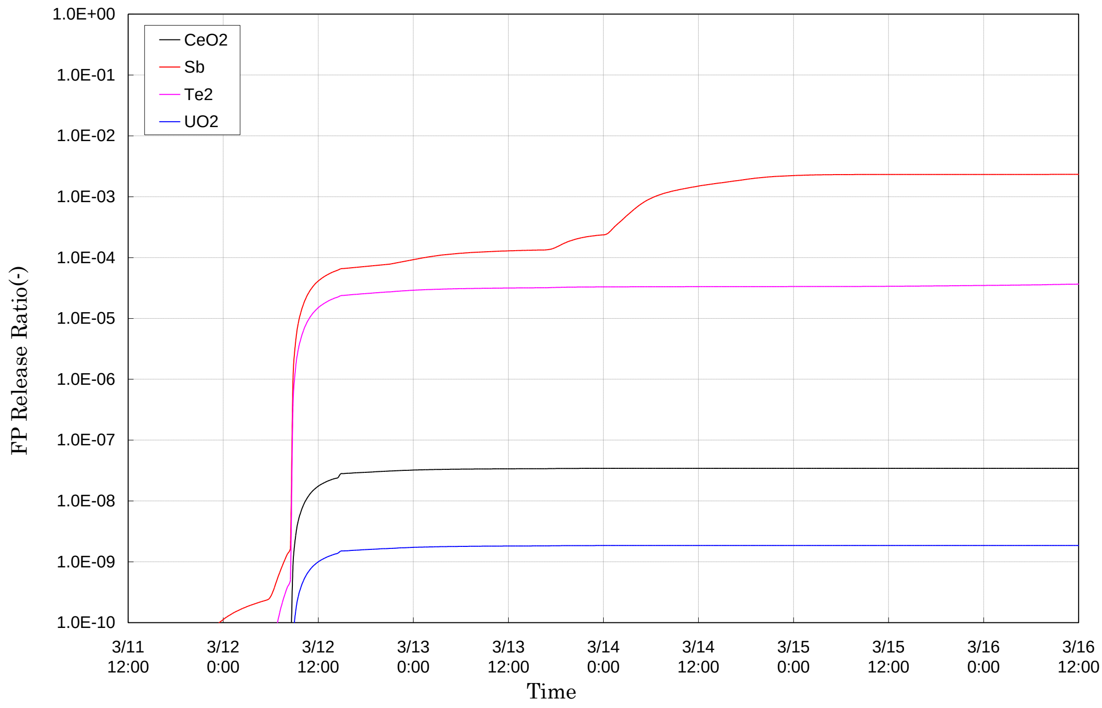


Figure3.1.7 Unit1 FP Release Ratio（３／３）

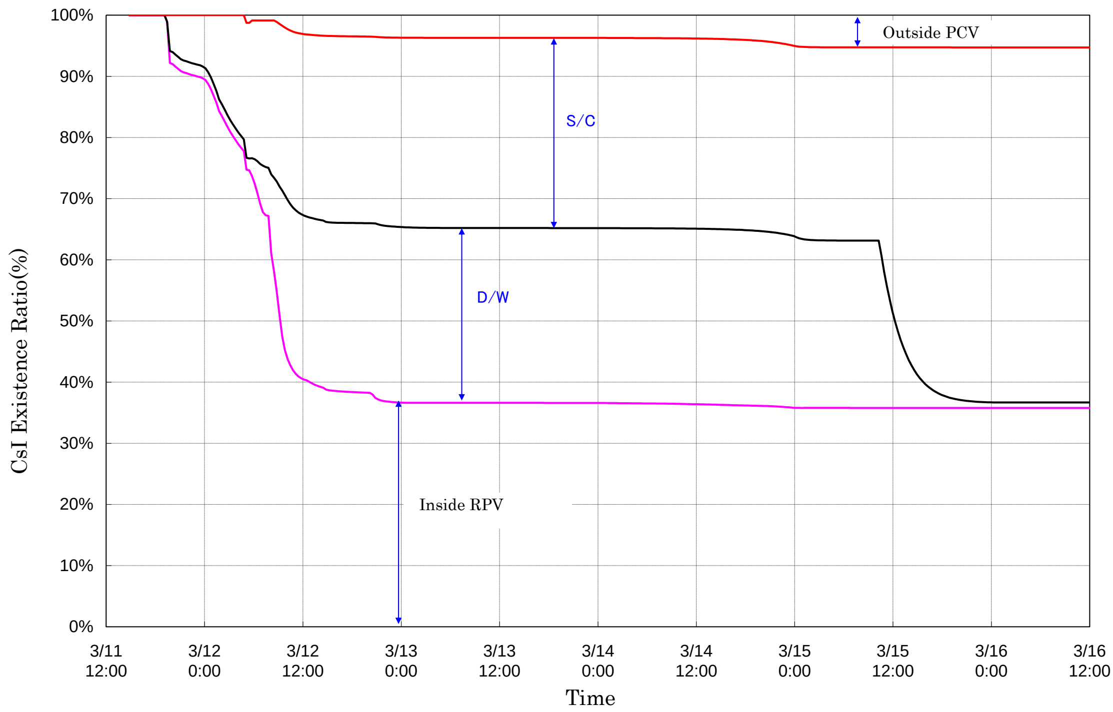


Figure3.1.8 Unit1 FP Existence Ratio（１／２）

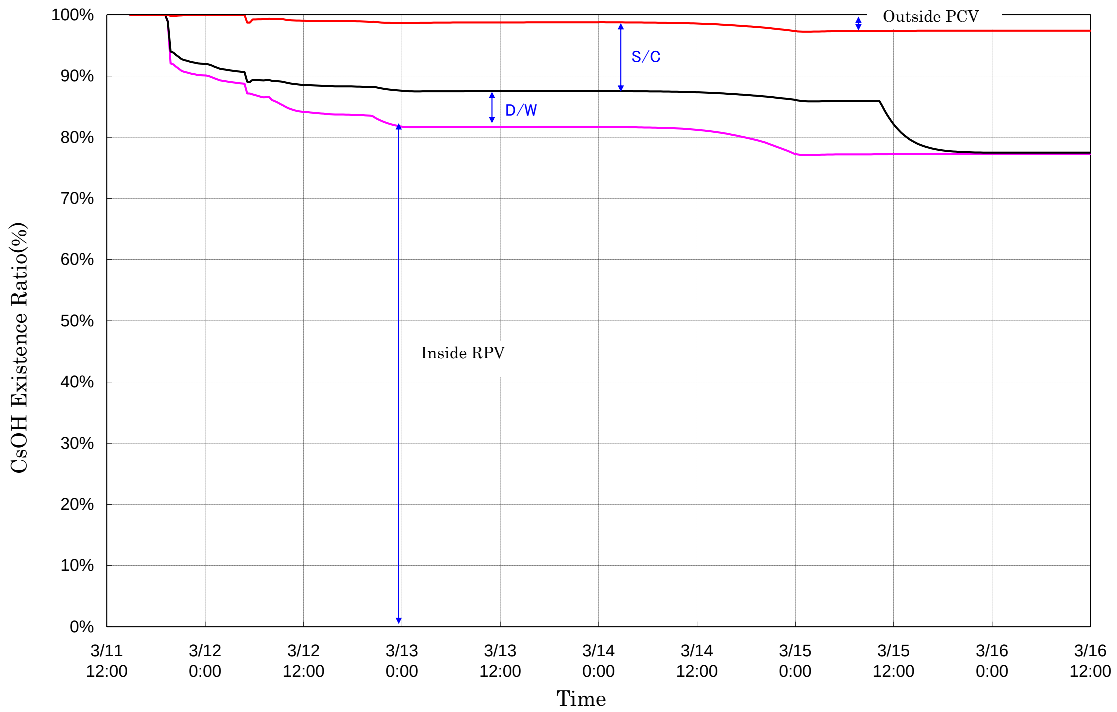
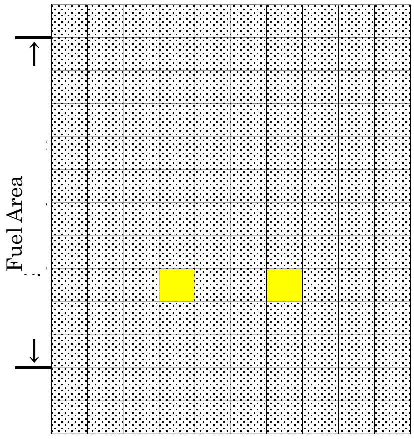
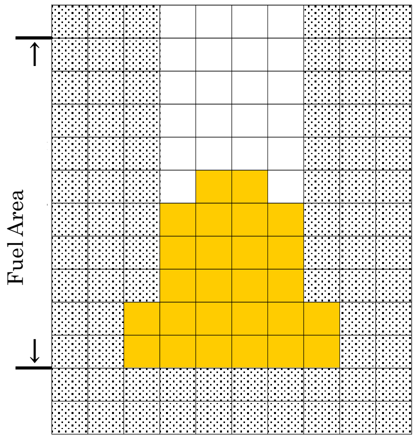


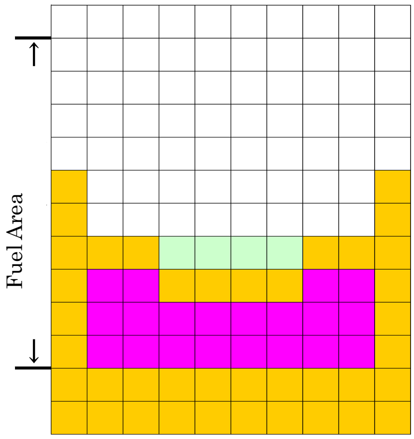
Figure3.1.8　Unit1　FP Existence Ratio（２／２）



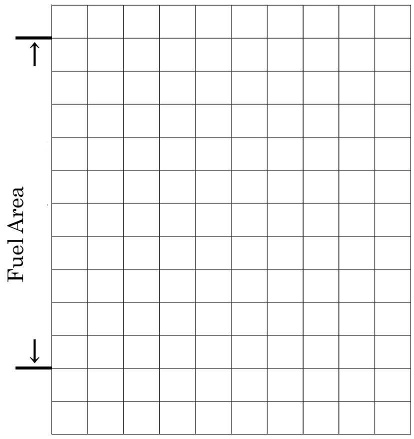
Approx. 4.7 hours after SCRAM



Approx. 5.3 hours after SCRAM



Approx. 14.3 hours after SCRAM



Approx. 15 hours after SCRAM

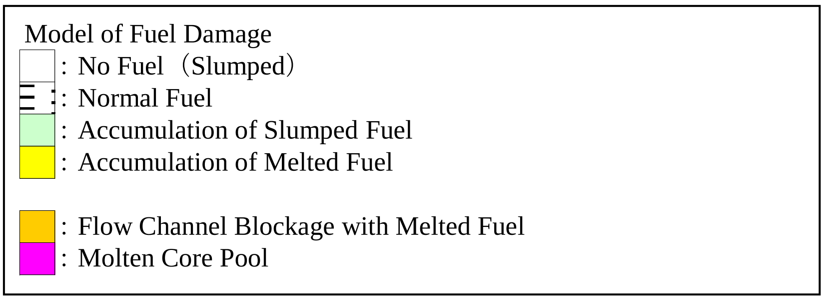


Figure3.1.9 Unit1 Core Status

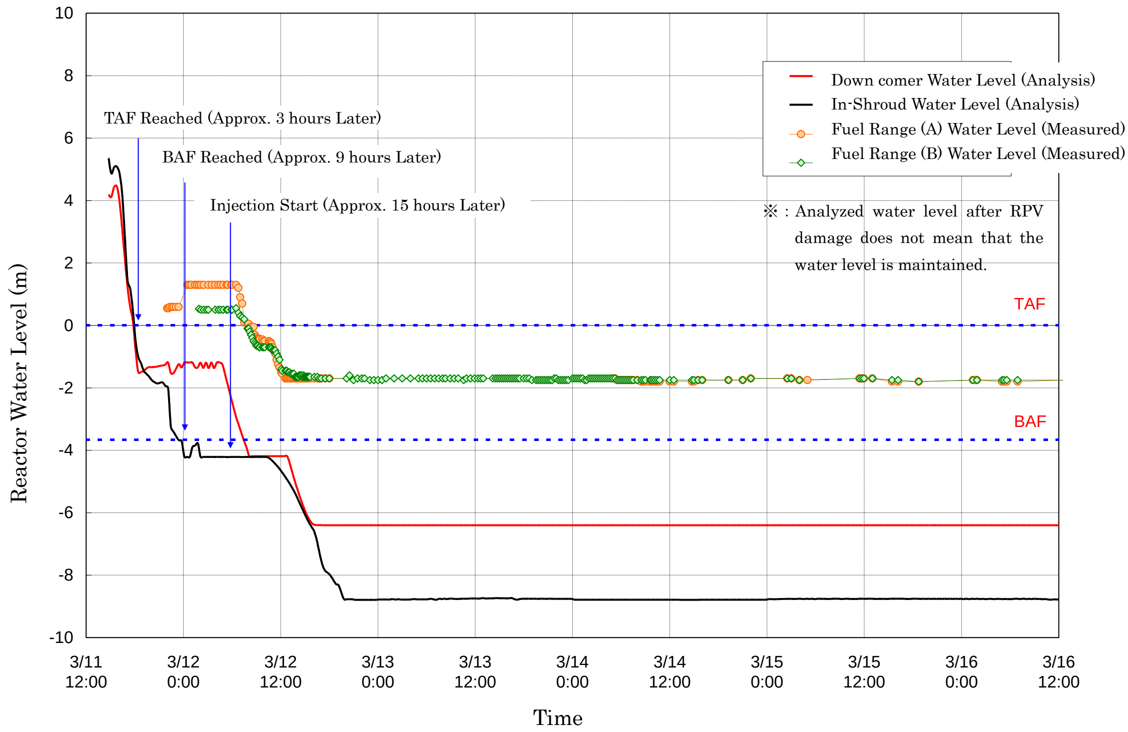


Figure3.1.10 Unit1 Reactor Water Level (IC Continued Operation Case)

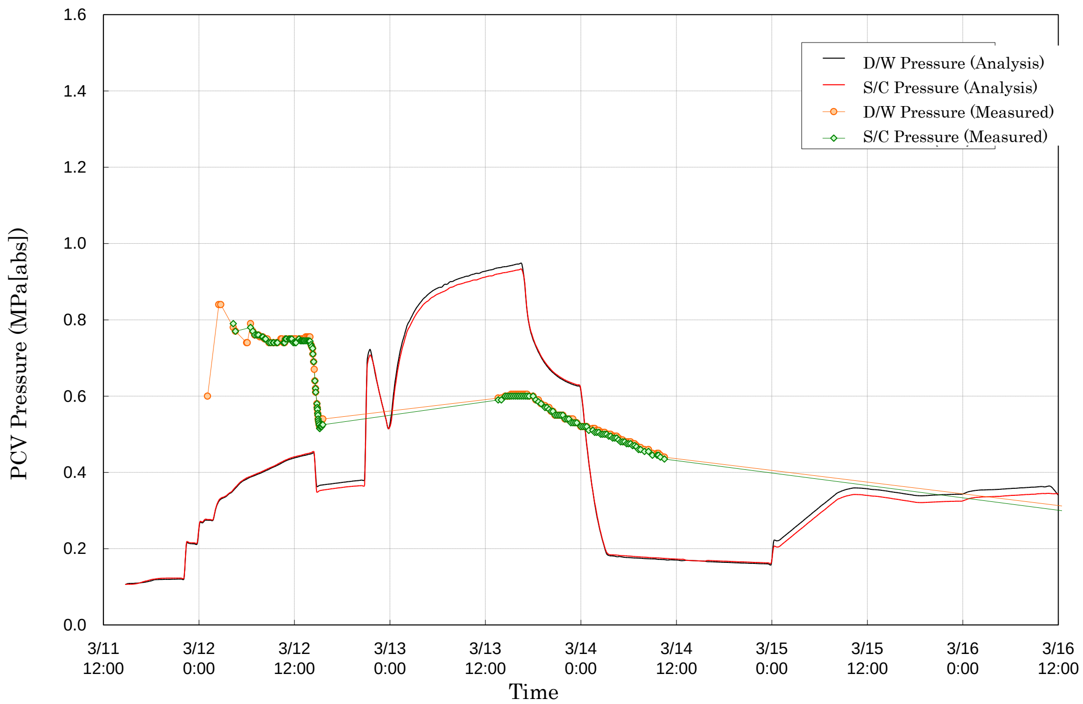


Figure3.1.11 Unit1 PCV Pressure (IC Continued Operation Case)

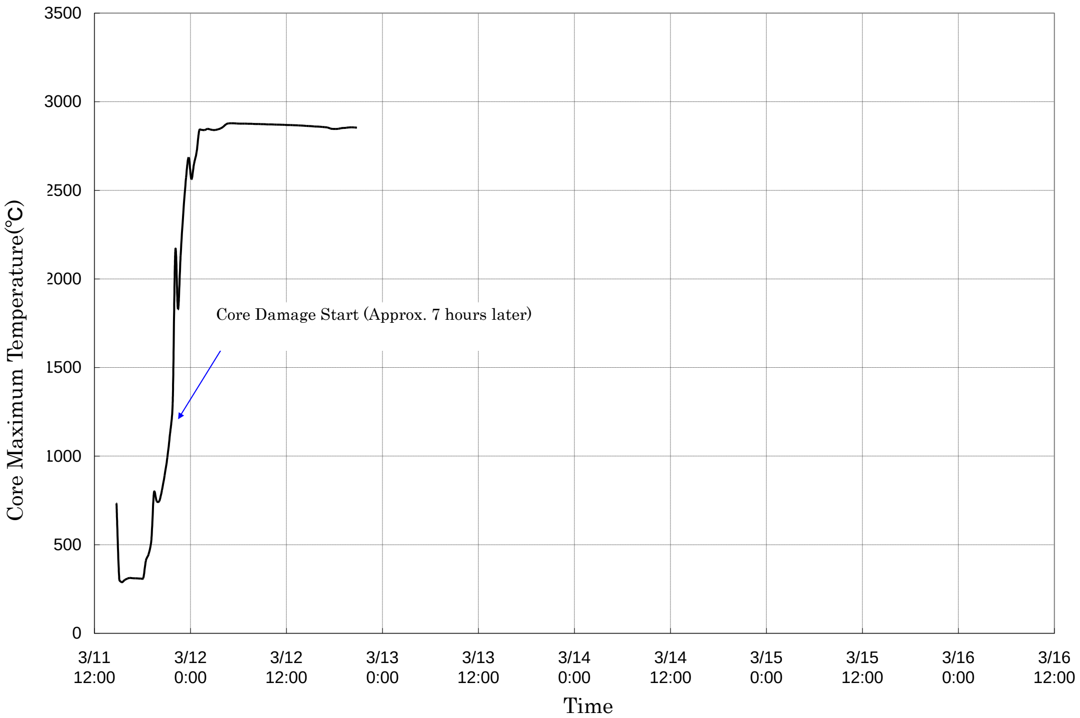


Figure3.1.12 Unit1 Core Temperature (IC Continued Operation Case)

3.2 Unit 2 of Fukushima Daiichi Nuclear Power Station

3.2.1 Analysis condition

Principal conditions of analysis regarding Unit 2 of Fukushima Daiichi Nuclear Power Station are shown in the Table 3.2.1 and 3.2.2.

We implemented the analysis in two cases below, and regarding a leak from PCV, we implemented the analysis based on the assumptions below.

① Cases of analysis

【Case 1】In order to match the measured value of reactor water level, we presumed a smaller amount of flow rate compared to the discharge flow of the fire pump that could maintain the reactor water level.

【Case 2】Based on the premise that it is impossible to maintain the reactor water level in the fuel range, we presumed an injection rate that can maintain slightly below the level of the fuel range, and not the flow rate of the discharge side of the fire pump.

② Assumption of the gas-phase leakage from PCV

In the analysis, we presumed that a leak (about ϕ 10 cm) from gas phase of D/W has occurred 21 hours after the earthquake occurred, in order to match the figure to the actually measured pressure of PCV to a certain extent. We also presumed a leak (about ϕ 10 cm) from gas phase of PCV since an abnormal sound occurred near the Suppression Chamber (S/C) on May 15th.

However, these assumptions were made for the sole purpose of analysis, and it is currently unclear whether the leak from D/W occurred actually, or it is a mismatch of measured figure and analyzed figure caused by the problem of the instrument.

Table 3.2.1 Plant Conditions

Items	Conditions
Initial reactor output	2381 MWt (rated power output)
Initial reactor pressure	7.03MPa[abs] (normal operation pressure)
Initial reactor water level	Normal level
Open space volume of PCV	D/W open space : $4240m^3$ S/C open space : $3160m^3$
Suppression pool water volume	$2980m^3$

Table 3.2.2 Events

Explanatory notes ○ : Records available △ : Estimates based on records □ : Assumption used on analysis

Analysis Condition				Classification	Notes	In case of ○ : Referred part of the records In case of △ or □ : Estimated, presumed reasons etc.
No	Time and Date		Analyzed Events			
1	March 11th	2:46 pm	Earthquake occurred	○	—	
2		2:47 pm	Reactor scram occurred	○	Report on May 16th, 4. Operation daily, Handover diary of shift supervisor	
3		3:02 pm	RCIC activated manually	○	Report on May16th, 7. Operation records	
4		3:28 pm	RCIC tripped (L-8)	○	Same as above	
5		3:41 pm	SBO occurred	○	Report on May 16th, 4. Operation daily, Handover diary of shift supervisor	
6	March 12th	4:20 am - 5:00 am	Changed the water source of RCIC from condensate storage tank to suppression chamber	○	Report on May16th, 7. Operation records	
7	March 14th	1:25 pm	RCIC stopped	○	Same as above	
8		4:34 pm	Started the operation of pressure reduction of Reactor Pressure Vessel (1 SRV open)	○	Same as above	
		4:34 pm	Started the injection of sea water through the fire protection system	○	Report on May16th, 7. Operation records ※1	

9		Around 6:00 pm	Confirmed the decrease of the reactor pressure	○	Report on May16th, 7. Operation records
10		7:20 pm	The fire pump stopped resulted from fuel run-out	○	Report on May16th, 7. Operation records ※1
11		7:54 pm	The fire pump activated	○	Report on May16th, 7. Operation records ※1 ※2
		7:57 pm	The second fire pump activated	○	Report on May16th, 7. Operation records ※1
12		9:20 pm	By opening 2 SRVs, reactor pressure decreased and water level recovered	○	Same as above
13		Around 11:00 pm	It is presumed that the 1 SRV was closed	□	As reactor pressure increased at around 11:00pm, it is presumed that 1 SRV was closed at this time.
14	March 15th	Around 6:14 am	An abnormal sound has occurred near the suppression chamber, and the pressure inside decreased	○	From the press release of Tokyo Electric Power Company（http://www.tepco.co.jp/index-j.html）

※1 There is a possibility that the certain amount of sea water was injected since 4:34pm, March 14th based on the record of 7:20pm, March 14th which describes "The fire pump has stopped", however, we presumed 7:54pm, March14th as the time which started the injection, as the water level has rose since then.

※2 We set up the injection amount of water and the time we changed the amount based on the daily injection amount described in 7.Operation records of the report on March 16th, in order not to exceed the daily average injection amount and total injection amount.

3.2.2.1 Analysis Result (Analysis Case 1)

Table 3.2.3 shows the result of analysis based on the condition shown in 3.2.1. And from Fig. 3.2.1.1 to Fig 3.2.1.10 show the result of analysis about the trend of reactor water level, etc.

Table 3.2.3 Summary of Analysis Result on Unit 2

Item	Analysis Result
Start of reactor core exposure	Approx. 75 hours after earthquake
Start of reactor core damages	Approx. 77 hours after earthquake
Start of reactor pressure vessel breach	(reactor pressure vessel breach did not occur in this analysis)

The details of analysis result are as follows.

The reactor water level gradually comes down after RCIC stops and the reactor core starts exposed, and the reactor core exposed completely by opening the SRV and the reactor core damage starts (see Fig.3.2.1.1). Although the water injection starts approximately at the same time, but in this analysis, because the water injection flow is assumed that it is commensurate with the reactor water level indicated by measurement equipment, so that the water injection flow is not enough and the water level is remained around a half level of the reactor core range. Therefore, the reactor core is damaged.

The reactor pressure is kept high around the pressure of SRV operation until RCIC stops. By opening the SRV after RCIC stopped the reactor pressure decreases rapidly, after that it goes down to around atmosphere pressure.

During the operation of RCIC, the measured value of the reactor pressure is lower than analyzed value, indicating the possibility that a leak path was formed through SRV to S/C, but currently it is not sure whether there was an actual path or it is only a problem of instrumentation. The behavior of analyzed value and measured value is in general agreement after SRV opening (see Fig.3.2.1.2).

The pressure of PCV increases with the increase of suppression pool water temperature, but because the leak from the PCV (D/W) is assumed, the increase from earthquake occurrence becomes slow which is same as measured value. After that, temporary increase of pressure occurs when the SRV opened on Mar. 14, and then at the measured value, the pressure changes decrease. Also regarding the analysis, the analysis was implemented on the condition that the timing of the abnormal sound noticed near S/C on Mar. 15 is set as a boundary; it means a leak occurred in the gas phase of S/C at this moment (see Fig. 3.2.1.3).

Regarding the assumption of leak from the PCV, considering that the vessel temperature already exceeded designed temperature at the timing of assumption, the increase of leakage from the vessel, which is caused by the influence of over-temperature could be assumed as one of the reasons (see Fig.3.2.1.5). In case there no assumption of leak for the PCV is made, the PCV pressure would reach 2Pd (twice the designed pressure) in relatively early stage (see Fig. 3.2.1.10). And after the moment of abnormal sound noticed near the S/C the pressure decreased rapidly, also the leak is assumed in the analysis, however currently it is not sure whether there is an actual leak in the PCV or it is only a problem of instrumentation.

Regarding reactor core temperature trend, after RCIC stop, the temperature increases with the decrease of reactor water level, and the fuel pellets start melting (see Fig.3.2.1.4).

A large amount of hydrogen is generated when the reactor core is exposed and the temperature of fuel cladding starts increasing. After 1 week of earthquake occurrence, the amount equal to the reaction of about 79% of fuel cladding in the active fuel range is generated (see Fig.3.2.1.6).

Regarding the release of FP, after reactor core damage, noble gases are released from the reactor pressure vessel to S/C, and according to the assumption of leak for this analysis, the result is that almost all noble gases are released. For cesium iodide, the release rate is about 1% and most exist in S/C. However, the release of FP out of the PCV is by the assumption of leak from the PCV, so it is possible that the result is different from actual situation (see Fig.3.2.1.7 and Fig.3.2.1.8).

The result says that, about the reactor core of Unit 2, there partially exists a molten core pool, but it remains in fuel range and does not lead to the reactor pressure vessel breach. The reason is that, the water injection by RCIC in the early stage was implemented continuously, and the period between when RCIC stopped and water injection begun was shorter than in the case of Unit 1 (see Fig.3.2.1.9).

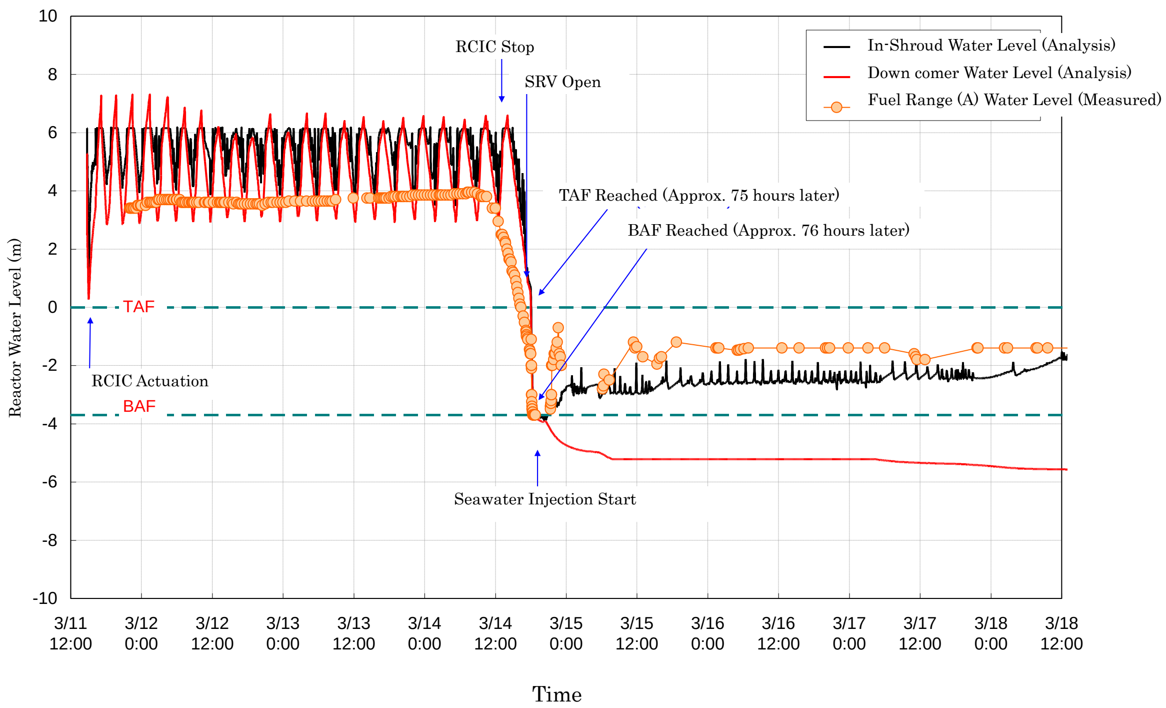


Figure3.2.1.1 Unit2 Reactor Water Level[Case1]

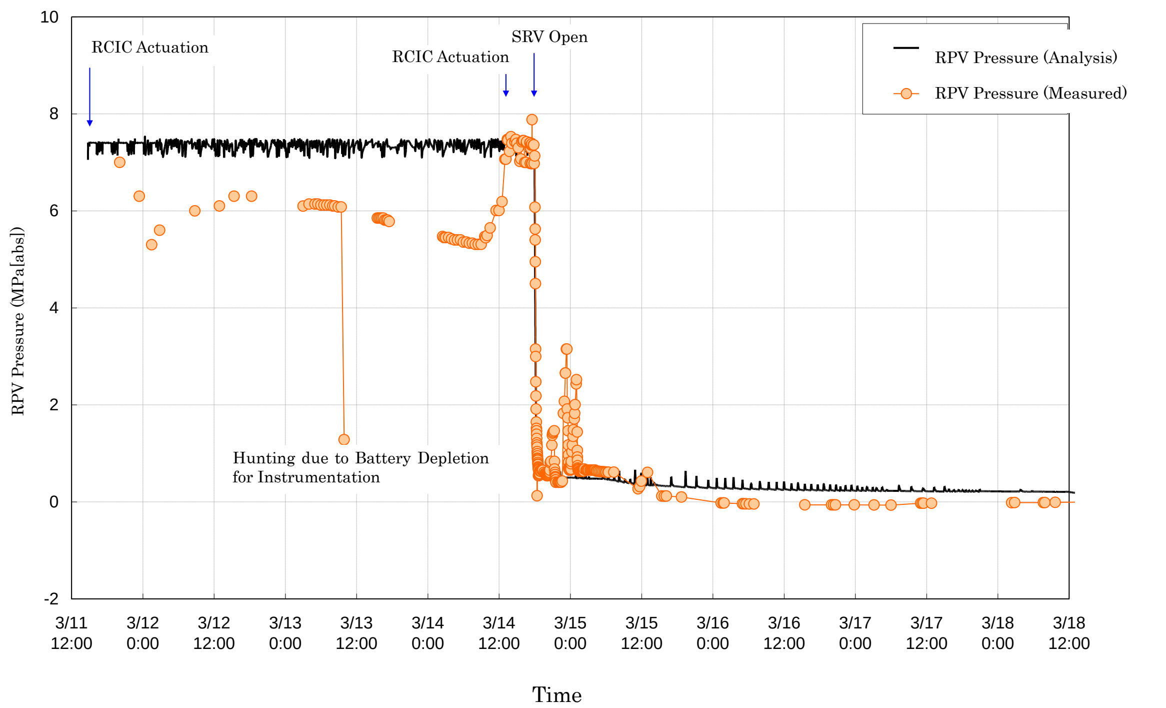


Figure3.2.1.2 Unit2 RPV Pressure[Case1]

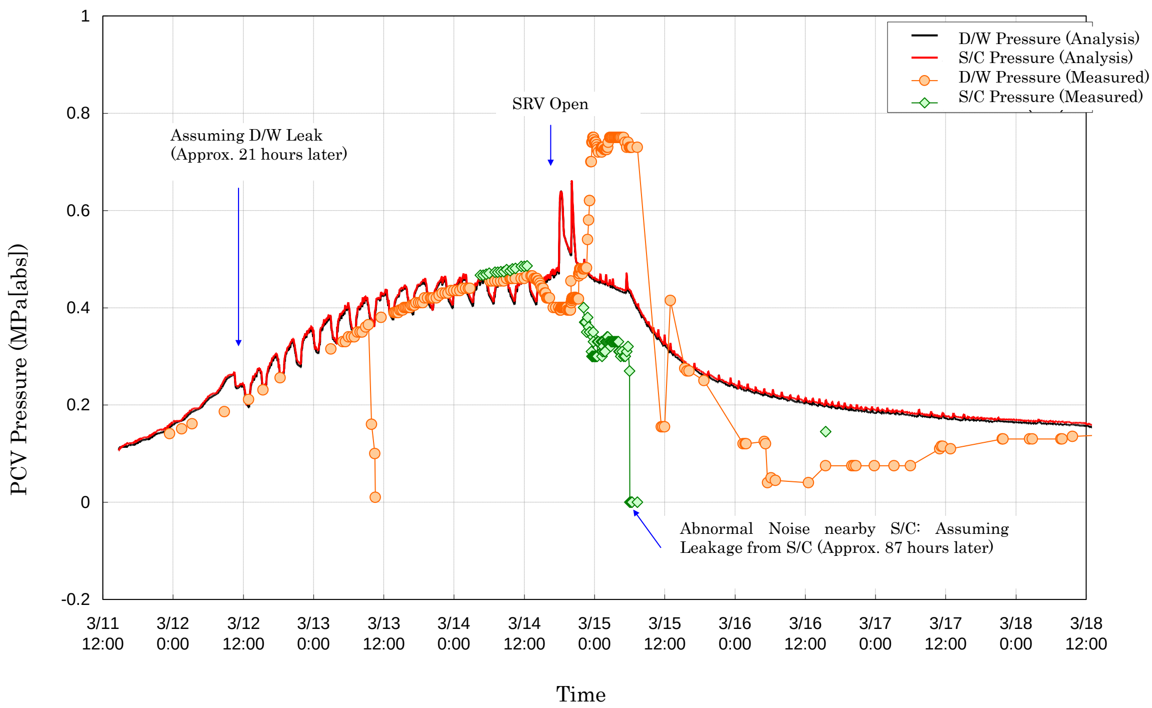


Figure3.2.1.3　Unit2 PCV Pressure[Case1]

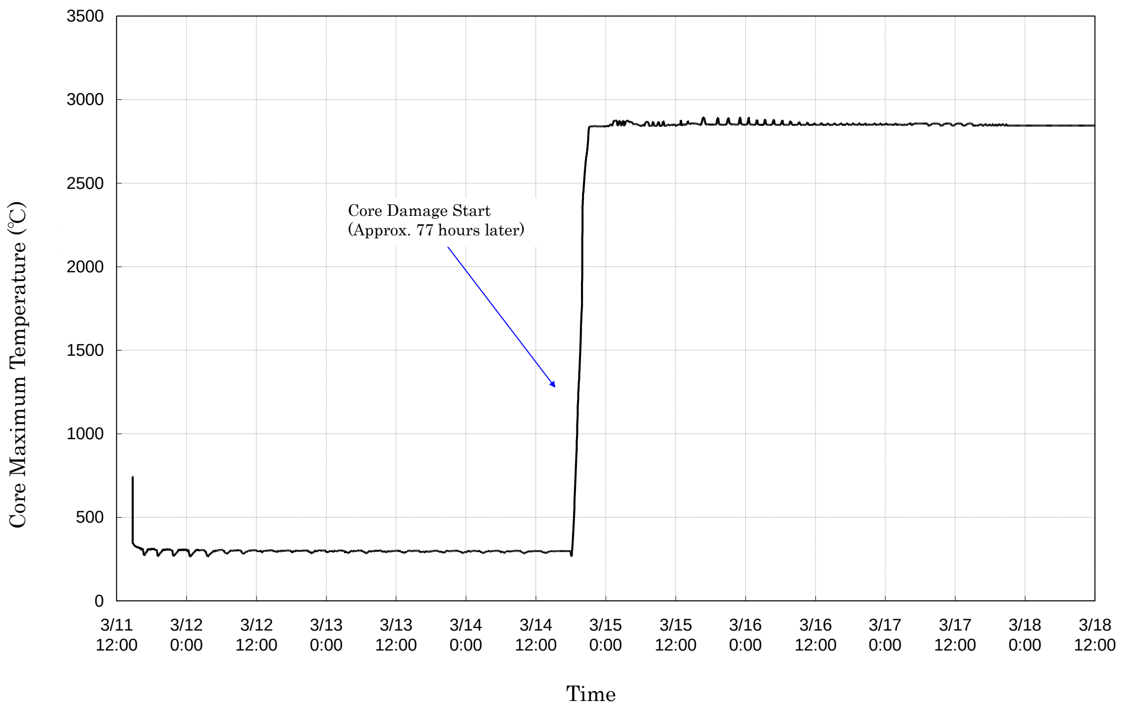


Figure3.2.1.4　Unit2 Core Temperature[Case1]

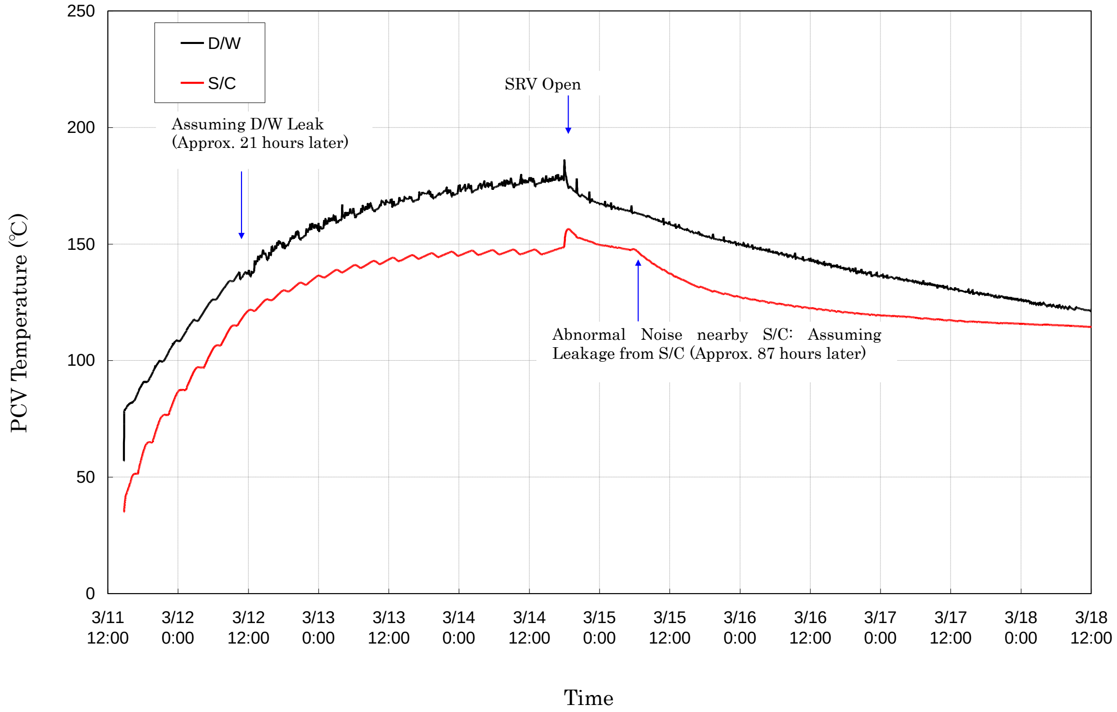


Figure3.2.1.5 Unit2 PCV Temperature[Case1]

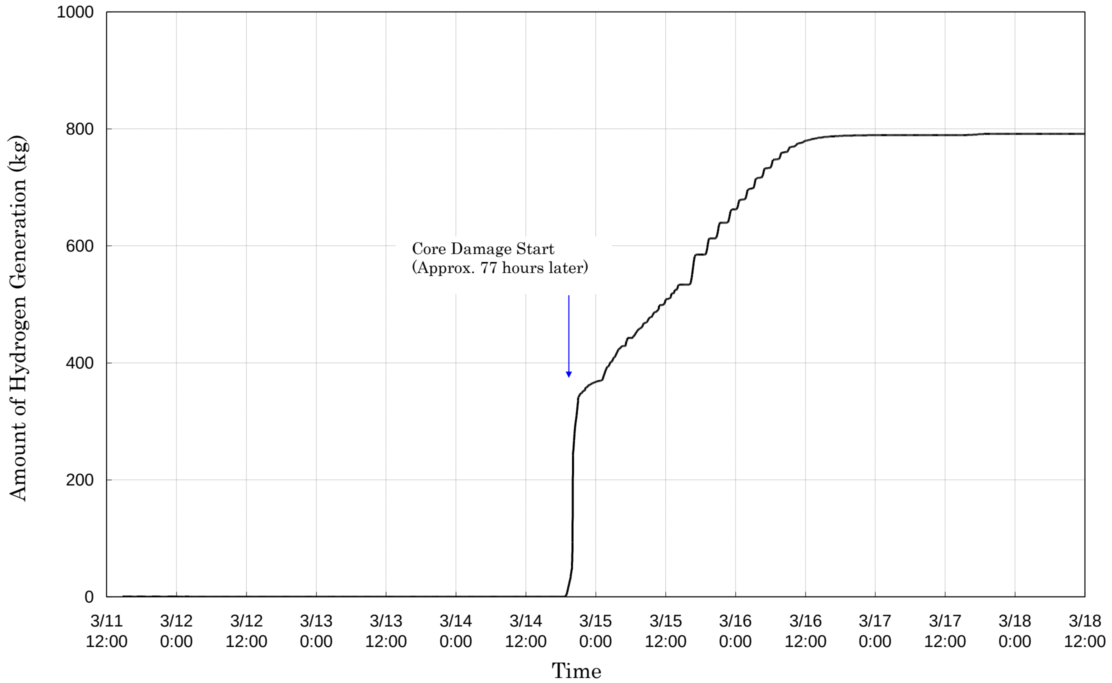


Figure3.2.1.6 Unit2 Amount of Hydrogen Generation[Case1]

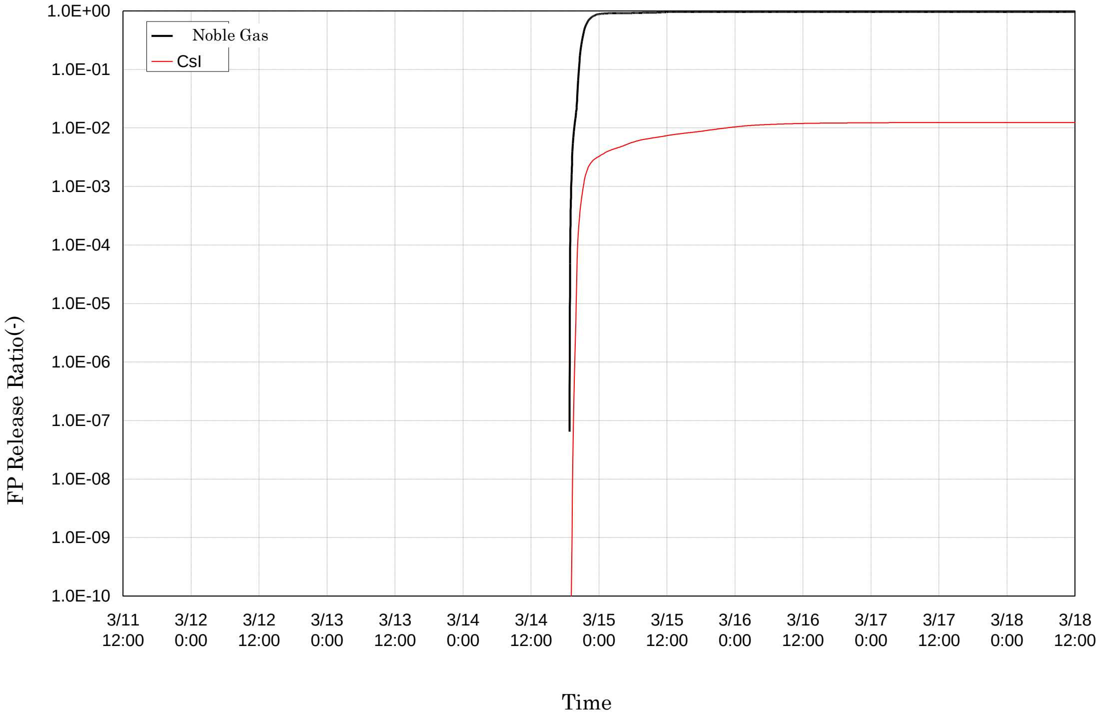


Figure3.2.1.7 Unit2 FP Release Ratio[Case1]

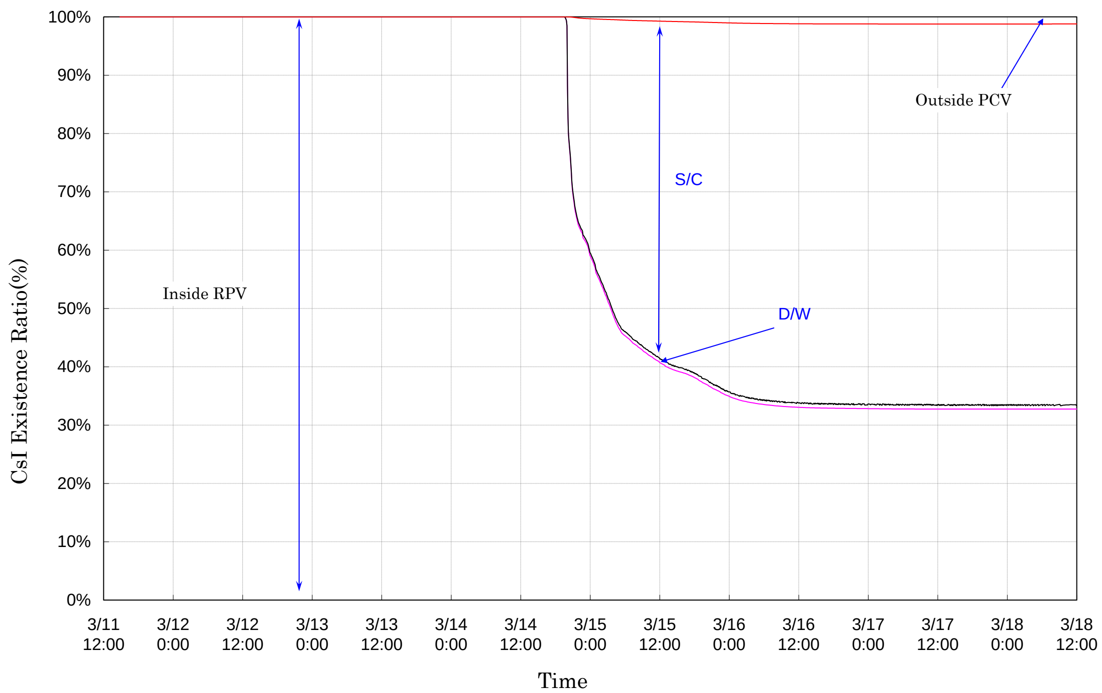


Figure3.2.1.8 Unit2 FP Existence Ratio(1/2)[Case1]

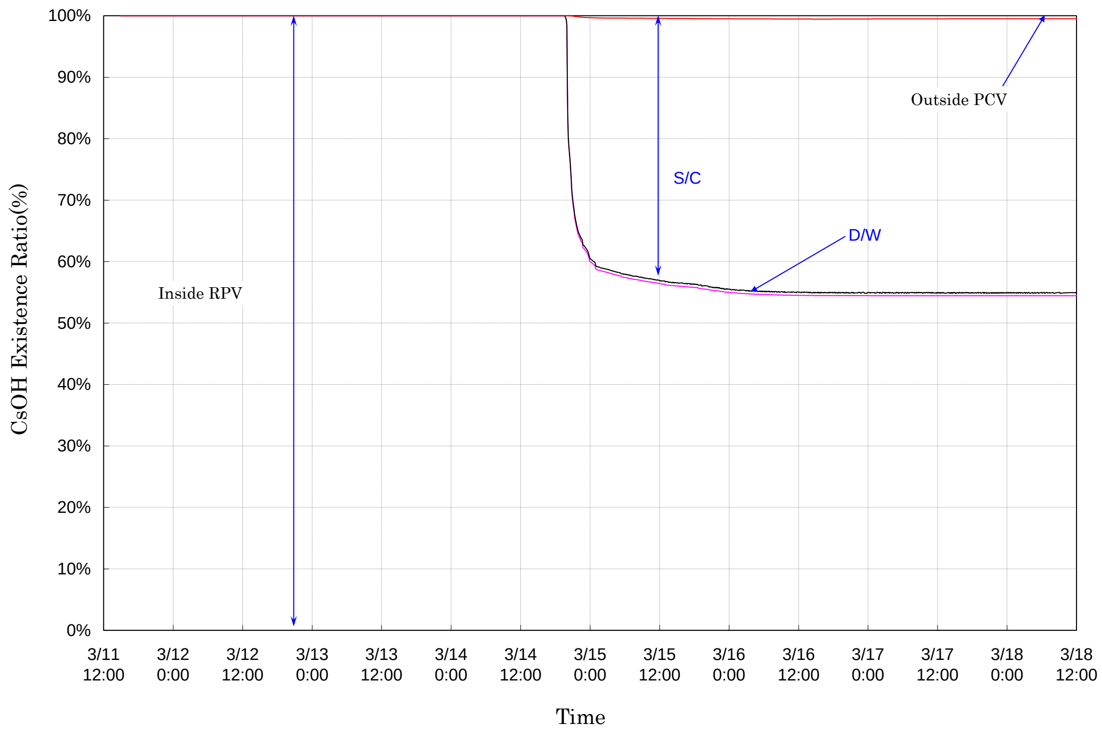
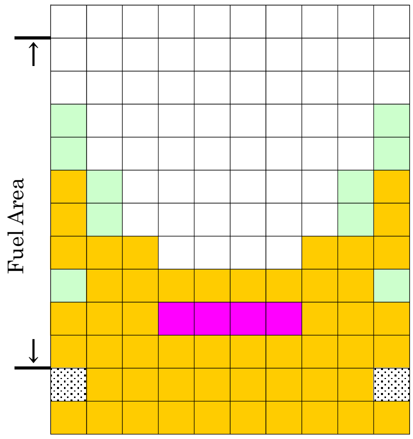
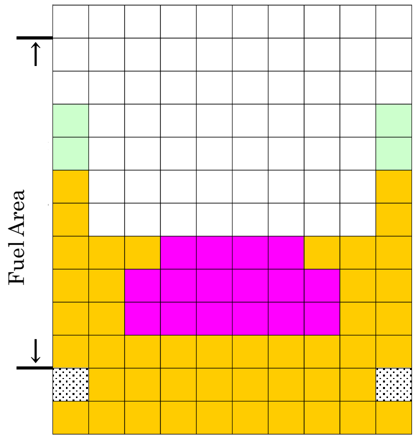


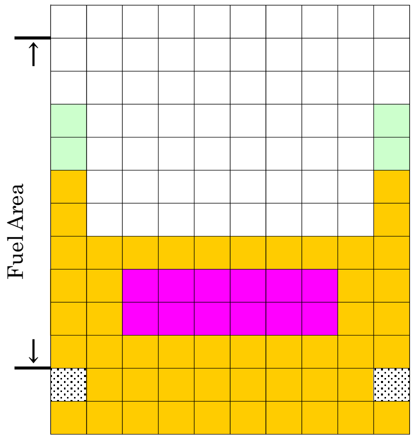
Figure3.2.1.8　Unit2 FP Existence Ratio(2/2)[Case1]



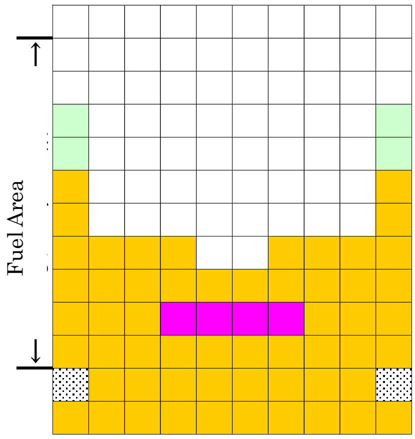
Approx. 87 hours after SCRAM



Approx. 96 hours after SCRAM



Approx. 120 hours after SCRAM



Approx. 1 week after SCRAM

Model of Fuel Damage
: No Fuel (Slumped)
: Normal Fuel
: Accumulation of Slumped Fuel
: Accumulation of Melted Fuel

: Flow Channel Blockage with Melted Fuel
: Molten Core Pool

Figure3.2.1.9 Unit2 Core Status[Case1]

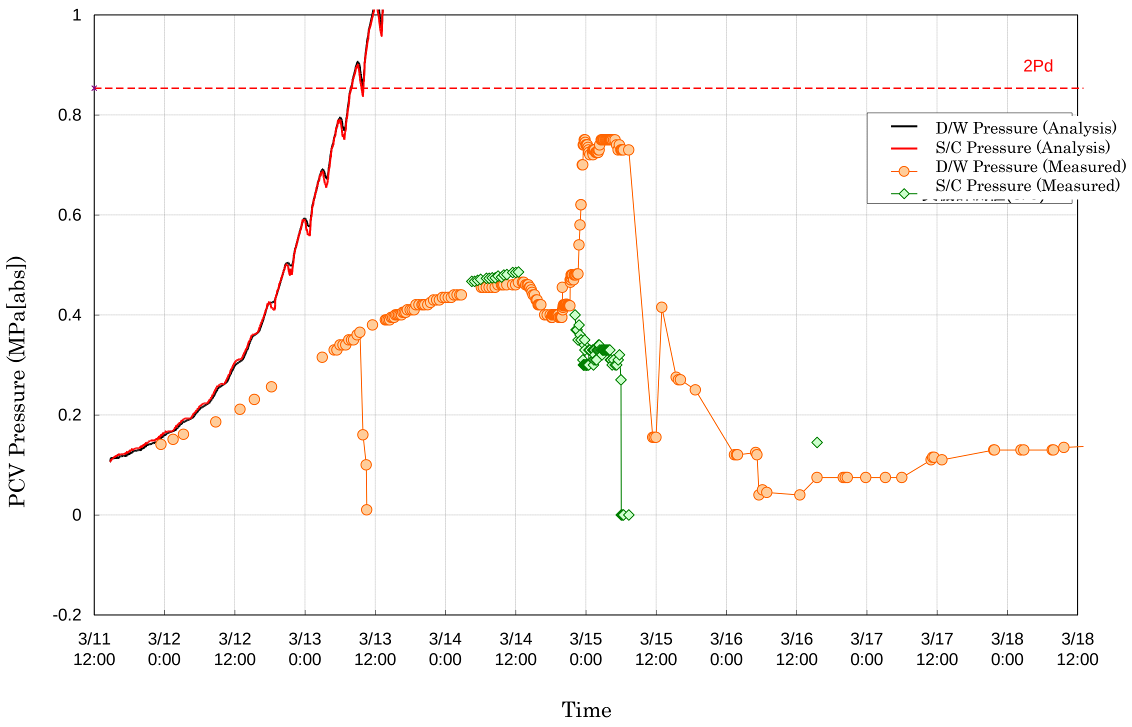


Figure3.2.1.10 Unit2 PCV Pressure[Case1] (Without Assuming Leak due to Superheat)

3.2.2.2 Analysis Result (Analysis Case 2)

Table 3.2.4 shows the result of analysis based on the condition shown in 3.2.1. And from Fig. 3.2.2.1 to Fig 3.2.2.9 show the result of analysis about the trend of reactor water level etc.

Table 3.2.4 Summary of Analysis Result on Unit 2

Item	Analysis Result
Start of reactor core exposure	Approx. 75 hours after earthquake
Start of reactor core damages	Approx. 77 hours after earthquake
Start of reactor pressure vessel breach	Approx. 109 hours after earthquake

The detail of analysis result is as follows.

The reactor water level gradually comes down after RCIC stop and the reactor core starts exposed, and the reactor core exposed completely by opening SRV and the reactor core damage starts. Although the water injection starts approximately at the same time, the assumed water injection flow is not enough so that the water level does not increase above the bottom of active fuel (see Fig.3.2.2.1).

Regarding the reactor pressure, there is a temporary increase by the steam generated by the relocation of reactor core to the lower plenum, but other behavior is almost same as the result of Case 1 (see Fig.3.2.2.2).

Regarding the pressure of PCV, it is similar to the reactor pressure, there is a temporary increase by the steam generated by the movement of reactor core to the lower plenum, but other behavior is almost same as the result of Case 1 (see Fig.3.2.2.3).

Regarding reactor core temperature trend, the temperature increases with the decrease of reactor water level, the fuel pellets start melting (see Fig.3.2.2.4).

A large amount of hydrogen is generated when the reactor core is exposed and the temperature of fuel cladding start increasing, the amount equal to the reaction of 36% of fuel cladding in the active fuel range is generated (see Fig.3.2.2.6).

Regarding the release of FP, for noble gases, the result is that almost all is released by the leak from S/C, which is similar to Case 1. For other materials such as cesium iodide, the release rate is under 1% (see Fig.3.2.2.7 and Fig.3.2.2.8).

The result says that, parts of the fuel remain in the reactor pressure vessel, but the reactor pressure vessel is breached. As the water injection flow in the early stage is assumed smaller than in Case 1, the result shows the damage of reactor core being more serious (see Fig.3.2.2.9).

3.2.3 Evaluation Result

In Analysis Case 1, we have had an analysis result that the reactor core of Unit 2 stays within the fuel range, though a partial fuel-melting pool exists, and the reactor pressure vessel will not be breached. In Analysis Case 2, we have had an analysis result that the reactor pressure vessel has been breached, though part of fuel stays within the reactor pressure vessel.

In addition, as a result of calibration of water-level gauge of Unit 1, we have found that the water level in reactor pressure vessel is not within fuel ranges. We cannot deny the possibility that a similar event has occurred in Unit 2.

Most of fuel is cooled down in the reactor pressure vessel, because, according to plant parameters, the temperatures at the bottom of the reactor pressure vessel currently change around between approximately 100 degrees Celsius and approximately 120 degrees Celsius and correlate changes in amount of water injection in multiple measuring points and temperature of the upper reactor pressure vessel is higher and its heat source is estimated to come from the inside of reactor pressure vessel.

Hence, according to the analysis and plant parameters, reactor core has been significantly damaged, relocated below or slumped to the lower plenums from fixed positions of fuel loading, however most of it is being stably cooled down around there.

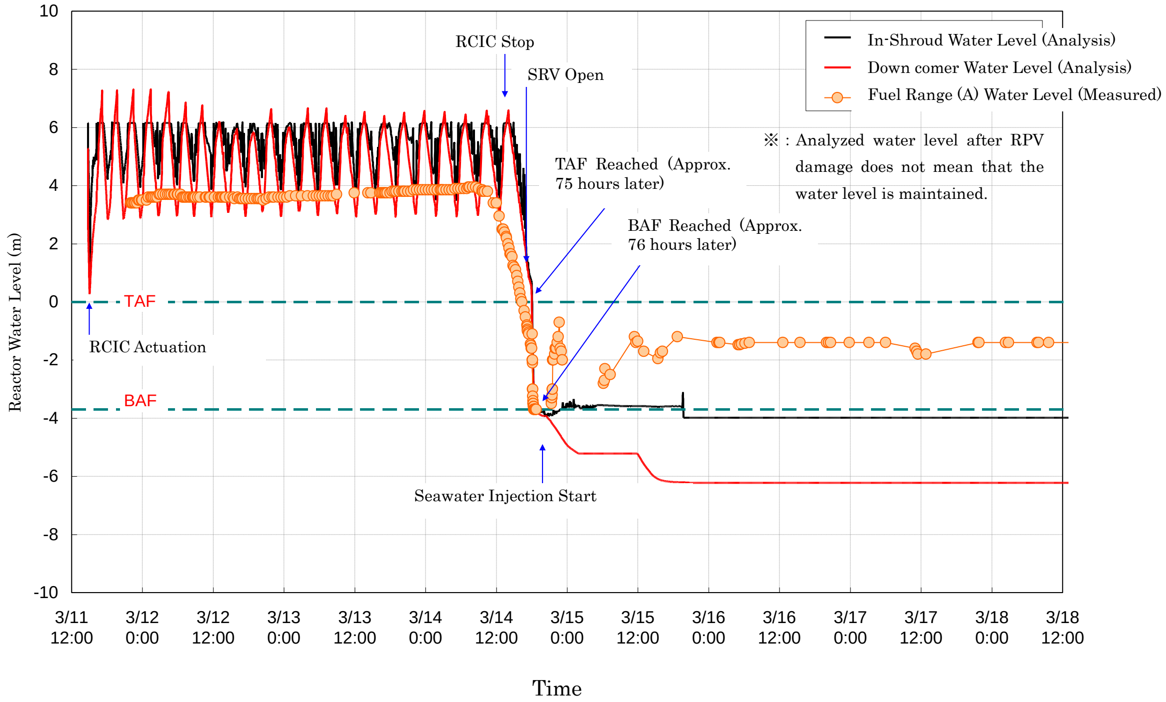


Figure3.2.2.1 Unit2 Reactor Water Level[Case2]

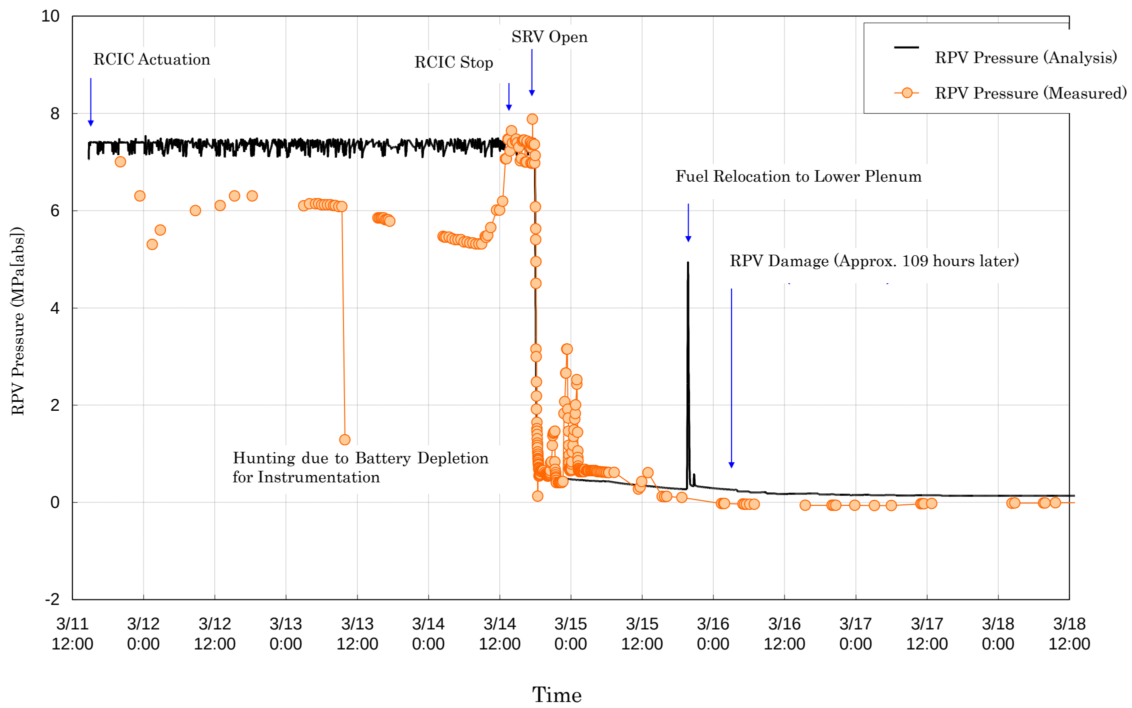


Figure3.2.2.2 Unit2 RPV Pressure[Case2]

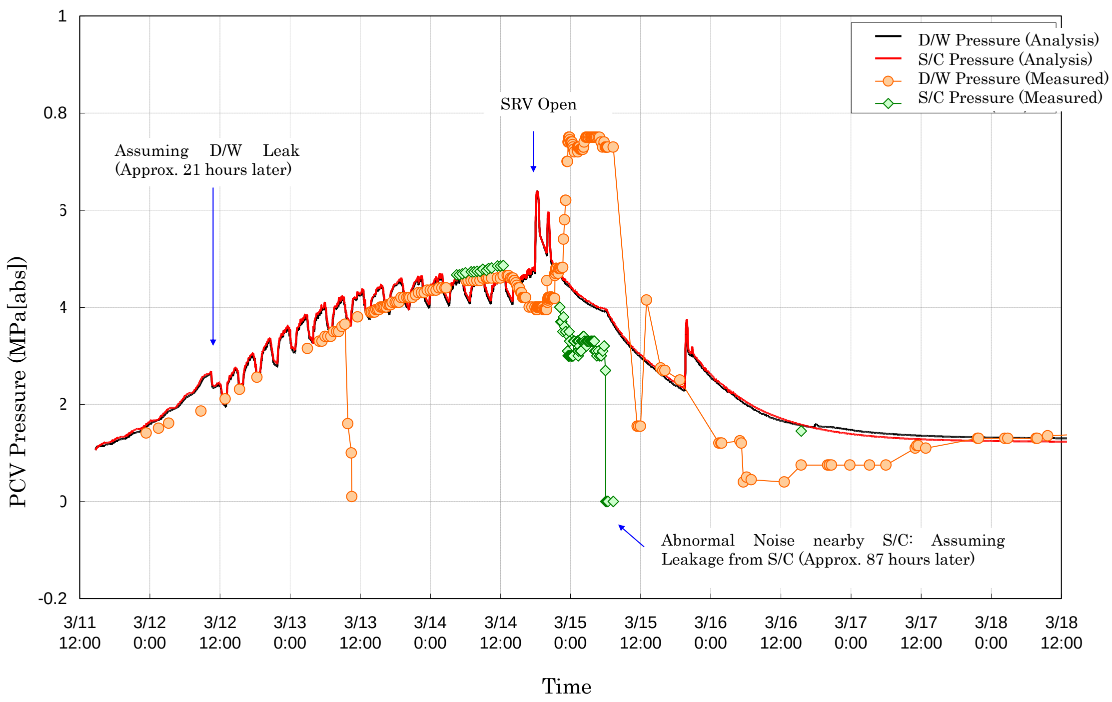


Figure3.2.2.3　Unit2 PCV Pressure[Case2]

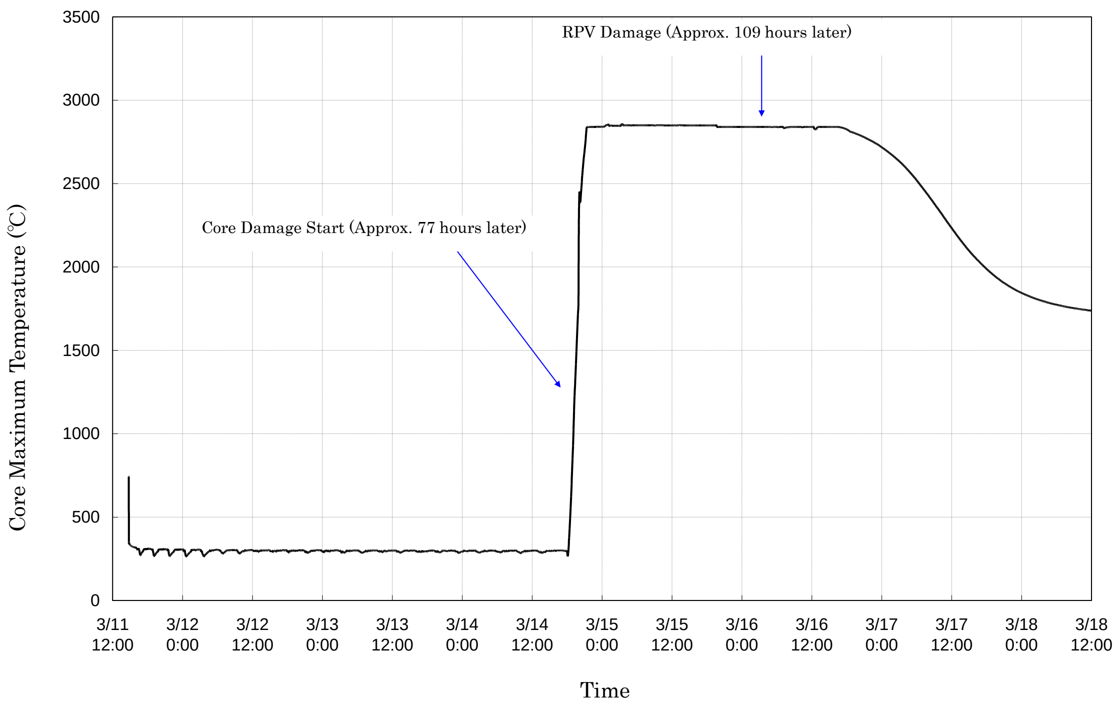


Figure3.2.2.4　Unit2 Core Temperature[Case2]

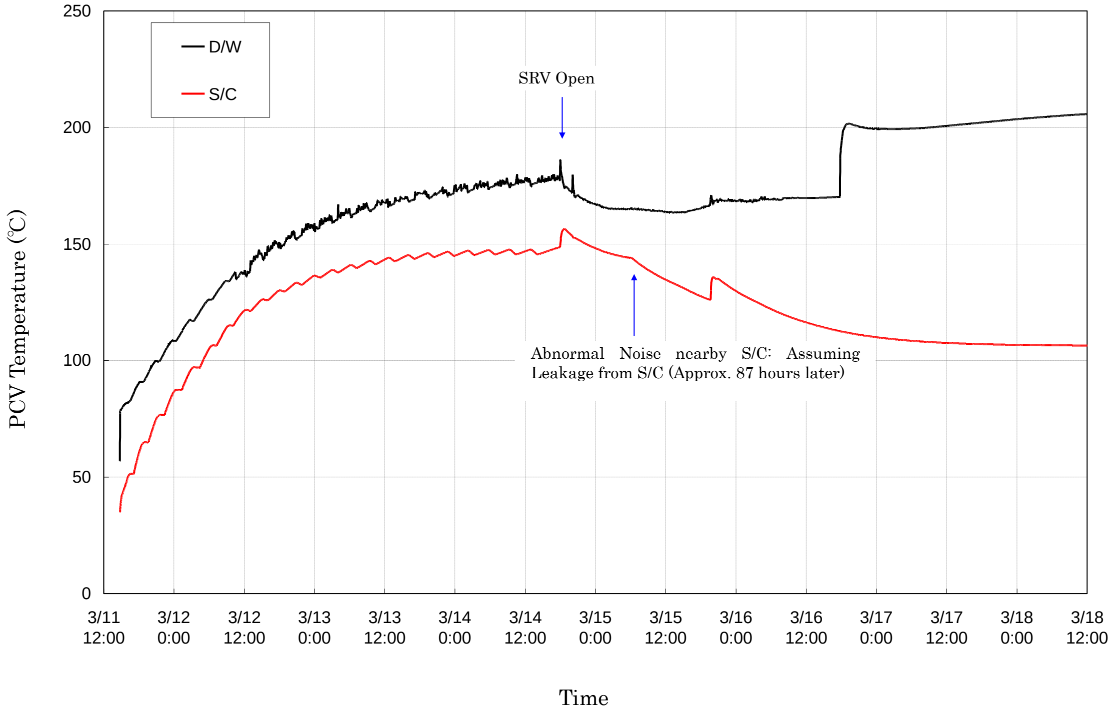


Figure3.2.2.5 Unit2 PCV Temperature[Case2]

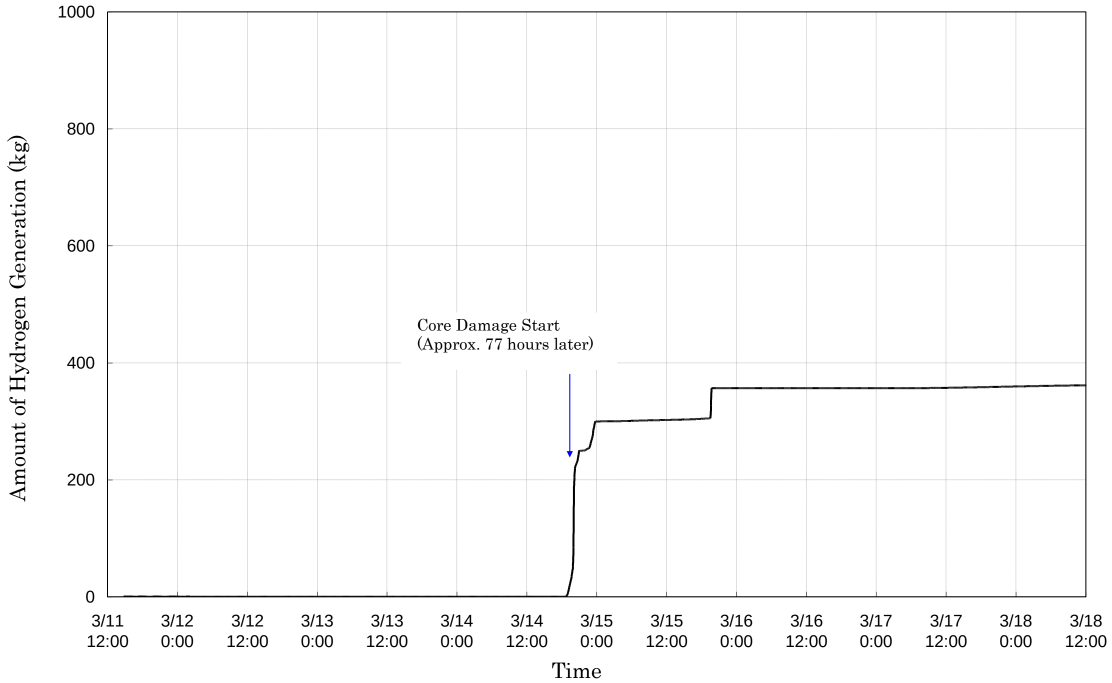


Figure3.2.2.6 Unit2 Amount of Hydrogen Generation[Case2]

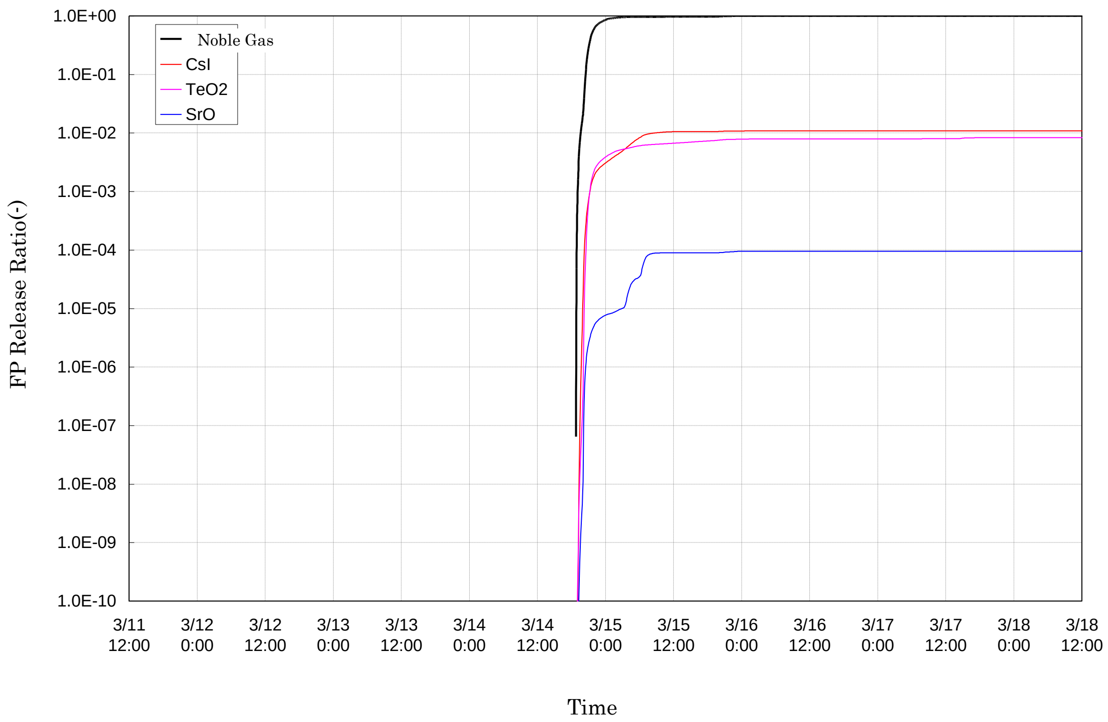


Figure3.2.2.7　Unit2 FP Release Ratio(1/3)[Case2]

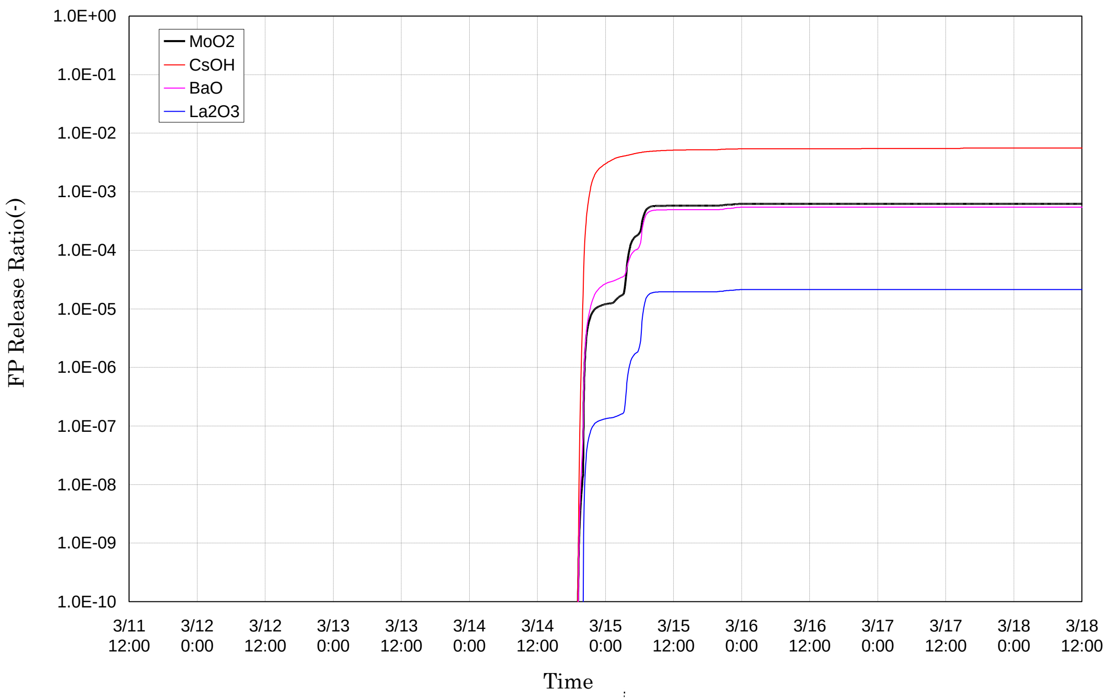


Figure3.2.2.7　Unit2 FP Release Ratio(2/3)[Case2]

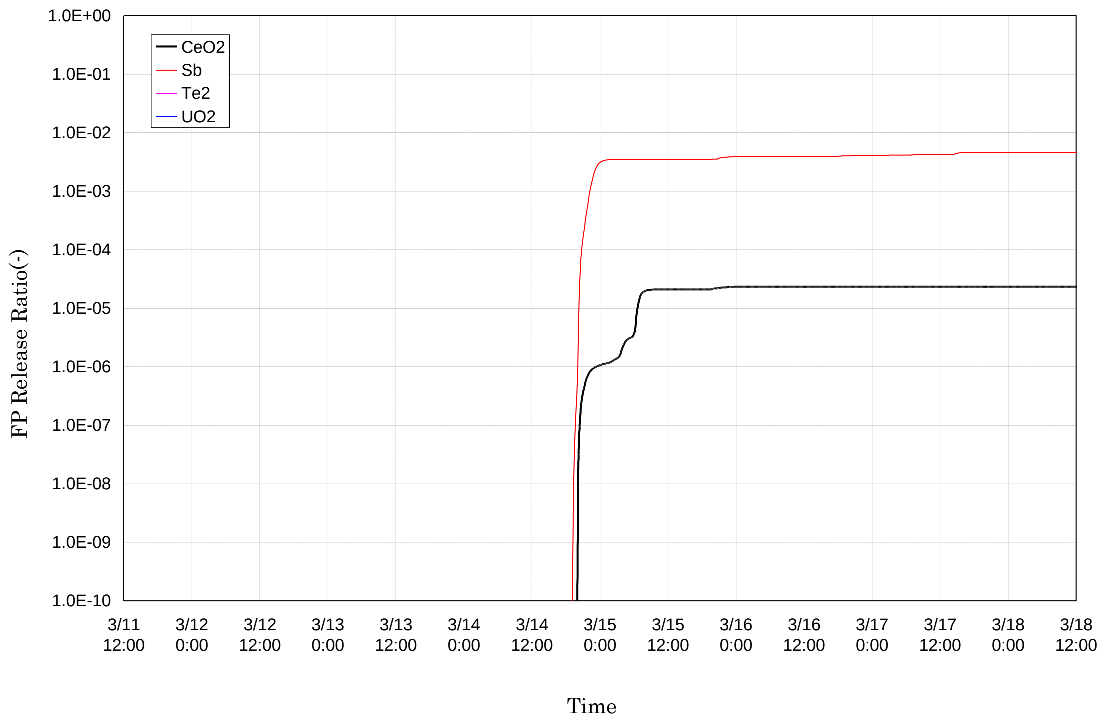


Figure3.2.2.7 Unit2 FP Release Ratio(3/3)[Case2]

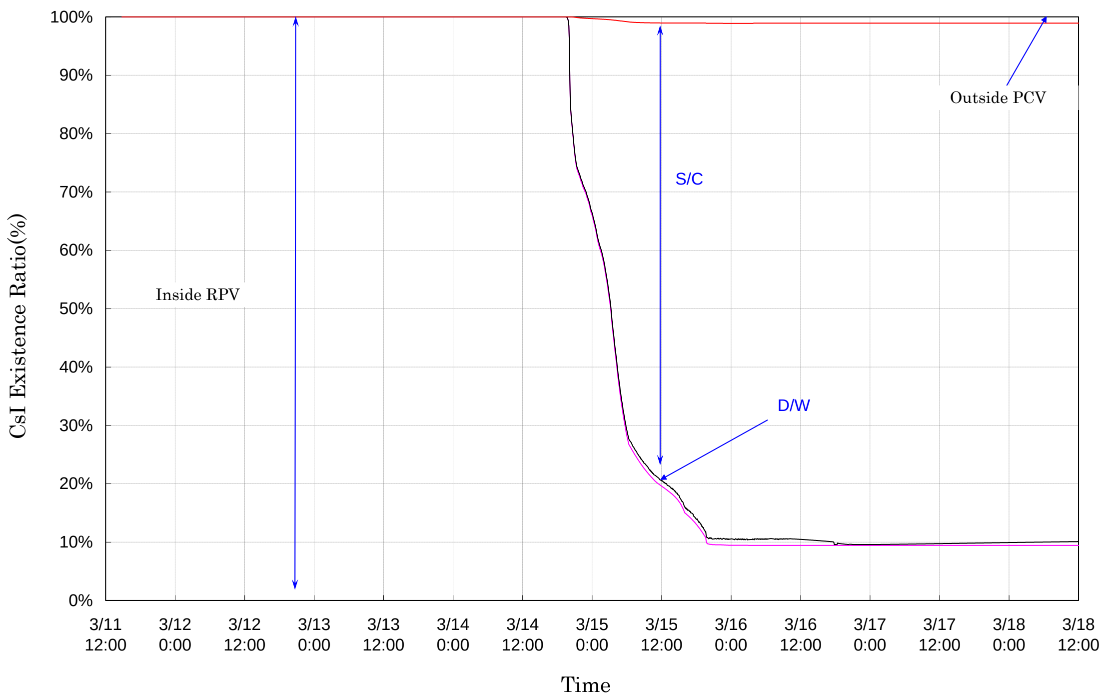


Figure3.2.2.8 Unit2 FP Existence Ratio(1/2)[Case2]

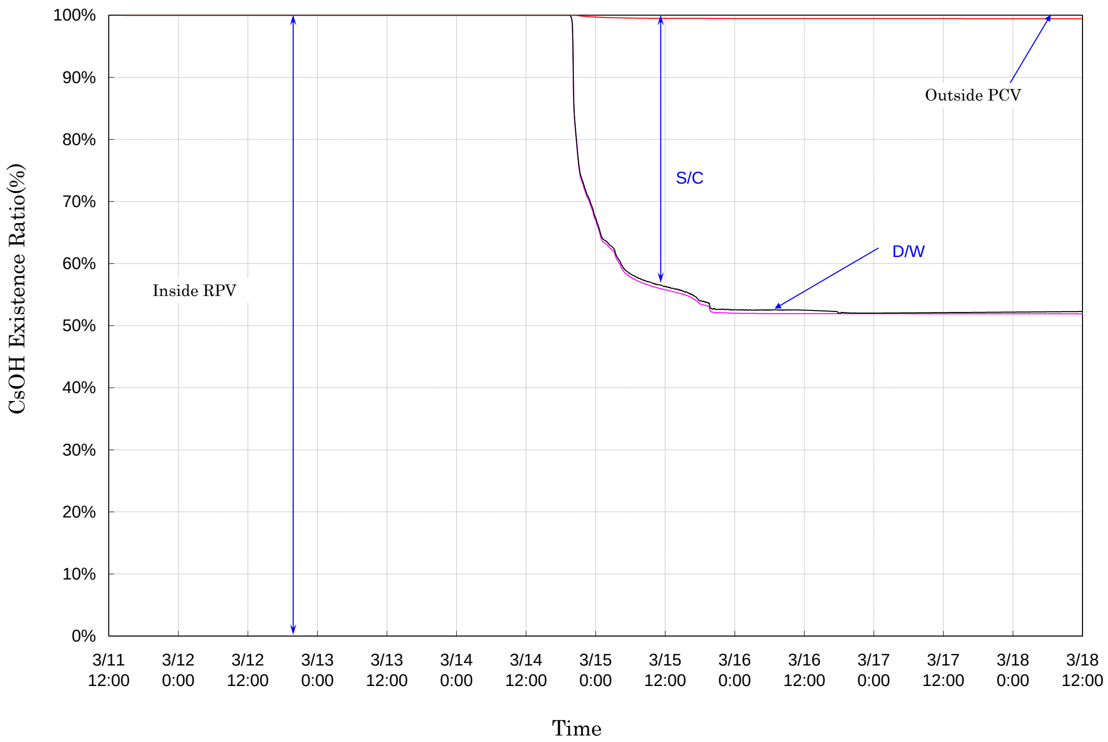
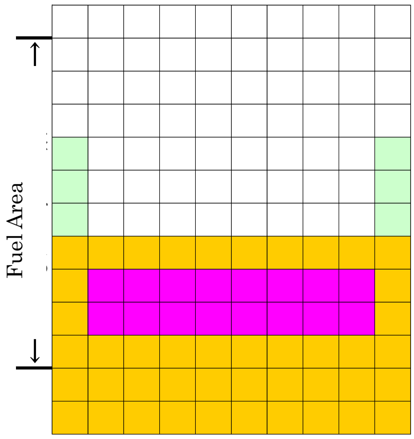
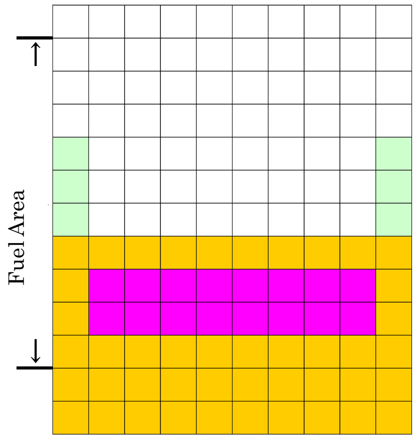


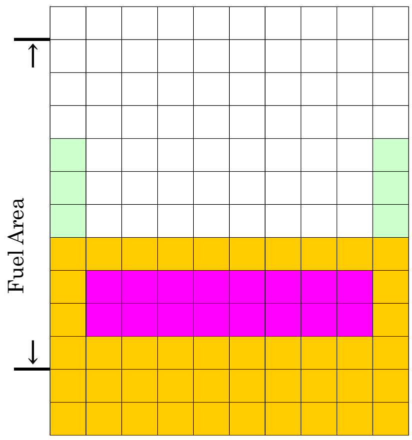
Figure3.2.2.8 Unit2 FP Existence Ratio(2/2)[Case2]



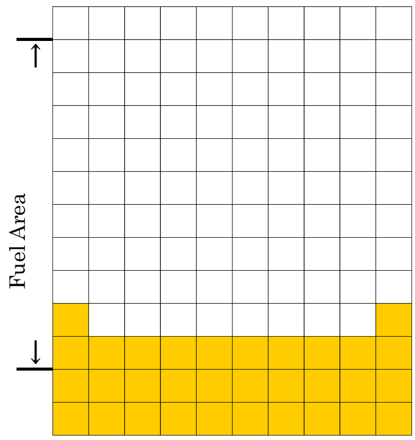
Approx. 87 hours after SCRAM



Approx. 96 hours after SCRAM



Approx. 100 hours after SCRAM



Approx. 109 hours after SCRAM

Model of Fuel Damage
: No Fuel (Slumped)
: Normal Fuel
: Accumulation of Slumped Fuel
: Accumulation of Melted Fuel

: Flow Channel Blockage with Melted Fuel
: Molten Core Pool

Figure3.2.2.9 Unit2 Core Status[Case2]

3.3 Unit 3 of Fukushima Daiichi Nuclear Power Station

3.3.1 Analysis condition

Principal conditions of analysis regarding Unit 3 of Fukushima Daiichi Nuclear Power Station are shown in the Table 3.3.1 and 3.3.2.

We implemented the analysis in two cases below.

① Cases of analysis

【Case 1】 In order to match the measured value of reactor water level, we presumed a smaller amount of flow rate compared to the discharge flow of the fire pump that could maintain the reactor water level.

【Case 2】Based on the premise that it is impossible to maintain the reactor water level in the fuel range, we presumed an injection rate that can maintain slightly below the level of the fuel range, and not the flow rate of the discharge side of the fire pump.

Table 3.3.1 Plant Conditions

Items	Conditions
Initial reactor output	2381 MWt (rated power output)
Initial reactor pressure	7.03MPa[abs] (normal operation pressure)
Initial reactor water level	Normal level
Open space volume of PCV	D/W open space : 4240m^3 S/C open space : 3160m^3
Suppression pool water volume	2980m^3

Chart 3.3.2 Events

Explanatory notes ○ : Records available △ : Estimates based on records □ : Assumption used on analysis

Analysis Condition				Classification	Notes	In case of ○ : Referred part of the records In case of △ or □ : Estimated, presumed reasons etc.
No	Time and Date		Analyzed Events			
1	March 11th	2:46 pm	Earthquake occurred	○	―	
2		2:47 pm	Reactor scram occurred	○	Report on May 16th, 4. Operation daily, Handover diary of shift supervisor	
3		3:06 pm	RCIC activated manually	○	Report on May 16th, 7. Operation records	
4		3:25 pm	RCIC tripped (L-8)	○	Same as above	
5		3:38 pm	SBO occurred	○	Report on May 16th, 4. Operation daily, Handover diary of shift supervisor	
6		4:03 pm	RCIC activated manually	○	Report on May 16th, 7. Operation records	
7	March 12th	11:36 am	RCIC tripped	○	Same as above	
8		0:35 pm	HPCI activated (L-2)	○	Same as above	
9	March 13th	2:42 am	HPCI stopped	○	Same as above	
10		Around 9:08 am	Started the pressure decrease of reactor pressure vessel by operating the SRV	○	Same as above	
11		9:20 am	Regarding the PCV vent, pressure decrease was	○	The vent line constitution is completed by the operation of AO valve in the 7.Operation records in	

			confirmed		the report on May 16th, however, we presumed the 9:20am as the start of the vent, since we confirmed the pressure decrease of PCV at this time.
12		9:25 am	Started the injection of fresh water	○	Report on May 16th, 7. Operation records ※1
13		11:17 am	Regarding the PCV vent, a closure of AO valve of vent line caused by the slip out of driving air pressure was confirmed	○	Report on May 16th, 7. Operation records
14		0:30 pm	Regarding the PCV vent, opening operation was implemented	○	Same as above
15		1:12 pm	Injection of water was changed from fresh water to sea water	○	Report on May 16th, 7. Operation records ※1
16		2:10 pm	Regarding the PCV vent, it is presumed that the vent valve is closed	△	We presume the termination of the vent which started 0:30pm, March 13th at this time based on the increase of D/W pressure. It is described in the 7.Operation records in the report on May 16th that the closure of the valve is confirmed at 4:00pm, May 15th.
17	March 14th	1:10 am	Injection of water was halted in order to supply water to water	○	Report on May16th, 7. Operation records

			source pit		
18		3:20 am	The supply to the water source pit was finished and the injection of water restarted	○	Report on May16th, 7. Operation records ※1
19		5:20 am	Regarding the PCV vent, AO valve of suppression chamber side was opened.	○	Report on May16th, 7. Operation records
20		0:00 pm	Regarding the PCV vent, it is presumed that the valve of suppression chamber side was closed.	△	We presume the termination of the vent which started 5:30am, March 14th at this time based on the increase of D/W pressure. It is described in the 7.Operation records in the report on May 16th that the closure of the valve is confirmed at 4:00pm, May 15th.
21		4:00 pm	Regarding the PCV vent, it is presumed that the valve of suppression chamber side was opened.	△	We presume the vent at this time based on the decrease of D/W pressure.
22		9:04 pm	Regarding the PCV vent, it is presumed that the valve of suppression chamber side was closed.	△	We presume the termination of the vent at this time based on the increase of D/W pressure.
23	March 15th	4:05 pm	Regarding the PCV vent, the valve of suppression chamber	○	Report on May16th, 7. Operation records

			side was opened.		
24	March 16th	1:55 am	Regarding the PCV vent, the valve of suppression chamber side was opened.	△	It is described in the 7.Operation records in the report on May 16th that the vent was implemented at this time, however, we presume that the vent was not implemented as the D/W pressure did not fluctuate.
25	March 17th	9:00 pm	Regarding the PCV vent, closure of the valve of suppression chamber side was confirmed.	△	It is described in the 7.Operation records in the report on May 16th that the closure of the valve was confirmed following the opening of the vent at 4:05pm March 15th, however, we presume that the valve was not closed based on the transition of the D/W pressure.
26		9:30 pm	Regarding the PCV vent, the valve of suppression chamber side was opened.	△	It is described in the 7.Operation records in the report on May 16th that the valve was opened, however, we presume that the valve was not opened based on the transition of the D/W pressure.
27	March 18th	5:30 am	Regarding the PCV vent, closure of the valve of suppression chamber side was confirmed.	—	Though it is described in the report on May 16th, it is out of the period for the analysis.
28		Around 5:30 am	Regarding the PCV vent, the valve of suppression chamber side was opened.	—	Same as above

29	March 19th	11:30 am	Regarding the PCV vent, closure of the valve of suppression chamber side was confirmed.	—	Same as above
30	March 20th	Around 11:25 am	Regarding the PCV vent, the valve of suppression chamber side was opened.	—	Same as above

※1 We set up the injection amount of water and the time we changed the amount based on the daily injection amount described in 7.Operation records of the report on March 16th, in order not to exceed the daily average injection amount and total injection amount.

3.3.2.1 Analysis Result (Analysis Case 1)

Table 3.3.3 shows the result of analysis based on the condition shown in 3.3.1. Fig. 3.3.1.1 to Fig 3.3.1.13 show the result of analysis about the trend of reactor water level etc.

Table 3.2.3 Summary of Analysis Result on Unit 3

Item	Analysis Result
Start of reactor core exposure	Approx. 40 hours after earthquake
Start of reactor core damages	Approx. 42 hours after earthquake
Start of reactor pressure vessel breach	(reactor pressure vessel breach did not occur in this analysis)

The detail of analysis result is as follows.

The reactor water level gradually comes down after HPCI stop and the reactor core starts exposed, and the reactor core exposed completely by opening SRV and the reactor core damage starts (see Fig.3.3.1.1). Although the water injection starts, but in this analysis, because the water injection flow is assumed that it is commensurate with the reactor water level indicated by measurement equipment, so that the water injection flow is not enough and the water level is remained around a half level of the reactor core range. Therefore, the reactor core is damaged.

The reactor pressure is kept high around the pressure of SRV action until RCIC and HCPI stop. By SRV opening after HCPI stops, the reactor pressure decreases rapidly, after that it goes down to around atmosphere pressure (see Fig.3.3.1.2). In the analysis, RCIC and HPCI are assumed to be in operation continuously, but there exist a pressure decrease in the part while HPCI is in operation. For example, if the leak of steam is assumed out of the PCV through the steam pipe of HPCI, the result is that the trend of pressure of reactor pressure vessel and PCV is almost similar (see Fig.3.3.1.10 and Fig.3.3.1.11). However, currently it is not sure whether there is an actual path at the HPCI system or it is only a problem of instrumentation.

Regarding the pressure of PCV, because the steam generated in the reactor is discharged to S/C, the pressure of D/W and S/C continue to increase. And the pressure increases temporary by SRV open, but the pressure decreases by S/C vent. After that the pressure repeats increasing and decreasing (see Fig.3.3.1.4).

A large amount of hydrogen is generated when the reactor core is exposed and the temperature of fuel cladding starts increasing. After 1 week of earthquake occurrence, the amount equal to the reaction of 70% of fuel cladding in the active fuel range is generated. In the analysis, most of the hydrogen is discharged out of PCV by S/C vent, but the amount of production assumed to be enough for causing explosion of the Reactor Building of Unit 3 (see Fig.3.3.1.6).

Regarding the release of FP, after reactor core damage, noble gases are released from the reactor pressure vessel to S/C, and the result is that approximately 86% of noble gas is released by vent. For cesium iodide, the release rate is about 0.5% and almost all exists in S/C (see Fig.3.3.1.7 and Fig.3.3.1.8).

Regarding the situation of reactor core, there partially exist a melted pool, but it remains in fuel range and does not lead to the reactor pressure vessel breach. The reason is that, the water injection by RCIC, HCPI in the early stage was implemented constantly, and the period between when the HPCI stopped and water injection begun was shorter than that in Unit 1 (see Fig.3.3.1.9).

Also, in this analysis, the water injection suspends 2 hours for the refill of water for the water source pit. We also implemented the analysis in case that the water injection went continuously. As a result, the reactor water level remains a little higher in the early stage, but is not enough to fill up the fuel range, so that the reactor core is damaged (see Fig.3.3.1.12 and Fig.3.3.1.13).

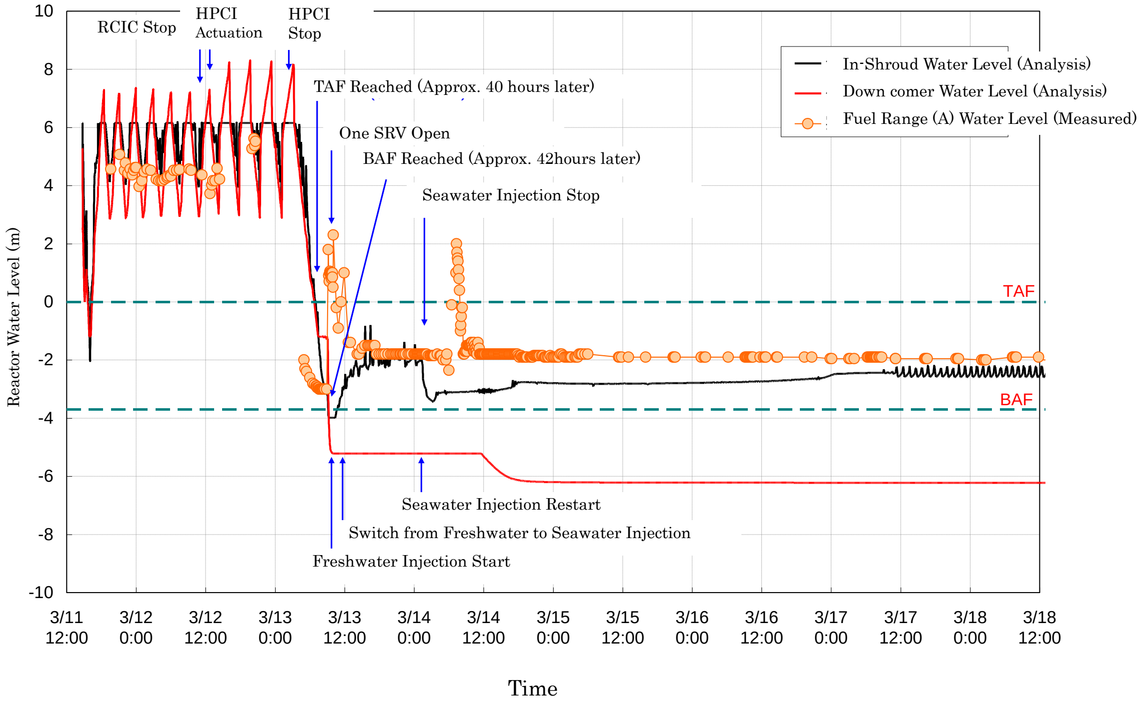


Figure3.3.1.1 Unit3 Reactor Water Level[Case1]

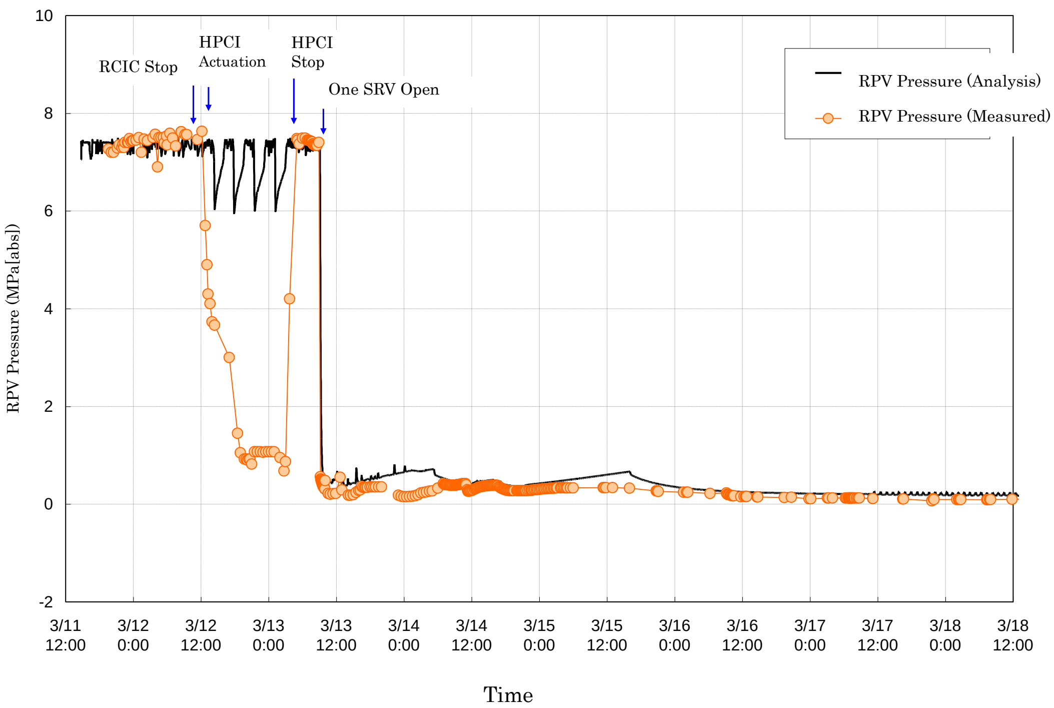


Figure3.3.1.2 Unit3 RPV Pressure[Case1]

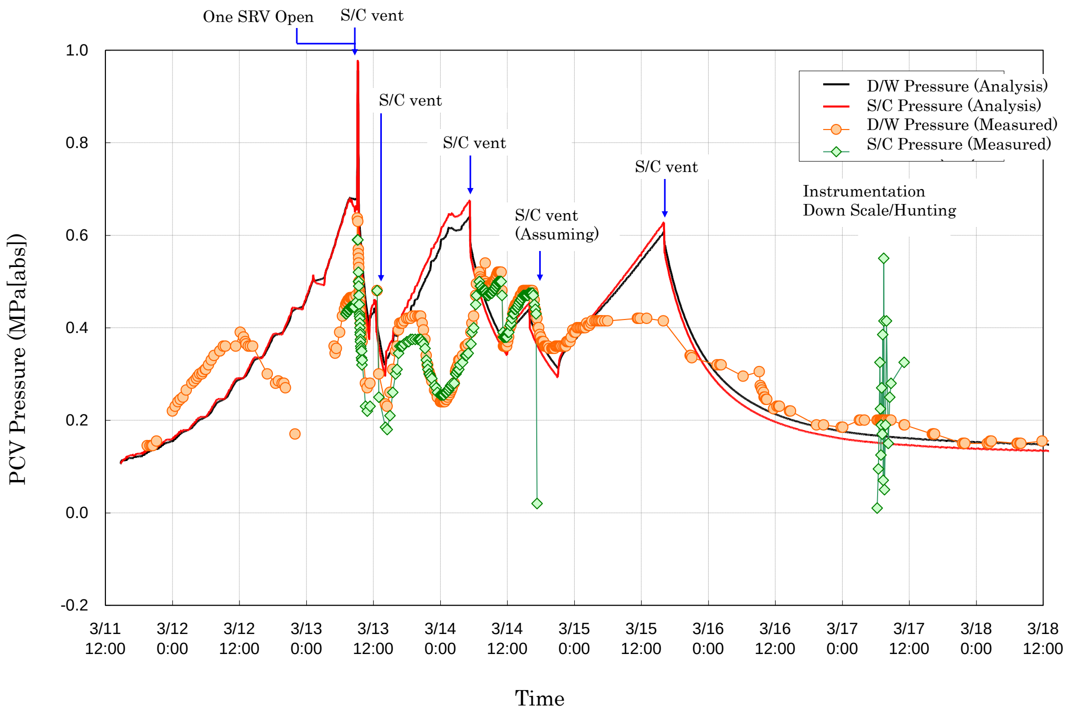


Figure3.3.1.3　Unit3 PCV Pressure[Case1]

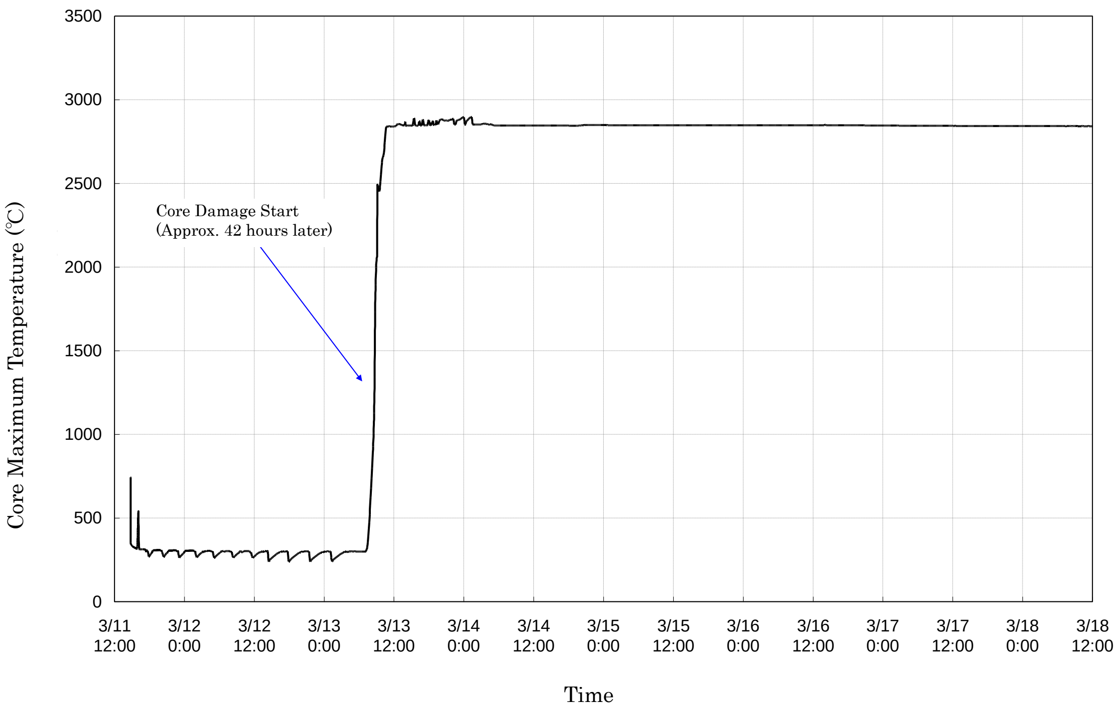


Figure3.3.1.4　Unit3 Core Temperature[Case1]

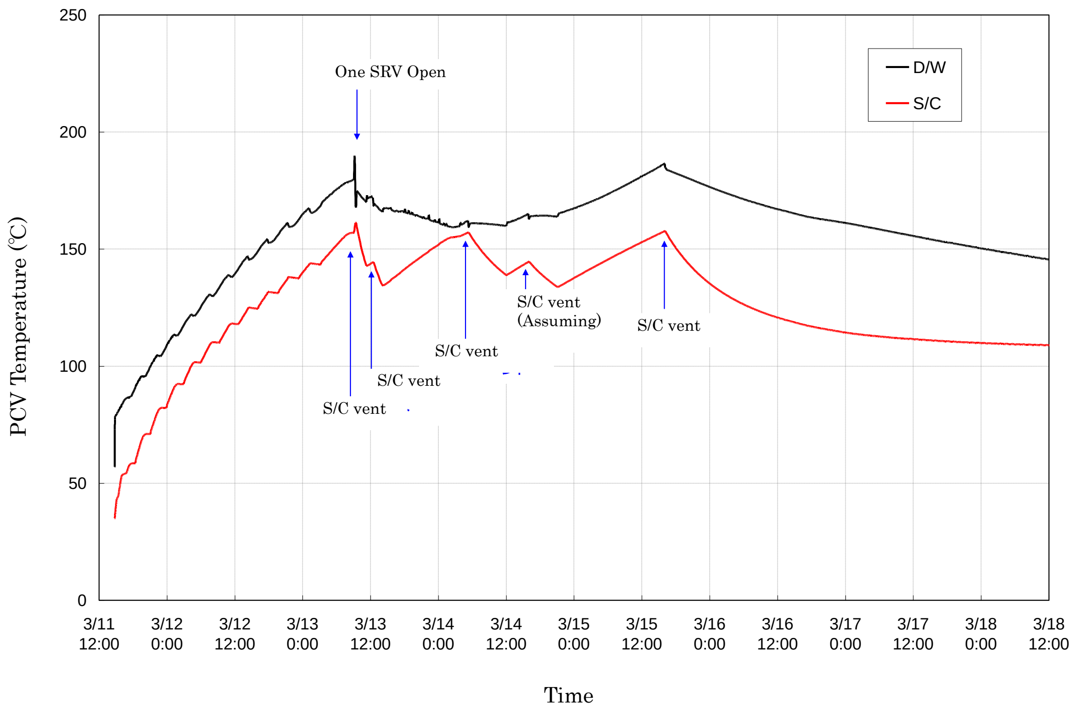


Figure3.2.1.5　Unit3 PCV Temperature[Case1]

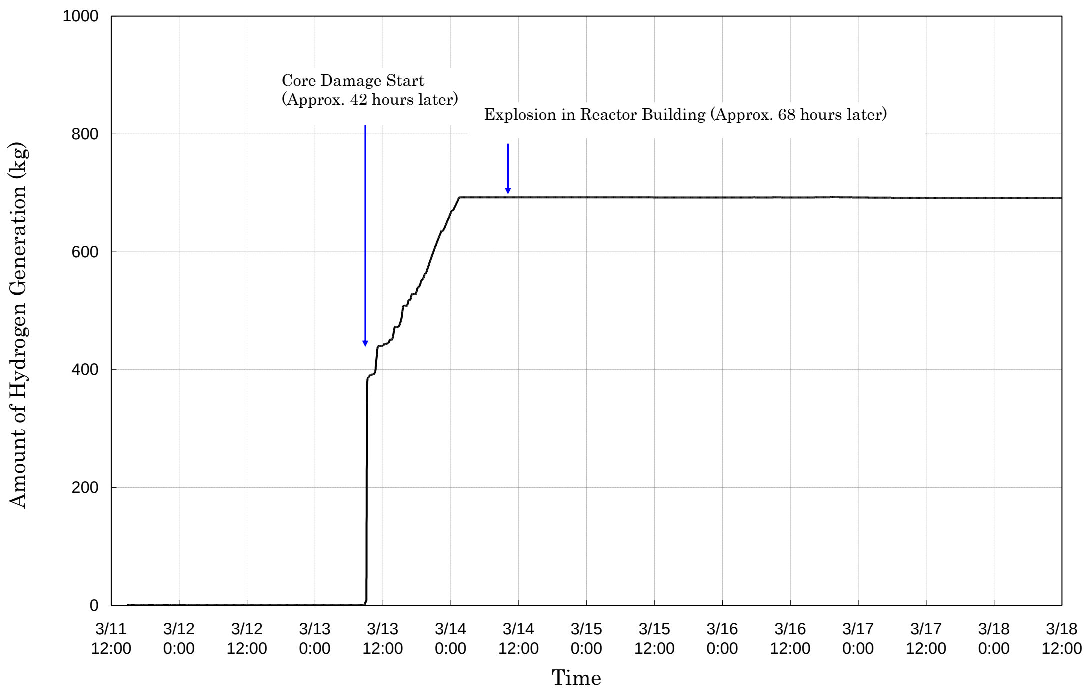


Figure3.3.1.6　Unit3 Amount of Hydrogen Generation[Case1]

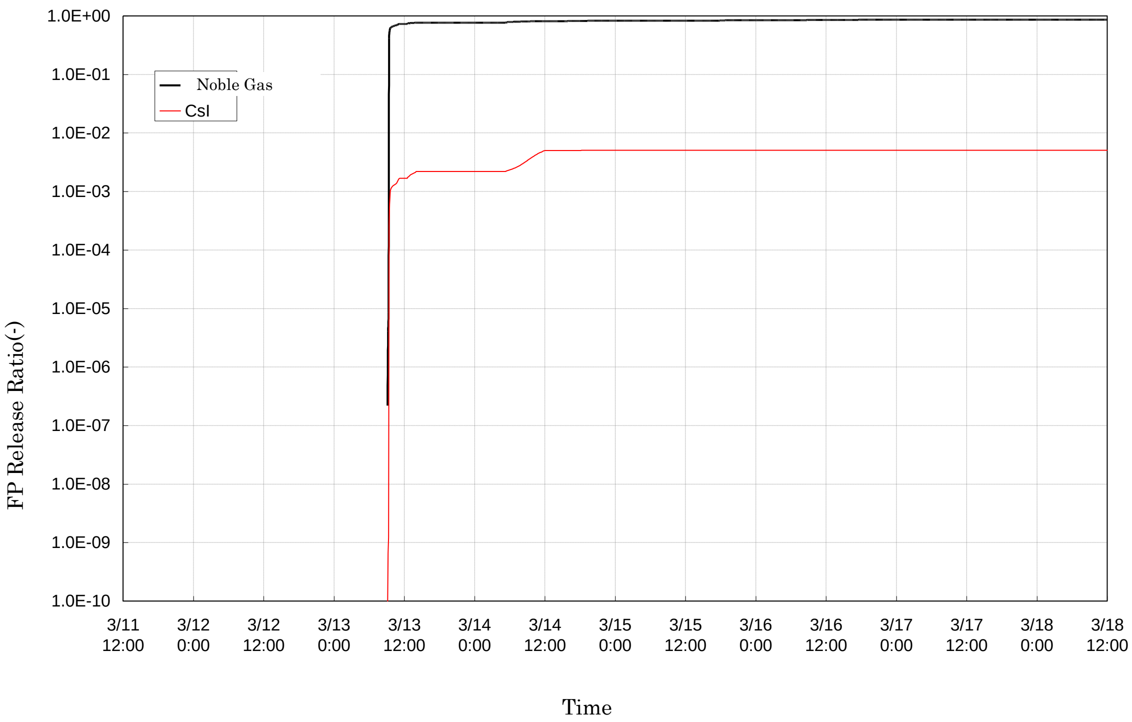


Figure3.3.1.7　Unit3　FP Release Ratio[Case1]

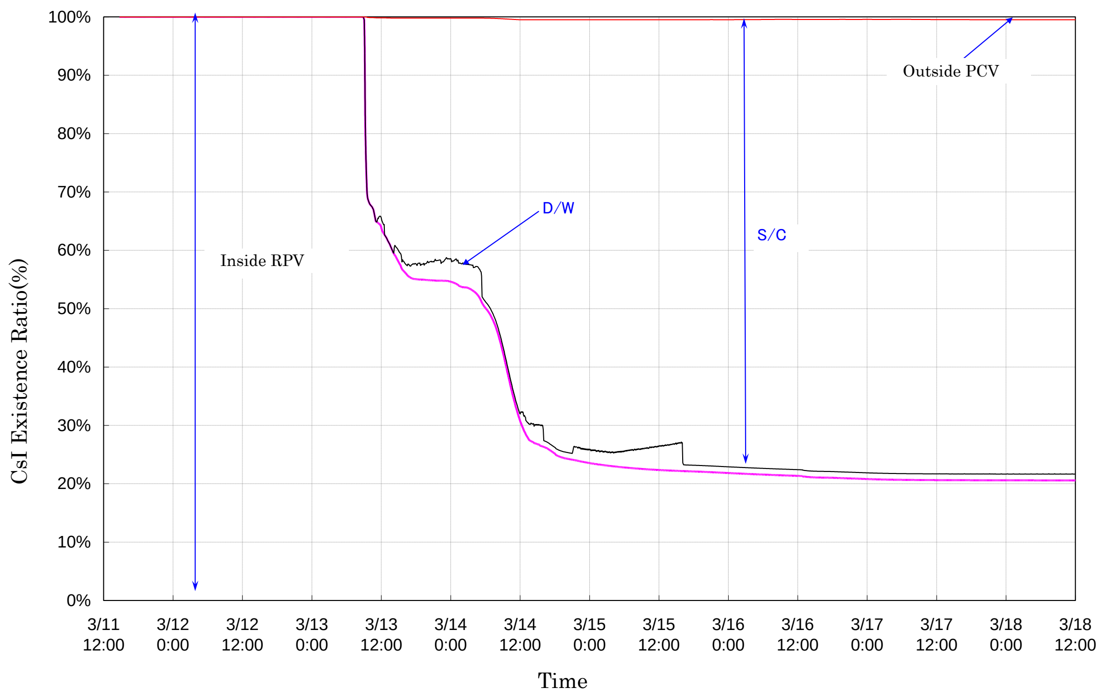


Figure3.3.1.8　Unit3 FP Existence Ratio(1/2)[Case1]

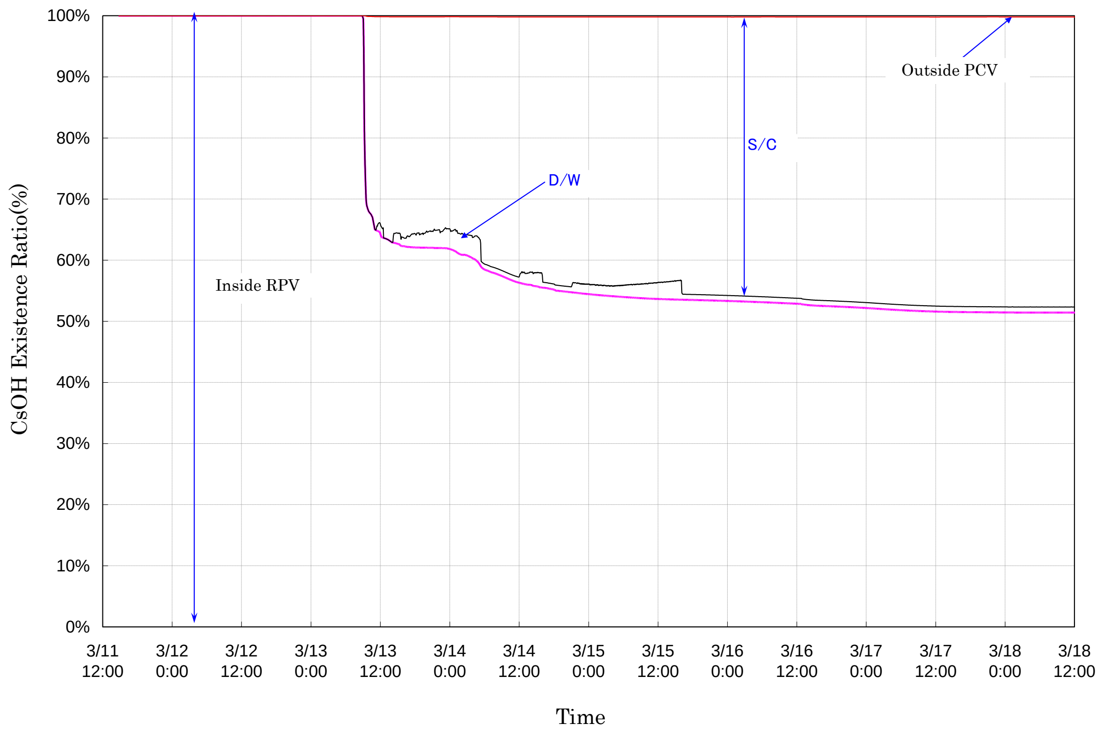
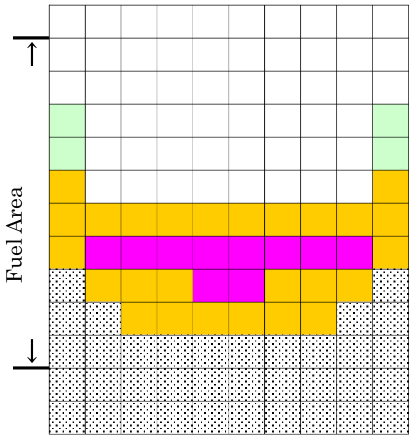
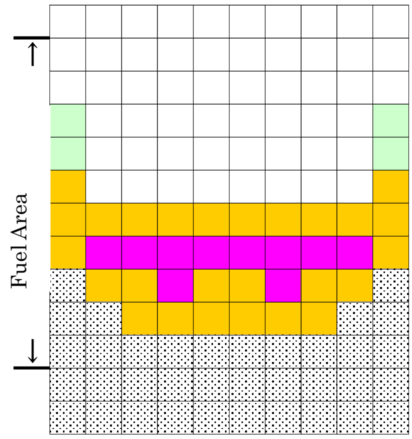


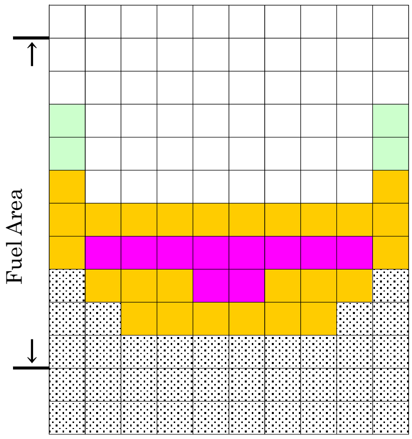
Figure3.3.1.8　Unit3 FP Existence Ratio(2/2) [Case1]



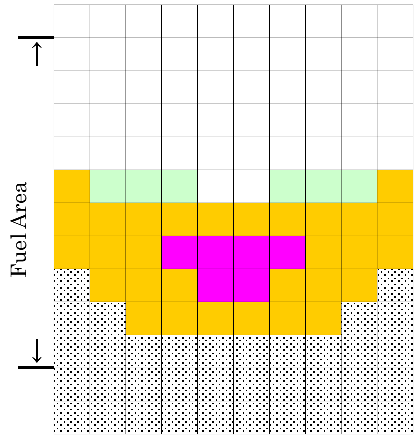
Approx. 64 hours after SCRAM



Approx. 68 hours after SCRAM



Approx. 72 hours after SCRAM



Approx. 1 week after SCRAM

Model of Fuel Damage
: No Fuel（Slumped）
: Normal Fuel
: Accumulation of Slumped Fuel
: Accumulation of Melted Fuel

: Flow Channel Blockage with Melted Fuel
: Molten Core Pool

Figure3.3.1.9　Unit3 Core Status[Case1]

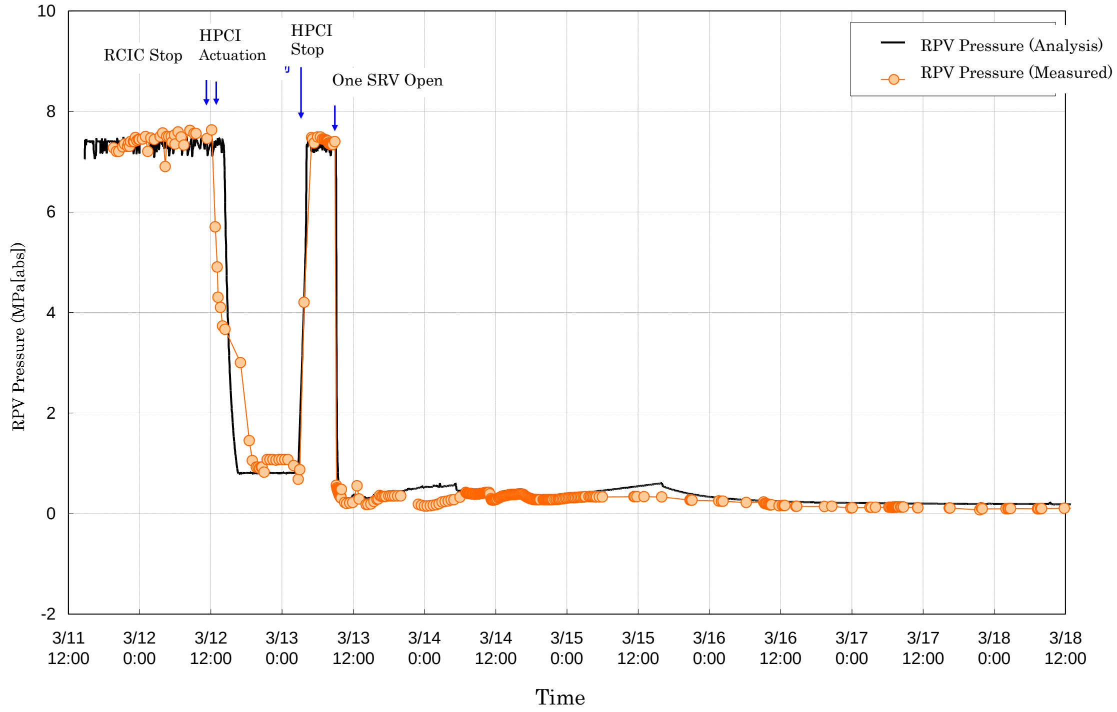


Figure3.3.1.10　Unit3 RPV Pressure[Case1] (Steam Leak)

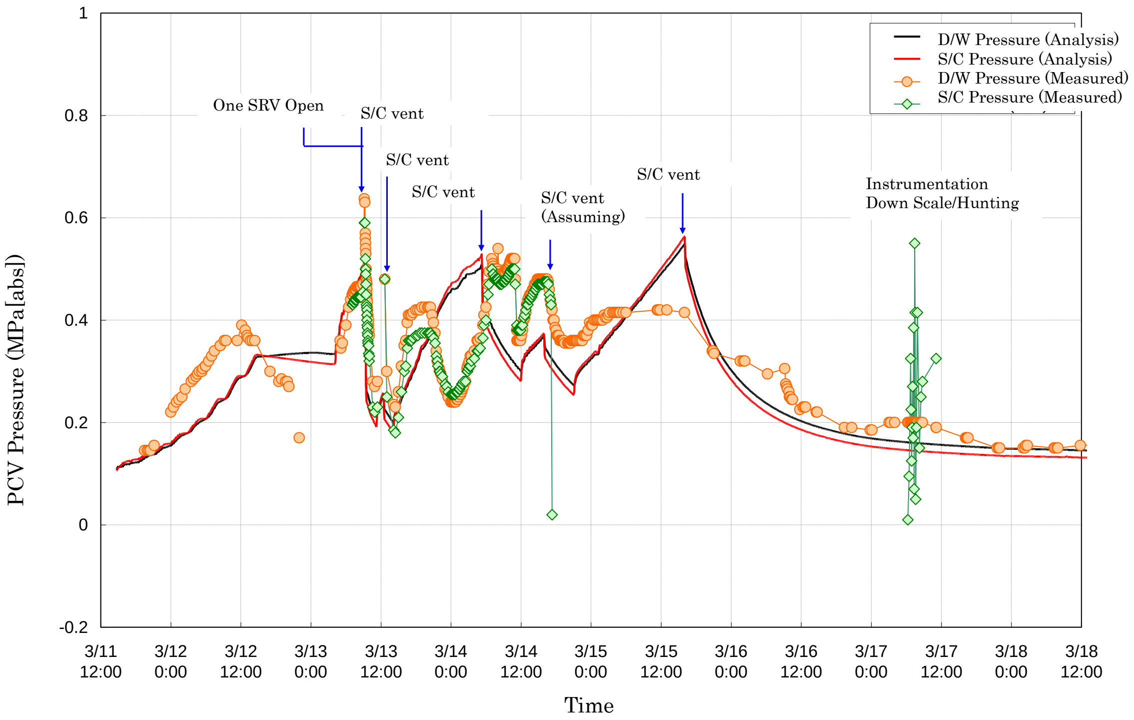


Figure3.3.1.11　Unit3 PCV Pressure[Case1] (Steam Leak)

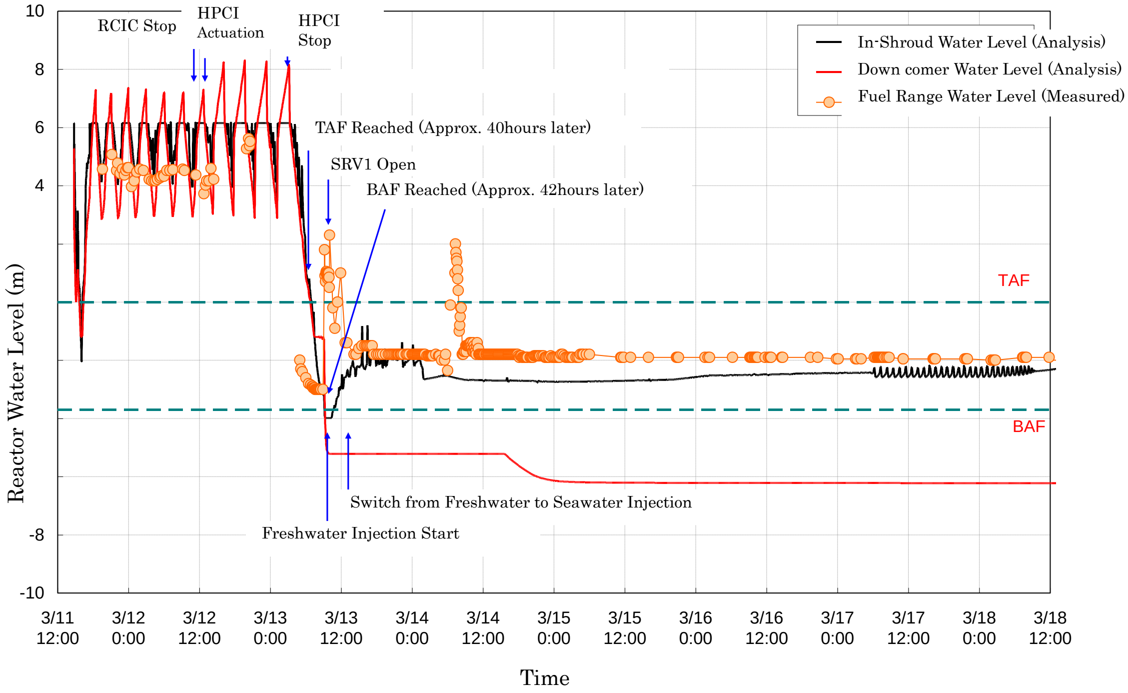


Figure3.3.1.12　Unit3 Reactor Water Level[Case1] (Continued Injection)

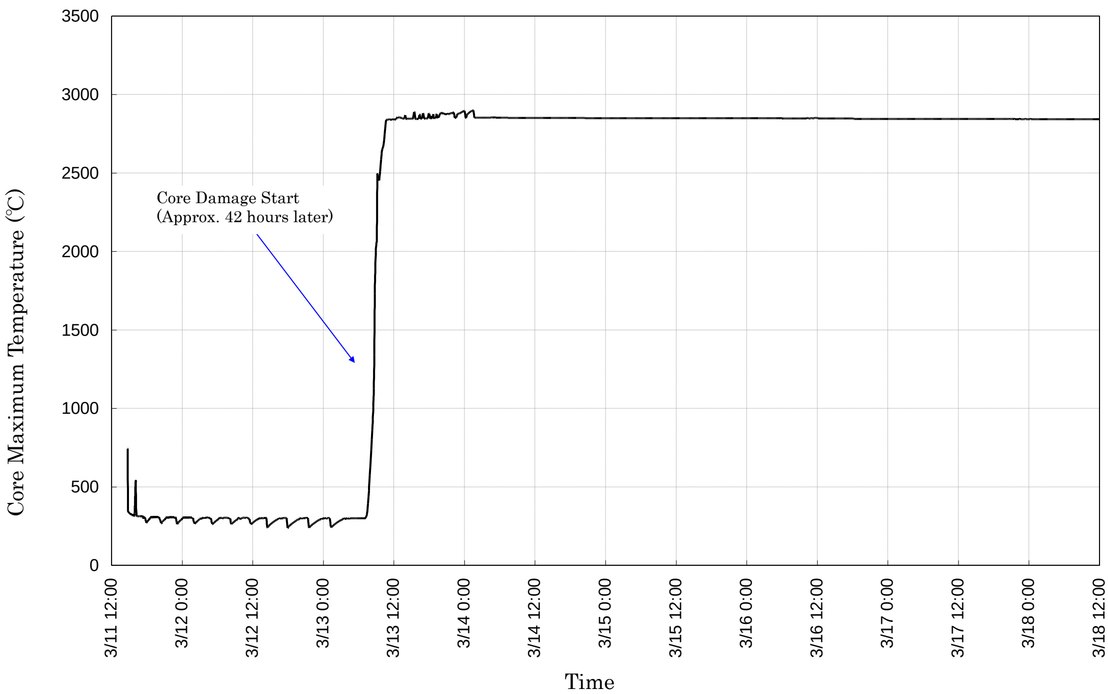


Figure3.3.1.13　Unit3 Core Temperature[Case1] (Continued Injection)

3.3.2.2 Analysis Result (Analysis Case 2)

Table 3.3.4 shows the result of analysis based on the condition shown in 3.3.1. And from Fig. 3.3.2.1 to Fig 3.3.2.9 show the result of analysis about the trend of reactor water level etc.

Table 3.3.4 Summary of Analysis Result on Unit 3

Item	Analysis Result
Start of reactor core exposure	Approx. 40 hours after earthquake
Start of reactor core damages	Approx. 42 hours after earthquake
Start of reactor pressure vessel breach	Approx. 66 hours after earthquake

The detail of analysis result is as follows.

The reactor water level gradually comes down after HPCI stops and the reactor core starts exposed, and the reactor core exposed completely by opening SRV and the reactor core damage starts (see Fig.3.3.2.1). Although the water injection starts, the assumed water injection flow is not enough so that the water level does not increase over the bottom of active fuel and as a result the reactor core is damaged more seriously than in case 1 (see Fig.3.3.2.1).

Regarding the reactor pressure, there is a temporary increase by the steam generated by the relocation of reactor core to the lower plenum, but other behavior is almost same as the result of Case 1 (see Fig.3.3.2.2).

Regarding the pressure of PCV, it is similar to the reactor pressure, there is a temporary increase by the steam generated by the relocation of reactor core to the lower plenum, but other behavior is almost same as the result of Case 1 (see Fig.3.3.2.3).

Regarding reactor core temperature trend, after HPCI stops, the temperature increases with the decrease of reactor water level, and reach to the melting point of the fuel pellets (see Fig.3.3.2.4).

A large amount of hydrogen is generated when the reactor core is exposed and the temperature of fuel cladding start increasing, the amount equal to the reaction of 59% of fuel cladding in the active fuel range is generated. In the analysis, almost all hydrogen is discharged out of PCV by S/C vent. The amount of production assumed to be enough for causing explosion of the Reactor Building of Unit 3 (see Fig.3.3.2.6).

Regarding the release of radioactive materials, after reactor core damage, noble gases are released from the reactor pressure vessel to S/C, and the result is that almost all noble gas is released by vent. For cesium iodide, the release rate is about 0.5% and most exists in S/C (see Fig.3.3.2.7 and Fig.3.3.2.8).

The result says that, parts of the fuel remain in the reactor pressure vessel, but the reactor pressure vessel is breached. As the water injection flow in the early stage is assumed to be smaller than in Case 1, the result says the damage of reactor core is more serious (see Fig.3.3.2.9).

3.3.3 Evaluation Result

In Analysis Case 1, we have had an analysis result that the reactor core of Unit 3 stays within the fuel range, though a partial fuel-melting pool exists, and the reactor pressure vessel will not be breached. In Analysis Case 2, we have had an analysis result that the reactor vessel has been breached, though part of fuel stays within it.

In addition, as a result of calibration of water-level gauge of Unit 1, we have found that the water level in reactor pressure vessel is not within fuel ranges. We cannot deny the possibility that a similar event has occurred in Unit 3.

Most of fuel is cooled down in the reactor pressure vessel, because, according to plant parameters, temperatures of the steel of the reactor pressure vessel currently change around between approximately 100 degrees Celsius and approximately 200 degrees Celsius and correlate changes in amount of water injected in multiple measuring points and temperatures in some points are increasing in May (though we keep following up conditions while the rate of inflow to the reactor increased) and due to the increase in temperatures their heat source is estimated to come from inside of reactor pressure vessel and the temperatures at the bottom of the reactor pressure vessel change around between approximately 100 degrees Celsius and approximately 170 degrees Celsius and correlate changes in temperatures in other parts of the reactor pressure vessel.

Hence, according to the analysis and plant parameters, it is evaluated that reactor core has been significantly damaged, relocated below or slumped to the lower plenum from fixed position of fuel loading, however most of it is being stably cooled down around there.

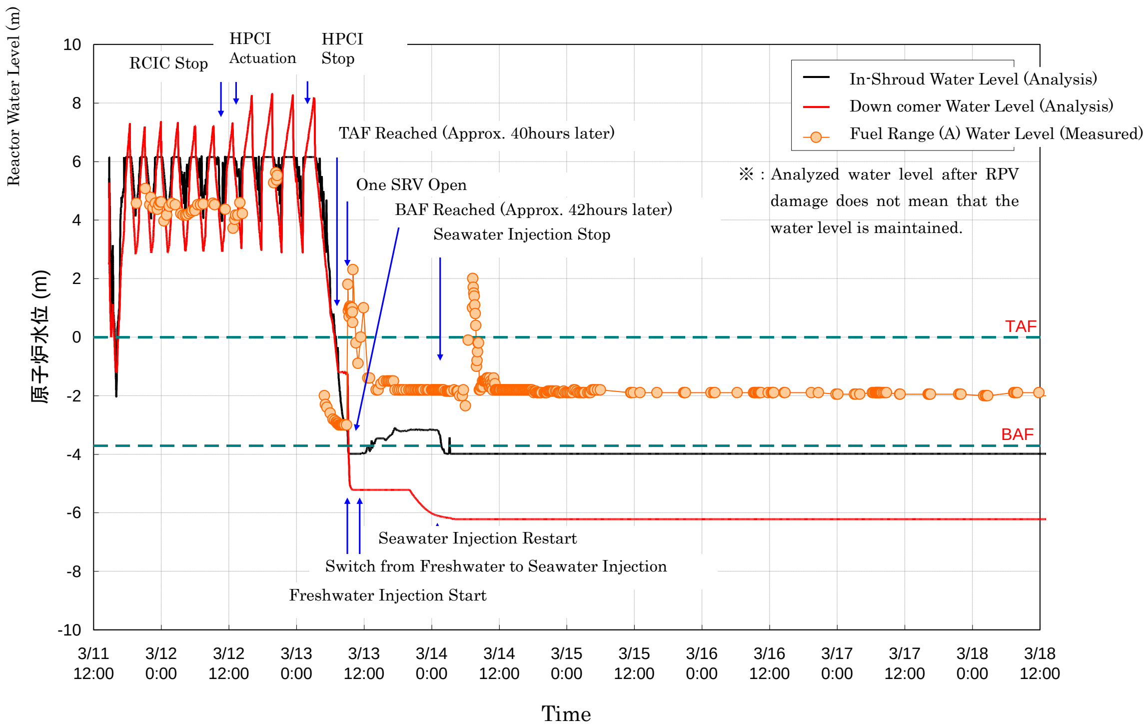


Figure3.3.2.1　Unit3 Reactor Water Level[Case2]

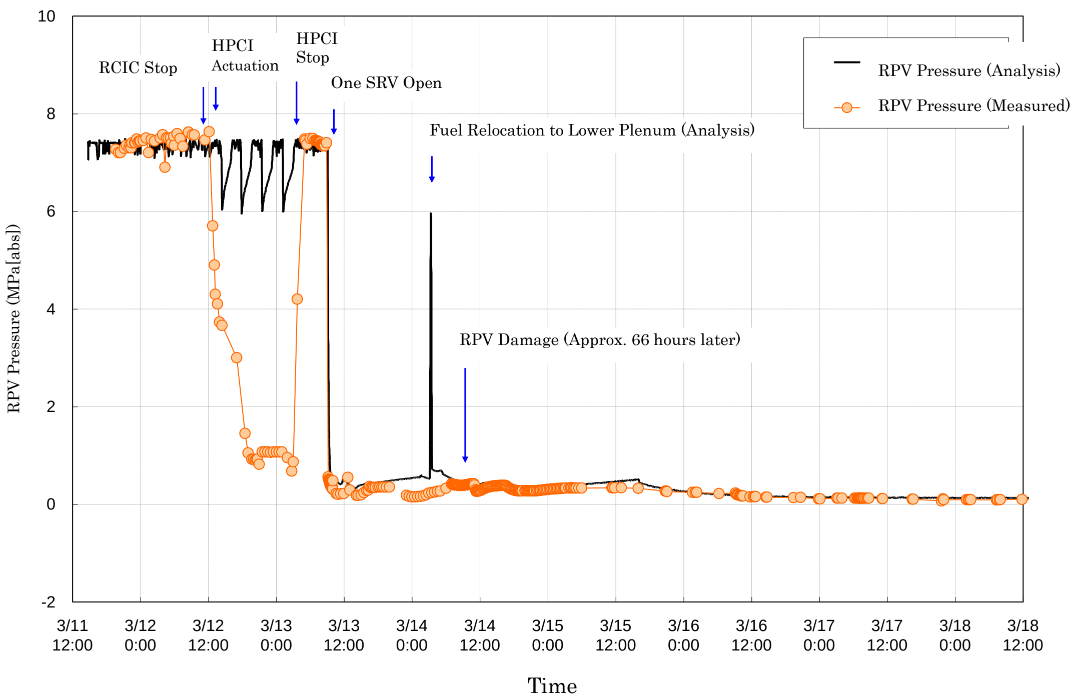


Figure3.3.2.2　Unit3 RPV Pressure[Case2]

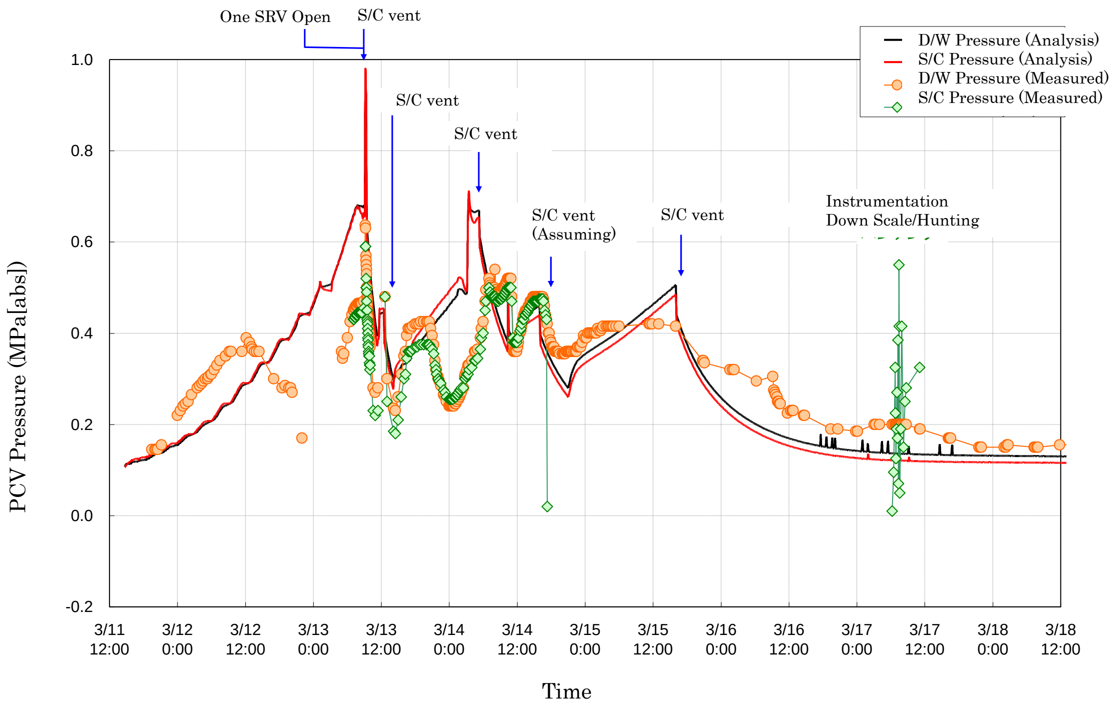


Figure3.3.2.3 Unit2 PCV Pressure[Case2]

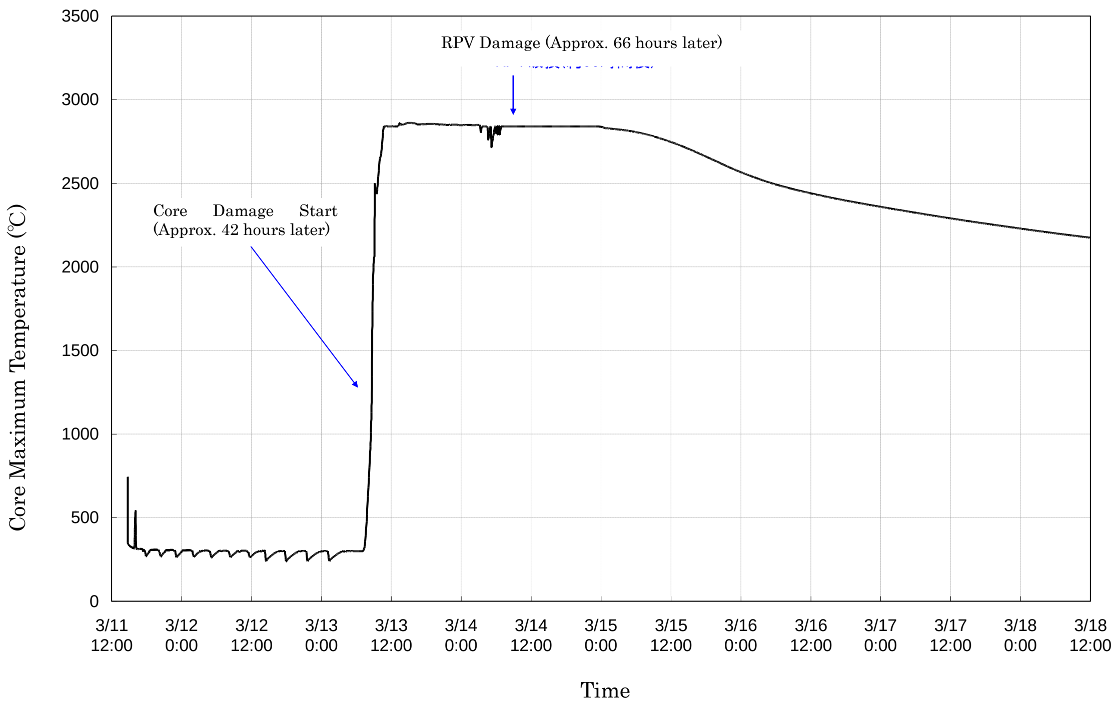


Figure3.3.2.4 Unit3 Core Temperature[Case2]

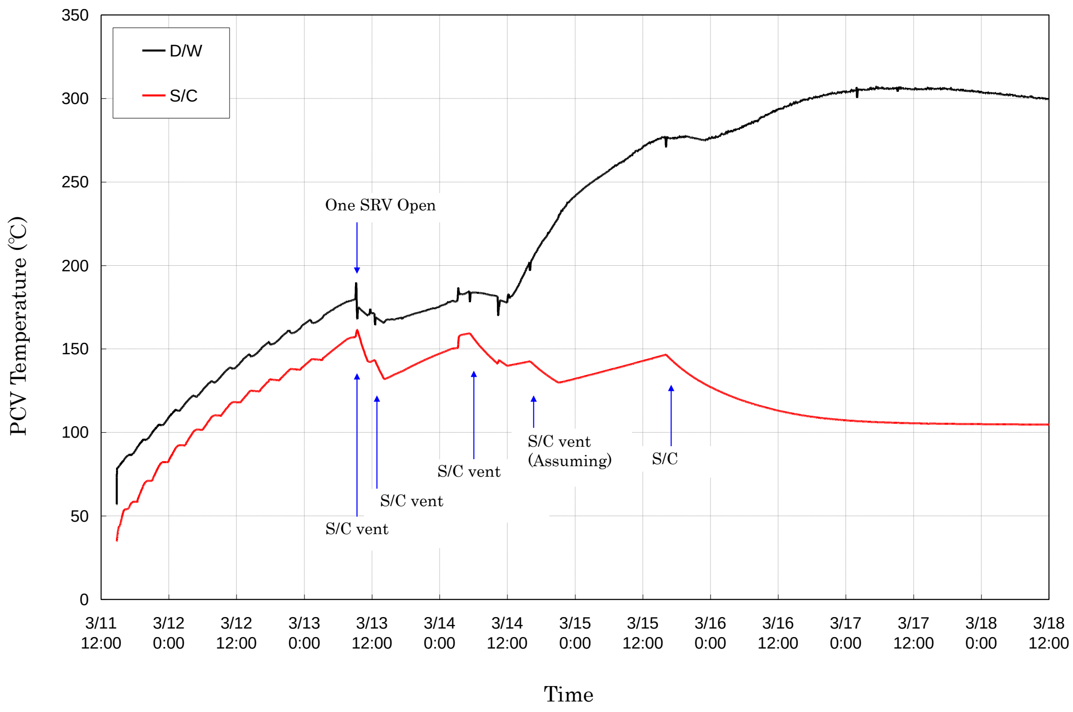


Figure3.3.2.5　Unit3 PCV Temperature[Case2]

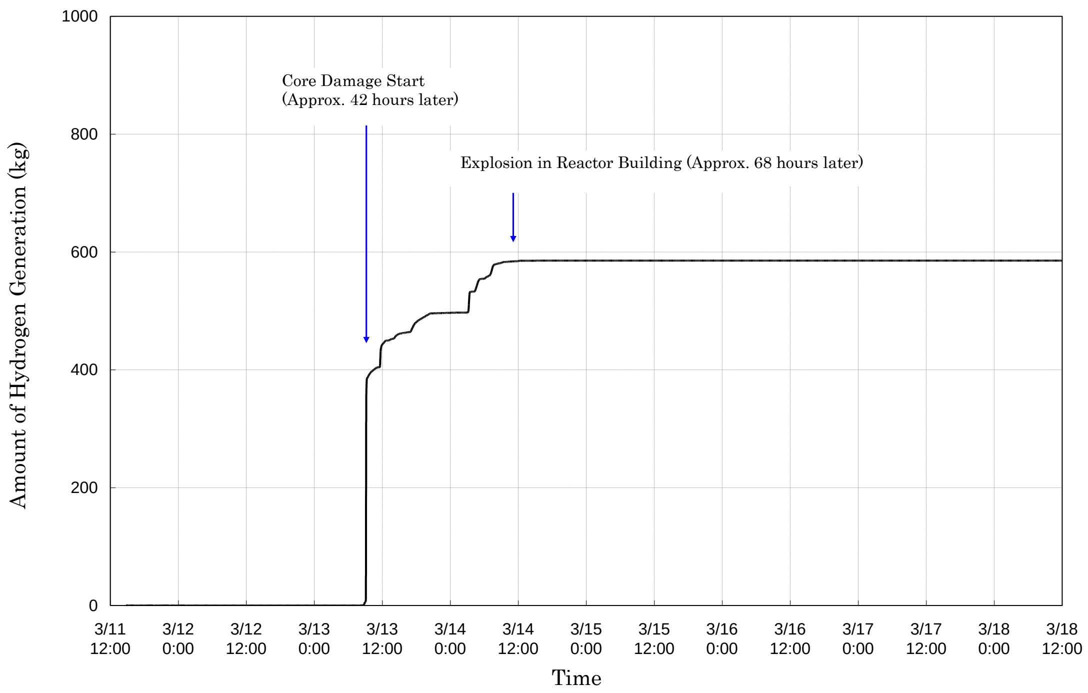


Figure3.3.2.6　Unit3 Amount of Hydrogen Generation[Case2]

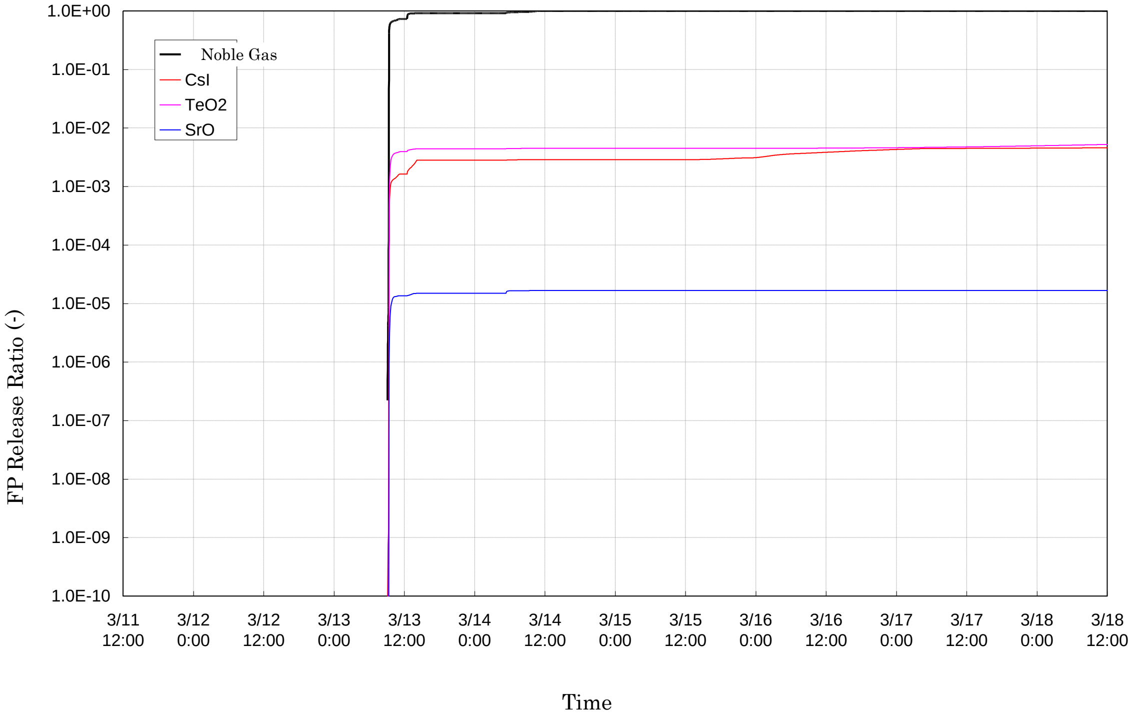


Figure3.3.2.7 Unit3 FP Release Ratio(1/3)[Case2]

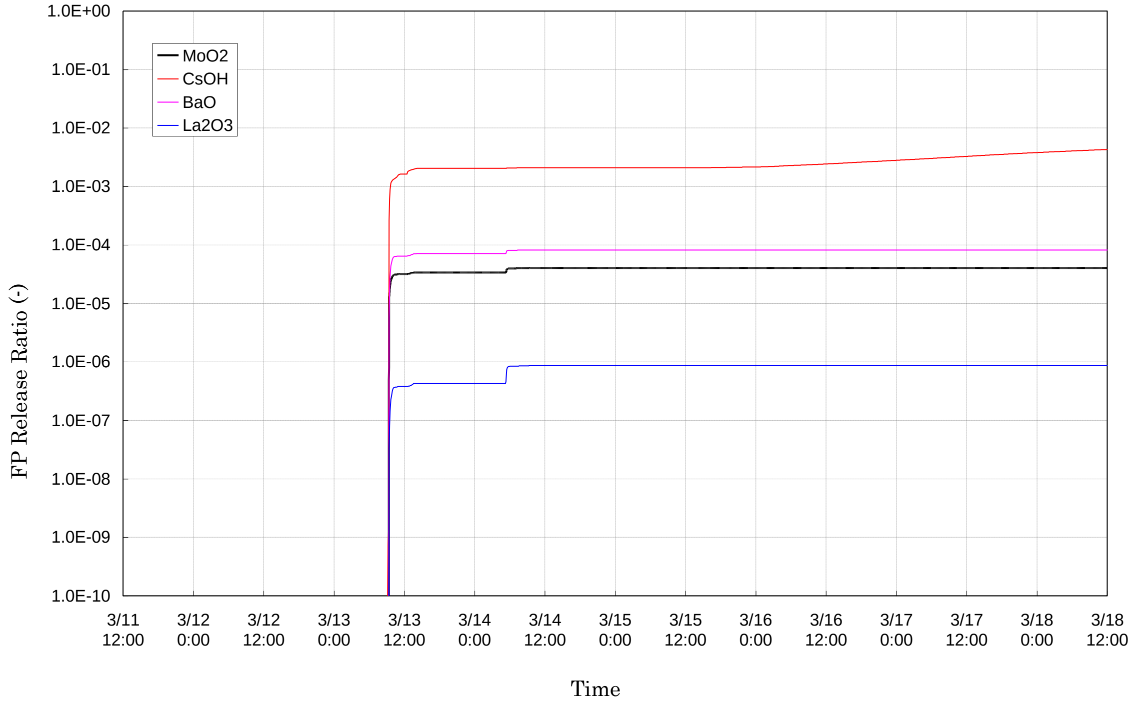


Figure3.3.2.7 Unit3 FP Release Ratio(2/3)[Case2]

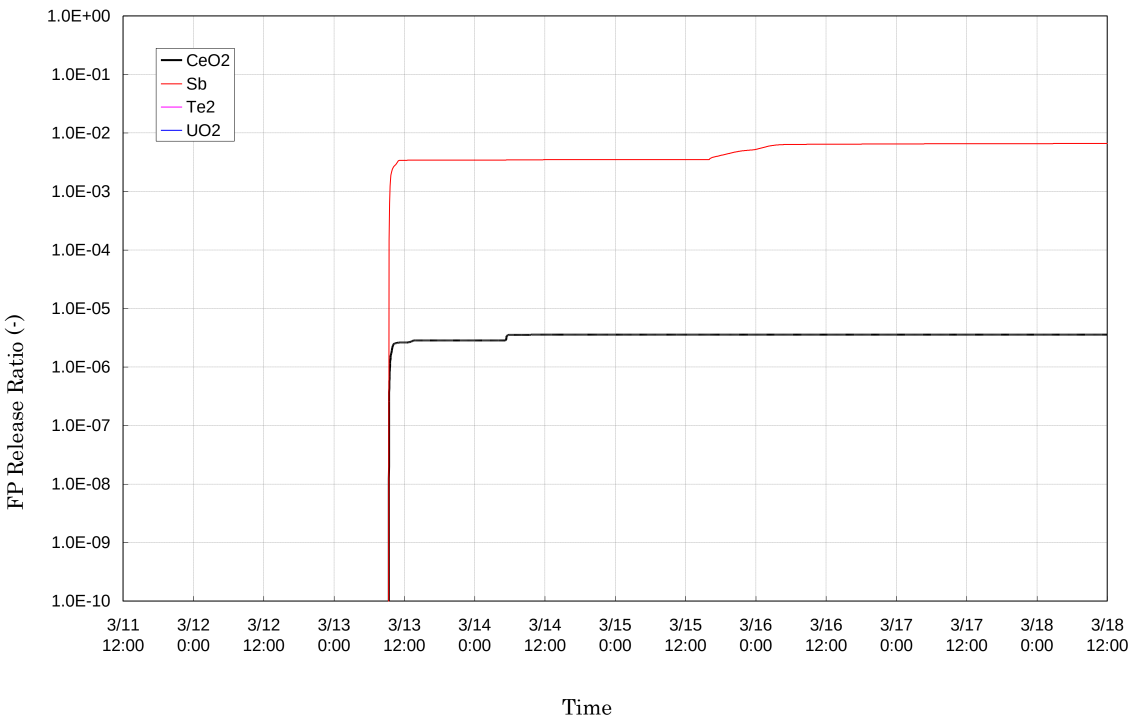


Figure3.3.2.7　Unit3 FP Release Ratio(3/3)[Case2]

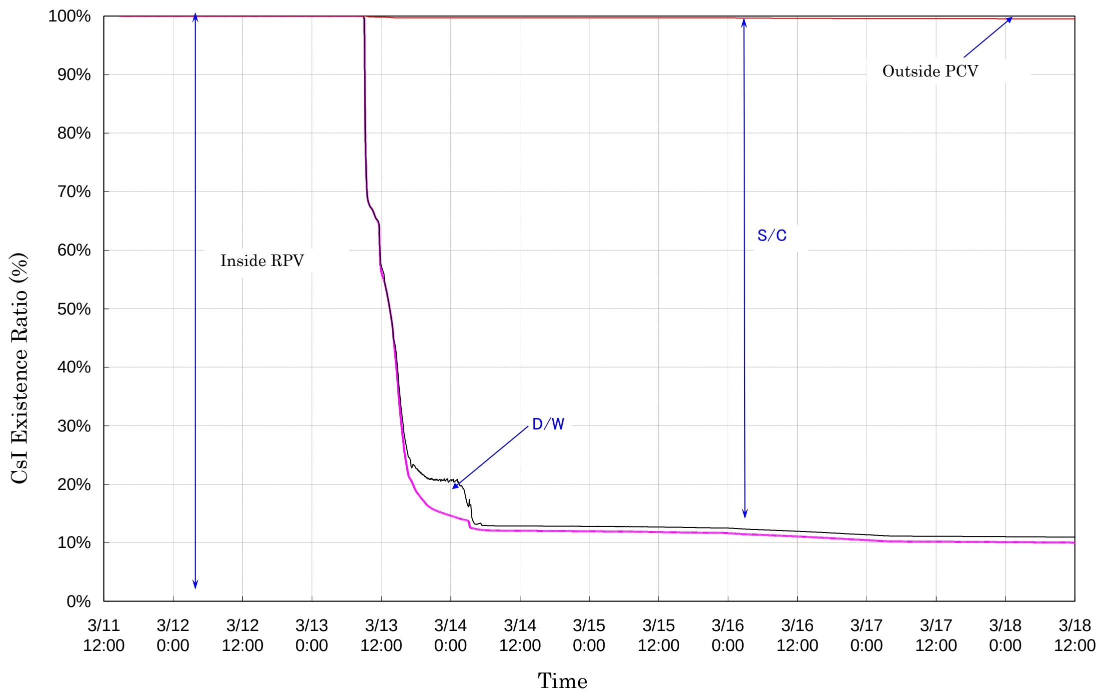


Figure3.3.2.8　Unit3 FP Existence Ratio(1/2)[Case2]

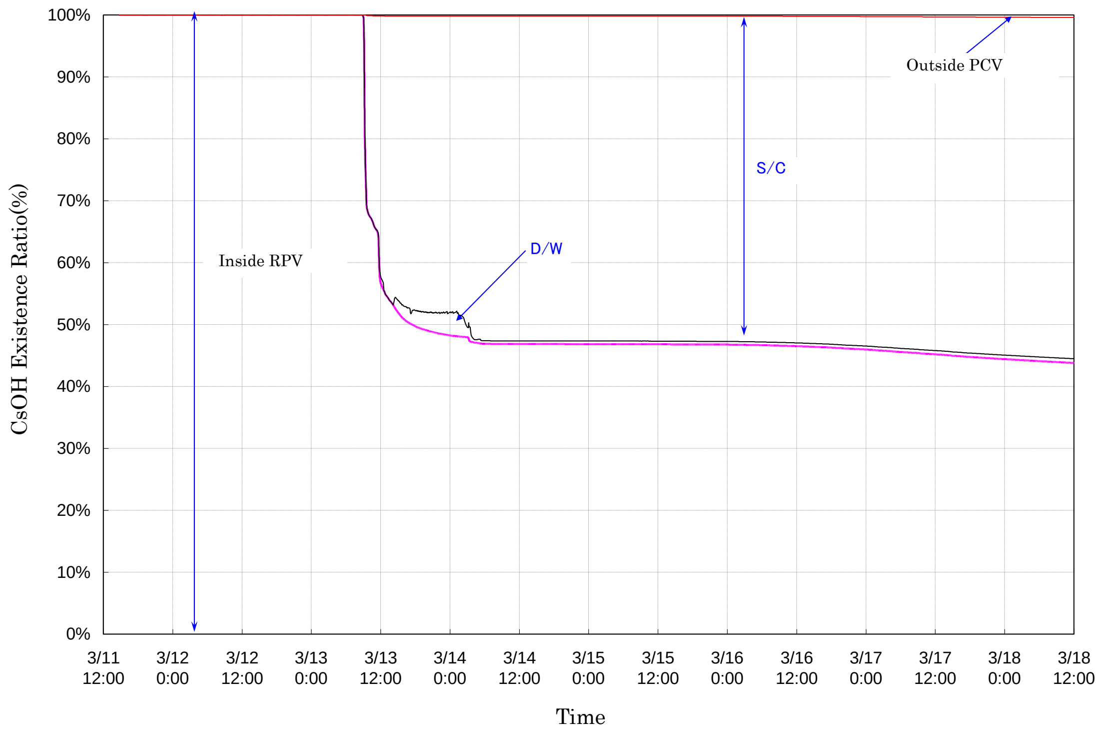
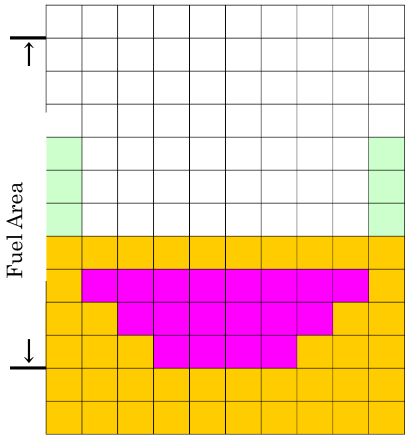
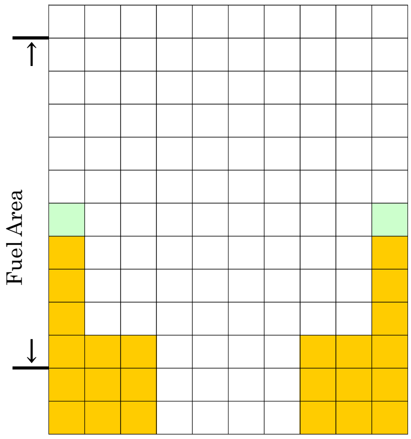


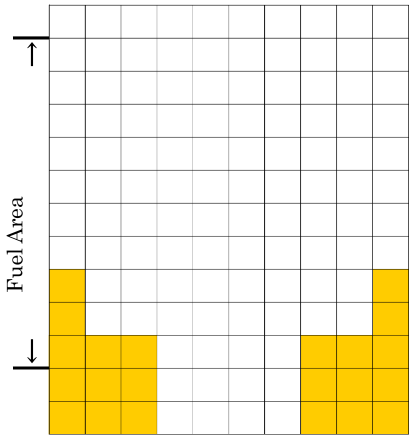
Figure3.3.2.8　Unit2 FP Existence Ratio(2/2)[Case2]



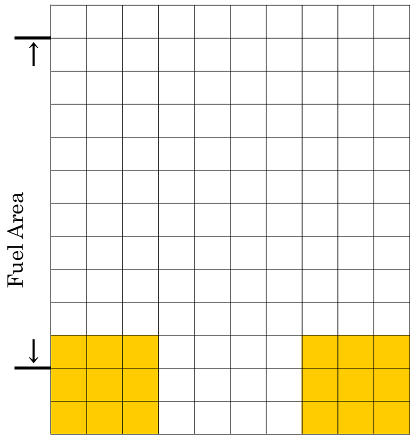
Approx. 58 hours after SCRAM



Approx. 62 hours after SCRAM



Approx. 66 hours after SCRAM



Approx. 96 hours after SCRAM

Model of Fuel Damage
: No Fuel (Slumped)
: Normal Fuel
: Accumulation of Slumped Fuel
: Accumulation of Melted Fuel

: Flow Channel Blockage with Melted Fuel
: Molten Core Pool

Figure3.3.2.9 Unit3 Core Status[Case2]