Estimation of accident progression based on the data measured for Unit-1 and the analyses available to date

* This document is generated based on the proposal and evaluation by TEPCO Systems Corporation [1] concerning the changes of water level indicator readings (Unit-1/Issue-3) in Attachment 2 “List of issues”.

1. Introduction

Attachment 1-2 “Evaluation of plant status by the fuel range water level indicators of Unit-1” described the mechanism of water level indicators and discussed the issue of changes in fuel range water level indicator readings. This document estimates the accident progression based on, in addition to the knowledge contained in that Attachment, the tendency of other data measured for Unit-1 (reactor pressures, containment vessel (PCV) pressures), and the knowledge obtained from the analyses available to date. An analysis code was used to reproduce the data measured in the estimated accident progression scenario. The estimated accident progression scenario was shown to be able to explain the tendency of the measured data.

2. Estimation of accident progression based on the data measured for Unit-1 and the analysis results available to date

The measured data (water level indicator readings, reactor pressures, and PCV pressures) and knowledge obtained in the analyses and other activities carried out to date are used for estimating the accident progression of Unit-1. Among such data, water level indicator readings, in particular, do not represent the correct water levels due to accident effects, but they may be helpful in acquiring useful information regarding the accident progression, i.e., by estimating the water level in the water level indicator line which can reproduce the readings. This is possible, since the readings correspond to the water head in a pipe of interest.

Table 1 highlights key events of the accident progression estimated for Unit-1 on the basis of such information. The grounds for the estimation of each event and its timing therein are given below in the estimation based on the measured data (2.1), and the estimation based on the analysis results available to date (2.2).

Table 1 Timeline of key events in the accident progression estimated for Unit-1

No.	Date	Time	Event	Grounds
E1	March 11	15:37	Station blackout	Ref. [2]
E2		18:10	Core water level reached TAF	2.2
E3		18:50	Start of fuel damage, hydrogen generation, and small leakage from RPV (e.g. instrumentation dry tube failure)	2.1.2 2.2
E4		19:40	Core water level reached BAF	2.2
E5		20:00 - 21:00	Major lekage from RPV (e.g. main steam line failure)	2.1.1 2.2
E6		21:00 - 22:20	Reactor pressure decreased to PCV pressure	2.1.1 2.2
E7		23:24 - 00:30	Molten fuel debris relocation to the lower plenum started (small scale)	2.2
E8	March 12	01:05 - 02:30	Molten fuel debris relocation to the lower plenum started (large scale)	2.1.2
E9		Around 04:00	Freshwater injection started	Ref. [2]
E10		Around 06:00	RPV lower head failure	2.1.1

2.1. Estimation based on the measured data

2.1.1. Estimation based on the fuel range water level indicator readings

Figure 1 shows the fuel range water level indicator readings from the station blackout till about 24 hours later. The water level indicator (Channel A) gave its reading of TAF+0.2m at 21:19 on March 11th, but by that time no water was being injected into the reactor which means the actual water level in the reactor was considered to have dropped below BAF by then. That is, the reading overstated the actual water level. Further, it should be noted that the readings were increasing at that time. But the only possible reason for increasing readings with no water injection into the reactor would be a decrease of the water level in the piping on the reference water chamber side (reference leg). From these deliberations the water level in the reference leg was considered to have largely dropped by this time and that it was further decreasing.

At around 21:30 the increase in the readings became slower and eventually stopped at 22:20 and stayed constant at TAF+0.59m till 23:24. In other words, changes in conditions to cause the water level decrease in the reference leg probably ceased by around 21:00 to 22:20 (a ground for Event E6 in Table 1). Thereafter at 00:30 on March 12th, the reading rose to TAF+1.3m and stayed there till around 06:00. The water level indicator readings at this time coincided roughly with the value for the situation in which the water level in the reference leg drops to the elevation of the PCV penetration, while the piping on the reactor side (variable leg) is almost completely filled and the reactor water level is below the connection part of the variable leg. Thereafter, the readings were gradually lowered to TAF-1.7m by around 12:00 and stayed there. The reading at this time coincided roughly with the value for the situation in which the water in the variable leg is completely lost by evaporation. Meanwhile, the PCV temperature measured on March 21st was about 400 deg C [2]. At this temperature the water in the water level indicators was likely to have completely evaporated. In addition, the water level indicator readings were almost constant by that time. Putting these together, it would mean that no water was left in the water level indicator piping by around 12:00 on March 12th. On the other hand, the water level indicator (Channel B) had been working since 02:30 on March 12th and its reading stayed roughly constant at about TAF+0.5m till around 06:00. Its reading was about 0.8m lower than that of Channel A. This might mean that the Channel B reference leg still had water to about an 80cm height at this time. Further it could be considered that, after 06:00 on March 12th the water in the variable leg evaporated as in Channel A and that by around 12:00 water in the variable leg as well as in the reference leg was completely lost.

If the water in the piping of water level indicators was lost by evaporation, it might mean that the water reached its saturation temperature because of the increased PCV temperature. Possible causes of the PCV temperature increase are: (a) heat transfer from the reactor pressure vessel (RPV) to the drywell (D/W); (b) coolant leaks; or (c) molten debris leaks.

(a) Heat transfer from the RPV is limited because its surface is covered by heat insulating material. Therefore, such heat transfer is unlikely to elevate the PCV temperature rapidly in a limited time.

(b) If the coolant had leaked from the RPV to the D/W, it would increase the PCV temperature and at the same time the water saturation temperature would drop due to the reactor depressurization. The water in the water level indicators would evaporate more easily. This might be a ground for the water level decrease in the reference leg at around 21:00 on March 11th (Event E5 in Table 1).

(c) If the molten debris had leaked from the RPV, heat transfer from the hot debris would directly elevate the PCV ambient temperatures in a limited time and the water in the water level indicators would evaporate. This might be a ground for the decreasing readings of water level indicators from around 06:00 on March 12th. To sum up, it was estimated that at around 06:00 on March 12th the RPV lower head failed and the molten debris relocated from the RPV to the PCV (Event E10 in Table 1).

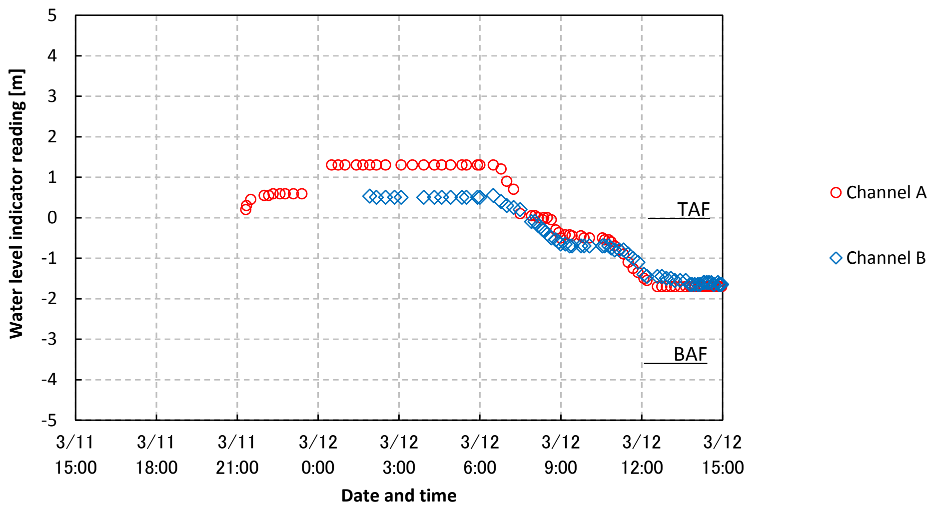


Figure 1 Fuel range water level indicator readings

2.1.2. Estimation based on the reactor pressure and PCV pressure

Figure 2 shows the reactor pressures and PCV pressures from the station blackout till about 24 hours later. The measured reactor pressures were 7MPa[abs] at 20:07 on March 11th and 0.9MPa[abs] at 02:45 on March 12th. After the station blackout, the safety function of the safety relief valves (SRV) worked. The working pressure was about 7.7MPa[abs] [9]. The reactor pressure as of 20:07 on March 11th was likely to be slightly below the minimum value of the SRV working pressures. This might indicate the possibility that at this time the RPV was already leaking (Event E3 in Table 1). As possible leakage causes, in-core instrumentation line damage or line melting and main steam line damage have been suggested [3][4][5][6].

The measured PCV pressures were 0.6MPa[abs] at 23:50 on March 11th and 01:05 on March 12th. Such pressures cannot be reached without leakage from the RPV to the D/W, as explained below in Section 2.2, i.e., leakage from the RPV to the D/W seems likely to have occurred by 23:50 on March 11th. Later at 02:30 on March 12th the measured PCV pressure was 0.84MPa[abs], which roughly balanced with the RPV pressure of 0.9MPa[abs] at 02:45 on March 12th. Thereafter, the PCV pressure gradually decreased and then to increase at around 06:00 on March 12th. In the current estimation, the RPV lower head failure was considered to have occurred at around 06:00 on March 12th, and therefore the PCV pressure increase at this time was considered to be caused by the relocation of molten debris to the PCV in the wake of RPV damage. In other words, the pressure increase prior to this time point was estimated to be caused by some other reasons, and the pressure increase at this particular period from 01:05 to 02:30 on March 12th was estimated to be caused by the large amount of molten debris relocation to the lower plenum (Event E8 in Table 1).

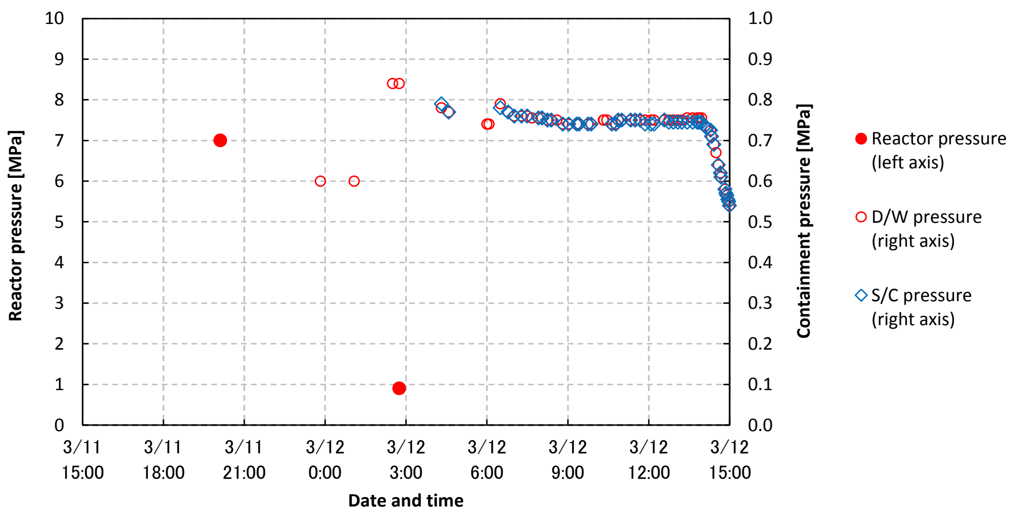


Figure 2 Measured reactor and PCV pressures

2.2. Estimation based on the analysis results available to date

The reactor level decrease can be fairly easily predicted by analysis, since the phenomenon simply depends on the residual decay heat and water inventory in the system. The results from analytical codes are generally reliable. The MAAP analysis could reproduce well the water level measured at 16:42 to 16:56 on March 11th [3] and it predicted a time around 18:10 on March 11th as the timing of the reactor water level to reach TAF, and around 19:40 on March 11th as the timing of the water level to reach BAF. In the current estimation, the MAAP prediction of the reactor water level changes till 19:40 on March 11th was used (Events E2 and E4 in Table 1).

According to the MAAP results [3], the core temperature starts to rise as soon as the reactor water level cuts the TAF level and it reaches about 1000 deg K at around 18: 50 starting to damage the fuel. Under such conditions, gaseous leaks may occur to the PCV bottom through the in-core instrumentation line. Meanwhile, the flange gaskets of the main steam line are said to possibly have lost their seal performance at about 450 deg C. The MAAP analysis predicts that the temperature reached this condition by around 20:00 to 21:00 [3][4][5]. With this background, relatively early gaseous phase leaks were assumed in the analysis from the RPV to the D/W. If the safety relief valves were stuck open, steam would move to the suppression chamber (S/C). This possibility is negated, though, in the analysis because in this case the D/W pressure at 23:50 on March 11th could not be reproduced [3][4][5][6].

In the current estimation, it was assumed that the core melt started at 18:50 on March 11th and a gaseous leak started at the same time to the PCV pedestal through the in-core instrumentation line (Event E3 in Table 1). Here, it was assumed that the scale of the leak had been limited, and the reactor pressure had been kept by the safety valve function of the SRVs until the reactor water level had dropped to BAF at 19:40 on March 11th, when the SRVs had been closed. From that time till 20:07 (about 30 minutes), the leak rate was assumed to be limited in volume to lower the reactor pressure from 7.5MPa[abs] (the average pressures of the SRVs while their safety function was working) down to 7.0MPa[abs].

As discussed later, the leak rate of this scale is not sufficient to raise the PCV temperature significantly. It is not sufficient either to let all water inventories in the water level indicator pipes evaporate. Other assumptions are necessary to interpret the behavior of water level indicator readings. It was assumed, therefore, in the current estimation that gaseous leaks had started at 20:00 to 21:00 on March 11th to the D/W from the main steam line (Event E5 in Table 1), by considering the afore-mentioned analysis results. Concerning the time span of leaks, it was assumed that the RPV and PCV pressures had been balanced by around 21:00 till 22:20, by when leaks of a sizable quantity were assumed to have ceased, since the reference leg water level decrease was considered to have almost terminated by that time (Event 6 in Table 1).

There are big uncertainties in the amount of hydrogen generated in the core, but the quantities have been estimated by various organizations as approximately 400 to 800kg [3][4][5][6]. In the current estimation, the quantity of generated hydrogen was assumed as 800kg. The hydrogen generation was assumed to have started at 18:50, the same as the time when the core damage had started (Event E3 in Table 1).

Concerning the timing when the molten debris started to relocate to the lower plenum or when the RPV lower head failed, different results have been reported for different analysis codes or different organizations carrying out the analysis. For example, as the timing for debris relocation to the lower plenum, one analysis predicted a large amount of debris relocated at around 22:00 to the lower plenum [3], while another analysis predicted only a limited amount of debris relcocated at around 20:00 and later at around 23:00 a larger amount relocated [5], or about one-third of the debris relocated at around 21:00 and the rest at around 23:00 [6], or a large amount of debris relocated at around 01:00 to 02:00 on March 12th [4][7], etc. Variations have also been found in the RPV lower head failure timing including occurring: at around 01:00 to 03:00 on March 12th [3][6][8], at around 05:00 to 06:00 on March 12th[4][5], at around 12:00 on March 12th[7], etc.

These disparities in the analysis results likely come from the complexity of the phenomena such as debris relocation to the lower plenum or the RPV lower head failure, and hence big uncertainties exist in the analytical models. There are certainly different modelings between analytical codes in allocating heat among structural parts or in damage conditions of structures, but all these analytical codes assume the same decay heat [2]. The timing of a large amount of debris relocation and the timing of the RPV lower head failure, which were assumed in the current estimation, lie in the range of values predicted by such existing analytical results, and therefore the results obtained in the current estimation can be considered as realistic.

Next, the deliberation turned to the possible cause of the increasing readings of water level indicator from 23:24 on March 11th to 00:30 on March 12th. It was estimated by this time that the gaseous leakage from the RPV to the D/W had almost ceased and the debris relocation in a large amount had taken place after 01:05 on March 12th. As a cause of the water level decrease in the reference leg of the water level indicator under such conditions, a limited amount of debris relocation was assumed to have occurred to the lower plenum (Event E7 in Table 1). Here, the debris relocation was assumed in a “limited” amount, because the amount of coolant leakage during the period from 23:50 on March 11th to 01:05 on March 12th was assumed to be limited, since the measured PCV pressure was 0.6MPa[abs] at both times.

3. Reproduction analysis of Unit-1 water level indicator readings based on the estimated accident progression

The reproduction analysis of Unit-1 water level indicator readings was conducted in order to check the appropriateness of accident progression assumed in Section 2 above. For simulating the water level changes in the water level indicator line, the changes of reactor pressures and local temperatures in the PCV need to be evaluated. In the current estimation, these conditions were set as boundary conditions based on the assumptions. Using these conditions, the thermal-hydraulic analysis code GOTHIC [10][11] was used to evaluate temperature changes in the PCV and water level behavior in the water level indicator line.

3.1. Geometry for analysis

Figure 3 shows the geometry for analysis[1]. The water level indictor line and PCV were modeled in several regions, flow paths (arrows) and heat structures (colored). Each heat structure exchanges heat with the adjacent region. The core internals in the RPV were not modeled. The temperature boundaries were the RPV inner wall and the space in the reactor building.

The water level indicator line was modeled in the PCV space as several regions and heat structures in the geometry, so that the water level changes therein due to evaporation could be simulated. The configuration was defined based on the design data. The PCV space was further divided as illustrated within the dashed lines in the figure in order to simulate temperature distribution and natural convection around the water level indicator line. This enabled simulation of the PCV temperatures as a function of elevation, etc. Although not illustrated in the figure, the regions were divided also along the circumferential direction (the direction into the page), and thus reference legs of Channel A and Channel B were separately modeled in different regions along the circumferential direction. This enabled simulation of the behavior difference in Channel A and Channel B water level indicator lines. Variable legs were modeled to the same for both channels, for simplicity.

The star marks in the figure represent the leak locations assumed in the current estimation. The leak location from the instrumentation line to pedestal corresponded to the leak in a limited amount from RPV in Table 1 (Event E3). As the location of the main leak

[1] The “Reference leg” in Figure 3 means the piping on the reference water level side, while “Variable leg” means the piping on the reactor water level side.

from the RPV, the upper side of the main steam line was assumed in the base case below in the current estimation. For comparison, the leak was assumed at the SRV position as well. It should be noted that the main steam line was not modeled in the analysis, but it is illustrated in the figure in order to help visualize positional relations in the simulation.

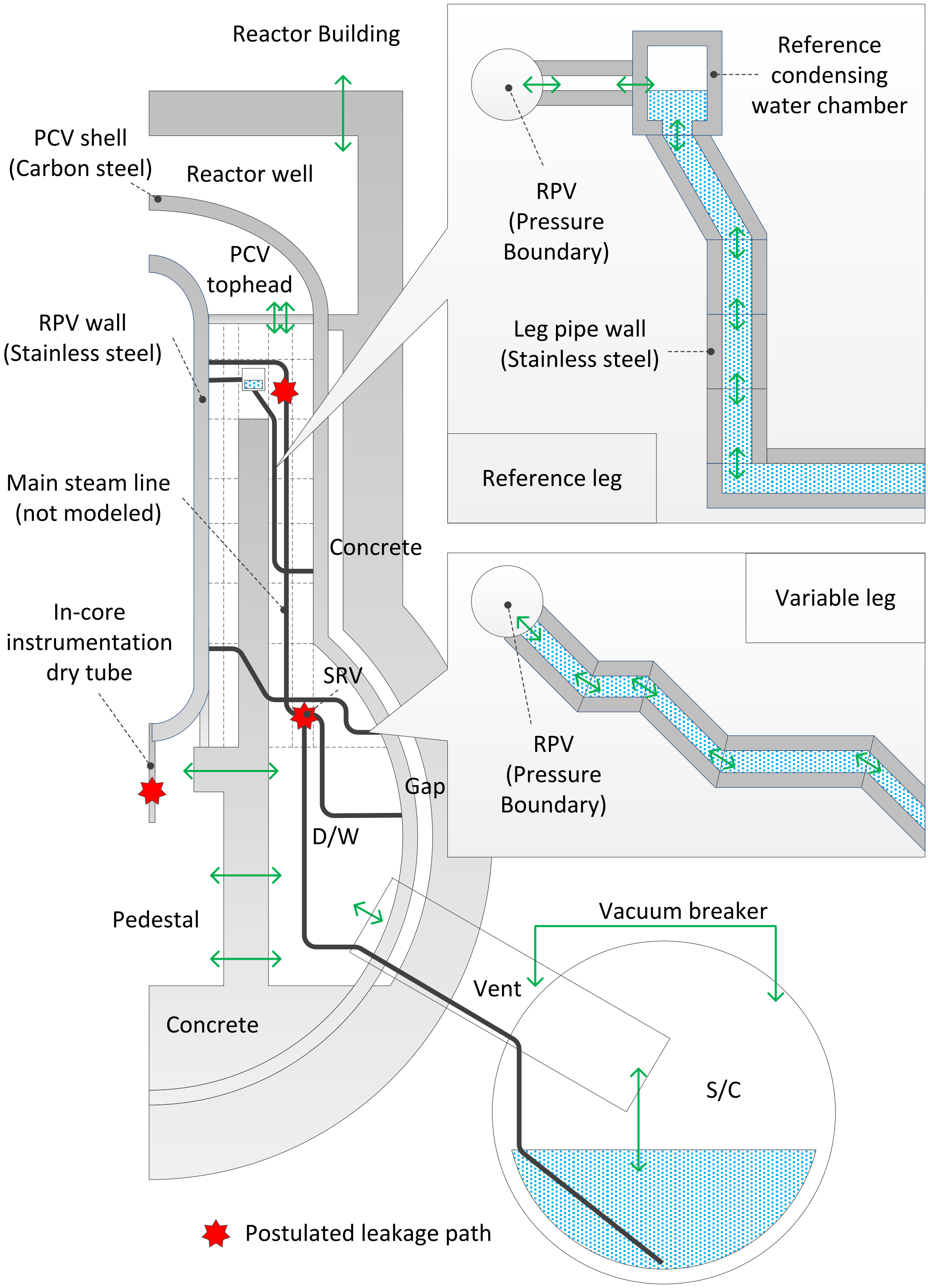


Figure 3 Geometry for analysis

3.2. Conditions for analysis

The analysis was carried out from the time of station blackout till 14:00 on March 12th. Table 2 gives the initial temperatures set, while Table 3 shows material properties of heat structures. The steam tables built into the code were used in the analysis.

The pressure boundary for the reference legs was set as the steam-hydrogen mixture. This was based on the estimate that, when the water level in the reference legs was rapidly decreasing (20:30 to 22:20 on March 11th), hydrogen had been being generated. The pressure boundary for the variable legs was switched from water to steam subject to the reactor water level postulated.

Table 4 gives boundary conditions (relevant to the changes of reactor pressures and PCV temperatures). These conditions are based on the estimated accident progression. The process and grounds for these settings are given at the end of this document for reference. It should be noted that the water injection into the reactor after 04:00 on March 12th was not considered in the analysis, since it can be considered to have little effect on the accident progression. Furthermore, gaseous leaks from the PCV were not considered either over the analysis time period.

Table 2 Initial temperatures set

Region	Values	Note
D/W	50 deg C	Structures and water in water level indicators, too
S/C	30 deg C	S/C water temperatures, too
Reactor well	50 deg C	Same as D/W
Reactor building	25 deg C	No temperature changes assumed over the time period

Table 3 Properties of heat structures

Material	Density	Heat conductivity	Specific heat
Stainless steel	7920 kg/m^3	16 W/m-K	0.499 kJ/kg-K
Concrete	2400 kg/m^3	1.2 W/m-K	0.9 kJ/kg-K
Carbon steel	7850 kg/m^3	51.5 W/m-K	0.473 kJ/kg-K

Table 4 Boundary conditions

Boundary conditions	Values
Reactor pressures	Reference Figure 1
RPV inner wall temperatures	Reference Figure 2
Gas flows from SRV to S/C	Reference Figure 3
Leaked gas flow from in-core instrumentation line to pedestal	Reference Figure 4

Leaked gas flow from main steam line to D/W	Reference Figure 5
Heat transfer to PCV atmosphere from molten debris fallen	About 15% of decay heat

3.3. Analysis results

The following results were obtained in the reproduction analysis as the water level indicator readings based on the geometry and conditions for the analysis set in Sections 3.1 and 3.2. Section 3.3.1 presents the base case results, in which the main leak from the RPV was assumed to have taken place at the upper side of the main steam line, while Section 3.3.2 presents the sensitivity analysis results assuming the leak location at the SRV position.

3.3.1. Analysis results of base case

Figure 4 presents the analysis results of temperature changes in the atmosphere surrounding the pedestal, the upper part of the D/W (near the reference legs of Channel A and Channel B), and the central part of the D/W (near the variable legs)[2]. The temperature increase is relatively gradual until 18:50 on March 11th, when a leak through the in-core instrumentation line was assumed. Further, even after the leak to the pedestal started, the temperature increase in the PCV atmosphere is at most about 100 deg C, not significantly high enough to evaporate water in the water level indicator line. This is because it is hard to increase the atmosphere temperature due to the low heat radiation from the RPV, the limited gaseous leaks to the pedestal, and the large heat capacity of heat structures in the PCV. Thereafter, upon initiation of large quantity, high temperature steam leaks from the main steam line at 20:30, all the PCV temperatures increase quickly. As the leaked amount decreases with time, the atmosphere temperature decreases because heat is absorbed by heat structures therein. Similar temperature changes of a spike-shape increase followed by a decrease are also noticed at 23:30 (March 11th) at the time of debris relocation in a limited amount and 01:30 on March 12th at the time of debris relocation in a big amount. Thereafter at 06:00 on March 12th, when the molten debris relocated because the RPV lower head had failed, heat from the molten debris is directly transferred to the atmosphere and it continues to increase the PCV temperatures.

Upon occurrence of leaks from the main steam line, the peak temperature of Channel A reaches a higher value than that of Channel B. The actual value of their difference is subject to the leak position. The position assumed in the current estimation was the upper side of the main steam line and it was close to the condensing water chamber of Channel B. In other words, in the analysis the leak above the Channel B piping raises the Channel A piping

[2] The “LP relocation” in Figure 4 illustrates the debris relocation to the lower plenum.

temperature. This is because the high temperature leaked gas causes upward and horizontal flows of gas above Channel B piping and these gas flows raise the temperature around Channel A piping. Figure 5 shows the temperature distribution in the gas phase in the D/W at 20:35 on March 11th, when the temperatures in the gas phase in the D/W peaked. The area around the variable leg is located at a fairly lower elevation than the leak position and therefore its peak temperature becomes lower than that around the reference leg, because the high temperature leaked gas loses some heat due to the surrounding structures before flowing down to the area around the variable leg.

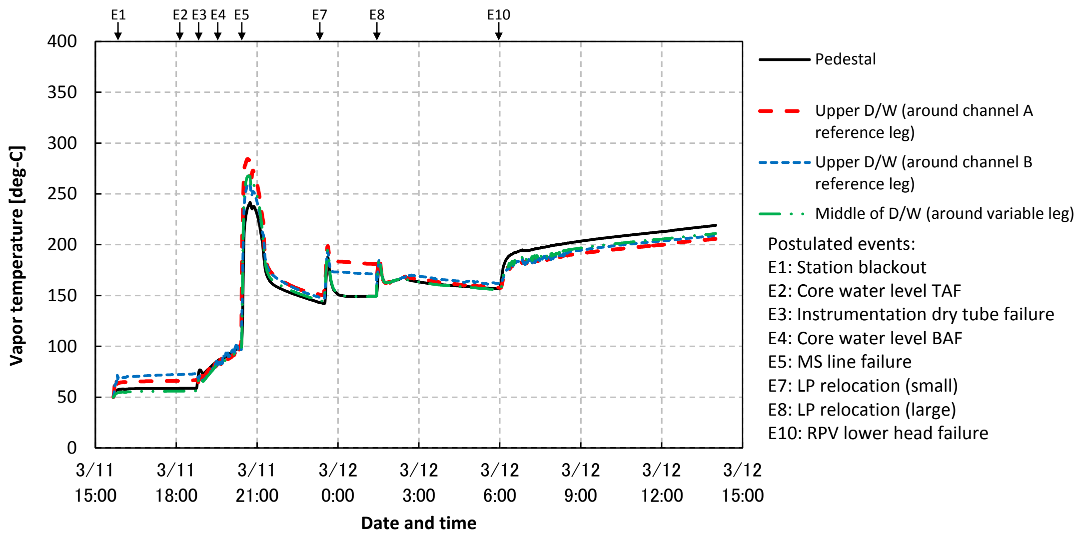


Figure 4 Temperature changes in the D/W (base case)

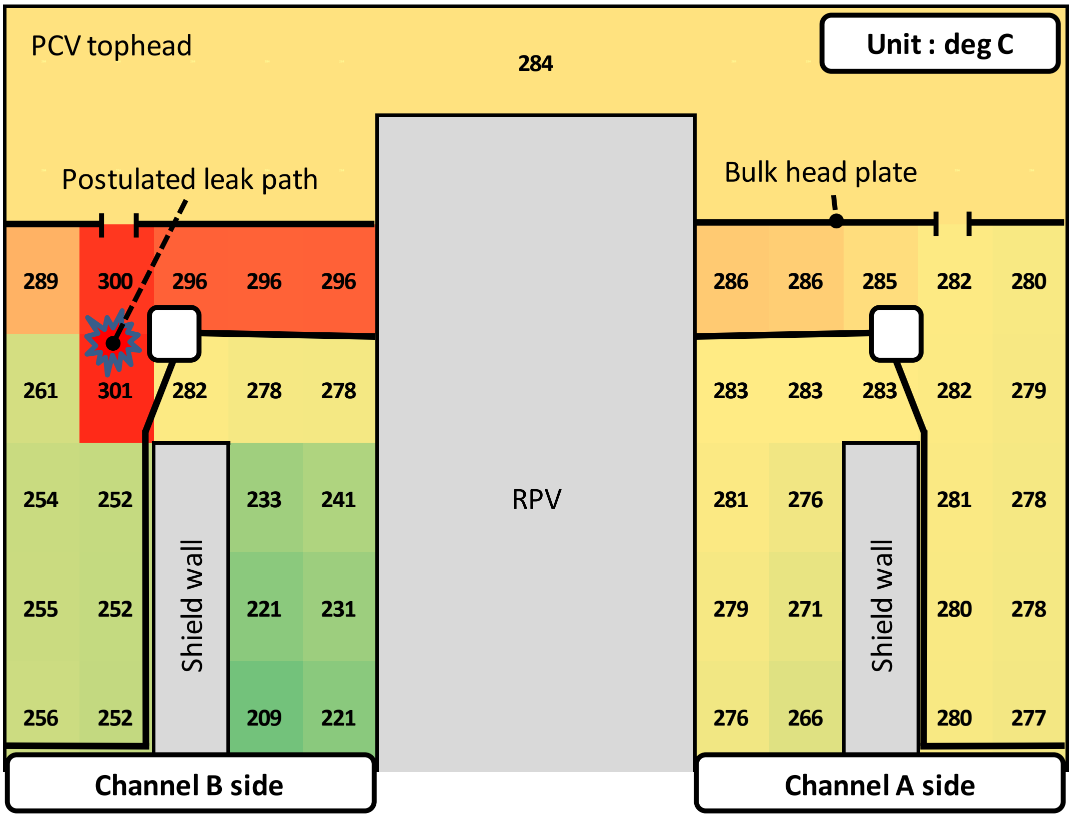


Figure 5 Temperature profile in the gas phase in the D/W
at 20:35 on March 11^{th} (base case)

Figure 6 presents the postulated reactor water levels and the analyzed water level changes in the piping of reference and variable legs. The reactor water level is assumed based on the estimated accident progression. The reactor water level is assumed to reach approximately the elevation of the variable leg inlet at the timing when the RPV depressurization ended and it maintains this level until 23:30 on March 11^{th} when the debris relocation took place in a limited amount. The reactor water level changes estimated now are used to evaluate the water level indicator readings.

The PCV atmosphere temperature increase raises the water temperature in the water level indicator lines. When this water temperature exceeds the saturation temperature, which was lowered by reactor depressurization, the water in the reference leg starts to evaporate. The temperature of the whole reference leg rises, and water levels decrease drastically by about 22:00 due to evaporation at its various parts. The water level further decreases thereafter due to the debris relocation to the lower plenum two times. Calculated water level decrease of Channel B is less than that of Channel A, because the ambient temperature around Channel B is relatively lower than that of Channel A (Figure 5). At some of the timings, the water level shows a spike-shaped increase. This is because the volume fraction of gaseous phase in the horizontal portion of the piping increases due to evaporation of water therein and this raises the water level in the upper portion of the piping temporarily.

On the other hand, the variable leg remains almost completely filled with water until about 06:00 on March 12th when the RPV lower head failure was assumed and then the variable leg water level gradually decreases until all its water is lost at around 12:00. The water level drops a few times after 21:00 on March 11th. This is due to a temporary temperature increase because of the gaseous leak from the main steam line. The water level in the variable leg recovers thereafter. This is because the temperature increase due to leaked gas is limited in time, as seen in Figure 4, the ambient temperature around the variable leg piping falls and this condenses the steam coming from the pressure boundary.

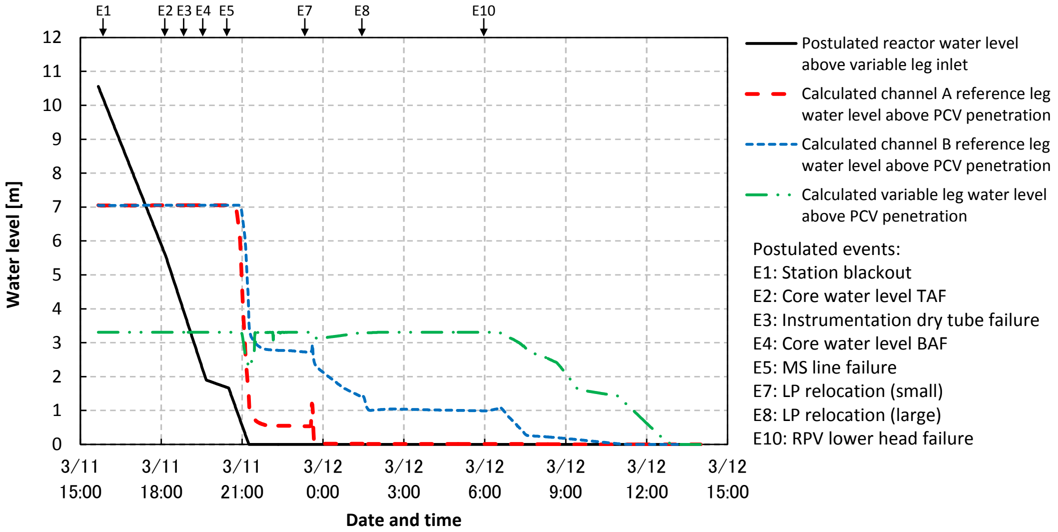


Figure 6 Water levels in the core and water level indicator piping (base case)

The water level changes in Figure 6 are converted to the water level indicator readings in Figure 7 and compared with the measured values. It should be noted that the fuel range water level indicators are calibrated to the correct water level at room temperature andatmospheric pressure, and therefore its reading is different from the real water level even if the water level indicator line is filled with water. The results obtained from the analysis, based on boundary conditions set from the accident progression assumed in Table 1, can reproduce approximately the general trend of measurement results including the difference in readings between Channel A and Channel B. The decreasing trend of readings becomes gradual at around 09:00 to 11:00 on March 12th. This time period corresponds to the timing when the water level in the variable leg was at the level of the variable leg horizontal portion. In the analysis, by modeling this horizontal portion of the water level indicator line, this gradual transition of water level indicator readings can be reproduced.

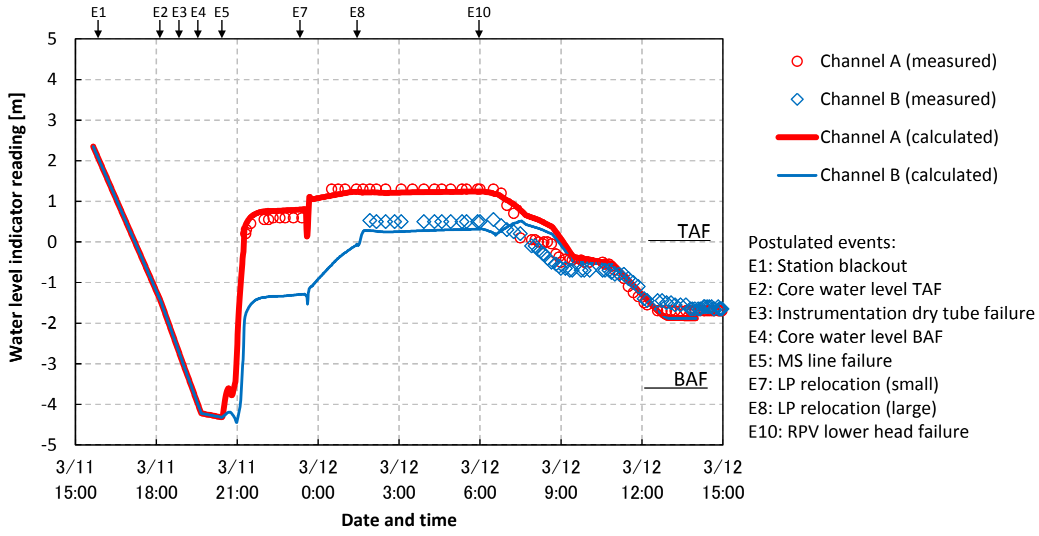


Figure 7 Comparison between water level indicator readings and water levels measured (base case)

Figure 8 compares the analysis results and measured values of D/W pressures and S/C pressures. In the analysis, the PCV pressures show a large increase until 21:15 on March 11th mainly due to hydrogen discharge to the S/C and steam/hydrogen mixture discharge from the main steam line. Thereafter, the pressures decrease in accordance with the ambient temperature (Figure 4). It increases again due to steam generated by the debris relocation (assumed as having occurred at 23:30) in a limited amount to the lower plenum, and then the pressures remain roughy constant due to stable steam leaks by decay heat. The pressures quickly increase after the debris relocation in a large amount to the lower plenum assumed as having occurred at 01:30 on March 12th. The pressures decrease gradually thereafter in accordance with the temperature decrease, and they turn upward due to the temperature increase of the PCV upon the debris relocation at around 06:00.

The comparison in Figure 8 shows the pressures of 0.6MPa[abs] observed at 23:50 on March 11th and 01:05 on March 12th are well reproduced in the analysis. The pressures over this time span remain roughly constant because the ambient temperature in the PCV in this period remained roughly constant (see Figure 4). This comes from the balanced heat supplied from the leaked steam generated by decay heat and heat loss absorbed by surrounding structures in the PCV. The pressure increases sharply at 01:30 on March 12th, when the debris relocation in a large amount was assumed, but it does not reach 0.84MPa[abs] measured at 02:30. In the current analysis, no hydrogen generation was assumed upon the debris relocation to the lower plenum. This may have led to underestimation of the containment pressures. In another existing analysis, hydrogen was newly generated upon debris relocation [5].

The observed pressures show an increase, by about 0.05MPa, from 06:00 to 06:30, but the pressure increase by the analysis during that time is less. If the RPV lower head was damaged and molten debris flowed out at this timing, non-condensable gas generated by the molten debris – concrete interaction (MCCI) would increase the pressure. The current analysis did not simulate the gas generation by MCCI, and this may have caused underestimation of the pressure increase. Thereafter the observed pressures remain roughly constant, while the analysis results show an increasing trend. This is probably because no leaks from the PCV were assumed in the analysis.

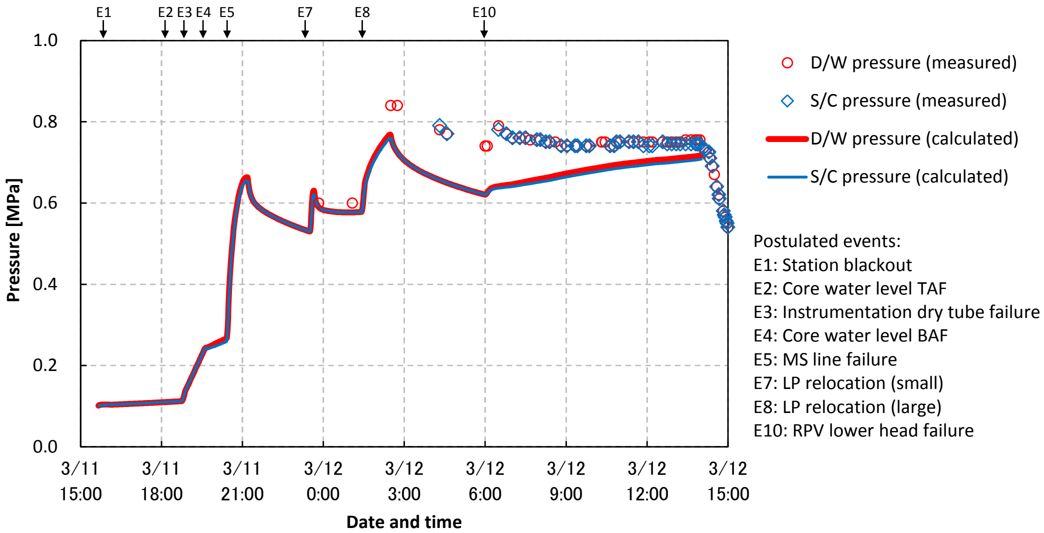


Figure 8 Comparisons between the analysis results and measured values of containment vessel pressures (base case)

3.3.2. Results of sensitivity analysis

A sensitivity analysis, assuming a leak at the SRV position, was conducted in order to check the dependence of analysis results on the leak location. As the leaked gas temperature from SRV to D/W, 620 deg C was setas a value which could reproduce well the reactor water level indicator readings of Channel A up to the time of 22:20 on March 11th. Figure 9 presents the analysis results of water level indicator readings. After the debris relocation to the lower plenum, assumed at 23:30 the water level in the variable leg falls and the analysis does not reproduce the measured levels. This can be explained by the geometric configuration of the equipment involved. As illustrated in Figure 3, the SRV is located close to the variable leg. The leaked gas raises the variable leg temperature quickly, evaporating the water therein. Another point to note is that the water level of Channel B decreases faster than that of Channel A. The reference leg of Channel B has a routing path of about 3m in the circumferential direction near the PCV penetration. The water in this portion of piping evaporates at high temperatures, causing a faster level decrease in the upper portion. This is also different from the data observed. Such difference is not seen in the base case, in which the leak position was located high in the PCV. These findings suggest that the main leak position from the RPV to D/W will be, if based on the estimated accident progression scenario, somewhere in the upper part of the PCV.

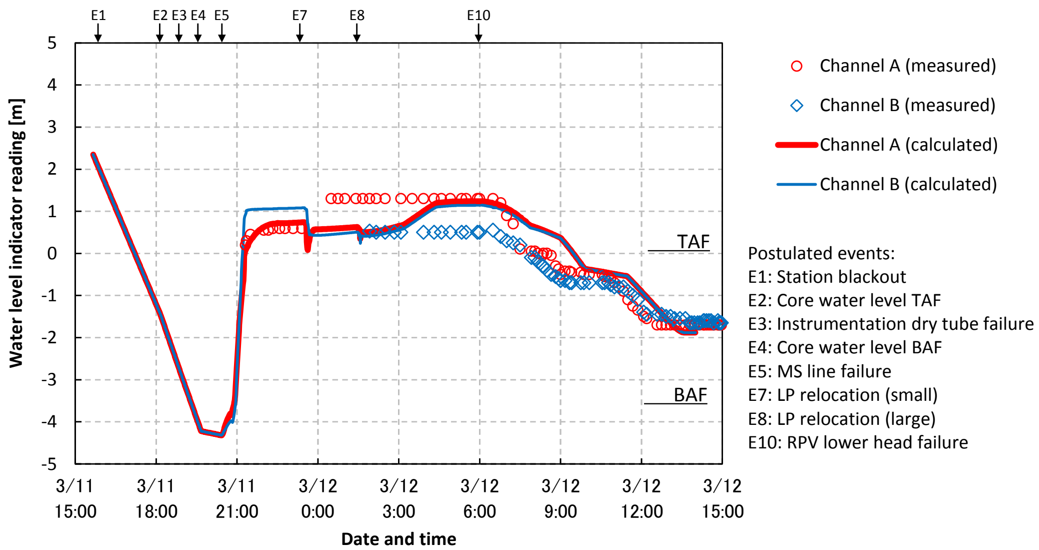


Figure 9 Comparison between the analysis results and measured values of water level indicator readings (sensitivity analysis case)

4. Conclusion

From the discussions above, it has been confirmed that the estimated accident progression scenario (Table 1) can explain the changes of measured data, including that of water level indicator readings. In addition, the possibility has been indicated that the main leaks from the RPV to the D/W took place in the upper part of the PCV.

(End)

(Supplement) Setting of boundary conditions for analysis

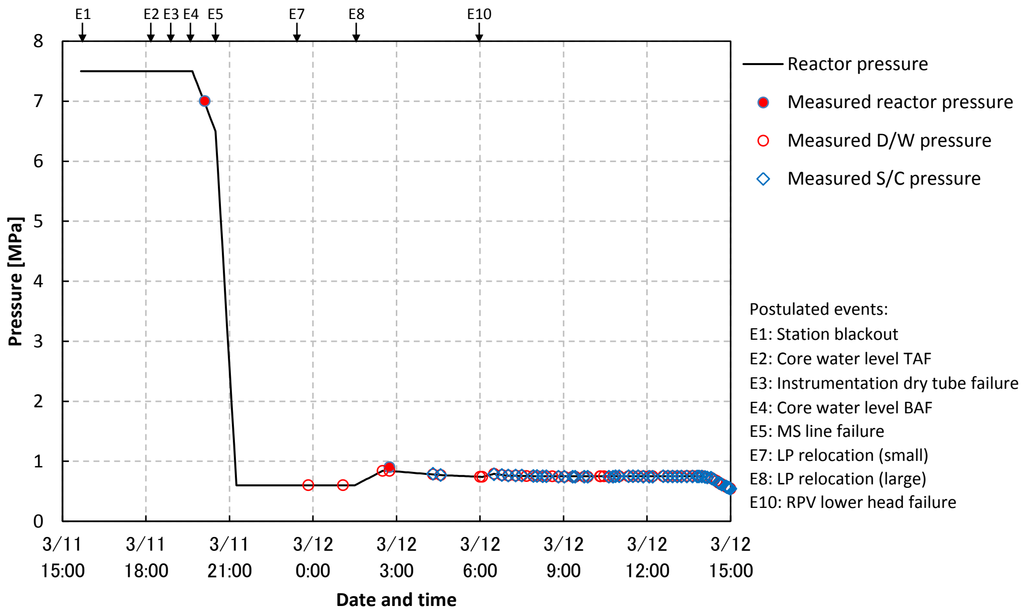
The setting procedures of boundary conditions (conditions relevant to reactor pressure changes and PCV temperature changes) are described in the following; they are based on the estimated accident progression scenario presented in Table 1. The term “preliminary analysis” used here refers to a series of parametric surveys in which the best values to reproduce measured data are sought for, by changing the parameters within the presumable range.

a. Reactor pressures

Reference Figure 1 explains how the reactor pressures were set in the current analysis. The reactor pressures from the time of station blackout till 19:40 on March 11th, when the reactor water level was lowered to the level BAF, were set as constant at 7.5MPa[abs] for simplicity by assuming that the reactor pressure was maintained at the pressure setpoint of the SRV safety function. The reactor pressure was assumed to decrease gradually to about 7MPa[abs], the measured pressure at 20:07, because, once the water level reached BAF, evaporation in the core was reduced and leakage through the instrumentation line took place. It was further assumed that a leak had occurred at 20:30 from the main steam line and, by 21:15, the pressure had decreased to 0.6MPa[abs] (the PCV pressure measured at 23:50). Thereafter, it was assumed that the pressure had shown only a small change for a while followed by an increase to 0.84MPa[abs] by 02:30 on March 12th as a result of debris relocation to the lower plenum in a large amount at 01:30, and then it accompanied the PCV pressure. The timings of the leak from the main steam line, ending of depressurization and debris relocation in a large amount to the lower plenum, respectively in the above transition, were decided through the preliminary analysis.

Some other analyses have reported that, upon the debris relocation in a large amount to the lower plenum, the reactor pressure showed a sharp increase [3][4][6][7]. But the current analysis did not consider such a pressure increase because of its large uncertainties[3].

[3] It should be noted that the current analysis may overestimate the water decrease in the water level indicator line if such a significant pressure increase occurs upon debris relocation to the lower plenum, because the saturation temperature increases and evaporation of the water in the water level indicator line is suppressed.



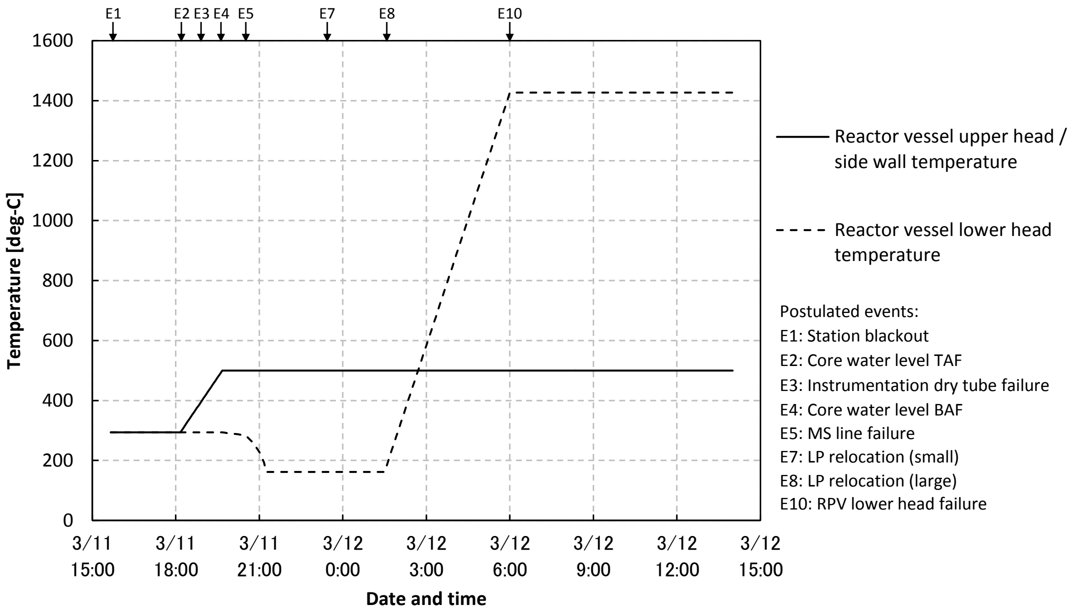
Reference Figure 1 Setting of reactor pressure changes

b. PCV temperature changes

In the current evaluation, heat transfer from the RPV to D/W, coolant leaks and debris leaks have been considered as the factors causing PCV temperature changes.

b.1. Heat transfer from RPV to D/W

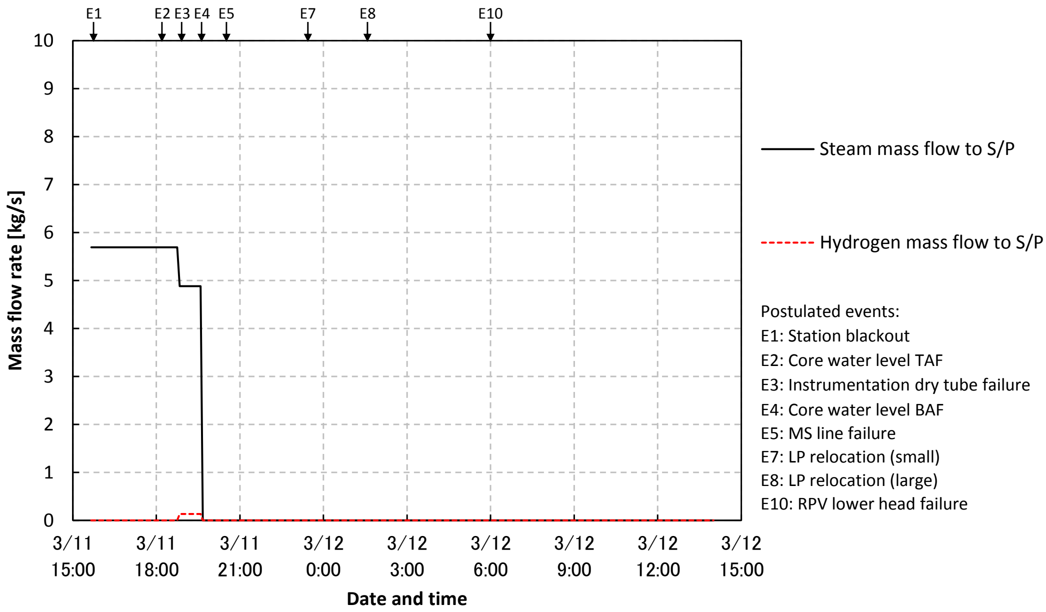
Upon the station blackout, the D/W coolers stopped and the D/W temperatures increased gradually by heat being transferred through the heat insulation material between the RPV wall and D/W atmosphere. Reference Figure 2 shows the RPV inner wall temperatures set in the current analysis. Until 18:10 on March 11th, the reactor water level was above TAF. Therefore, the RPV inner wall temperatures were set as the saturation temperatures corresponding to the RPV pressures. Thereafter, the inner wall temperatures above the reactor water level would increase due to superheated steam or heat radiation from the core. In the current evaluation, reference was made to the MAAP analysis results [3][4] and the RPV side wall temperatures after 19:40 above the lower plenum were assumed as 500 deg C, while from 18:10 to 19:40 they were linearly interpolated. Concerning the lower plenum inner wall, its temperatures were set as the saturation temperatures corresponding to the RPV pressures set above, until 01:30 on March 12th, when the debris relocation in a large amount was estimated to have occurred. After that, the lower plenum temperatures were considered to increase due to the molten debris leaked in. Reference was made to the analysis results [5], and the lower plenum temperatures were assumed to increase linearly up to 1700 deg K, the melting temperature of stainless steel, till 06:00, when the RPV lower head failure was assumed, and they remained at that temperature thereafter. The heat transfer coefficients of insulation materials were estimated from the heat transfer from the reactor under the normal reactor operation conditions.



Reference Figure 2 RPV inner wall temperatures

b.2. Gas flow from SRV to S/C

Reference Figure 3 illustrates gas flow rates to the S/C through the SRV that were set in the analysis. It was assumed that the water inventory between BAF and the level at the station blackout (close to the level at normal operations) had flowed out as steam at a constant flow rate over about a 4 hour period to 19:40 on March 11th. The leakage rate through the instrumentation line assumed after 18:50 (described later) was subtracted from the steam flow rate above. Hydrogen gas generation was assumed to have started at 18:50 and 400kg of hydrogen gas, half of the total hydrogen gas assumed as having been generated, was assumed to have flowed out at a constant rate through the SRV during 50 minutes till 19:40. This assumption of 50% release of the total hydrogen gas generated was based on the MAAP analysis finding [3][4] that about 60% of the total hydrogen gas was generated before depressurization and the remaining 40% was generated during it. The discharged gas temperatures were assumed to be the saturation temperatures at the corresponding pressures. It should be noted that the temperatures of gas discharged to the S/C have little influence on the analysis of water level in the water level indicator lines.



Reference Figure 3 Gas flow rates from SRV

b.3. Gas leaks to the pedestal through in-core instrumentation line

Reference Figure 4 illustrates the mass leak rates to the pedestal through the instrumentation line that were set in the analysis. The leak through this line was assumed only as steam and hydrogen gas was assumed to flow out only through the SRV. The steam mass generated upon depressurization was obtained in the following equation.

$$M = V \rho x \quad (1)$$

$$x = \frac{h_f(P_1) - h_f(P_2)}{h_g(P_2) - h_f(P_2)} \quad (2)$$

Here;

M is the amount of evaporation [kg];

V is the amount of water in the RPV (including in the downcomer and recirculation line) [m^3];

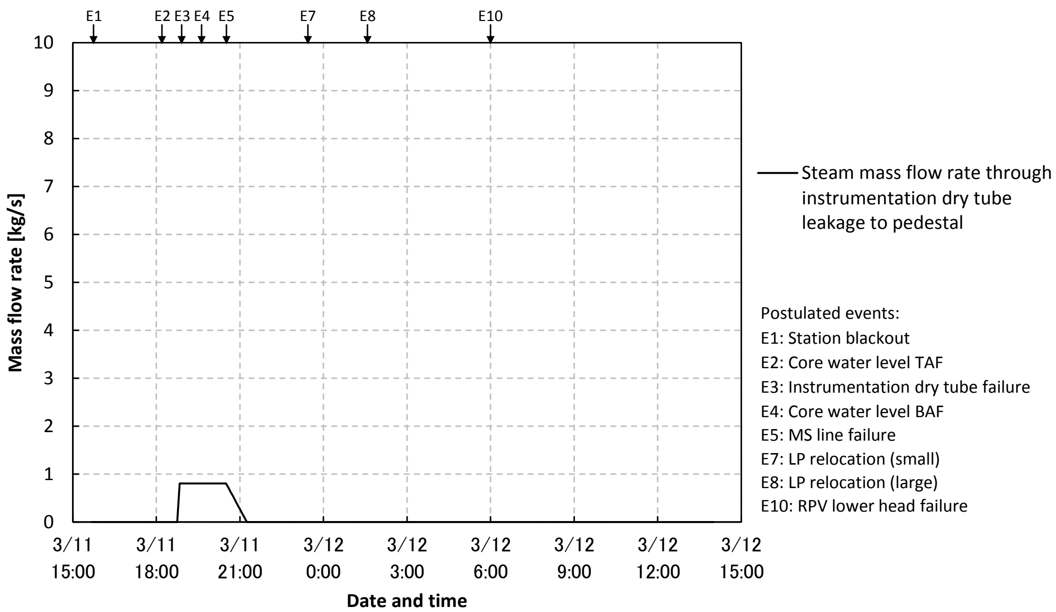
ρ is the average water density during depressurization [kg/m^3];

x is the evaporation rate [-];

hg, hf are saturated steam enthalpy and saturated water enthalpy [J/kg], respectively; and

P_1, P_2 are the pressures before and after depressurization [Pa], respectively.

The SRV was assumed to have closed at 19:40 on March 11th, when the reactor water level was estimated to have decreased to BAF, and then steam was assumed to have leaked through the instrumentation line, depressurizing the reactor from 7.5MPa[abs] at 19:40 to 7.0MPa[abs] at 20:07. Thus, the amount of evaporation was estimated at about 1,300kg from Equations (1) and (2). This led to the average leak rate of about 0.8kg/s. This leak rate was assumed to remain constant for simplicity from 18:50, when the leak through the instrumentation line was assumed to have started, to 20:30, when the leak from the main steam line was assumed to have started. Thereafter, until 21:15, when the leak from the instrumentation line was assumed to have stopped, the leak rate was assumed to have decreased linearly. The leaked gas temperatures were assumed to be the saturation temperature at the RPV pressure estimated. This was based on the deliberation that the leaked gas would be cooled down by the reactor water around the instrumentation line.



Reference Figure 4 Leak rates through instrumentation line

b.4. Gas leaks to D/W through main steam line

Reference Figure 5 illustrates the leaked gas flow rates to the D/W through the main steam line that were set in the analysis. The leaks occurred mainly in three phases: flashing upon depressurization from 20:30 on March 11th, when the main steam line was damaged, till 21:15; steam generation upon debris relocation in a limited amount (set at 23:30 based on the preliminary analysis); and steam generation upon debris relocation in a large amount at 01:30 on March 12th. The leak rates during each time period were set as explained below.

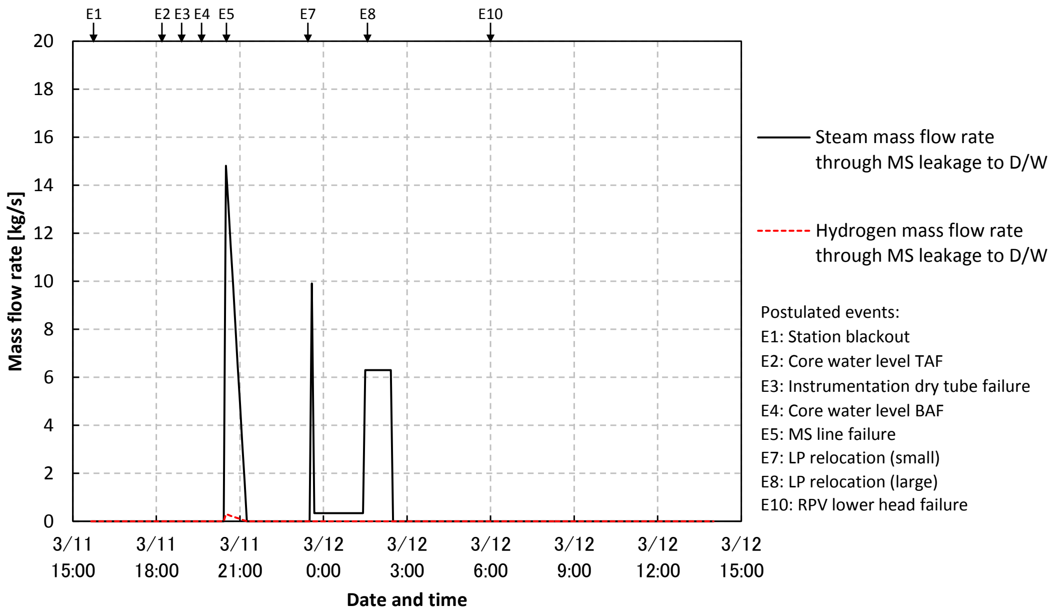
The leak rates from 20:30 to 21:15 on March 11th were set as follows. The amount of evaporation while the pressure was lost from 7.5MPa[abs] to 0.6MPa[abs] was evaluated by

Equations (1) and (2), from which, by subtracting the leak amount through the instrumentation line shown in Reference Figure 4, the steam amount leaked through the main steam line was estimated at about 20,000kg. In order to simulate the leak rate changes responding to the rapid pressure decrease, the leak rates were assumed to have decreased linearly from the value at 20:30 when leaking began to the value at 21:15, while the total leaked amount was kept at the volume estimated above (20,000kg). In the meantime, 400kg of hydrogen out of the estimated total hydrogen gas generated (800kg) was assumed to have leaked to the D/W. The steam generated upon depressurization is heated up as it goes through the core or downcomer, and therefore the leaked gas temperatures from the main steam line become more or less elevated. But the temperatures have big uncertainties. In consequence, a fixed temperature of 600 deg C was set as the leaked gas temperatures, since this value reproduced the observed values best in the preliminary analysis.

At the time of molten debris relocation in a limited amount to the lower plenum at 23:30, part of the whole core was assumed to have moved to the lower plenum as molten debris. When relocating, the debris was assumed to be quickly cooled in fine grain forms. The amount of water evaporation in the lower plenum was estimated by dividing by latent heat of evaporation the calorific heat of debris (2500 deg K in molten state and 300J/kg-K as specific heat were assumed) being cooled down to the saturation temperature at 0.6MPa[abs]. The molten debris is quickly cooled in about 10 minutes, and thereafter till 01:30 on March 12th, mild evaporation due to decay heat was assumed to continue. According to the preliminary analysis about 7% of the whole molten core (10,000kg) was assumed to move to the lower plenum as molten debris. The leakage gas temperature was assumed to be constant at 450 deg C.

At the time of debris relocation in a large amount at 01:30 on March 12th, all of the residual molten material in the core was assumed to relocate to the lower plenum. From this relocation, all the residual water in the lower plenum was assumed to evaporate by 02:30 and its evaporation rate was set as the average rate over the time period. The temperature of leaked steam was assumed to be the saturation temperature at the corresponding reactor pressure. This was because the generated steam was considered as not easily heated up until it reached the leak outlet, because all molten fuel relocated to the lower plenum.

It should be noted that when integrated, the steam leak rates in Reference Figures 3 to 5 is equal to the water inventory in the RPV (excluding the water in the recirculation loop line) at the time of the station blackout.



Reference Figure 5 Leak rates at the main steam line

b.5. Debris relocation from RPV to pedestal

About 15% of the decay heat was assumed to be transferred to the PCV atmosphere from the molten debris that leaked out into the pedestal, through the preliminary analysis. The incondensable gas generation from the molten debris and pedestal concrete interaction (MCCI) was not taken into account, based on the consideration that the gas would have little influence on reproduction of the water level indicator readings. Here, the molten debris properties were ignored, because the debris was treated only as a heat source for heat transfer.

(End)

Reference materials

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[2] "Information Portal for the Fukushima Daiichi Accident Analysis and Decommissioning Activities," URL：https://fdada.info/index

[3] Tokyo Electric Power Company, Inc., "Evaluation of the situation of cores and containment vessels of Fukushima Daiichi Nuclear Power Station Units-1 to 3 and examination into unsolved issues in the accident progression, Progress Report No. 1," December 13th, 2013

[4] Tokyo Electric Power Company, Inc., "Evaluation of the situation of cores and containment vessels of Fukushima Daiichi Nuclear Power Station Units-1 to 3 and examination into unsolved issues in the accident progression, Progress Report No. 2," August 6th, 2014

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[7] Electric Power Research Institute, "Modular Accident Analysis Program (MAAP) 5.03 Simulation of Accidents at Fukushima Dai-ichi Units 1, 2, and 3," Report No. 3002002670 (2014).

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