RECENT FINDINGS ON THE DAMAGED REACTORS AND CONTAINMENT VESSELS OF FUKUSHIMA DAI-ICHI NPS

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ABSTRACT

On March 11th 2011, Great East Japan Earthquake and ensuing tsunami hit Fukushima Dai-ichi nuclear power station. The Fukushima Dai-ichi units 1 to 3 finally lost all DC and AC power, which resulted in the core melt and release of radioactive materials to environment. In order to step forward with the decommissioning, it is necessary to know the final damage status and fuel debris distributions but the remaining records and field investigations are limited and there are still large uncertainties especially in core melt progression.
Now activities towards decommissioning are in progress and investigation results inside reactor buildings and containment vessels are being obtained step by step. We have identified current water levels and water leakage locations in containment vessels. It is important to integrate all of the on-site observations as well as simulation-based analysis for understanding accident progression, damaged reactors, and fuel debris distributions. In this paper, updated findings on damaged containments are shown focusing on damage status of containment vessels.

KEYWORDS
TEPCO Fukushima Daiichi, Severe accident progression, Debris distribution, Containment vessel damage, On-site investigation

1. INTRODUCTION

On March 11th 2011, Great East Japan Earthquake and ensuing tsunami hit Fukushima Dai-ichi nuclear power station. The Fukushima Dai-ichi units 1 to 3 finally lost all DC and AC powers, which resulted in the core melt and release of radioactive materials to environment. TEPCO released our accident analysis report in June 2012 [1] and many reports have been released in the world. Thanks to their efforts, we believe that the investigations and analyses conducted so far have made progress in ascertaining many of the facts related to the progression and cause of the Fukushima nuclear accident.
However, the remaining records and field investigations are limited and there are still large uncertainties especially in core melt progression and fuel debris distributions. TEPCO released the series of progress reports [2, 3] addressing the unsolved issues related to the detail of accident progression and damaged cores and containment vessels. It is important to integrate the obtained on-site information as well as simulation-based analysis and severe accident researches for understanding accident progression and final damage status, which we believe will contribute to further nuclear safety improvement and efficient decommission planning such as prioritization of fuel removal process and investigation in the future.
In the first year of post severe accident, various efforts were made to stabilize damaged reactors and containment vessels, and monitor its cooling conditions and additional radioactive release, as follows:

- Water treatment system was installed to remove Cs and salinity from accumulated water in the buildings which consists of injected water leaking from containment vessels, in-coming underground water and accumulated tsunami.
- Water circulation system from the building basements into cores via the water treatment system was established and enough water was injected to maintain sub-cool conditions inside reactors and containment vessels.
- Two water injection lines through Feed water system (FDW) and Core spray system (CS) were established with redundant motor pumps and power supply lines to enhance the reliability of water injection system.
- Original instrumentation including thermocouples, pressure gages and water level gages was restored and used as long as we could to monitor the cooling conditions in reactors and containment vessels.
- Nitrogen injection systems into containment vessels and reactor vessels were installed to restrain hydrogen concentration.
- Containment gas exhaust systems were installed to minimize gaseous radioactive release and monitor the radioactivity and hydrogen concentration in the exhaust gas.
- Reactor building cover for Unit-1 and Reactor building blow out panel gas exhaust system for Unit-2 were installed to mitigate and monitor the radioactive material scattering from reactor building. Buildi ng rubbles removal works above refueling floor for Unit-3 had been performed.

Since then, on-site works and government-led R&D projects have been in progress to ensure and improve safety and step forwards with fuel removal process based on the mid and long term roadmap towards the decommissioning, which was made firstly in December 2011 and updated with its progress[4]. It is important to extract the insight from these activities for fuel removal and decommissioning such as decontamination in reactor buildings, repair containment boundary for filling containment vessels with water, inspection inside containment vessels and reactor vessels, evaluation by enhanced severe accident simulation codes, reactor imaging with cosmic ray muon, and so on.

Visual inspection in reactor buildings and containment vessels is direct information to know the damage status. Other than that, the plant responses towards sub-cooling conditions are also indicative of estimating damaged core, which were the temperature and pressure changes with the change of water injection rate, for example. Furthermore the information on dose rate distribution in reactor buildings, radioactivity measurement of sampled water, gas, dust and soil could be a trace of fission product release.

In the previous paper [5], TEPCO described our evaluation status of fuel debris distribution based on the plant records during the accident progression, temperature behavior observed towards sub-cooling conditions, and MAAP simulation analysis. It is presumed that almost all of fuel would drop down to containment vessel floor in Unit-1, while, in Units-2 and 3, some of the fuel was left at the original core region and the rest dropped to the bottom of the RPV or to the containment vessel floor.

Investigation results inside reactor buildings and containment vessels are being obtained step by step. We have been identified the water levels and water leakage locations from the lower part of containment vessels. Although it is difficult to specify when the water leakage had occurred, the information on the water leakage can provide some insights for accident progression. In this paper, on-site information is summarized and current estimation is shown focusing on damage status of containment vessels.

2. OBSERVATIONS ON CONTAINMENT VESSEL STATUS

2.1. Unit-1

2.1.1. Current Water Level in Drywell And Suppression Chamber

The water level in drywell was found to be about 2.8m above the floor as of October 10th, 2012, as described below (Figure 1). An investigation was conducted into the status of the drywell of Unit-1 by inserting survey devices into the containment through a hole dug at the drywell penetration (X-100B, on the first floor of the reactor building). In the investigation, video was filmed by cameras, the level of water retained in the drywell was confirmed, dose rate was measured with ion chamber dose meter (Table I), and retained water was sampled and analyzed. Then thermocouples and water sensors were installed to monitor the temperatures and water level. The level of water retained was measured by lowering the CCD camera cable down to the water surface through the grating above in the drywell.

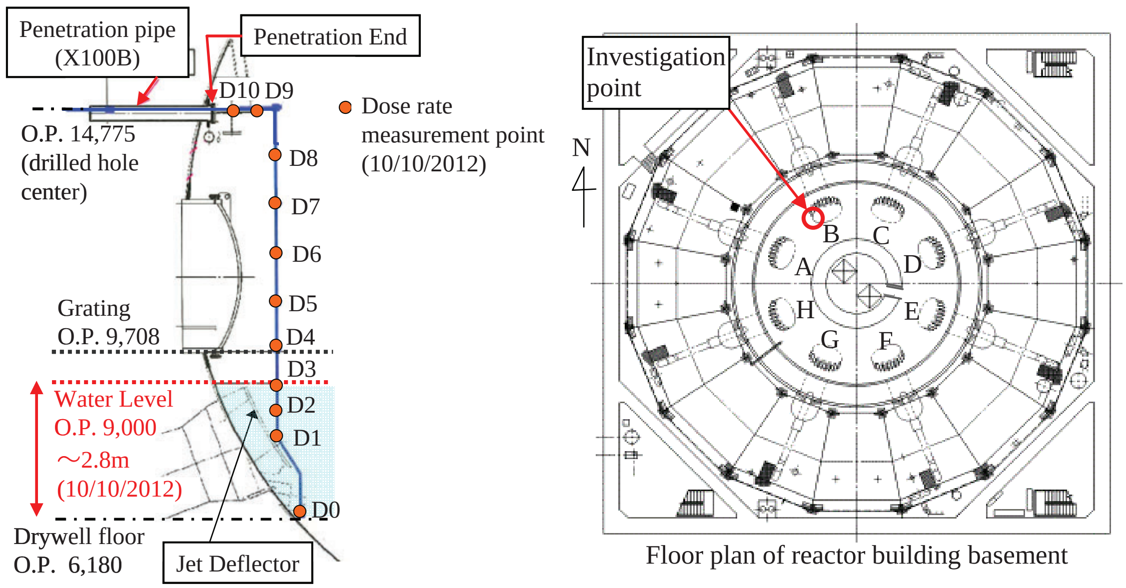


Figure 1. Illustrations of water level and dose rate measurement in inspection inside containment vessel of Unit-1 (Here, OP. 1000 means the elevation of 1000 mm from Onahama Port construction standard surface.)

Table I. Dose rate measured in inspection inside containment vessel of Unit-1

Measurement Point	Distance from penetration end (mm)	Distance from drywell floor (mm)	Dose Rate (Sv/h)
D10	495	8,595	11.1
D9	695	8,595	9.8
D8	870 (1.2m from drywell wall)	7,595	9.0
D7		6,595	9.2
D6		5,595	8.7
D5		4,595	8.3
D4		3,595	8.2
D3		2,795	4.7
D2		2,217	0.5
D1		Not measured	Not measured
D0	Not measured	0	Not measured

The water level in suppression chamber (S/C) was shown to be at around the lowest end of the vacuum breaker tube by a nitrogen injecting test into S/C in September 2012, as described below (Figure 2). The nitrogen injection test was conducted with an intention to explain the phenomenon of the intermittent increase of hydrogen gas concentration and Kr-85 radioactivity measured by the containment gas exhaust system of Unit-1 that has been seen since April 2012. The S/C pressure rose after the injection of nitrogen started into the S/C, the hydrogen concentration and Kr-85 radioactivity monitored by the containment gas exhaust system started to increase, which decreased when nitrogen gas injection was halted. This is interpreted to be that the injected nitrogen pressurized the closed space of the S/C upper part, which lowered the S/C water level and formed a gas discharge channel to the drywell through the vacuum breaker tubes, thus the retained gas in the space was discharged together to the drywell by the injected nitrogen. The volume of the closed space in upper S/C is about 340 m^3 and compressed to about 40 kPag measured by original S/C pressure gage. Most of the hydrogen gas retained in the S/C has been purged by continuously injecting nitrogen into the S/C since October 2012.

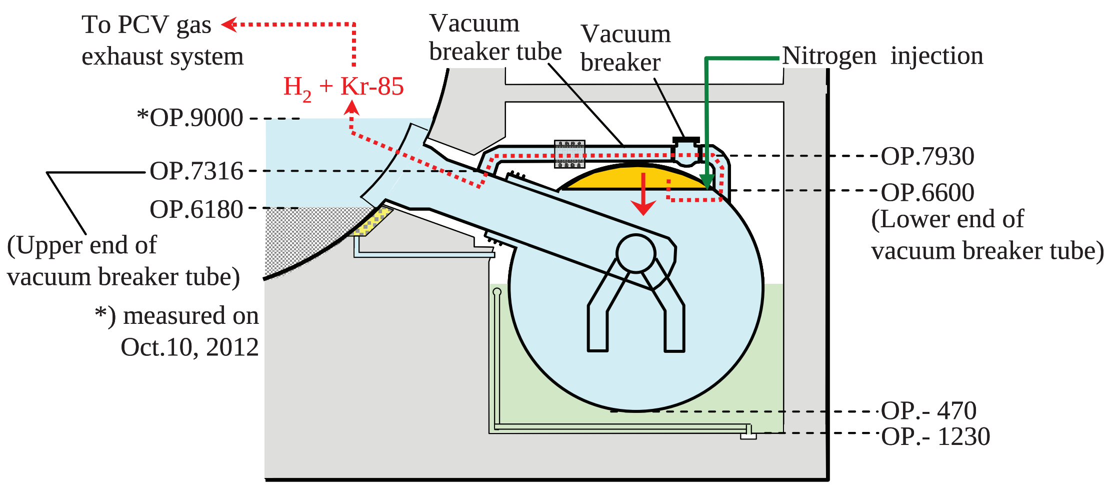


Figure 2. Illustration of S/C water level in Unit-1

In addition, the drywell water level can be evaluated by nitrogen injection pressure into drywell, which was started from April 7, 2011. Here we describe the behavior of drywell water level in 2011.
The drywell water level is calculated by converting the differential pressure between nitrogen injection pressure and drywell pressure to water head as following equation,

$$H = (P_I - P_L - P_{D/W}) / wg + H_0 \tag{1}$$

where H is drywell water level, P_I is nitrogen injection pressure, P_L is pressure loss in nitrogen injection line to the drywell inlet, $P_{D/W}$ is drywell pressure, w is water density, g is the gravity acceleration and H_0 is elevation of the drywell inlet. Nitrogen is supplied by PSA nitrogen generators and injected through the AC system piping into drywell. The nitrogen injection pressure is measured by newly installed bourdon pressure gage in the nitrogen injection line and responding to the change of drywell pressure. If the drywell water level goes over the elevation of injection inlet into drywell (OP.6930 at the upper end), its water head is added to the nitrogen injection pressure.
On the other hand, drywell pressure is measured by original diagram pressure gage and transferred to the main control room. The measurement value was confirmed to be precise by comparing it to another calibrated bourdon pressure gage in reactor building on May 11th, 2011. And also, the pressure loss in nitrogen injection line to the drywell inlet, which was calculated as 2 kPa, was confirmed to be valid by comparing it to the drywell pressure behavior.
The drywell water level calculated started to increase above the OP. 6930 after increasing water injection rate from 6 to 8 m^3/h on May 6th, 2011. After increasing injection rate from 8 to 10 m^3/h on May 15th, however, the drywell water level was saturated around OP.7000 to 8000. These results indicated that

water leak path existed at this elevation, which agrees with the elevation of the vacuum breaker tube. Considering the accumulation of contaminated water in building basements, we gave up the attempt to flood the containment vessel and decreased water injection rate from 10 to 6 m^3/h and drywell water level started to decrease on May 17th. Since May 18th, nitrogen injection pressure had been almost same as drywell pressure, which means drywell water level had been under the nitrogen injection inlet of OP.6930. When we started to depressurize drywell by gas exhaust system from December 8th, 2011, drywell water level started to increase to current level.

2.1.2. Water Leak Location

Water leakage locations have been identified through the series of visual inspections in the torus room. One was a sand cushion drain pipe coming from sand cushion area under the containment vessel and the other was an expansion joint connecting vacuum breaker tube of S/C, as described below (Figure 3).

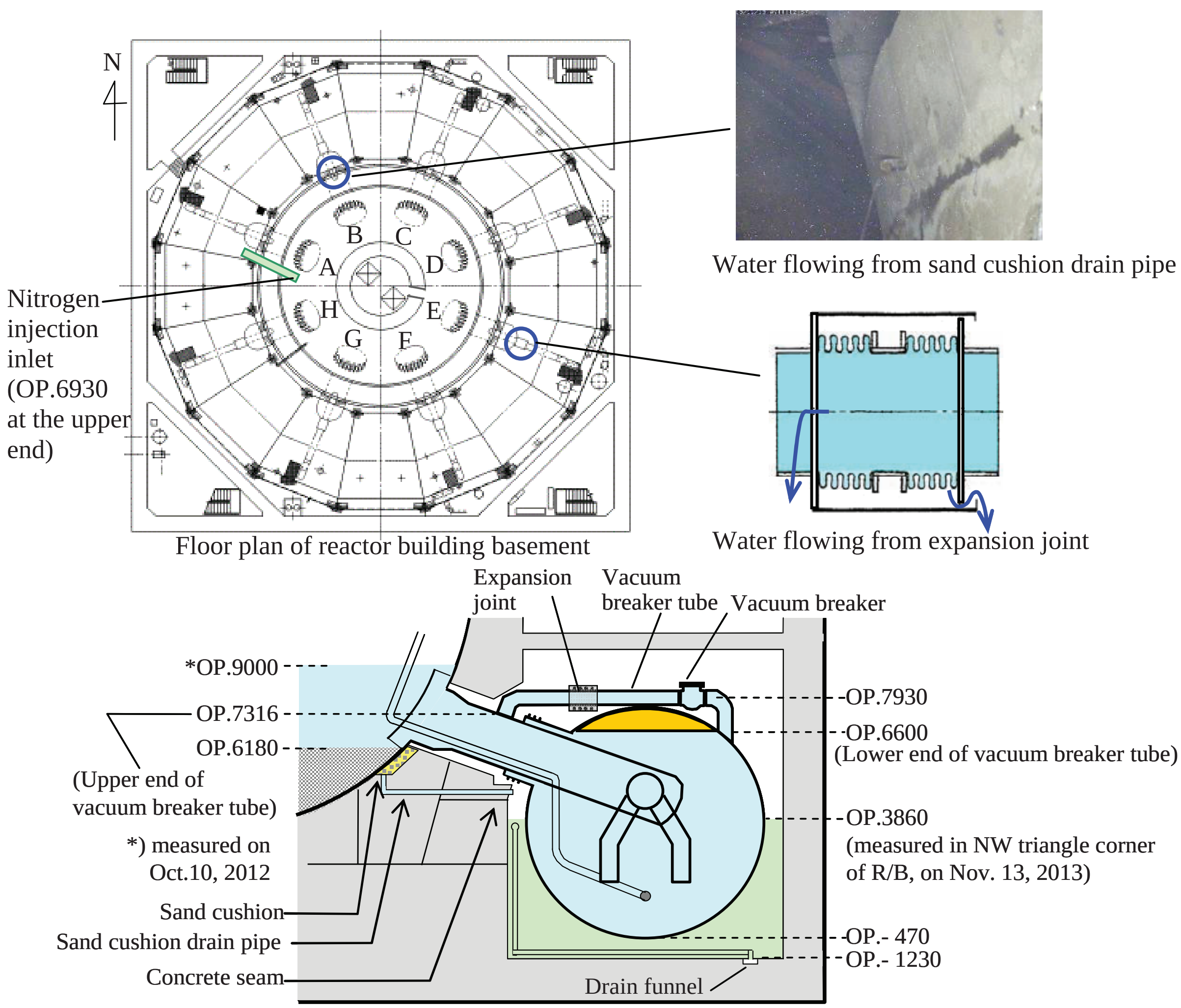


Figure 3. Illustrations of water leakage location in Unit-1

In the torus room investigation in November 2013, a compact automated instrumentation boat, on which a camera and dose meters were mounted, was lowered into the torus room through a 510mm diameter hole drilled into the flooring of the first floor of the Unit 1 reactor building in the northwest corner. The boat was lowered to check visually for water leaks from the vent tube sleeve terminals and the sand cushion drain pipes, and to make dose measurements.

Camera imaging confirmed water flowing from the displaced sand cushion drain pipe of the vent tube X-5B and water flowing down on the S/C surface around the vent tube X-5E as shown in Figure 3. The former was confirmed since the vinyl chloride pipe connecting the sand cushion drain tube and drain funnel with an insertion-type joint had been displaced. Water leaks could not be confirmed at other sand cushion drain pipes, since the drain tubes had not been displaced. The concrete seams below the sand cushion drain piping were observed to be wet all around on the concrete wall, which indicates that leaked water is permeating through the concrete seams from sand cushion area.

In May 2014, survey instrumentation robot was introduced through a 615 x 615mm hole drilled into the northwest area of the first floor of the Unit1 reactor building to explore the S/C top area in order to locate the leak source near the vent tube X-5E. By using the outer catwalk, the instrumentation robot made a camera survey around the vent tube X-5E, and the water leak was confirmed to be from the protective cover of the expansion joint on the vacuum breaker tube as shown in Figure 3. No leaks were noticed from the vacuum breaker valve, torus hatch, Shutdown cooling system (SHC) piping or Atmospheric control system (AC) piping.

2.1.3. Other Information

(1) Drywell temperature history

It was also found during the drywell investigation in October 2012 that shielding lead plates attached in the penetration end inside drywell disappeared, which indicates that gaseous temperature inside drywell exceeded the melting point of lead, 328 deg.C and the plates melted down.

(2) Possibility of leakage from RCW system

High dose rates were noticed around the piping and heat exchangers of the reactor building closed cooling water system (RCW). RCW piping is running into the equipment drain sump pit on the pedestal floor to cool the drain water. If molten core slumped down into equipment drain sump pit, there is a possibility that corium would rupture the RCW piping and highly contaminated water or steam flow out through RCW piping. However some observations that water remained in RCW surge tank located in 4th floor are not totally consistent with this possibility and further investigation is needed.

2.2. Unit-2

2.2.1. Current Water Level in Drywell And Suppression Chamber

The water level in drywell was roughly measured to be about 60 cm above the floor as of March 26th 2012, and finally found to be about 30cm by more precise measurement on June 11th 2014, which was about lower end of vent tube jointing above the drywell floor, as described below (Figure 4).

In January 2012, first investigation was conducted into the drywell of Unit-2, when videos were taken by cameras and temperature was measured above grating by inserting survey devices into the drywell through a hole dug at the drywell penetration (X-53, on the first floor of the reactor building). In the second investigation in March 2012, videos were taken by cameras, the level of water retained in the drywell was confirmed, and dose rates and temperatures were measured by ion chamber dose meter (Table II). The level of water retained was roughly measured by lowering the camera cable down to the water surface through the grating above along the drywell wall. In the third investigation in August 2013, retained water was sampled and analyzed. In June 2014, thermocouples and water sensors were installed to monitor temperatures and water level. At this time, water level was precisely measured by vertically lowering the camera cable down to drywell floor. From these investigations, we cannot estimate the highest temperature in drywell during the accident because concrete block was originally used in the penetration pipe for shielding instead of lead plate.

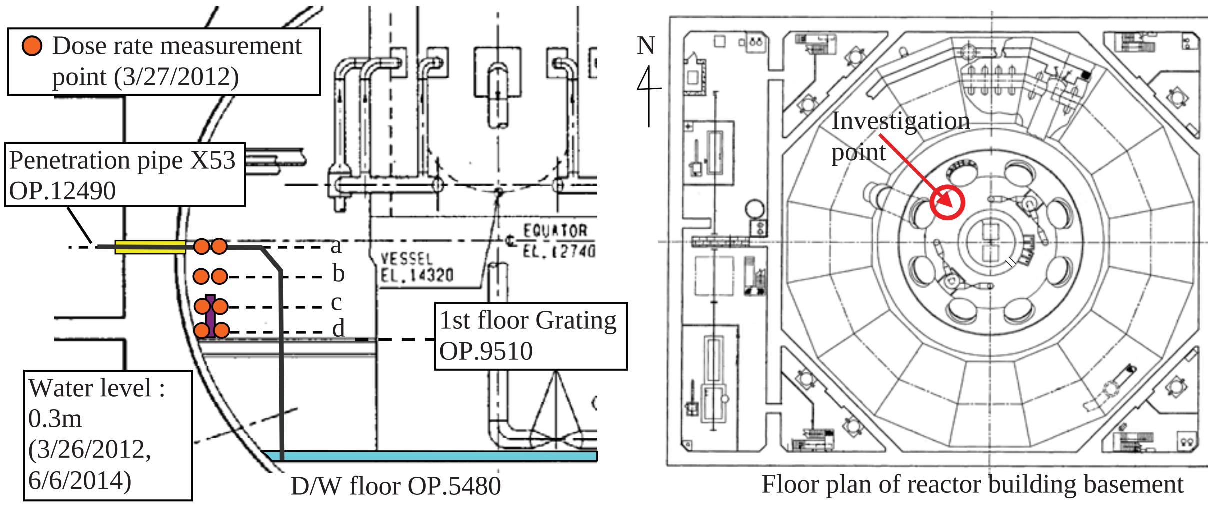


Figure 4. Illustrations of water level and dose rate measurement in inspection inside containment vessel of Unit-2

Table II. Dose rate measured in inspection inside containment vessel of Unit-2

Measurement Point	Distance from Drywell floor (mm)	Dose Rate (Sv/h)	
		500mm from PCV wall	1000mm from PCV wall
A	7,010	31.1	39.0
B	6,010	48.0	54.1
C	5,010	41.4	57.4
D	4,180	37.3	72.9

The S/C water level, which was measured by ultrasonic technique in January 2014, found to be about middle of S/C and several cm lower than the torus room water level changing together with torus room water level, as shown in Figure 5 and Table III. The water level was remotely measured using ultrasonic techniques from the chamber outer surface in January 2014. That is, the ultrasonic waves reflected by the S/C internal structures as well as the opposite wall were continuously measured. The water level could be determined by observing where the reflective waves disappeared.
The original S/C pressure gages malfunctioned but a new pressure gage was installed for nitrogen injection test, as shown in Figure 5. The pressure measured was 3 kPag as of May 14th 2013 and increased to about 13kPag with increase in drywell pressure during the nitrogen injection. This positive S/C pressure is consistent with the fact that the S/C water level is lower than the torus room water level and indicates that upper S/C structure is not largely damaged.

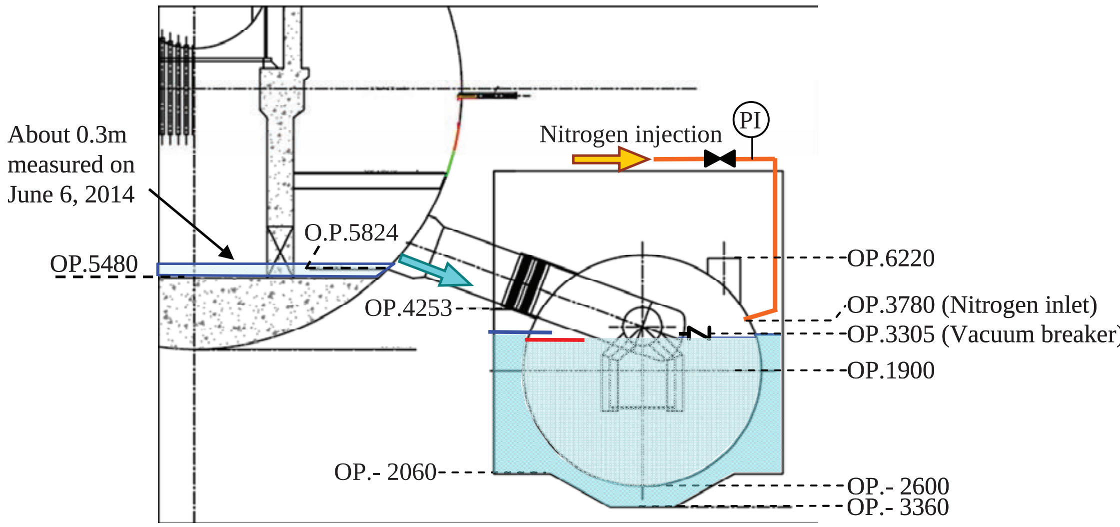


Figure 5. Illustration of S/C water level in Unit-2

Table III. Water level in S/C and torus room measured in Unit-2

	January 14, 2014	January 15, 2014	January 16, 2014
Water level inside S/C	OP. 3210mm	OP. 3160mm	OP. 3150mm
Water level in torus room	OP. 3230mm	OP. 3190mm	OP. 3160mm

2.2.2. Water Leak Location

There is no direct evidence to specify water leak path from containment vessel of Unit-2, as of the end of 2014. However, following observations confirm that injected water into reactor is flowing into S/C and water leaks occur at the S/C lower position including pipes connected to S/C.

- Drywell water level is about 30cm, which is about the height of lower end of vent tube jointing above the drywell floor.
- S/C water level is about middle of S/C and several cm lower than the torus room water level changing together with torus room water level.
- No leak was confirmed from the lower end of 8 venting tubes within the visible range, including the end part of venting tube sleeve, sand cushion drain pipe, and lower part of bellows cover of venting tube, as described below.

In the Unit-2 torus room investigation in April 2012, a robot accessed the gallery (catwalk) inside and videotaping, dose rates measurement, acoustic checks, etc. were carried out. No leak trace was confirmed on the flange of the two S/C manholes, as far as the camera photos show. Further investigations were made in December 2012 and March 2013, and the area around the lower end of venting tubes was surveyed by a robot. A small patrol vehicle, which was mounted on the tip of an arm of a four-leg robot, was set on the S/C, from which it accessed the lower end of the venting tube and took photos. At least, no leak was confirmed from the lower end of 8 venting tubes within the visible range (Figure 6).

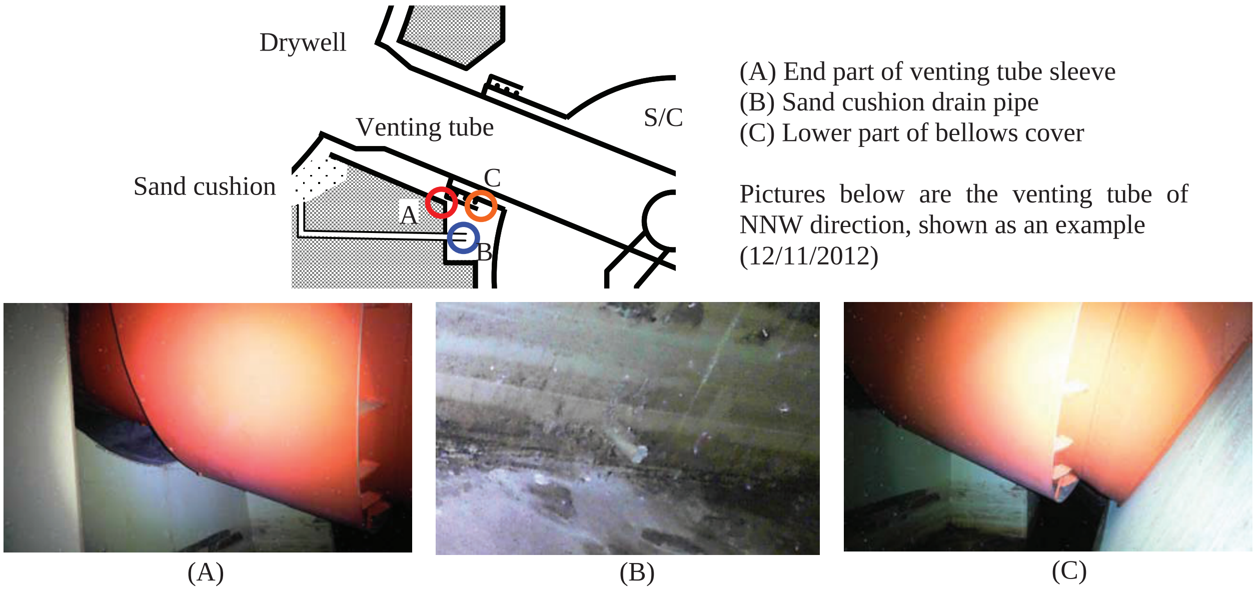


Figure 6. Pictures showing no leakage from lower end of venting tubes of Unit-2

2.2.3. Other Information

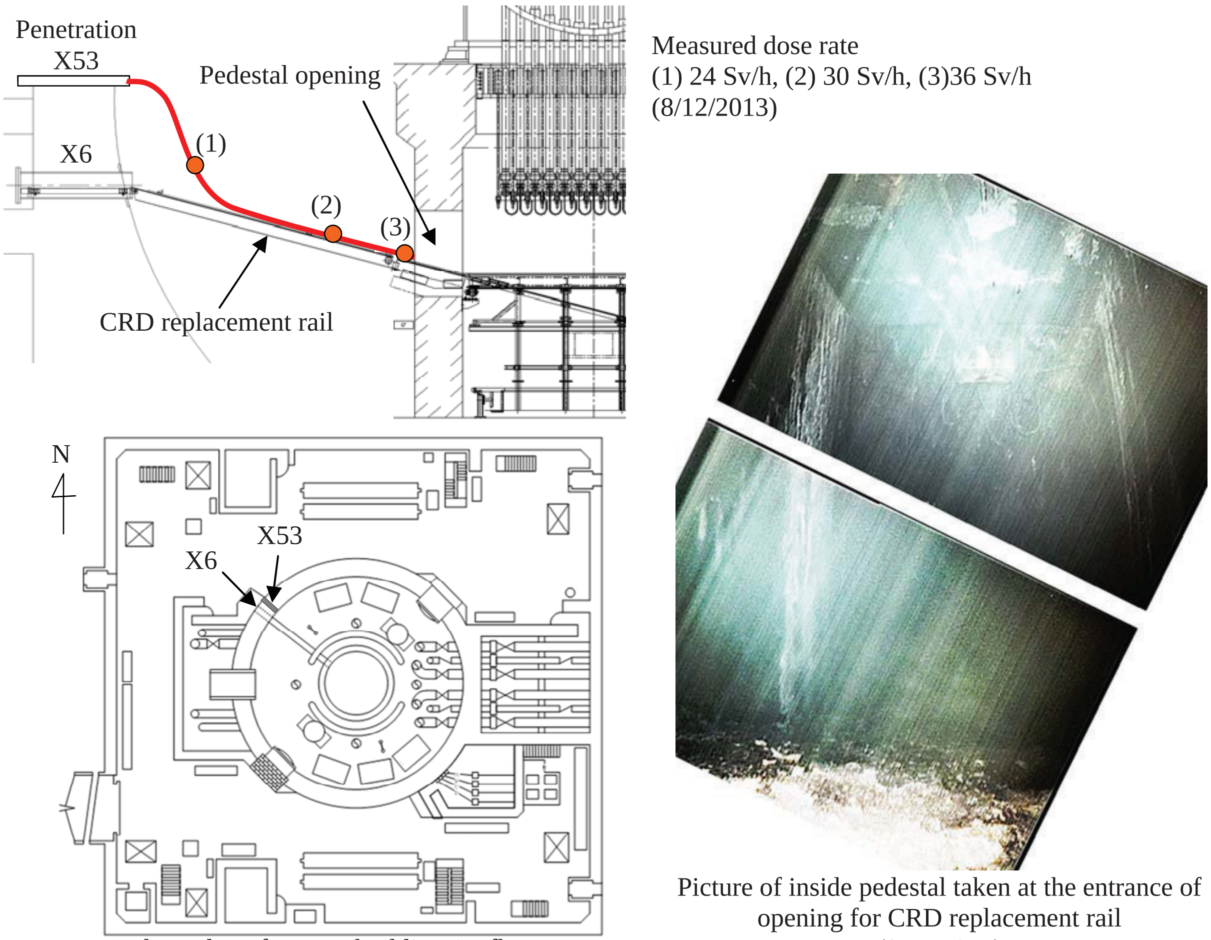
(1) Access to pedestal

In July and August 2013, a survey was conducted inside the drywell of Unit-2, when instrumentation was introduced through the drywell piping penetration X-53 in reactor building first floor to take video images and make dose and temperature measurements along the replacement rail for control rod drive mechanism (CRD) and pedestal opening, as shown in Figure 7.

Camera images were taken at the pedestal opening into its inside and after photo processing for noise and contrast. They confirmed the position of the control rod position indicator probe (PIP) cables in the upper part of the pedestal opening, but no clear information was obtained regarding what was in the lower part inside the pedestal.

Dosimeters measured the dose rates as far as the top of the CRD replacement rails. The values were about 45 Sv/h just after entering drywell. Because the dosimeters were out of order on the way to pedestal, dose rates were evaluated from the camera image noises; they were about 30 Sv/h near the landing point in replacement rail and about 36 Sv/h near the pedestal opening. No clear indication was obtained about gaining access to fuel debris, even via the pedestal opening on the CRD replacement rail because access to fuel debris will result in rapid dose rate increase.

Measured dose rates can be interpreted as the summation of radiation from deposited FP like Cs on the drywell wall and various internal structures. The radiation from fuel debris cannot be detected in the current dose rate level because it is shielded by water, pedestal wall, or biological shielding wall around RPV. If fuel debris exists nearby the pedestal opening for CRD replacement in pedestal, rapid dose rate increase can be detected in access to the entrance. However, no clear indication was obtained about gaining access to fuel debris, even via the pedestal opening on the CRD replacement rail.



Floor plan of reactor building 1st floor

Picture of inside pedestal taken at the entrance of opening for CRD replacement rail (8/12/2013)

Figure 7. Illustrations of investigation along with CRD replacement rail in Unit-2

(2)Dose rate distribution on refueling floor

Dose rate distribution on the refueling floor of Unit-2 was measured by robot walking on the floor in 2012. The dose rate tended to be high nearby the shield plug above the drywell top head and maximum dose rate of about 880mSv/h was obtained on June 13, 2012. In addition to observed steam blowing from the displaced blow-out panel on the wall of refueling floor since March 15 2011, these observations show that steam had been leaking from drywell top head.

2.3. Unit-3

2.3.1. Current Water Level In Drywell And Suppression Chamber

There is no direct measurement of drywell water level so far, but first investigation inside drywell is planned in 2015. However, the drywell water level can be evaluated as about OP.12000 as of May 15 2014, which is about 2 m above the reactor building first floor or 6.5m above the drywell floor, by converting the differential pressure between S/C and drywell to water head as following equation.

$$H = (P_{S/C} - P_{D/W}) / wg + H_0 \tag{2}$$

Where H is drywell water level, $P_{S/C}$ is S/C pressure, $P_{D/W}$ is drywell pressure, w is water density, g is the gravity acceleration and H_0 is elevation of S/C pressure measurement point. The drywell pressure has been nearly atmospheric pressure since the end of March 22, 2011, as described in section 2.3.3. The S/C pressure is measured by the original diaphragm pressure gage which instrumentation pipe is connected to S/C at the elevation of OP.3780 and the measurement is transmitted to the main control room. Here, S/C pressure of 180 kPa(abs) corresponds to drywell water level of about OP. 12m using 1 kg/m^3 as water density. The absolute value of the measured S/C pressure would not be precise because it has not been calibrated since the accident. However, it is thought that the calculated water level is not largely deviated from following observations.

- The measured S/C pressure has been stable around 180 to 190 kPa(abs) without drift behavior and has been responding with the change of water injection rate into reactor.
- Water leak was confirmed from near the expansion joint of the drywell penetration pipe for a Main Steam line on May 15 2014, which indicates drywell water level is over the elevation of the penetration (OP. 11670).
- There was no water at the elevation of drywell penetration pipe X-53 (OP.12490), which was confirmed by ultrasonic technology in October 2014.

If we look back on the past measurement, S/C pressure showed about 200 kPa(abs) when measurement was started by supplying AC power on March 24th. After that, the S/C pressure rapidly decreased to 180kPa(abs) on March 27th and then showed gradual change between 165 kPa(abs) and 180 kPa(abs) until May 4th. And the S/C pressure responded with the change in water injection rate on May 4th and May 12th. S/C temperature measured from March 22nd showed around 100 deg.C at first, but then monotonously decreased towards 40 deg.C on May 4th. On the other hand, the drywell and RPV temperatures measured was over 100 deg.C and drywell pressure measured was about atmospheric pressure. From these observations, drywell water level is thought to have been high since then, although it is difficult to evaluate accurate level due to the uncertainty in drywell water pressure and temperature.
This would be because that various attempts had been done to increase water injection such as increasing discharge flow rate of fire trucks from March 17th, adding one more fire truck in series from March 20th, and closing valves to avoid water leak to other system. Also it should be noticed that the injected water had been seawater until water source was switched to fresh water on March 25th.

On the other hand, there is no direct information on water level in S/C. So far, the water level in S/C is expected to be almost full from following observations although it should be noted that there is uncertainty in volume of the residual non-condensable gas in S/C because it was uncertain to have been able to keep opening wetwell venting valves.

- Drywell water level has been relatively high compared to that of unit-1 and unit-2, which indicates there would be no significant leak path in S/C.
- The non-condensable gas including hydrogen, which was generated by in-vessel water zirconium reaction and transferred into S/C during accident progression, is thought to be exhausted by repeated wetwell venting operation until April 8, 2011.

2.3.2. Water Leak Location

Water leak was confirmed from near the bellows seal for the pipe penetration of Main Steam (MS) line on May 15 2014, which indicates that drywell water level was over the elevation of the penetration (OP. 11670), as described below (Figure 8).
In January 2014, while camera photos taken by the wreckage removal robot were being checked, water was seen to be flowing from near the main steam isolation valve (MSIV) room door in the northeast area of the reactor building 1st floor. The water was flowing towards a nearby floor drain funnel and falling down to the torus room.

The MSIV room is located east area of the 1st floor in reactor building and there are 9 pipe penetrations through drywell wall in the MSIV room including MS lines. There are 2 kinds of pipe penetration except for electric cable penetration. One is the bellows seal type penetration, which is used for high temperature piping or other necessary piping to allow its dislocation due to thermal expansion or other reasons at the penetration. In the case of high pressure piping, protection tube is also inserted for the bellows seal. The other is the welding type penetration where piping is welded to nozzle.
Instrumentation was inserted into the MSIV room from the upper floor in April and May 2014, in order to locate the water flows in the room. It was found that water leaks were from near the bellows seal for pipe penetration of MS line D. It was concluded that the leakage had occurred only from the MS line D, based on: (1) confirmation of no leaks from the MS lines A, B and C, and their main steam drain pipes; and (2) the flow directions of leaked water on the floor. Dose rate was not measured nearby these pipes in the room. Besides, the calculated water level from S/C pressure was about OP.12m, which agreed with the elevation of pipe penetration for MS line.

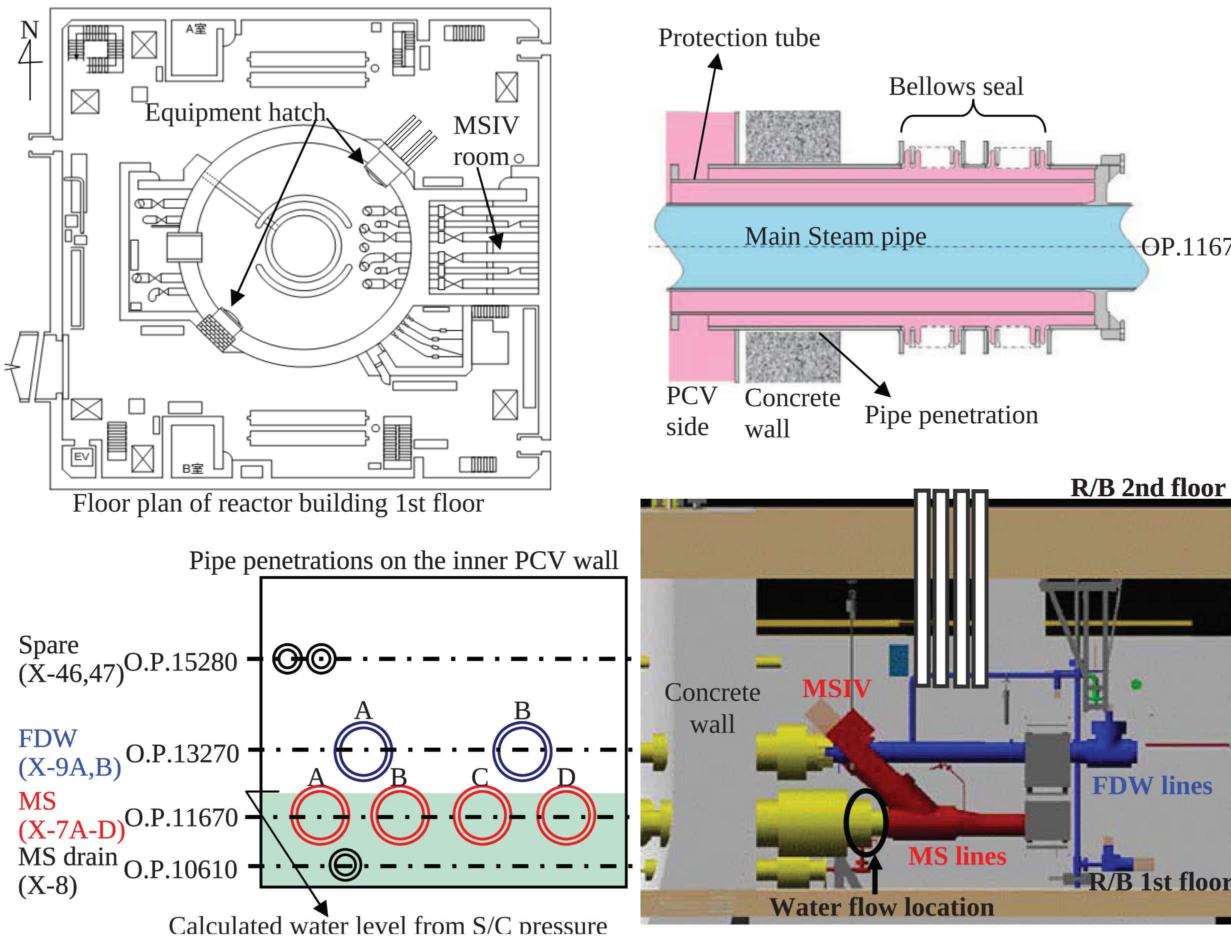


Figure 8. Illustrations of water leakage location in Unit-3

In addition, water leak trace was observed in the equipment hatch in northeast area of 1st floor in the Unit-3 reactor building in April 2012, as described below (Figure 9). It was found that part of the floor in front of the equipment hatch was wet by inserting video imaging scope into the entrance space on April 19th, 2012. The entrance space penetrating into the concrete outer drywell wall is normally closed by a shielding plug but it was found that the shielding plug was displaced by robot inspection in September 2011. The displacement of shielding plug might be attributed to the impact of hydrogen explosion of reactor building as it was observed that TIP room door or torus room door were blown off. The video imaging scope was inserted through this opening between displaced shielding plug and concrete wall. Furthermore, extremely high dose rate was observed at the rail of shield plug where water remained. These observation indicated drywell water could seep through the equipment hatch.

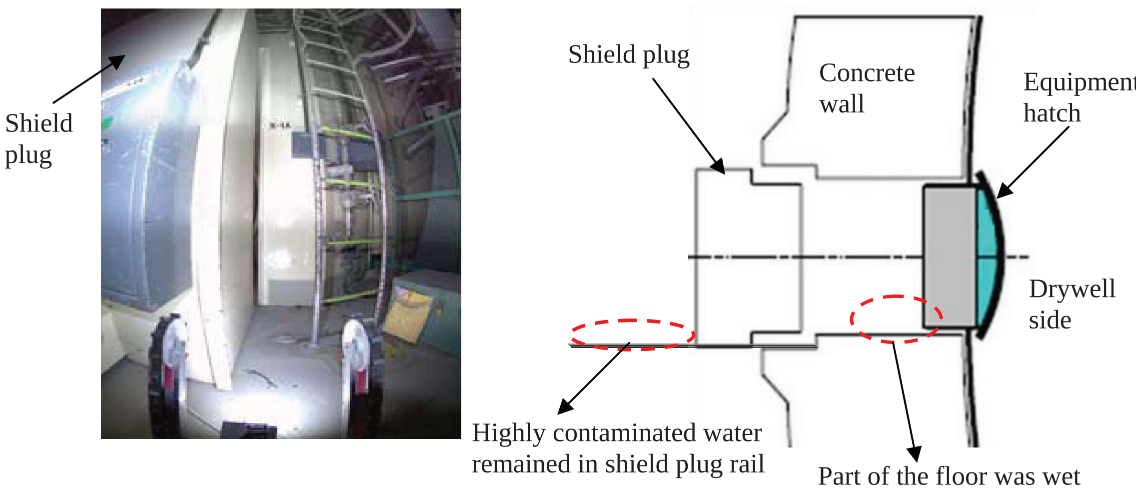


Figure 9. Illustrations of observed water leakage trace in front of the equipment hatch of Unit-3

On the other hand, no water leaking position in the S/C was located yet, as described below. In the Unit-3 torus room investigation in July 2012, a robot accessed the gallery inside or catwalk. Videotaping, dose rates measurement, acoustic checks, etc. were also carried out to the extent possible. At least, no leak was confirmed on the flange of the access hatches, as far as the camera photos show.

2.3.3. Other Information

(1)Complete loss of leak tightness

Nitrogen is being sent to the drywell and reactor in order to maintain an inert atmosphere, while the containment gas exhaust system discharges the same amount of gas from the drywell. It was confirmed through analyzing the discharged gas that the oxygen concentrations in the drywells of Unit-1 and Unit-2 were nearly zero, while that in Unit-3 was about 8%.

Unit-3 drywell pressure has been almost constant at the atmospheric pressure since the end of March 2011 and showed no change in spite that steam kept being generated in drywell. Furthermore the drywell pressure showed no change after nitrogen injection into drywell from July 2011 while Unit-1 and 2 drywell pressure show several kPa and responded after water injection change or nitrogen injection. Consequently, the gas leak rate of the Unit-3 drywell was confirmed to be the highest.

(2)Dose rate distribution on refueling floor

Dose rate distribution above the refueling floor of Unit-3 has been measured by dropping dosimeter from crawler crane. The dose rate tended to be high around the shield plug above the drywell top head and maximum dose rate of about 400mSv/h at 5m above the floor was obtained on November 6-7, 2013, which was measured after completing removal works of building rubbles and before starting the decontamination works in the refueling floor. In addition to observed steam blowing from top of the building, these observations show that steam had been leaking from drywell top head.

3. DISCUSSION

3.1. Water Leak Path In Unit-1

As described in previous section, we have identified current water level in drywell and S/C and water leakage locations: Sand cushion drain pipe and expansion joint of S/C vacuum breaker tube. We now discuss when and where these leakages would occur.

(1) Water leak from sand cushion drain pipe
There are two kinds of possible water leak path leading to the sand cushion drain pipes in the first place: (a) drywell liner damage due to shell attack by molten core and (b) pipe penetrations located under the drywell water level (OP.9000). From observed behavior of nitrogen injection pressure into drywell as described in previous section, it is inferred that drywell water level had been lower than the elevation of nitrogen inlet into drywell (OP.6930) since May 18th until December 8th, 2011. There is no other pipe penetration under the nitrogen inlet except vent tubes connecting drywell with S/C. Furthermore the elevation of nitrogen inlet into drywell is lower than the S/C vacuum breaker tube (OP.7678 at lower end). In this period, therefore, injected water would be leaking from somewhere in drywell shell under OP.6930, which would be flowing into the sand cushion drain pipes. On the other hand, generated steam was leaking through the expansion joint of vacuum breaker tube as well as other upper leak path like drywell top head, which is also supported by the fact observed in June 2011 that steam was rising from the piping penetration in the southeast 1st floor of reactor building, which was same direction as the expansion joint with leakage confirmed.
Water flow rate from the sand cushion drain pipe was evaluated by mockup test where we analyzed angles and width of the water flow in the photo and reproduced the situation. As a result, estimated water flow rate is about 0.15m^3/h per a drain pipe. Assuming equal flow rates for other drain pipes, total leakage flow rate from sand cushion would be 1.2m^3/h. Therefore estimated leak flow rate from sand cushion is about 3 to 30% of total injected water flow into reactor (4.4m^3/h at the time), which shows that this flow path is not main leakage.
According to our previous MAAP simulation result [3], RPV rupture occurred about 15 hours after the SCRAM or at about 05:40 on March 12th. At the time of RPV rupture, water injection from fire truck is thought to have been ineffective [3], so there would be a possibility of shell attack. On the other hand, measured drywell pressure was kept as high as 0.7MPa(abs) until wetwell venting was performed around 14:10 on March 12th. It is inferred that, even if drywell liner was attacked by molten core, this shell attack might not result in catastrophic damage on containment vessel and might create high resistance leak path. In the NUREG/CR-5423 report addressing the Mark I liner attack issue, it was described that one of the important phenomena which was not fully understood was a possibility that the relief path through a gap between the shell and concrete wall would be easily blocked [6]. It should be noticed that there is a possibility of shell attack on the bottom as a result of vertical propagation of MCCI or shell attack on the side wall as a result of spread on the floor. It is needed to identify the leak path and damage on drywell shell by inside investigations in the future.

(2) Water leak from expansion joint of vacuum breaker tube
The venting tube of the vacuum breaker tube with water leakage faces the pedestal opening for access way in the drywell floor, which could be molten core discharge path from the pedestal. This leakage location might be related to the thermal affect by the discharged molten core although it is unknown when this leakage occurred. There is also a possibility of damage by long-term corrosion progression. At latest, however, this water leak path had been already formed before we increased water injection to try to re-flood containment vessel in early May as discussed above. According to the dose rate measurement in

torus room, relatively high dose rate was obtained around the leakage area. More detail dose rate distribution along the vacuum breaker tube could be useful to estimate how this leakage occurred including wetwell venting behavior because wetwell venting line is connected to this vacuum breaker tube. This will be also informative to know the behavior of fission product deposition during wetwell venting because high dose rates were found along the wetwell vent line such as AC piping in reactor building 1st floor, vicinity of SGTS (Standby Gas Treatment System) room in the reactor building 2nd floor, and the SGTS piping connecting to exhaust stack. The contamination level is much higher than that of Unit-3, which indicates that scrubbing effect might be limited in the case of Unit-1 wetwell venting.

When it comes to gaseous leakage, it should be noted that the dose rate increases in the reactor building were observed earlier as follows,

- High dose rate in the 1st floor of reactor building; when operators entered the reactor building at about 21:00 on March 11th, in order to check the water levels of the IC shell tank and the reactor, their alarm pocket dosimeters showed 0.8 mSv shortly thereafter (about 300mSv/h) and they reported that upon returning to the main control room at 21:51.
- Dose rate increase in the site; the dose rate started to increase at about 4:00 March 12th, monitored by monitoring post-7, 8 and a monitoring car deployed nearby main gate at the time.
- High dose rate in torus room; when operators entered the torus room around 9:30 March 12th, in order to open a valve for wetwell venting, they found extremely high dose rate more than 1Sv/h.

3.2. Water Leak Path In Unit-2

As described in previous section, there is no direct evidence to specify water leak path from containment vessel of Unit-2, as of the end of 2014. However, it was confirmed that injected water into reactor was flowing into S/C and water leaks occur at the S/C lower position including pipes, so that drywell water level was quite low and S/C water level was connected to the water level in the reactor building basement. This shows that relatively large leak path is formed compared to Unit-1 and 3. We now discuss when and where these leakages would occur.

The S/C temperature, which was measured by original resistance thermometers (RTD) from April 2011, also showed the movement of S/C water level. There are two measurement elevations: one is OP.350 located in the lower hemisphere and originally measuring S/C pool water temperature and the other is OP.2885 located in the upper hemisphere and originally measuring S/C gas temperature. In starting the temperature measurement from April 2nd, the both thermometers indicated same temperature about 100 deg.C and showed gradually decreasing transient. Then the temperature of lower-positioned thermometers at OP.350 started rapidly to decrease on April 8th and finally reached 59 deg.C in the end of June, while the upper-positioned thermometers maintained the gradual decreasing trend.

This temperature behavior can be explained as follows. Drywell temperature was over 100 deg.C and steam was generated at that time. At first, both thermometers measured the temperature of gas flowing into S/C from drywell. Then lower-positioned thermometers got submerged by increasing water level in S/C, which resulted in the temperature decrease. Thereafter the temperatures in two positions have repeated on and off transient following drywell pressure change, water inflow change, or steam inflow change, which indicated the up and down movement of S/C water level. It is difficult to specify when this leakage occurred. At latest, however, it is thought that this water leak path had been already formed in the early April, 2011. Furthermore the S/C water level at that time was too low (under OP.350), which indicates S/C water might be pushed out while drywell pressure or S/C pressure was kept high.

Among potential leak paths, first of all, seal parts of valves or components on a pipe connected to the lower S/C are expected rather than the shell or piping structures considering their robustness. There are 6 suction lines for CS system (sub train-A, B), RHR system (sub train-A, B), HPCI system, RCIC system, which are located in the basement of reactor building, from the bottom part of S/C. Indeed these systems are designed to inject water into high pressurized reactor and unlikely to have damage to cause the current main water leak path. Nevertheless, RCIC was operated for almost 3 days without DC power and its water source was switched from CST to S/C on March 12th, 2011. Therefore high temperature water

beyond design limitation was supplied to RCIC pump and used as cooling water for auxiliary. Water leak would be limited if seal parts were intact. Still there is no direct information on final status of RCIC turbine and pump after the 3 days operation and one possibility would be a leak path through RCIC suction line.
There are also 4 instrumentation pipes for level water gages in lower part of S/C, although its pipe diameter and valves are small. Further investigation is needed in order to specify the leak path forming the high leak flow.

3.3. Water Leak Path In Unit-3

As described in previous section, water leak was confirmed from near the bellows seal for the pipe penetration of MS line, which indicated that drywell water level was over the elevation of the penetration (OP. 11670) and agreed with the water level calculated by S/C pressure gage. We now discuss when and where these leakages would occur.
According to the observed S/C pressure and drywell temperature behavior, as described in previous section, the drywell water level had been already as high as the current water level before the end of March, 2011, and the bellows seal might have been exposed to sea water under high temperature conditions. Many other plants have experienced cracking of the stainless steel bellows commonly used at reactor building or containment pipe penetrations [7]. One possibility is that there could have been cracks already initiated and the hot seawater added a stressor that accelerated the crack growth. We have not performed an investigation to look into the water leakage from other bellows seals for pipe penetration (HPCI etc.) and lower end of venting tubes including bellows cover or sand cushion drain pipe as of the end of 2014. Further investigation is needed.

4. CONCLUSION

We have been identified the current water levels in drywells and S/Cs and locations of water leakage from the lower part of containment vessels in Fukushima Dai-ichi Unit-1 to 3, based on on-site investigations.
In Unit-1, water leakage from sand cushion drain pipe indicates the possibility of drywell liner damage due to shell attack by molten core. However, it is inferred that, even if drywell liner was attacked by molten core, this shell attack might not result in catastrophic damage on containment vessel and might create high resistance leak path. Water leakage from expansion joint of vacuum breaker tube faces the pedestal opening for access way in the drywell floor and this leakage might be related to the thermal affect by the discharged molten core.
In Unit-2, it was confirmed that injected water into reactor was flowing into S/C and water leaks occur at the S/C lower position including pipes. And low water level in both drywell and S/C shows that relatively large leak path is formed compared to Unit-1 and 3. One possibility would be a leak path through RCIC suction line but there is no direct information on final status of RCIC turbine and pump after the 3 days operation with high temperature water beyond design limitation. Further investigation is needed in order to specify the leak path.
In Unit-3, water level in drywell is the highest among 3 units and water leakage was confirmed from bellows seal of pipe penetration for MS line. The drywell water level seems to have been as high as current level since the end of March 2011 and bellows seal was exposed to seawater. One possibility is that there could have been cracks already initiated and the hot seawater added a stressor that accelerated the crack growth. On the other hand, gas leak rate of the Unit-3 drywell was confirmed to be the highest as a final state but it is unknown how gaseous leakage progressed.
Investigation results inside reactor buildings and containment vessels are being obtained step by step although these efforts are primarily focused on obtaining data required to support decommissioning activities. We are planning following investigations in the near future. There remain many unclear issues and further forensic examination is needed to know the detail.

- Containment vessel investigation in Unit-1
  - Fiscal Year 2015: Walking around on the 1st grating floor in drywell (performed in April 2015)
  - Fiscal Year 2015: Going down to the bottom floor in drywell outside pedestal.
  - TBD: Accessing inside pedestal through CRD replacement rail
- Containment vessel investigation in Unit-2
  - Fiscal Year 2015: Accessing inside pedestal through CRD replacement rail
- Containment vessel investigation in Unit-3
  - Fiscal Year 2015: First investigation inside drywell from the penetration X-53

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