Attachment 2-9

# Examination into the reactor pressure increase after forced depressurization at Unit-2, using a thermal-hydraulic code

* This document is generated based on the evaluation upon contract using an analysis code by TEPCO Systems Corporation concerning the reactor pressure increase after forced depressurization (Unit-2/Issue-7).

## 1. Background

In Attachment 2-7, it was shown that the reactor pressure increase after forced depressurization at Unit-2 might have reflected the results of the sequence of water injections by fire engines which caused water-zirconium reactions and advanced the core damage and core melting. But the elaboration was limited to a qualitative analysis. Examination was not sufficient into the possibility of multiple combinations of SRV opening/closing, quantitative evaluation of steam generation or hydrogen generation, and the feasibility of such a progression scenario. Therefore, this document examined an accident progression scenario, using the thermal-hydraulic analysis code GOTHIC 8.0(QA) (hereafter simply GOTHIC), which could reproduce the changes with time of reactor pressures and primary containment vessel (PCV) pressures. But GOTHIC cannot simulate water evaporation behavior and water-zirconium reactions in its analysis appropriately. They should be provided as input conditions. This means it is possible to identify the amount of water evaporation and hydrogen gas production which can well reproduce reactor pressures and PCV pressures. The GOTHIC analysis in this document took this methodological approach for the examination.

## 2. Contents of examination

### 2.1. Estimation of plant situation concerning the reactor pressure changes

The accident progression of Unit-2 of the Fukushima Daiichi Nuclear Power Station of Tokyo Electric Power Company (TEPCO) was estimated concerning the opening situations of safety relief valves (SRVs) after the forced depressurization, the amount of hydrogen generation and its timings, and the leak situations of the reactor pressure vessel (RPV) and the containment vessel (PCV). The TEPCO investigation reports on the accident and various plant data made open to date were used in the estimation [1]. Basic consistency with measured plant data, other than the reactor pressure (PCV pressure, water level indicator readings, etc.), was maintained.

Figure 2.1 gives the data measured at Unit-2 after forced depressurization, while Table 2.1 presents an accident scenario estimated from the measured data. In Figure 2.1, identifier numbers are given to the important timings for estimating the accident progression, and in Table 2.1 the estimated plant situation and its grounds are given for each number.

From among the pressure data given in Figure 2.1, the RPV pressure and drywell (D/W) pressure showed similar changes after about 21:30 on March 14$^{th}$. This can be attributed to the gas leaks from the RPV to the PCV balancing both pressures. On the other hand, after about 22:00 on March 14$^{th}$, the suppression chamber (S/C) pressure greatly deviated from the D/W pressure. Since this S/C pressure is unlikely to be correct, it was ignored in the current estimation of the accident progression.

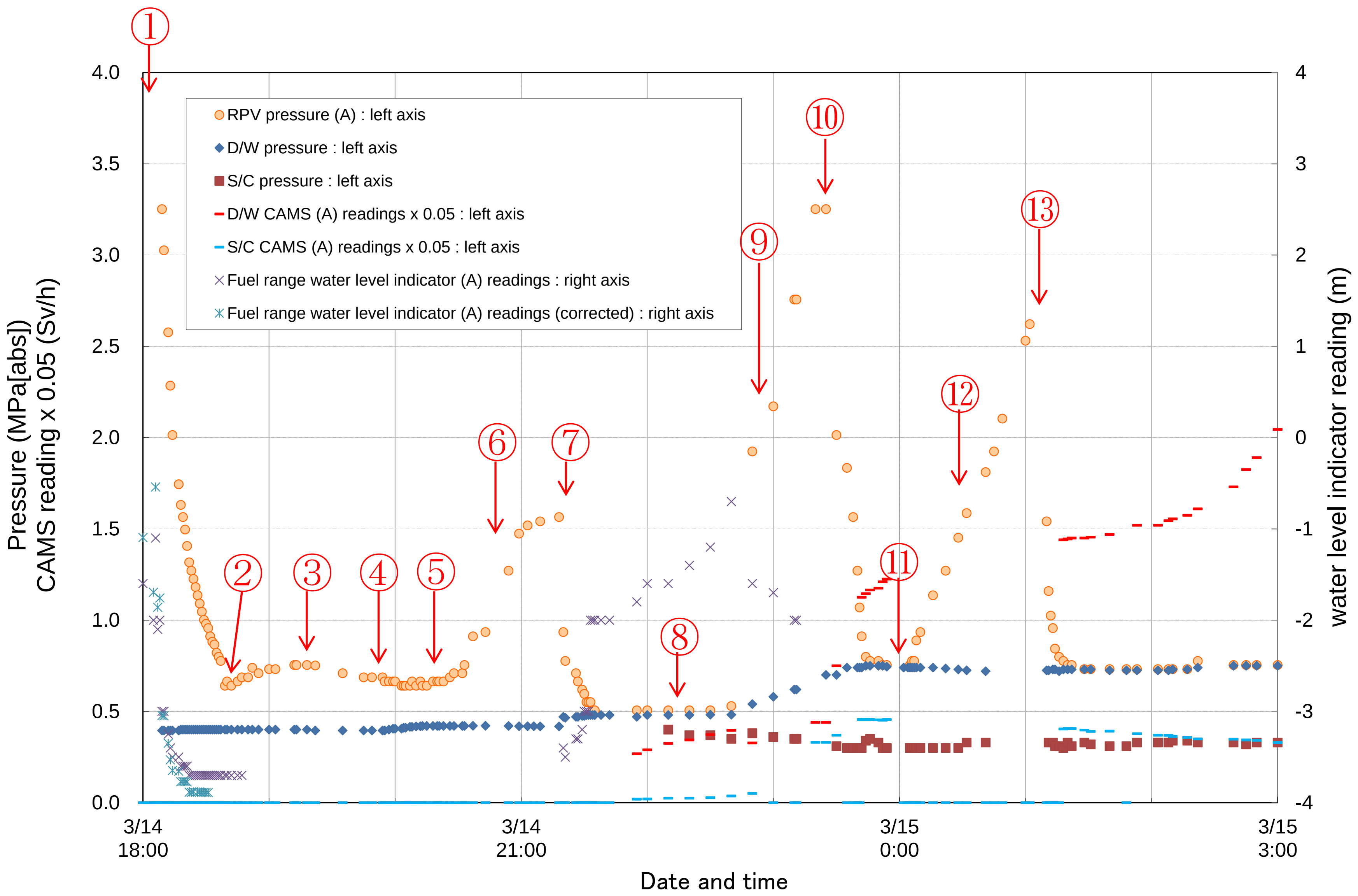


Figure 2.1 Measured data after Unit-2 forced depressurization

Table 2.1 Estimated accident progression after forced depressurization at Unit-2

| No. | Date & time | Estimated situation | Grounds for the estimation |
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| ① | March 14th 18:02 | SRV(s) opening forced (1 or 2) | • Operational records (July 17, 2013) |
| ② | about 18:40 | SRV(s) closed by their dead load | • The pressure difference of RPV and D/W was 241kPa (2.4atg). When the difference cut 343kPa (SRV dead load closure pressure), the SRV(s) were closed by their dead load.<br>(Note 1) Another possibility was the pressure increase due to steam generation from water injection by fire engines, but this was estimated to be unlikely because the water level indicator readings showed no changes and a record existed that said the water injection pump had been inoperable 30 to 60min before 19:20 [2][3].<br>(Note 2) The RPV pressure increase in ②-③ was estimated to come from the increased RPV vapor temperatures |
| ③ | about 19:20 | SRV(s) slightly opened | • The pressure difference of RPV and D/W reached 354kPa (3.5atg), exceeding the SRV dead load closure pressure.<br>• The pressure decrease thereafter was slow. If the SRV(s) were fully opened, the decrease should be faster. |
| ④ | 19:54 | Water injection resumed | • Reference materials [2][3]<br>(Note) The D/W pressure increased thereafter by about 20kPa (0.2atg) till ⑤ (about 20:15). This was estimated to be because fuel element temperatures became elevated, causing water-metal reactions, hydrogen generation and discharge of hydrogen through the slightly opened SRV(s) to the S/C. The amount of steam flow to the S/C predictable for slightly opened SRV(s) was considered to be insufficient to cause this D/W pressure increase. |
| ⑤ | about 20:15 | SRV(s) closed | • The D/W pressures remained unchanged from ④ to |

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| | | | ⑤, nonetheless the RPV pressures increased. The slightly opened SRV(s) were estimated to have been closed for unknown reasons. |
| ⑥ | about 20:15 to 21:20 | Steam and hydrogen generated in the core | • The RPV pressure showed a rapid increase.<br>• At RPV depressurization (⑦), the D/W pressure increased by about 50kPa (0.5atg). Steam discharge to the S/C would not be enough to cause this pressure increase. Therefore, hydrogen generation at this timing was assumed. |
| ⑦ | about 21:20 | SRV(s) opening forced | Reference materials [2][3] |
| ⑧ | about 21:30 - 22:40 | SRV(s) held open | • The pressure difference between the RPV and D/W was below the SRV dead load closure pressure. Therefore, the SRV(s) were assumed to have been held open for some unknown reasons. SRV(s) were assumed to have held the opened position thereafter.<br>(Note 1) The SRV working mechanisms were the same at depressurizations ①and ⑦. The possibility of dead load closure was taken into account.<br>(Note 2) If leaks occurred from the RPV to the D/W, their pressures should have balanced, but the pressure difference between these two during ⑧ remained at about 25kPa (0.25atg), indicating a low possibility of such leaks. This pressure difference was considered to correspond to the water head difference from the S/C quencher and S/C water surface.<br>(Note 3) The S/C CAMS readings during ⑧ were lower than those of the D/W. The S/C CAMS located outside the S/C might have caused lower readings due to, e.g., S/C wall shielding effect. It was also possible that FPs deposited on SRV piping in the D/W caused higher D/W CAMS readings.<br>(Note 4) Water level indicator readings increased during ⑧. It was possible that heat transfer from the RPV and gas flow from the S/C increased the D/W temperatures, causing the water in the reference leg to evaporate and readings to |

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| | | | increase. |
| ⑨ | about 22:40 - 23:25 | Steam and hydrogen generated in the core | • RPV pressure and D/W pressure sharply increased.<br>• D/W pressure increased about 270kPa (2.7atg) by about 23:40. As steam inflow was considered to be insufficient to account for this D/W pressure increase, hydrogen generation and inflow was assumed.<br>(Note) D/W and S/C CAMS readings increased from about 23:00 March 14th to about 00:00 on March 15th. It was considered that a large amount of non-condensable gas (hydrogen) flowed from the RPV to the S/C at this timing and FPs left behind after S/C scrubbing moved to the S/C gaseous phase and then, after a while moved to the D/W via the vacuum breakers. Lower S/C CAMS readings than those of the D/W CAMS might have the same reasons behind them as those during ⑧. The S/C CAMS readings during ⑧ were about 1/10 of those of the D/W CAMS, but during ⑨ they were only about one fifth. It was possible that key FP nuclides flowing into the S/C changed. The shielding effect of structures and others for gamma rays are nuclide dependent. |
| ⑩ | about 23:25 | Steam and hydrogen generation in the core declined | • This was so assumed because RPV pressure started to decrease, but the SRV(s) opening was not recorded as being confirmed. |
| ⑪ | March 15th about 00:06 | Steam and hydrogen generation started in the core.<br>Leaks from D/W to R/B started | • The RPV pressure increased, while the D/W pressure decreased slightly. It was assumed, therefore, that gas was generated in the core and leaks occurred from the D/W to the R/B.<br>(Note) The radiation level near the main gate showed no increasing trend at this timing. Leaks from the D/W to the R/B were assumed to be in a limited scale. |
| ⑫ | 0:06 to 1:10 | Steam generation in the core (with | • There was no big D/W pressure increase during ⑫, when RPV pressure increased. Therefore, hydrogen generation during ⑫ was assumed to be on a limited |

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| | | limited Hydrogen generation) | scale and the RPV pressure increase was mainly due to steam generation. |
| ⑬ | about 01:10 | SRV(s) forced opening | • Reference materials [2][3] |

### 2.2. The analysis of reactor pressure changes

A reproduction analysis was carried out, based on the accident progression estimated in Section 2.1, on the reactor pressure changes starting at 18:00 on March 14th, the timing of reactor forced depressurization. Consistency of the analysis results with the D/W pressures and other measured data was also reviewed. The thermal-hydraulic code GOTHIC [4][5] was used.

#### 2.2.1. Geometry for analysis

Figure 2.2 shows the geometry for the analysis. The RPV, PCV and reactor building (R/B) were modeled as several numbers of regions (volumes). The flow path between each volume was a junction, and structures were modeled as heat structures. Each heat structure exchanged heat with its adjacent volumes.

The RPV was divided into five sections (core region, upper plenum and separator region, upper head and downcomer region, lower plenum region, and recirculation loop region) in order to estimate the accident progression with in-RPV temperature distributions being taken into account. Instead of simulating water injection into the reactor, a mixture of steam and hydrogen was “injected” into the RPV by the injection boundaries. That means, the reactor water level changes due to water injection by fire engines were not included in the analysis. The chronological amount of steam and hydrogen generation in the core region was provided in the analysis in the form of a separate, independent table. The decay heat was provided at the fuel pellet position. It should be noted that fuel pellets and claddings were modeled in two different heat structures with a gap in between. This was because of a need to provide the heat of water-metal (zirconium) reactions at the fuel cladding.

The PCV was divided into the D/W region, the venting line region and the S/C region. In order to take heat transfer from the S/C and the D/W into consideration, the torus room (a room in the R/B where the S/C is installed) and the R/B were modeled and the heat structures were set in between (the S/C wall and the D/W wall). Considering the possibility of water being present in the torus room, heat removal through the S/C wall was taken into consideration. Further, the D/W and the S/C were connected by the vacuum breakers (V/Bs) as well as by the venting line. Therefore, if the S/C pressure exceeded the D/W pressure, the pressure was released to the D/W via the V/Bs.

The leak holes from the SRV and from the D/W to the R/B were set as a valve junction which could adjust its cross-sectional area with time. This enabled simulation of the leak changes with time during the accident progression. Material properties of heat structures were defined from general values of each component material.

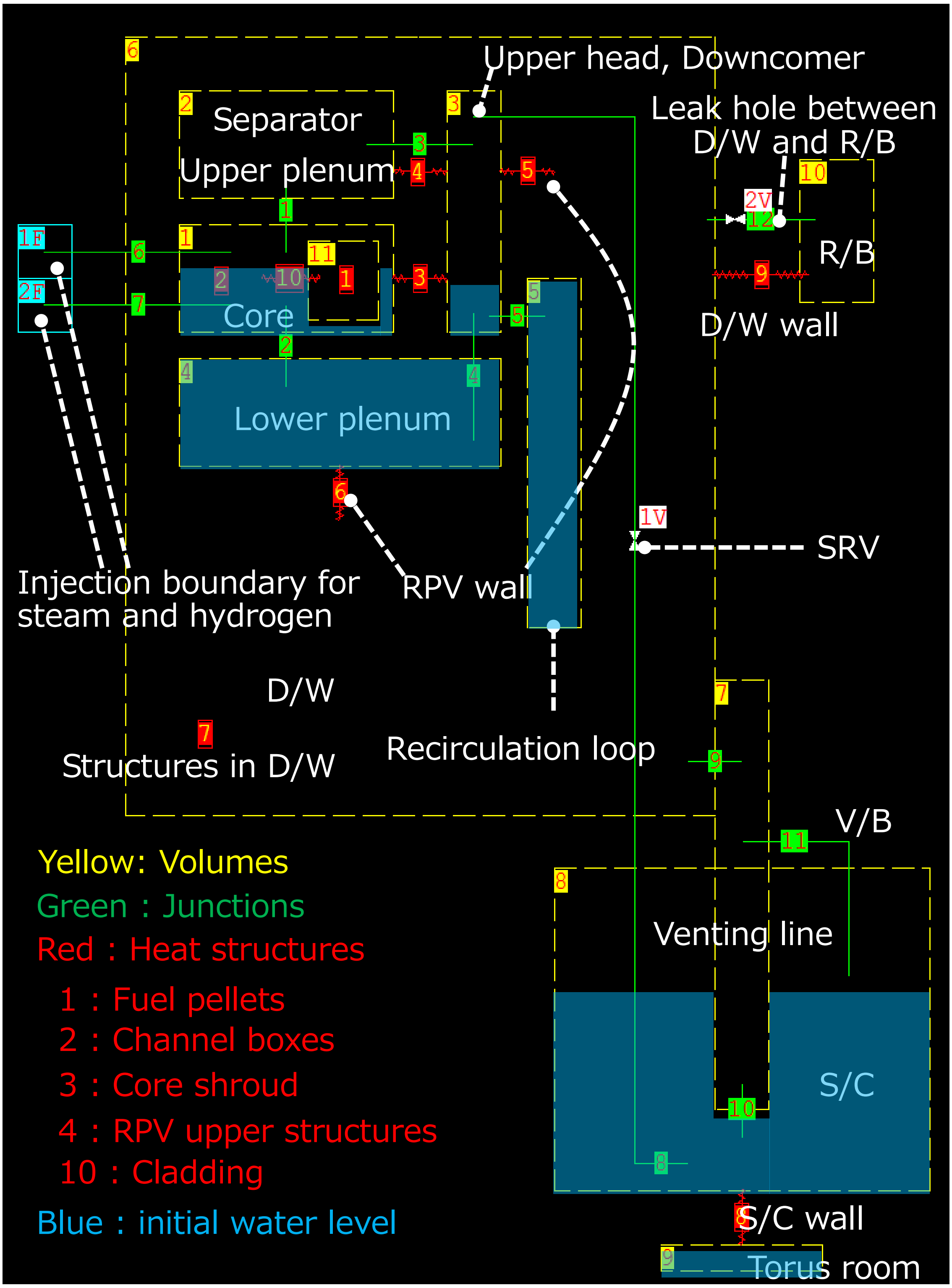


Figure 2.2 Geometry for GOTHIC analysis

### 2.2.2. Conditions for analysis

Table 2-2 summarizes key conditions for the analysis, while Figure 2-3 shows the sizes of SRV opening and the sizes of leak holes of the D/W to the R/B, and Figure 2-4 presents the amount of steam and hydrogen generated. Table 2-3 explains the grounds for setting the changes shown in Figures 2-3 and 2-4.

Table 2.2 Key conditions for analysis

| Item | Setting | Grounds for the setting |
|---|---|---|
| Time span of analysis | From 18:00 on March 14th to 02:00 on March 15th | From forced depressurization by SRV(s) opening until the big changes of RPV pressures ceased |
| Initial conditions of pressure and temperature | RPV: 7.234MPa/ saturation temperature<br>D/W: 0.4MPa/ saturation temperature<br>S/C: 0.386MPa<br>Vapor: 143 deg C (saturation temperature)<br>Liquid: 139.2 deg C | The RPV and D/W pressures were set based on the measured data. The S/C pressure was set as the water head difference assuming that all the water in the venting line was discharged to the S/C by the pressure difference between the D/W and S/C which was cooled by heat removal to the torus room (See separate item “S/C external cooling.” The S/C liquid temperatures were set by searching for the best value to reproduce the D/W pressures. |
| Initial water inventory | RPV: about 120m$^3$<br>S/C: about 60% of S/C volume | RPV water inventory was set based on the water level indictor readings. S/C water inventory was set from the initial inventory and water inflow from the RPV considering the water injected from condensate storage tank (CST). |
| Decay heat | about 7.74MW at 18:00 on March 14th →about 7.43MW at 02:00 on March 15th | The decay heat [6] during the analysis time span was given for the fuel pellet position. |
| S/C external cooling | Heat transfer area: 300m$^2$ | The S/C cooling by the residual water in the torus room could provide good reproducibility |

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| | | of PCV pressures in the MAAP analysis and other analyses [2][3]. This approach was used to set the S/C heat transfer area which could well reproduce the D/W pressure changes. |
| Depressurization conditions | Size of SRV opening and size of leak hole from D/W to R/B: Figure 2-3 | Conditions were searched which could reproduce depressurization behavior based on the estimated accident progression (Table 2.1) |
| Steam/hydrogen generation | The amount: Figure 2-4<br>Steam temperature: Saturation temperature at RPV pressure (measured)<br>Hydrogen temperature: 1000 deg C | Conditions were searched which could reproduce RPV and D/W pressures based on the estimated accident progression (Table 2.1) |
| Heat of water-metal reactions | 293kJ per mol of hydrogen | $Zr+2H_2O \rightarrow ZrO_2+2H_2+586kJ$[7]<br>After subtracting the heat carried by hydrogen the rest was equally provided, depending on the area size at the fuel cladding and channel box. |

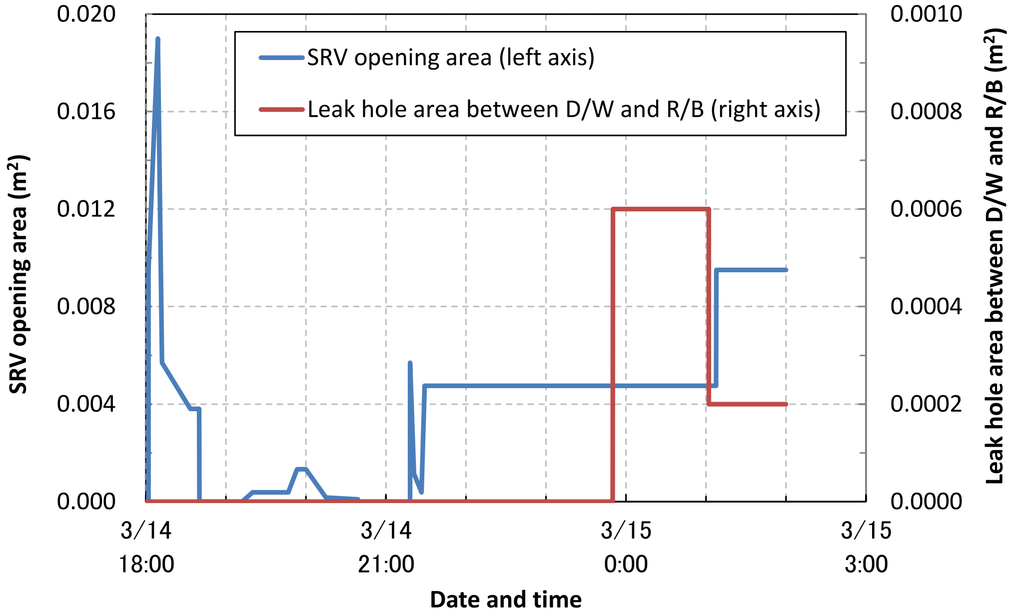


Figure 2.3 Area of leak holes at SRV(s) and from D/W to R/B

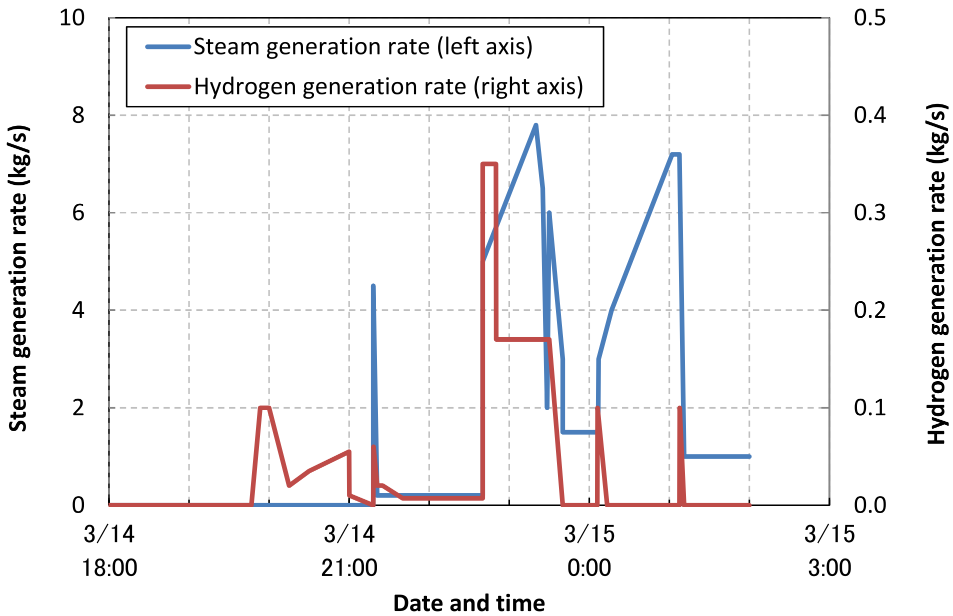


Figure 2.4 Amount of steam and hydrogen generated[1,2]

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[1] GOTHIC calculates the heat transfer in the gaseous phase to the water in the core as of 18:00 on March 14th and the amount of steam generated by flashing. Steam amount shown here is the steam generated by other processes (injected water, debris falling, etc.). Hydrogen is given as the net amount.

[2] The integrated amount of hydrogen provided was 274kg at 22:40 on March 14th and 975kg at 02:00 on March 15th.

Table 2.3 Appropriateness of condition setting for depressurization and steam/hydrogen generation

| Item | Setting | Grounds for the setting |
|---|---|---|
| Size of SRV opening | Size changes of SRV opening from being opened until closed | Size increase of SRV opening at the beginning<br>A recorded message "18:02 SRV2 opened because RPV pressure decrease insufficient" in the Operational Records (July 17, 2013) was reflected.<br>Until 18:40 on March 14th<br>The RPV pressures were reproduced by decreasing the SRV opening size subject to the RPV pressure decrease. This means a possibility of SRV opening size changes subject to the pressure difference at the actual system. There could be other reasons to change the RPV pressures.<br>From about 19:20 to about 20:40 on March 14th<br>The RPV pressure did not increase during this time period, but it was necessary to assume an SRV opening size increase in order to reproduce the RPV pressures measured. A possible reason for this was that the gas compositions (fractions of steam and hydrogen) flowing through the SRV(s) after hydrogen generation were different in the actual situation and in the analysis (the lower hydrogen fraction and the higher steam fraction in the analysis because they were averaged in each node). The amount of gas flows discharged from the RPV during this period depended on the speed of sound of the gas components, because the flow was considered to have been in critical flow conditions between the RPV and S/C. The speed of sound of hydrogen is about 3 times that of steam [8]. A lower hydrogen fraction would decrease the amount of discharge gas from the RPV and cause an underestimation of the RPV pressure decrease rate. In order to compensate for this, a size increase of the SRV opening would become necessary in the analysis. In the accident progression scenario in Table 2-1 the SRV(s) were estimated to have been closed at about 20:15. But in the analysis, the SRV(s) were assumed to have been slightly open until about 20:40 for better reproduction of D/W pressure |

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| | | changes.<br><u>A few minutes from about 21:20 on March 14th</u><br>Set as in the "Until 18:40 on March 14th" above.<br><u>After about 01:10 on March 15th</u><br>The SRV opening size change after opening was ignored for the following reasons: The opening operation at this time was not to activate the SRV relief functions as in other opening operations, but to activate the ADS functions and the appropriateness to change the SRV opening size subject to the pressure difference was unclear; and the influence to the analysis results were minor. |
| | Size of SRV opening in opened position | The size was chosen which could reproduce the decreasing tendency of RPV pressure (below the SRV dead load closure pressure) between 21:21 and 21:34 on March 14th. |
| Leak hole size from D/W to R/B | Leaks considered | Leaks from the D/W to the R/B had to be assumed to reproduce the D/W pressure decrease after around 00:00 on March 15th. See the analysis results below (Base Case, Sensitivity Cases). |
| | Leak size changes (decrease) | In the current evaluation, it was necessary to assume the leak size reduction during the time of D/W pressure increase at about 01:30 on March 15th, in order to reproduce the D/W pressures. The reason for this could be (as in "From about 19:20 to about 20:40 on March 14th" above) that the gas compositions (fractions of steam and hydrogen) flowing through the leak hole were different in the actual accident and in the analysis (lower hydrogen fraction and higher steam fraction in the analysis, because of averaging them in each node). The hydrogen concentration in the upper part of the PCV depends on the extent to which it is mixed while it is transferred, after being generated in the core, to the SRV(s), S/C, V/Bs and D/W. This is hard to simulate appropriately in the analysis code GOTHIC used in the current evaluation. But the above setting seems reasonable for the objective of analyzing leaks from the PCV. |
| The amount of steam | The amount | The amount of steam generation set in the current evaluation was adjusted mainly for use in reproducing the RPV pressure |

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| generated | | changes. The amount of steam increase increases the RPV pressure. While the steam was condensed in the S/C, the amount of steam generation does not increase the PCV pressure.<br>It should be noted that the heat transfer via vapor phase to the reactor water remaining as of 18:00 on March 14th and the amount of steam generated by flashing were calculated by the analysis code. The amount of steam generation to be set here was that by other reasons (water injection, fuel debris falling, etc.).<br>Until about 21:20 on March 14th<br>No steam generation was assumed. Water injection to the RPV was resumed at 19:54, but the water was injected via the downcomer and only a little water is considered to have reached the core for generating steam. Therefore, the assumption above of "no steam generation" would be reasonable.<br>From about 21:20 to about 22:40 on March 14th<br>A spike-shaped steam generation was assumed at about 21:20. This would correspond to the flashing of the water injected into the RPV. A series of steam generations thereafter were assumed. This assumption would be reasonable, because part of the injected water would evaporate due to elevated temperatures in the RPV.<br>From about 23:40 on March 14th to about 00:00 on March 15th<br>A large amount of steam generation was assumed. The RPV pressure was considered to have exceeded 1MPa during most of this period impeding the water injected by fire engines to reach the core. But it was necessary to set a large amount of steam generation during this period in order to reproduce the RPV pressure changes. In other words, the amount of steam generation set here was the amount which enabled reproduction of the RPV pressure increase observed after about 23:40 on March 14th, when the situation of the afore-mentioned "SRV(s) in opened position" was assumed (the size which could reproduce the decreasing tendency of RPV pressure between 21:21 and 21:34 on March 14th). |

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| | | A possible reason for such a large amount of steam generation is that the water in the lower plenum was evaporated by falling of part of the fuel debris. When the fuel debris fell into the lower plenum, a large amount of steam was generated because of a big temperature difference between the fuel debris and water. When the temperature of the fuel debris was lowered, the amount of steam generation also was lowered to the level corresponding to the decay heat. It should be noted that the total amount of steam generated during 22:40 to 23:40 set in the current evaluation corresponded roughly to half of the water inventory in the lower plenum.<br>The amount of steam generation changed up and down from about 23:25 to about 23:40. It was so set to reproduce the observed tendencies of the RPV pressures during this period: from 23:25 to 23:30 it had decreased drastically and then the decrease became slower from 23:30 to 23:40. A possible reason for this is that some fuel debris newly fell into the lower plenum.<br>It should be noted that the amount of steam generated during about 23:40 on March 14$^{th}$ to about 00:00 on March 15$^{th}$ corresponded to the amount when about 40% of the total decay heat had been transferred to water. The background for this would be that the decay heat of the fuel debris that had fallen to the lower plenum until this time point had been transferred to the water therein.<br>In order to check the reality of the thus estimated accident progression, the fraction X of fuel debris fallen to the lower plenum by this period was calculated by the following equations.<br>$Q_{evap}=Q_{quench}+Q_{decay}+Q_{H2}$<br>$Q_{evap}=M_{evap}*h_{fg}$<br>$Q_{quench}=M_{core}*X*Cp*\Delta T$<br>$Q_{decay}=q*Time$<br>$Q_{H2}=0$<br><br>Here, $Q_{evap}$ is the evaporation latent heat of water in the lower plenum, $Q_{quench}$ is the heat discharged from the fallen fuel debris before its temperature was lowered to the saturation |

| | | temperatures, $Q_{decay}$ is the decay heat, and $Q_{H2}$ is the water-metal reaction heat, all of which contribute to the heat transfer in the water of the lower plenum. But $Q_{H2}$ (water-metal reaction heat in the water) was ignored in the current evaluation, since its magnitude was unknown. Other values were set as follows. |
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| Notation | Meaning | Value | Remarks |
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| $M_{evap}$ | Amount of water evaporated in the lower plenum | 21000kg | Total amount of steam generated during 22:40 to 23:40 |
| $h_{fg}$ | Latent heat of evaporation | 2000kJ/kg | |
| $M_{core}$ | Total weight of fuel | 160000kg | About 300kg per assembly x 548 assemblies |
| Cp | Specific heat of fuel | 0.3kJ/kg-K | |
| $\Delta T$ | Temperature difference between fuel debris and saturated water | 1600K | Fuel debris at 1800 deg C and water at 200 deg C were assumed |
| q | Decay heat | 7500kW | |
| Time | Duration | 3600s | 22:40 to 23:40 |

From the equations above X=0.4 was obtained. In other words, the amount of steam generation set in the current evaluation could be explained by assuming that about 40% of the fuel had fallen into the lower plenum as fuel debris. On the other hand, as noted above, the amount of steam generated from about 23:40 on March 14th to about 00:00 on March 15th corresponded to the amount when about 40% of the total decay heat had been transferred to water. This consistency indicates that the estimation for the accident progression scenario set was realistic and the values set in the current analysis were reasonable.

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| | | But it should be noted that the fraction of fallen fuel debris derived above is simply a rough value and is subject to the uncertainties of the SRV opening size, uncertainties of the contribution of water-metal reaction heat to water evaporation, etc.<br><u>After about 00:00 on March 15th</u><br>A large amount of steam generation was assumed. Like the period from about 22:40 on March 14th to about 00:00 on March 15th, part of fuel debris was considered to have fallen into the lower plenum. At around 01:10, the total amount of steam generated after about 22:40 on March 14th reached the total water inventory in the lower plenum, which would mean that the entire water inventory in the lower plenum had evaporated and the amount of steam generation had declined. The condition settings in the current evaluation would be more or less reasonable, because such a scenario as described above turned out to be feasible for interpreting the accident progression.<br>The amount of steam generated thereafter had little influence on the analysis results, but it should be noted that, in the current evaluation, about 4 t/h of the water was assumed to have been injected into the reactor and totally evaporated, based on the MAAP results [2][3]. |
| Amount of hydrogen generation | The amount | The amount of hydrogen generation in the current evaluation was set by adjustment so that the RPV pressures during the period assuming "SRV(s) closed (between about 20:15 and 21:20 on March 14th)" and the D/W pressures for other time periods assuming "SRV(s) open" could be reproduced. While the SRV(s) were open, more hydrogen was discharged from the RPV because of its smaller molecular weight (higher speed of sound) and it contributed less to the RPV pressure increase.<br><u>Until about 21:20 on March 14th</u><br>A moderate amount of hydrogen generation was assumed. It was possible that the fuel temperatures increased with the core being uncovered and the water-metal reactions started at around 20:00. This hydrogen generation might have been caused by the steam present in the core and the steam generated by, for |

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| | | example, the water in the lower plenum being evaporated by the heat of the core. The condition settings in the current evaluation would be more or less reasonable, because such a scenario as described above turned out to be feasible to interpret the accident progression.<br>From about 21:20 to about 22:40 on March 14th<br>A spike-shaped hydrogen generation was assumed at about 21:20. This would correspond to the steam generation upon the reactor depressurization. Gradual hydrogen generation was assumed to follow thereafter until about 21:40. This would correspond to the flashing of the water in the RPV. A small amount of continuous hydrogen generation was assumed thereafter until about 22:40. The steam generated by the continuous water injection to the reactor would have contributed to the hydrogen generation. The condition settings in the current evaluation would be more or less reasonable, because such a scenario as described above turned out to be feasible to interpret the accident progression.<br>From about 22:40 to about 23:40 on March 14th<br>A large amount of hydrogen generation was assumed. This would correspond to the steam generation due to: (i) the increased reactor water level above the bottom of active fuel (BAF) by water injection of fire engines; and (ii) the fuel debris falling into the lower plenum. It is also possible that hydrogen was generated by the continuous water–metal reactions for a while immediately after the fuel debris had fallen into the water in the lower plenum. In the current evaluation, a large amount of hydrogen generation was assumed immediately after the debris fell and a little less thereafter. This would indicate a possibility that the oxide film thickened during the early violent reaction period and reduced the hydrogen generation rate, zirconium (Zr) was cooled in the water or other reasons. The condition settings in the current evaluation would be more or less reasonable, because such a scenario as described above turned out to be feasible to interpret the accident progression.<br>It should be noted that, if the total amount of Zr in the reactor |

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| | | (including cladding, water rods, spacers, channel boxes) had reacted with water, about 1900kg of hydrogen should have been generated (or about 1000kg when only Zr in the cladding had reacted). The net amount of hydrogen which could be generated in the reactor would be less than 1900kg, since it can be considered that the Zr in the surface layers of structures is easily oxidized upon contact with steam, while that in the deep layers is less oxidized. The total amount of hydrogen generated till about 23:40 on March 14th was about 940kg, It is possible, therefore, that the water-metal reactions mostly ended by this time.<br><u>After about 00:00 on March 15th</u><br>A spike-shaped hydrogen generation was assumed twice, at about 00:06 and at about 01:10. It is possible to consider an accident progression, in which, at about 00:06, another falling of fuel debris to the lower plenum took place again and part of the residual Zr reacted with a large amount of steam, but hydrogen generation thereafter calmed down; at about 01:10, new steam flow upon opening the SRV(s) caused additional water-metal reactions. The condition settings in the current evaluation would be more or less reasonable, because such a scenario as described above turned out to be feasible to interpret the accident progression. |

### 2.2.3. Results of analysis

The following are the results obtained based on the geometrical configuration and the conditions of analysis given in 2.2.1 and 2.2.2. In addition, the results of sensitivity analysis are also given for the sensitivity cases: (i) when ignoring leaks from the D/W to the R/B; (ii) the D/W temperatures; and (iii) the steam amount generated.

#### 2.2.3.1. Results of the Base Case

Figures 2.5 and 2.6 compare the results of analysis and measured values of the D/W pressures and S/C pressures. Both RPV and D/W pressures were well reproduced by appropriate adjustment of the SRV leak size, the leak size from the D/W to R/B, and the amount of steam/hydrogen provided. The RPV pressure increase was also well reproduced between 18:40 and 19:20 by taking into account the decay heat.

It should be noted that, while the RPV pressure dropped to the level of the D/W pressure, the results of RPV pressure analysis were slightly higher than the measured pressures. This could have come from the effect of water evaporation in the water level indicator line. The RPV pressure gauge was located at the end of the water level indicator line. It is known that when the water level decreases in the reference leg, the RPV pressure is underestimated by the amount equivalent to its water head (about 1 [atg] maximum).

Figure 2.7 presents the results of RPV vapor temperatures analysis. The temperature increased due to the decay heat and water-metal reactions, while it decreased due to core cooling by steam. It should be noted that the vapor temperature at the upper part of the RPV (steam dome and downcomer region) reached about 600 deg C at the time of the second SRV opening (about 21:20). In the current examination it was assumed that the status of this SRV opening had been maintained. Part of the SRV components might have been affected by these high temperature gases.

The results of analysis are shown in Figure 2.8, concerning the temperatures in the PCV. The vapor temperatures in the D/W might have been affected by the uncertainties of heat (heat transfer coefficients) transferred to the D/W from the RPV. In the current examination, such conditions were searched which could reproduce the PCV pressures with consideration of the uncertainties of D/W vapor temperatures. The S/C vapor temperatures followed basically the vapor saturation temperatures, but when a large amount of hydrogen was discharged from the RPV the S/C vapor temperatures exceeded, and then decreased to, the saturation temperatures. It should be noted that the tendency of the vapor temperatures becoming lower than the liquid phase temperatures after about 00:00 on March 15$^{th}$ might have come from the interfacial heat transfer model used in GOTHIC.

Consideration should be given to the relationship of the results with the water level indicator readings shown in Figure 2-1. After 21:20 on March 14th, the water level indictor readings showed an increasing trend. This suggests a possibility that the D/W temperatures rose at this timing and evaporated the water in the water level indicator line. The saturation temperature at the RPV pressure (about 0.5MPa) between 21:30 and 22:30 was 152 deg C, while in the analysis the D/W temperature reached as high as about 140 deg C at the timing of SRV(s) opening at about 21:20. This temperature was below the saturation temperature in the RPV, but it is still feasible that the water in the water level indicator line partly evaporated during this time period, when the uncertainties of the D/W temperatures are considered or considerations are given to the local distribution of D/W temperatures

The results of analysis concerning the vapor phase leaks from the D/W to R/B are also given in Figure 2.9 as reference information.

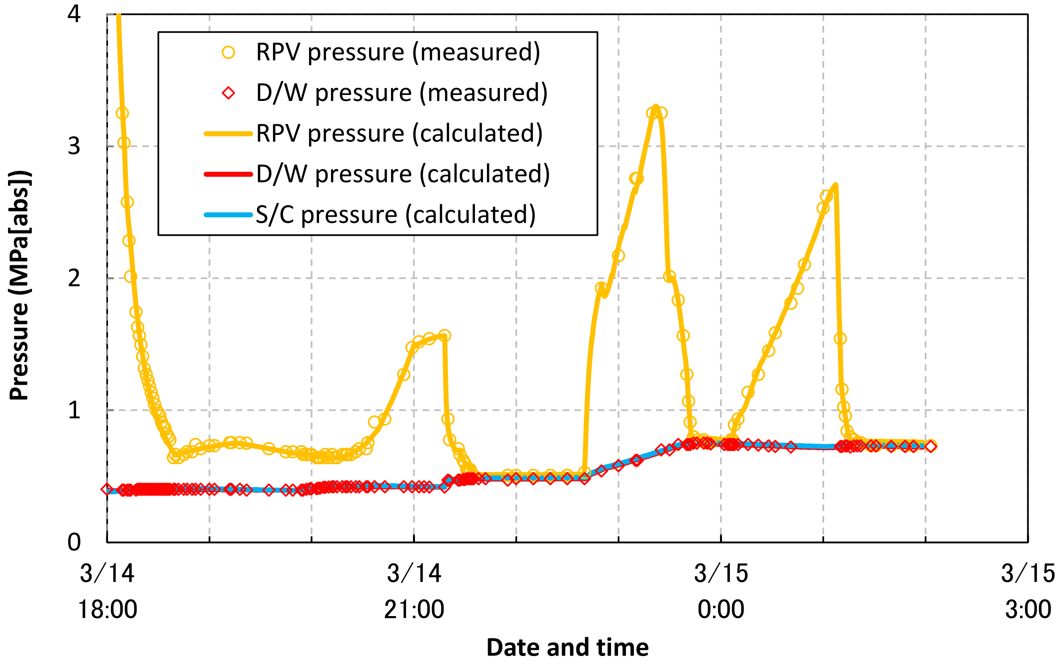


Figure 2-5 Base Case analysis results (Pressure): 0 – 4MPa

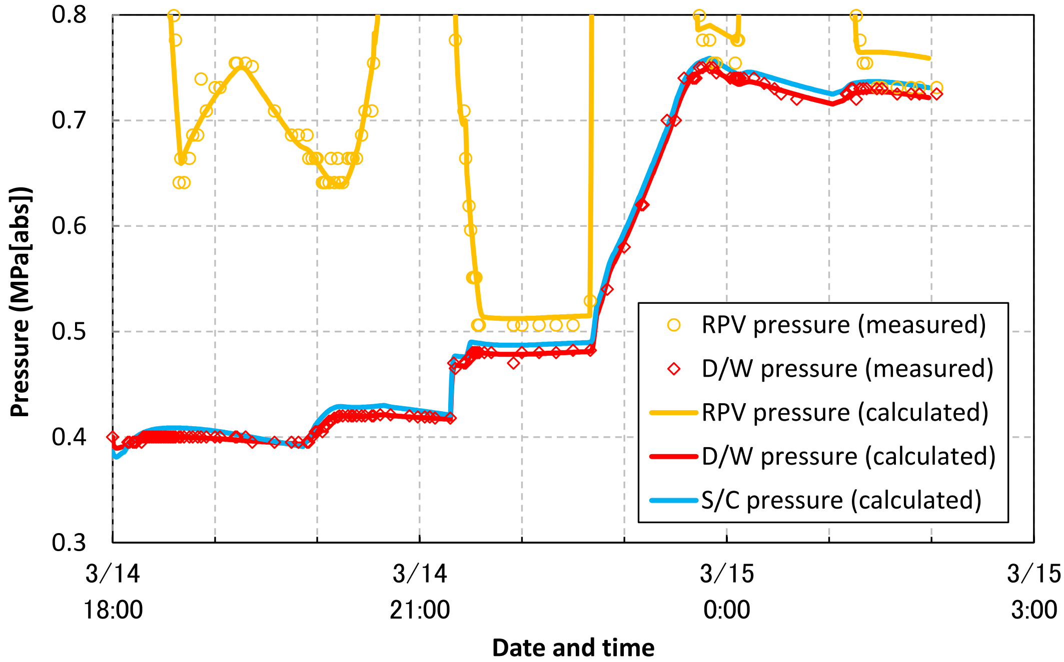


Figure 2-6 Base Case analysis results (Pressure): 0.3 – 0.8MPa

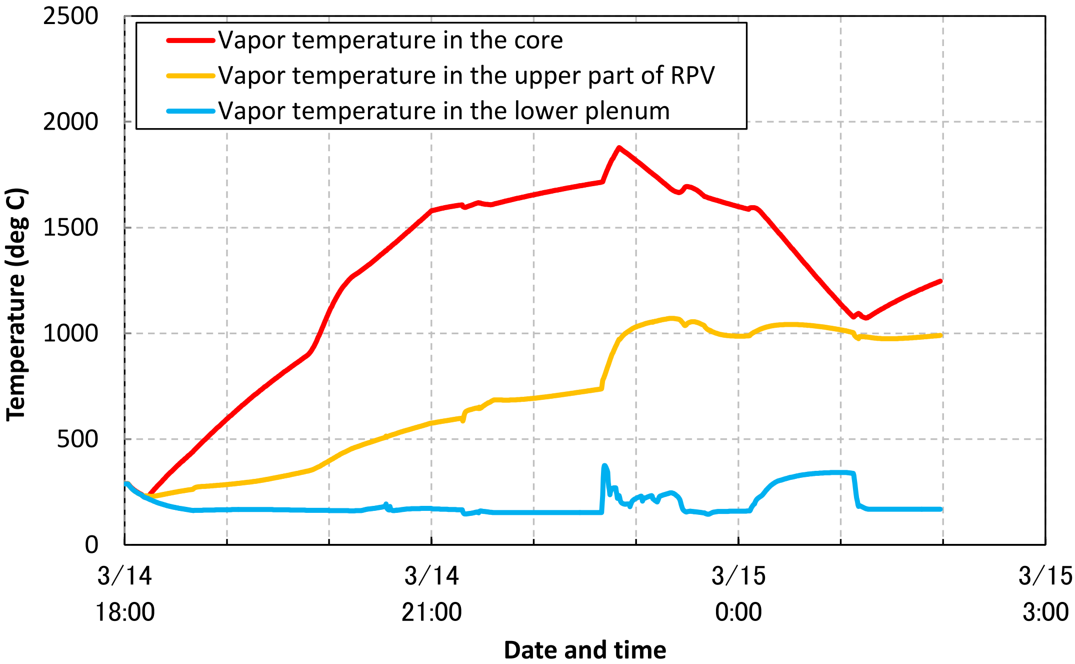


Figure 2-7 Base Case analysis results (RPV vapor temperatures)

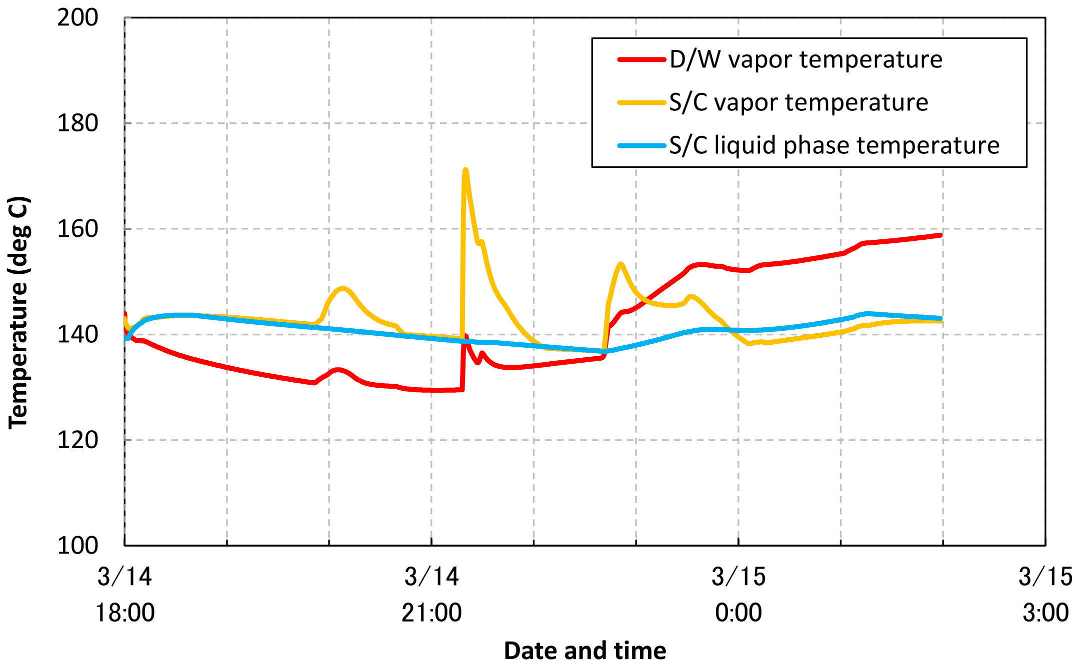


Figure 2-8 Base Case analysis results (PCV temperatures)

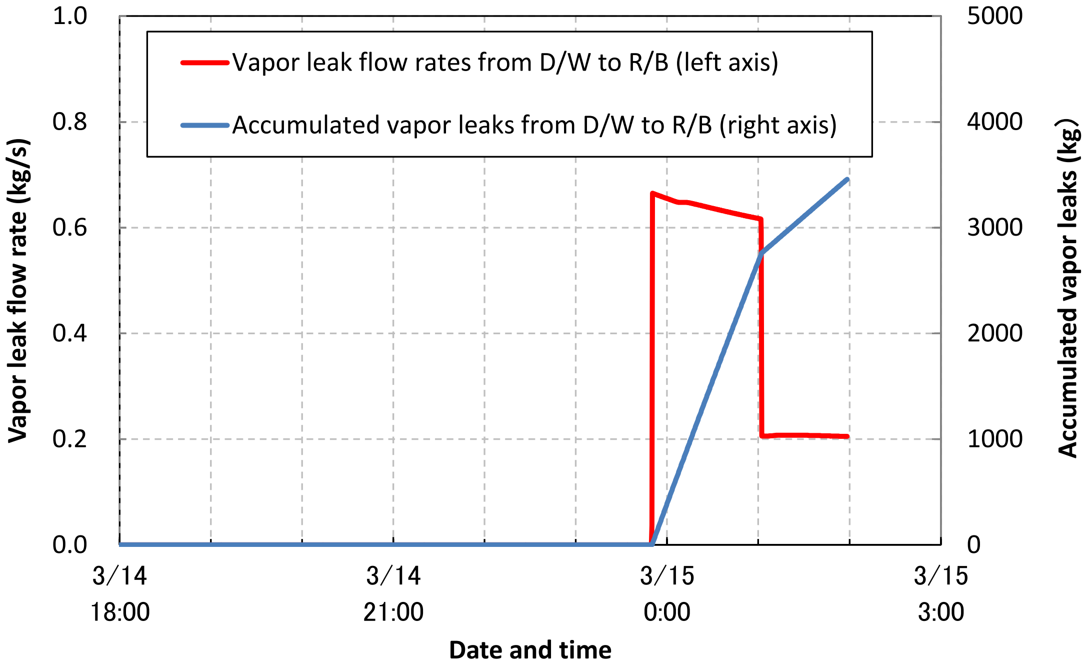


Figure 2-9 Base Case analysis results (Leak rates in gaseous phase from D/W to R/B)

#### 2.2.3.2. Results of sensitivity analysis

The following are the results of sensitivity analysis for the cases mentioned in Table 2.4.

The analysis in the Base Case confirmed that the D/W pressure changes (the decreasing trend after about 00:00 on March $15^{th}$) could be reproduced by assuming gas leaks from the D/W to the R/B. In the sensitivity analysis, the reproducibility of the D/W pressure changes without assuming the leaks from the D/W to R/B was checked. There are two big possibilities to cause the PCV pressure to decrease: vapor leaks or steam condensation. The steam condensation depends on the fraction of steam in the PCV and cooling situation. With this consideration, the following three cases were examined in the sensitivity analysis: Sensitivity Case ① (no vapor leaks assumed), Sensitivity Case ② (changes of vapor leaks to PCV taken into consideration) and Sensitivity Case ③ (changes of cooling situation in the PCV taken into consideration).

In addition, two other cases were considered: Sensitivity Case ④ (for obtaining relevant information on the effect of D/W temperature uncertainties), and Sensitivity Case ⑤ (changing the amount of steam generation from about 23:40 on March $14^{th}$ to about 00:00 on March $15^{th}$ as part of verifying the fuel debris falling scenario to the lower plenum).

It should be noted that no adjustment was made to reproduce measured values in these sensitivity cases concerning the conditions of depressurization or steam/hydrogen generation. This is because the sensitivity analysis was to evaluate only qualitative impacts on the D/W pressures and temperatures. .

Table 2.4 Sensitivity Analysis Cases

| Case | Contents | Objectives |
| --- | --- | --- |
| ① | No leaks from D/W to R/B assumed vs. the Base Case | To check the D/W pressure reproducibility without leaks |
| ② | Vapor leaks from RPV to D/W from 22:40 to 23:50 assumed vs. Case ① | To check the impacts of changing steam release to the PCV on the D/W pressures |
| ③ | Heat removal by S/C after 23:50 increased vs. Case ① | To check the impacts of changing the PCV cooling situation on the D/W pressures |
| ④ | Heat transfer coefficients from RPV to D/W changed vs. Base Case | To check the impacts of changing heat transfer situations from RPV to D/W on the D/W vapor temperatures |
| ⑤ | The amount of steam generation changed from 23:40 on March $14^{th}$ to 00:00 on March $15^{th}$ vs. Base Case | To check appropriateness of the fuel debris falling scenario to the lower plenum |

Sensitivity Analysis Case ①: Leaks from D/W to R/B excluded from the Base Case

Case ① checked the reproducibility of D/W pressure changes without assuming the leaks from the D/W to R/B after around 23:50 on March 14th assumed in the Base Case.

Figures 2.10 and 2.11 compare the results of analysis with the measured data of the RPV pressure, the D/W pressure and the S/C pressure. The D/W pressure after around 00:00 on March 15th was an increasing trend. The decreasing trend of measured data could not be reproduced by the steam condensation only in the condition of the Base Case.

Sensitivity Analysis Case ②: Leaks from RPV to D/W assumed from 22:40 to 23:50 in Case ①

This case assumed vapor leaks from the RPV to the D/W between 22:40 and 23:50 in Case ①, in order to check the impacts of steam release changes to the PCV on the D/W pressures[3]. A leak path was set from the core to the D/W, simulating leaks through the in-core instrumentation line. The leak size of $10cm^2$ was assumed.

Figures 2.12 and 2.13 compare the results of analysis and measured values of the RPV pressure, the D/W pressure and the S/C pressure, while Figure 2.14 gives the analysis results of the PCV temperature. By adding a leak path, the RPV pressure decreased and the D/W pressure increased (Case ② vs. Case ①). The measured D/W pressure slightly dropped after it increased. This is because, as shown in Figure 2.14, the D/W temperature was increased instantaneously by direct leaks from the RPV to the D/W and then was cooled down by, for example, the D/W internal structures. Thereafter, the D/W temperature was increased again due to the steam discharge from the RPV to the S/C. Thus, the decreasing trend of D/W pressure could not be reproduced, even when the steam discharge to the PCV was changed.

Sensitivity Analysis Case ③: Heat removal by the S/C at 23:30 increased in Case ①

In Case ③ more heat was removed from the S/C than in Case ① after 23:30 on March 14th, in order to check the effects of changes of PCV cooling conditions on the D/W pressure. The inundation situation in the torus room at that time is still unknown. In the analysis, an increased water level in the torus room was assumed, thus removing more heat. Increased

[3] The reason for limiting the time span of leaks is as follows: Before 22:40 and after 23:50, the measured RPV pressure ≧the measured D/W pressure. When a possibility of underestimated RPV pressure due to water evaporation in the water level indicator line was considered, it was possible that during this time, too, the RPV pressure>the D/W pressure. Should leaks occur from the RPV to D/W, the RPV pressure and D/W pressure would balance, when the RPV pressure decreased. Therefore, no leaks would have occurred before 22:40 or after 23:50. Concerning the blockade of leak holes, the leak holes of in-core instrumentation line could have been blocked by the debris [9].

heat removal was simulated by the increased heat transfer coefficient of the S/C wall, i.e., 10 times. This corresponds to a situation in which the S/C walls were entirely immersed in the saturated water and at atmospheric pressure.

Figures 2.15 and 2.16 compare the results of analysis and measured values of the RPV pressure, the D/W pressure and the S/C pressure. As compared with the results in Case ①, the D/W pressure increase was slower, but still its decreasing trend could not be reproduced by the increased heat removal only from the S/C.

From the sensitivity analysis cases ①, ②, and ③, it became certain that the leaks from the D/W to the R/B needed to be assumed in order to reproduce the D/W pressure decreasing trend after around 00:00 on March 15th in the current estimated accident progression scenario.

Sensitivity Analysis Case ④: Heat transfer coefficients from the RPV to D/W increased from the Base Case

In Case ④, the heat transfer coefficients from the RPV to the D/W were doubled (x2) or quadrupled (x4), in order to check the impacts of heat flow from the RPV to the D/W on the D/W vapor temperatures. The heat transfer coefficients in the Base Case were set based on the estimated heat balance in normal operations. This Sensitivity Analysis Case ④ was to consider a possibility of increased heat transfer coefficients when the RPV wall temperatures increased.

Figure 2.17 shows the results of analysis of the PCV temperatures. The increased heat transfer coefficients from the RPV to D/W increased the D/W vapor temperatures. The increased water level indicator readings at about 21:20 could have been caused by evaporation of the water in the reference leg (saturation temperature is about 152 deg C). Figure 2.17 indicates a possibility of the D/W vapor temperature increase up to this level. It should be noted that the analysis results of the Base Case, in which the conditions were searched to reproduce the D/W pressures including the consideration of uncertainties of vapor temperatures, were in line with the objective of this examination.

Sensitivity Analysis Case ⑤: The amount of steam generation changed from the Base Case between about 23:40 on March 14th and 00:00 on March 15th

In the condition settings of Table 2.3, the fuel debris was assumed to have fallen into the lower plenum between around 22:40 to 23:40 on March 14th followed by steam generation due to its decay heat from around 23:40 on March 14th to 00:00 on March 15th. In the Base Case, the amount of steam generation between around 23:40 on March 14th and 00:00 on

March 15th was set corresponding to the about 40% of total decay heat in the core, and this could well reproduce the RPV pressure decreasing trend in the limited time period after 23:40. Sensitivity Analysis Case ⑤ checked the impacts on the reproducibility of measured values by changing the amount of steam generation to zero (assuming no debris falling to the lower plenum) or doubling it (assuming more fuel debris falling to the lower plenum); in other words, Case ⑤checked the appropriateness of the fuel debris falling scenario to the lower plenum.

Figure 2.18 shows the analysis results of the RPV pressures between 23:40 and 23:43. When the steam generation was nullified, the RPV pressures were underestimated compared to the measured values, while, when doubled, the RPV pressures were overestimated compared to the measured values. Although the RPV pressure decreasing speed depends on the size of the SRV opening, it would be appropriate to estimate, with some uncertainties, that a certain amount of fuel debris fell down to the lower plenum.

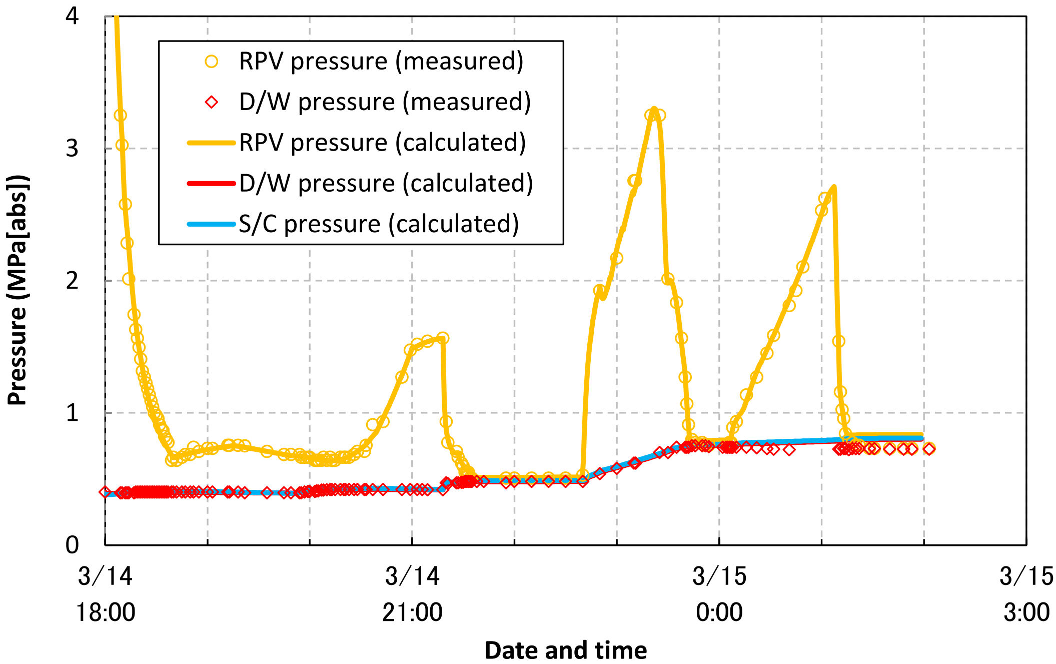


Figure 2.10 Results of Sensitivity Analysis Case ① (pressures): 0 – 4 MPa

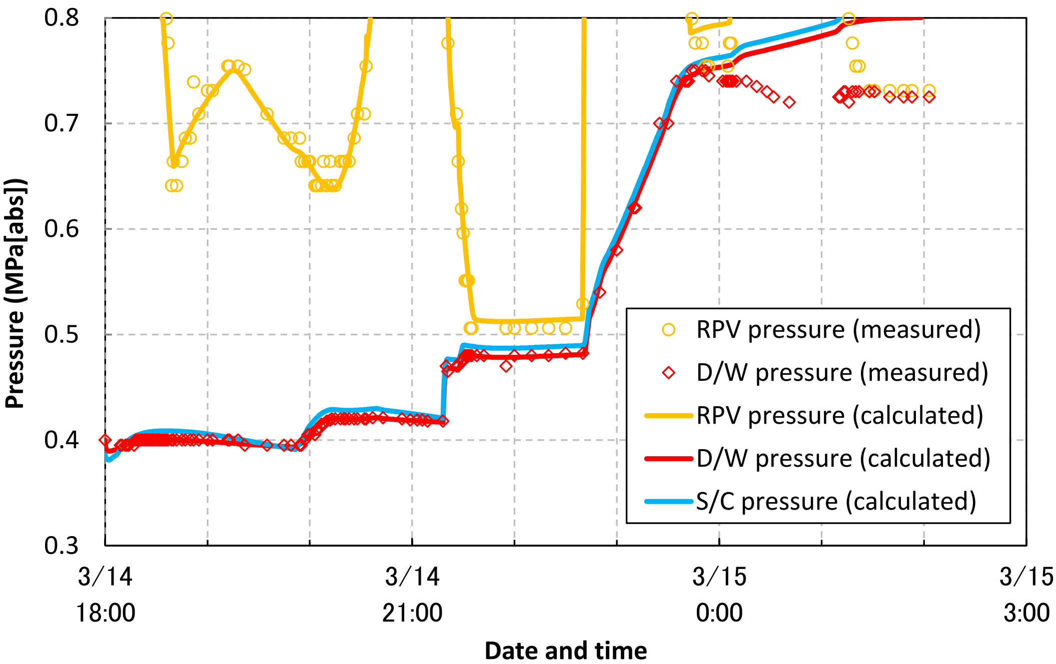


Figure 2.11 Results of Sensitivity Analysis Case ① (pressures): 0.3 – 0.8 MPa

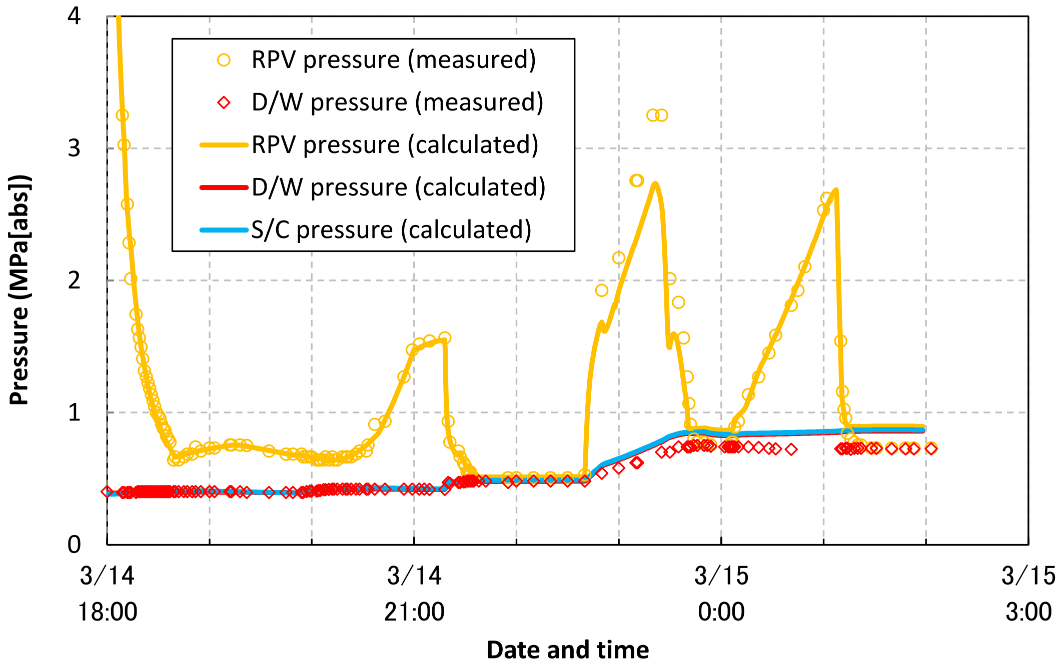


Figure 2.12 Results of Sensitivity Analysis Case ② (pressures): 0 – 4 MPa

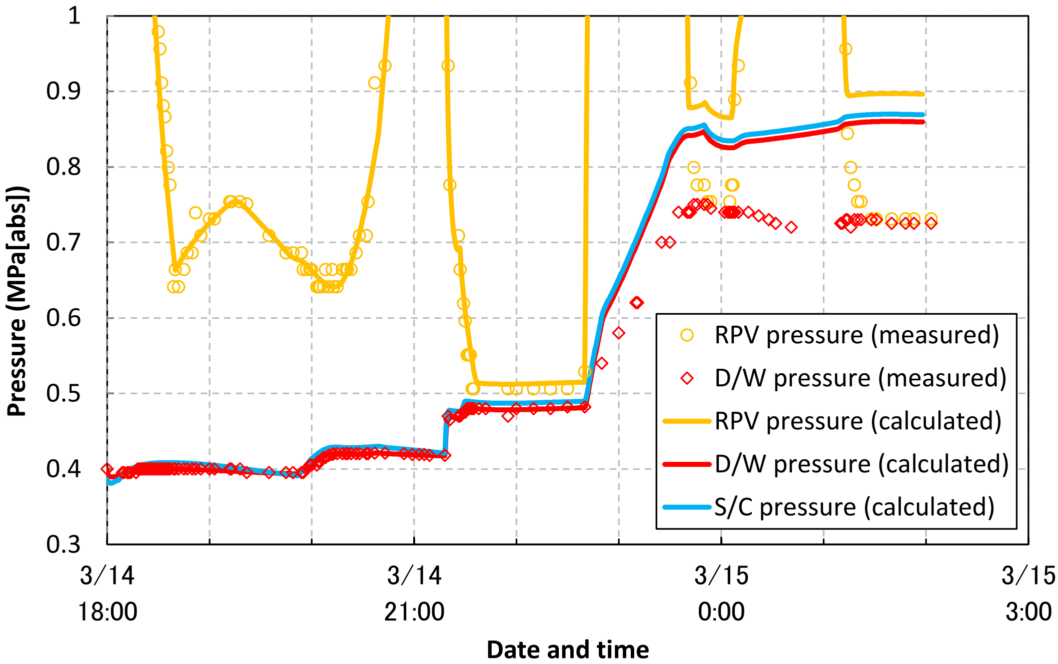


Figure 2.13 Results of Sensitivity Analysis Case ② (pressures): 0.3 – 1.0 MPa

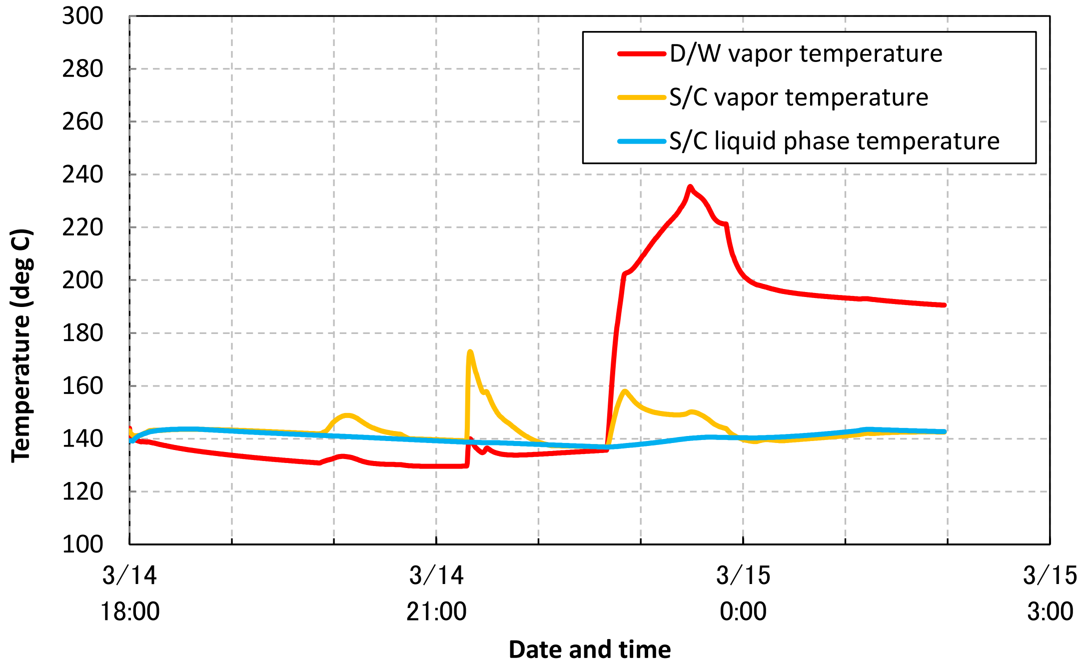


Figure 2.14 Results of Sensitivity Analysis Case ② (PCV temperatures)

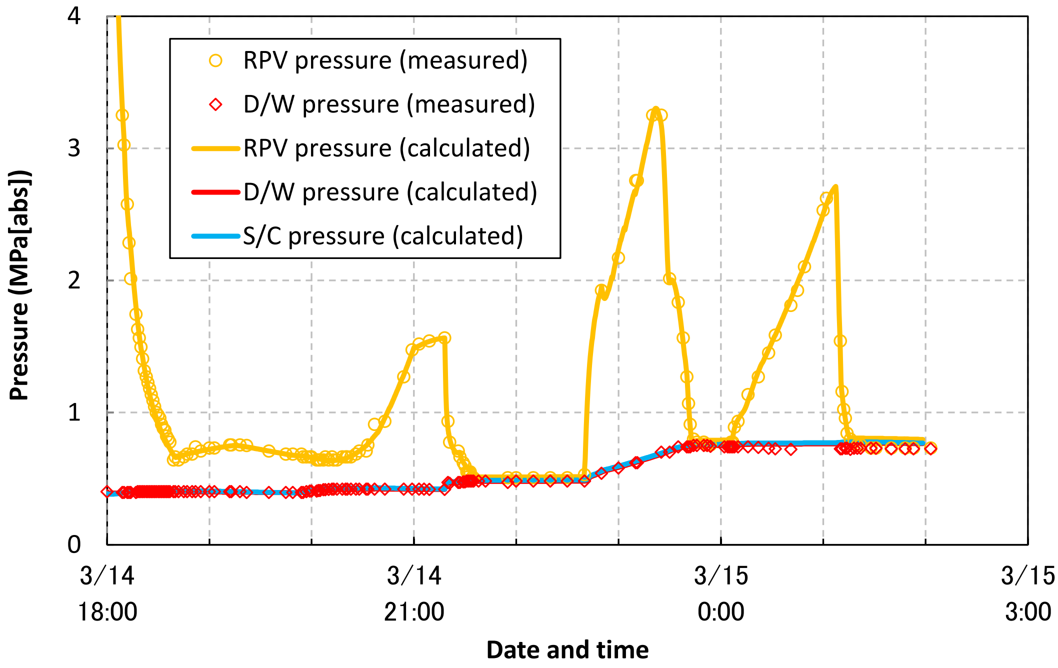


Figure 2.15 Results of Sensitivity Analysis Case ③ (pressures): 0 – 4 MPa

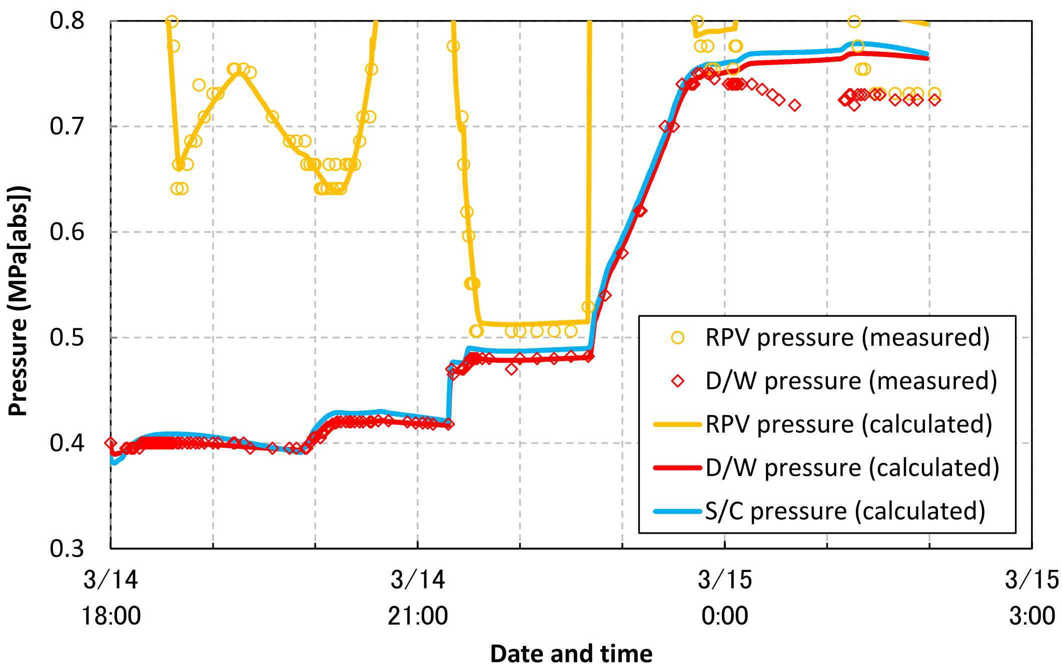


Figure 2.16 Results of Sensitivity Analysis Case ③ (pressures): 0.3 – 0.8 MPa

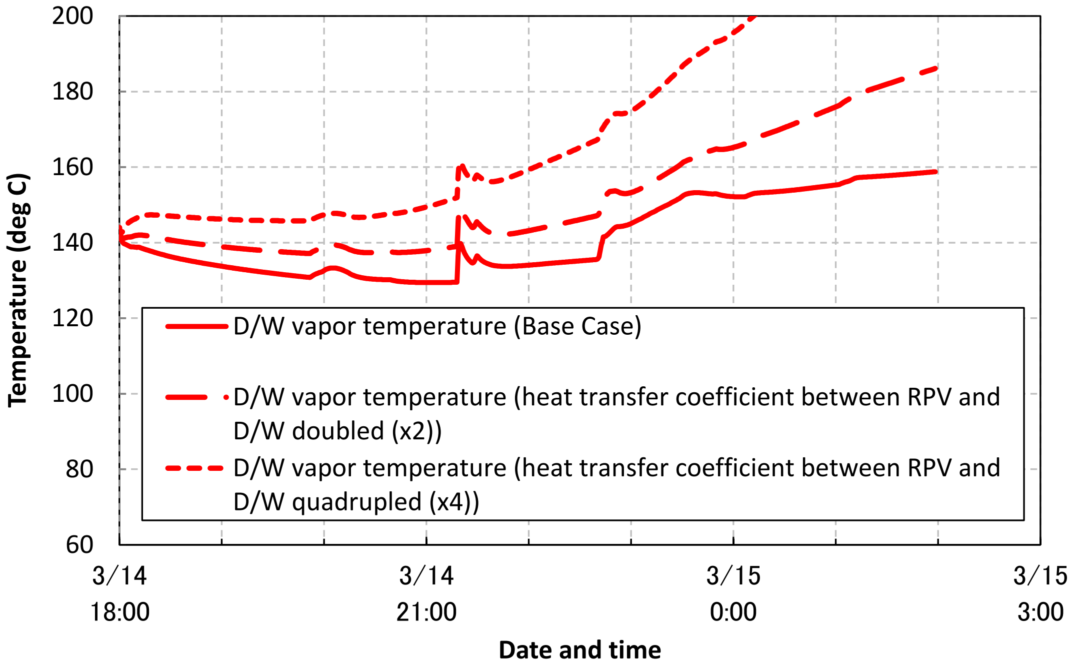


Figure 2.17 Results of Sensitivity Analysis Case ④ (D/W vapor temperatures)

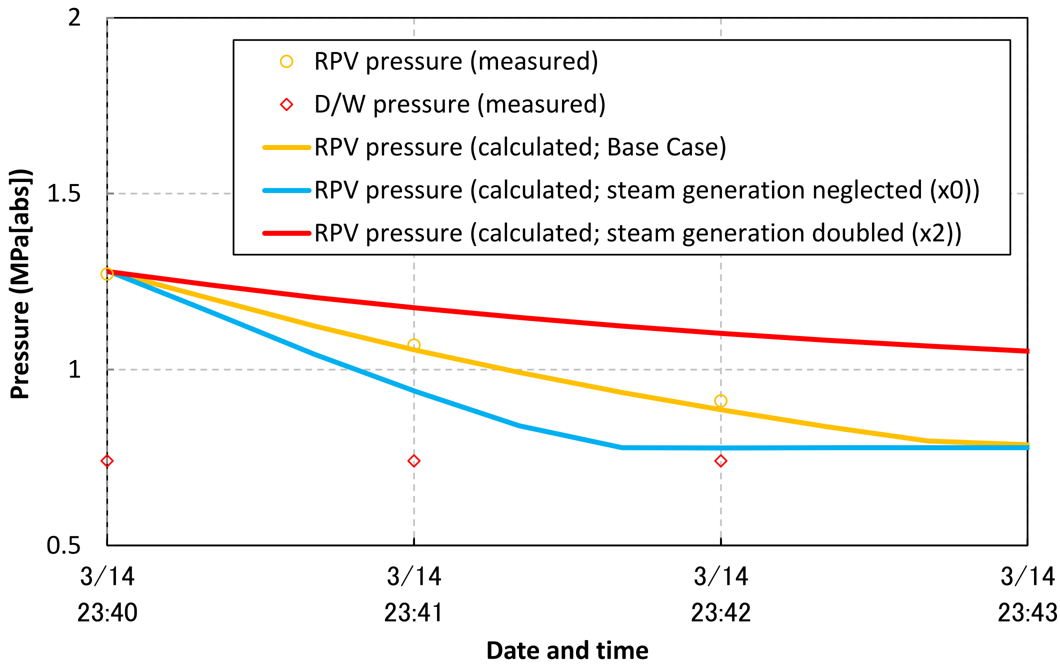


Figure 2.18 Results of Sensitivity Analysis Case ⑤ (RPV pressures)

## 2.3. Deliberation on the accident progression after forced depressurization

Based upon the deliberations above, the grounds for the reactor pressure changes and in this connection the accident progression in the core and PCV were estimated.

An analysis scenario was estimated based on the accident progression scenario summarized in Table 2.1. A set of conditions for analysis (the depressurization conditions and the amount of steam/hydrogen generation) could be found which could well reproduce the measured reactor pressures and PCV pressures. The set of conditions were obtained by adjusting relevant parameters. Their appropriateness was provided in Table 2.3. It was confirmed that the trend of measured reactor pressures and PCV pressures was explicable from the accident progression scenario given in Table 2.1

Furthermore, in connection with the accident progression, the following points were noted in the process of verifying appropriateness of conditions for the analysis shown in Table 2.3.

- ✧ The RPV pressure increase around 22:40 to 23:40 on March 14th can be considered to have been caused by the discharge of a large amount of steam due to the debris falling to the water in the lower plenum. By around 23:40 the debris temperature decreased to about the saturated water temperatures and the amount of steam generation decreased. But again the fuel debris dropped at around 00:06 on March 15th and that increased the RPV pressure. In addition, the water inventory in the lower plenum might have been totally lost by around 01:10 on March 15th.
- ✧ The amount of hydrogen generation set in the current examination was about 940kg by 00:00 on March 15th and about 975kg by 02:00 on March 15th. This amount corresponds to the amount to be generated when most of the zirconium in the core that was available for oxidization was oxidized. This indicates a possibility that by 00:00 on March 15th most water-metal reactions in the core came to an end.

Furthermore, the following points were confirmed, from the sensitivity analysis cases in 2.2.3.2, concerning the accident progression.

- ✧ In the accident progression scenario estimated in the current examination, leaks from the D/W to the R/B needed to be assumed so that the decreasing trend of the D/W pressures after around 00:00 on March 15th could be reproduced.
- ✧ It is possible that the increase of water level indicator readings at around 21:20 on March 14th was caused by evaporation of water in the water level indicator line.

Table 2.5 summarizes the examination results of Unit-2 accident progression scenario after the forced depressurization. Concerning the SRV(s) opening/closing situation, leaks from the D/W to R/B and the amount of steam/hydrogen generated, Figures 2.3 and 2.4 are cited as the examination results of their magnitudes and chronological changes.

Table 2.5 Examination results of Unit-2 accident progression scenario after forced depressurization

| Date | Time | Accident progression | Grounds |
|---|---|---|---|
| March 14th | 18:02 | SRV(s) opening forced | Table 2-1 |
| | about 18:40 | SRV(s) closed by their dead load | Table 2-1 |
| | about 19:20 | SRV(s) slightly opened | Table 2-1 |
| | 19:54 | Water injection resumed | Table 2-1 |
| | about 19:54 | Hydrogen generation started in the core | Table 2-3 |
| | about 20:40 | Slightly opened SRV(s) closed<br>(RPV pressure increased mainly due to hydrogen generation in the core by 21:20) | Table 2-3 |
| | about 21:20 | SRV(s) opening forced | Table 2-1 |
| | about 21:30 | SRV(s) opened and remained opened thereafter | Table 2-1 |
| | about 22:40 | Part of the fuel debris collapsed and fell into the water in the lower plenum | Table 2-3 |
| | about 23:25 | The fallen fuel debris was quenched and the hydrogen generation was slowed down<br>(Hydrogen generation in the core mostly ended by this time.) | Table 2-3 |
| | about 23:50 | Leaks started from D/W to R/B | Table 2-3 |
| March 15th | about 00:06 | Part of the fuel debris collapsed and fell into the water in the lower plenum | Table 2-3 |
| | about 01:10 | SRV(s) opening forced<br>Steam generation slowed down due to water depletion in the lower plenum | Table 2-1<br>Table 2-3 |

## 3. Conclusion

The reactor pressure changes and the containment vessel pressure changes at Unit-2 after the forced reactor depressurization were analyzed. This analysis was intended to clarify the accident progression behavior in the core and the containment vessel. To this end, the reactor and the containment vessel pressure changes measured in the accident at Unit-2 were examined. Through the examination of the accident progression based on the measured data and the analysis by the analysis code, the following findings were obtained.

### (1) In the evaluation of plant conditions relevant to the reactor pressure changes

The accident progression scenario at Unit-2 was estimated, based on the accident investigation report and plant data published by TEPCO, concerning the situation of opening/closing after the forced depressurization for the main steam safety relief valve(s) (SRV(s)), the amount and timing of hydrogen generation, the leaks from the reactor pressure vessel (RPV) or primary containment vessel (PCV), etc.

### (2) In the analysis of the reactor pressure changes

Based on the accident progression scenario estimated in (1) above, the reproduction analysis was conducted, using the thermal-hydraulic analysis code GOTHIC 8.0(QA), for the Unit-2 reactor pressure changes starting at the forced reactor depressurization at 18:00 on March 14th. Analysis conditions (the depressurization conditions and the amount of steam/hydrogen generation) were searched which could well reproduce the reactor pressure changes and the containment vessel pressure changes. The appropriateness was also shown based on the findings that those condition settings were reasonably explicable. In consequence, it was confirmed that the reactor and containment vessel pressure changes were explicable from the accident progression scenario estimated in (1) above.

### (3) In the examination of accident progression after the forced reactor depressurization

The accident progression scenario was derived by reflecting the knowledge obtained from the analysis results in (2) into the accident scenario estimated in (1). The knowledge obtained from the analysis results in (2) includes that of the accident progression scenario which could be estimated from the condition settings and that obtained from the sensitivity analysis.

In addition to the accident progression scenario estimated in Attachment 2-7 (steam generated by injecting water to the reactor by fire engines → the steam-Zr reactions releasing a large amount of energy and hydrogen, and increasing the reactor pressure → the increased reactor pressure impeding water injection by fire engines → termination of the hydrogen generation → the reactor pressure decrease), a possibility of the fuel debris contribution to the accident progression scenario was confirmed (the fuel temperatures elevated by water-metal reactions → fuel melting → the fuel debris falling to the RPV lower plenum → steam generation by evaporation of water therein → water-metal reactions). Further, the current examination showed that, if the SRV(s) had been kept open, nearly 1000kg of hydrogen could have been generated and that this amount was more than the amount obtained in the earlier evaluation.

## 4. References


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