Status of investigation on estimating situation of cores and containment vessels

1. Introduction

The conditions of Unit-1 and Unit-3 containment vessels (PCVs), and the situation of damaged and fallen fuel were estimated, at a technical workshop held on November 30, 2011, based on comprehensive evaluation of then-available knowledge, such as temperature changes, etc. due to water injection by the core spray systems. The workshop (organized by the former Nuclear Industry and Safety Agency) was for estimating the conditions of core damage at Unit-1 to Unit-3 of the Fukushima Daiichi Nuclear Power Station.

Thereafter, TEPCO has continued estimating core and in-containment conditions and updated them by incorporating accumulated knowledge. Since FY2016, the estimation of fuel debris distribution at Unit-1 to Unit-3 has been undertaken in collaboration with a government subsidized project of “Decommissioning and Contaminated Water Management (Upgrading Level of Grasping State inside Reactor)” being managed jointly by the International Research Institute for Nuclear Decommissioning (IRID) and the Institute of Applied Energy (IAE). TEPCO is continuing examination even after the end of the project.

The latest illustrations for the estimated situations in cores and containment vessels incorporating new knowledge obtained thereafter by the results of accident progression analysis, field investigations, etc., are summarized in Figures 1-1 to 1-3.

In addition, as a separate document to this attachment, a summary is given for how the estimation has been updated from the state estimation diagram presented on November 30, 2011 (Supporting information 2, "Estimation of status inside reactor pressure vessels and containment vessels after the accident at the Fukushima Daiichi Nuclear Power Station").

In the next and subsequent sections, the information obtained at the site since November 30, 2011 is summarized.

Refer to Chapter 10. Supplement to the Main Body of the Progress Report for usage of O.P.

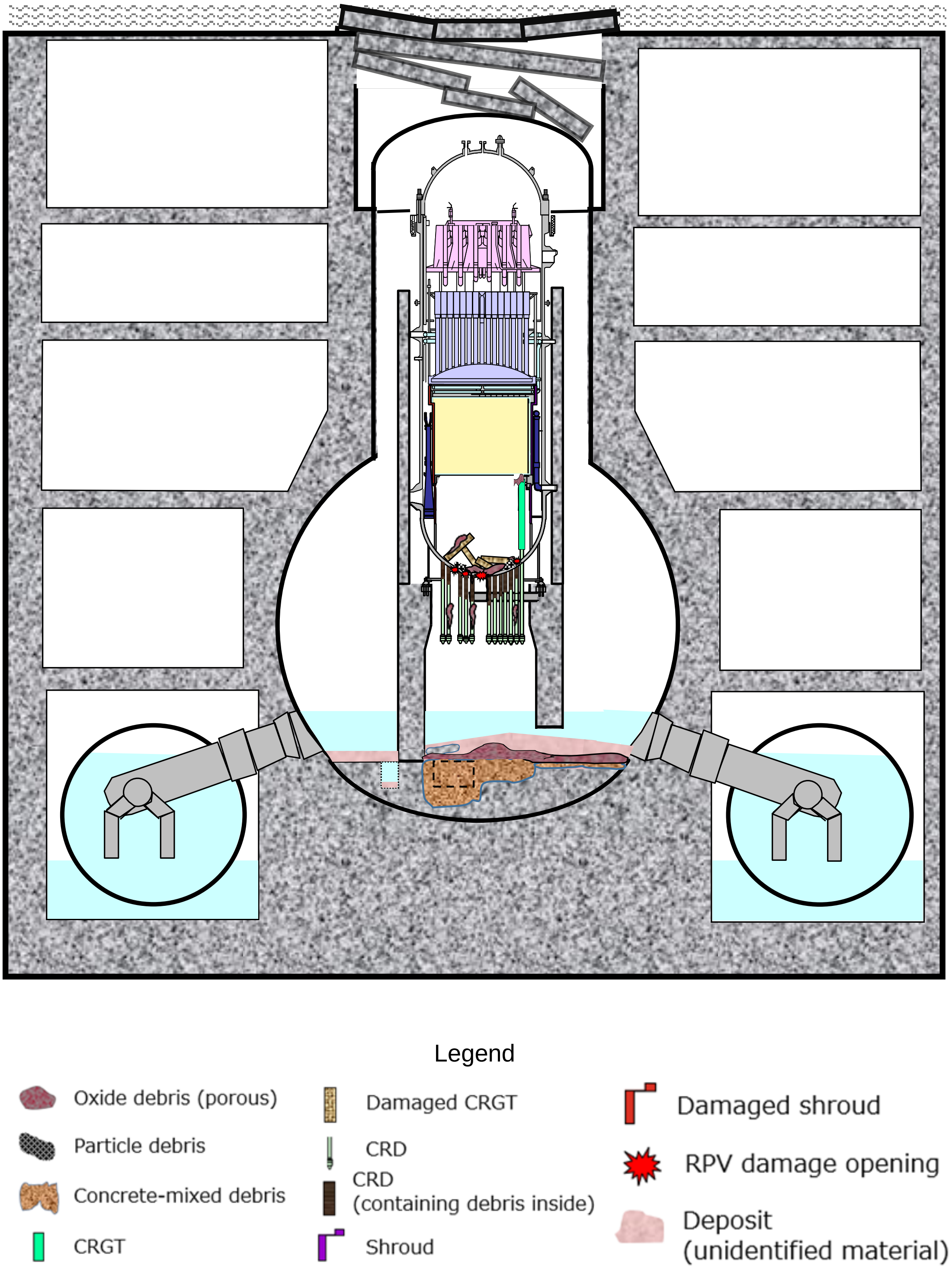
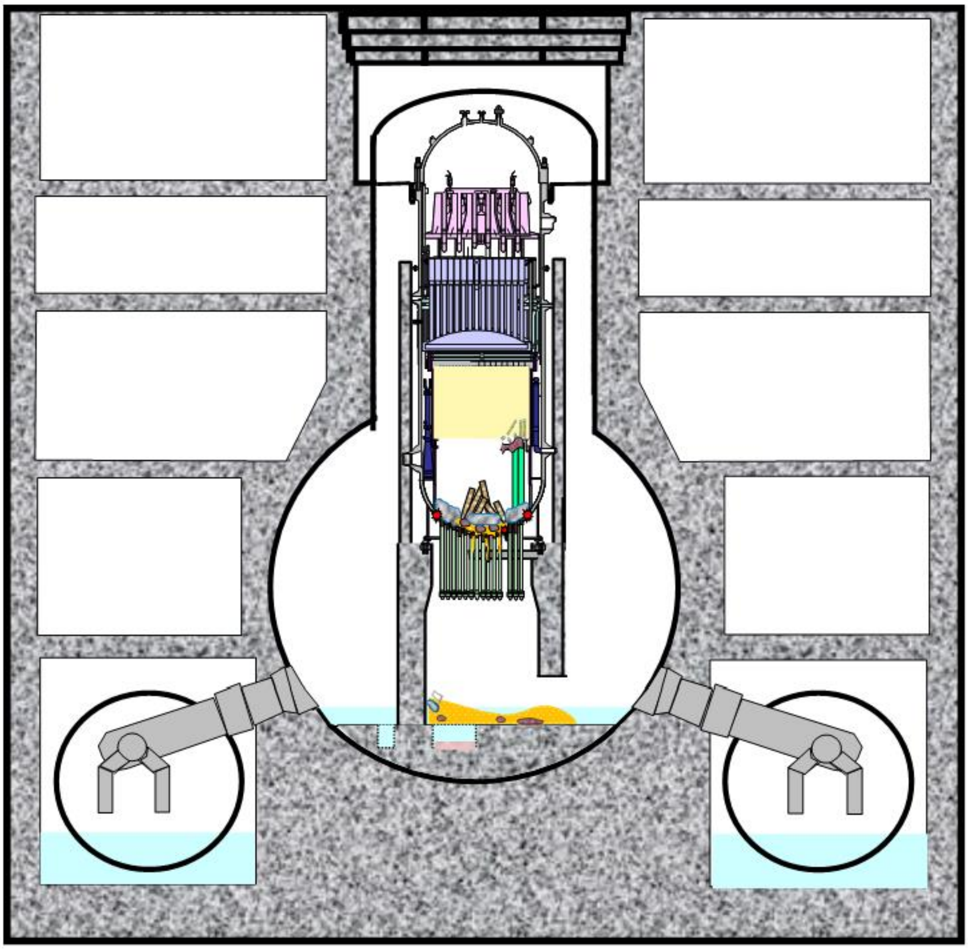


Figure 1-1 Estimated condition of the reactor core and containment vessel of Unit-1

(Note) The figures shown here are images and do not represent the quantitative reality of the size of the fuel debris.



Legend

- Oxide debris (porous)
- Particle debris
- Concrete-mixed debris
- CRGT
- Damaged CRGT
- CRD
- CRD (containing debris inside)
- Shroud
- Damaged shroud
- RPV damage opening
- Deposit (unidentified material)

Figure 1-2 Estimated condition of the reactor core and containment vessel of Unit-2

(Note) The figures shown here are images and do not represent the quantitative reality of the size of the fuel debris.

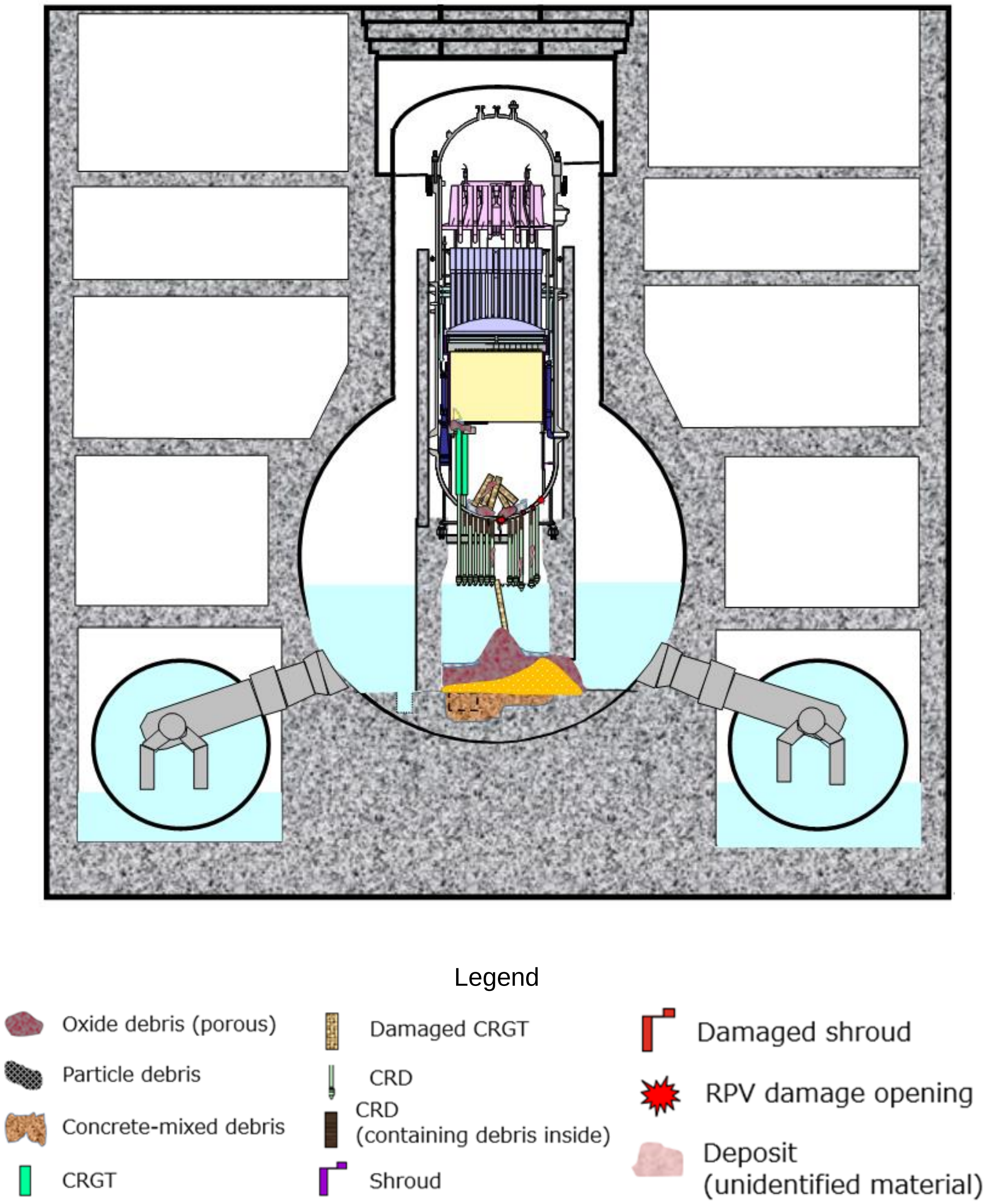


Figure 1-3 Estimated condition of the reactor core and containment vessel of Unit-3

(Note) The figures shown here are images and do not represent the quantitative reality of the size of the fuel debris.

2. Conditions of Unit-1 core and PCV

(1) In-containment water level measured

In October 2012, an investigation was conducted into the status of the PCV of Unit-1, when photos were taken by cameras, the level of water retained in the PCV was confirmed, dose rates and temperatures were measured, and retained water was sampled and analyzed [1] by inserting survey devices into the containment through a hole dug at the PCV penetration (X-100B, on the first floor of the reactor building). [2-1]

The level of water retained was measured by lowering the CCD camera cable down to the water surface through the grating above in the PCV. The water level was found to be about 2.8m above the D/W floor (as of October 10, 2012) (Figure 2-1).

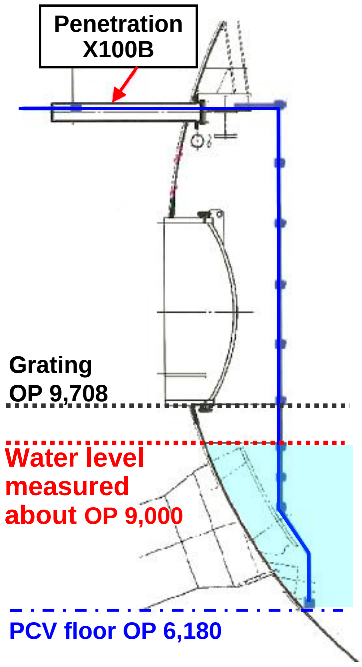


Figure 2-1 Measured level of water retained in Unit-1 PCV

(2) Test results of injecting nitrogen gas into the suppression chamber of Unit-1

In September 2012, a nitrogen gas injecting test was conducted into the suppression chamber (S/C), in which the theory was demonstrated that Kr-85 and hydrogen gas formed early in the accident were retained in the S/C upper space, pushed down the S/C water level and were discharged to the D/W through the vacuum breakers. This helped to confirm that the S/C was currently almost filled with water (the level at around the lowest end of the vacuum breaker tube (Figure 2-2). [2-2]

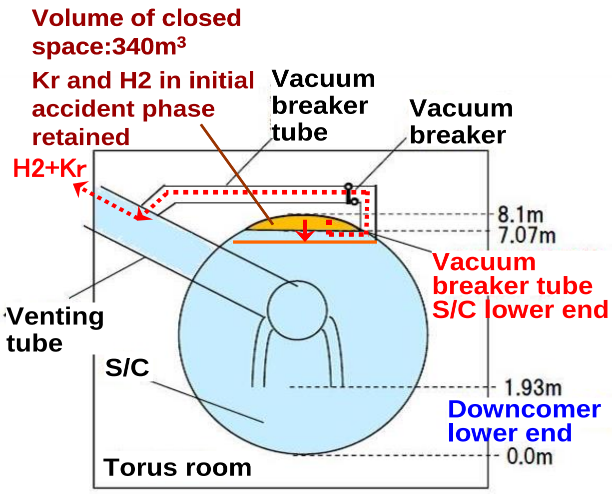


Figure 2-2 Status of the closed space in the gas phase section in S/C Unit 1

This 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 control system of Unit-1 that has been seen since April 2012. This intermittent increase was assumed to occur in the following sequence: When the S/C water level

[2-1] Handout document: Investigation results of Unit 1 Primary Containment Vessel (PCV) inside, 11th Steering Committee, Government – TEPCO Joint Board on Mid- and Long-term Response Policy, October 22, 2012.
[2-2] Handout documents: Nitrogen injection into Unit-1 S/C, 9th and 10th Steering Committees, Government – TEPCO Joint Board on Mid- and Long-term Response Policy, August 27 and September 24, 2012.

drops, residual gas left in the closed space in the upper S/C is discharged to the D/W through vacuum breakers, and then the S/C water level rises and stops the gas discharge. In this hypothesis, Kr-85 is understood to originate in the early phase of the accident, because Kr-85 is a long half-life fission product and its amount cannot be explained as being newly produced by spontaneous fission, etc.

In the test to verify the mechanism hypothesis, the S/C pressure (being monitored by the existing instrumentation) rose after the injection of nitrogen gas started into the S/C, the hydrogen gas concentration and Kr-85 radioactivity monitored by the containment gas control system started to increase, which decreased when nitrogen gas injection was halted. This is interpreted to be that the nitrogen gas injection 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 D/W through the vacuum breakers, thus the retained gas in the space was discharged together to the D/W by the injected nitrogen gas.

Most of the hydrogen gas retained in the S/C has been purged by continuously injecting nitrogen gas into the S/C since October 2012. Further tests are now underway to verify a mechanism of hydrogen production in the S/C by water radiolysis.

(3) Investigation of the torus room of Unit-1

The torus room was investigated in February 2013, when photos were taken by cameras, dose rates and temperatures were measured, and retained water was sampled and analyzed by inserting thermometers, dosimeters and cameras through a ϕ200 hole dug on the northeast corner on the first floor of the reactor building. [2-3]

Figure 2-3 Photo of an S/C vacuum breaker in the torus room of Unit-1 (part)

No water leaking position in the S/C has been located yet. At least, no leak was confirmed on the flange of one of the eight vacuum breakers, as far as the photos showed (Figure 2-3).

(4) Investigation of the situation at the bottom of the vent tubes in the torus room of 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

[2-3] Handout document: Progress in preparations for decommissioning of Units-1 to -4 of the Fukushima Daiichi Nuclear Power Station, Decommissioning Measures Steering Panel, March 7, 2013.

510 mm 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, to check visually the condition of the sand cushion drainpipes, and to make dose measurements.[2-4]

Camera imaging confirmed water leaks at the following locations (Figure 2-4).

- Vent tube X-5B (① in the figure): water flowing from the displaced sand cushion drainpipe*
- Vent tube X-5E (④ in the figure): water flowing down on the S/C surface with 2 streams around both sides of the vent tube

*Water leaks at ① were confirmed since the vinyl chloride pipe (connecting the sand cushion drainpipe and drain funnel with an insertion-type joint) had been displaced. Water leaks could not be confirmed at locations ② to ⑧, since the drainpipes had not been displaced. The concrete seams (joints) below the sand cushion drainpipe were observed to be wet all around on the concrete wall.

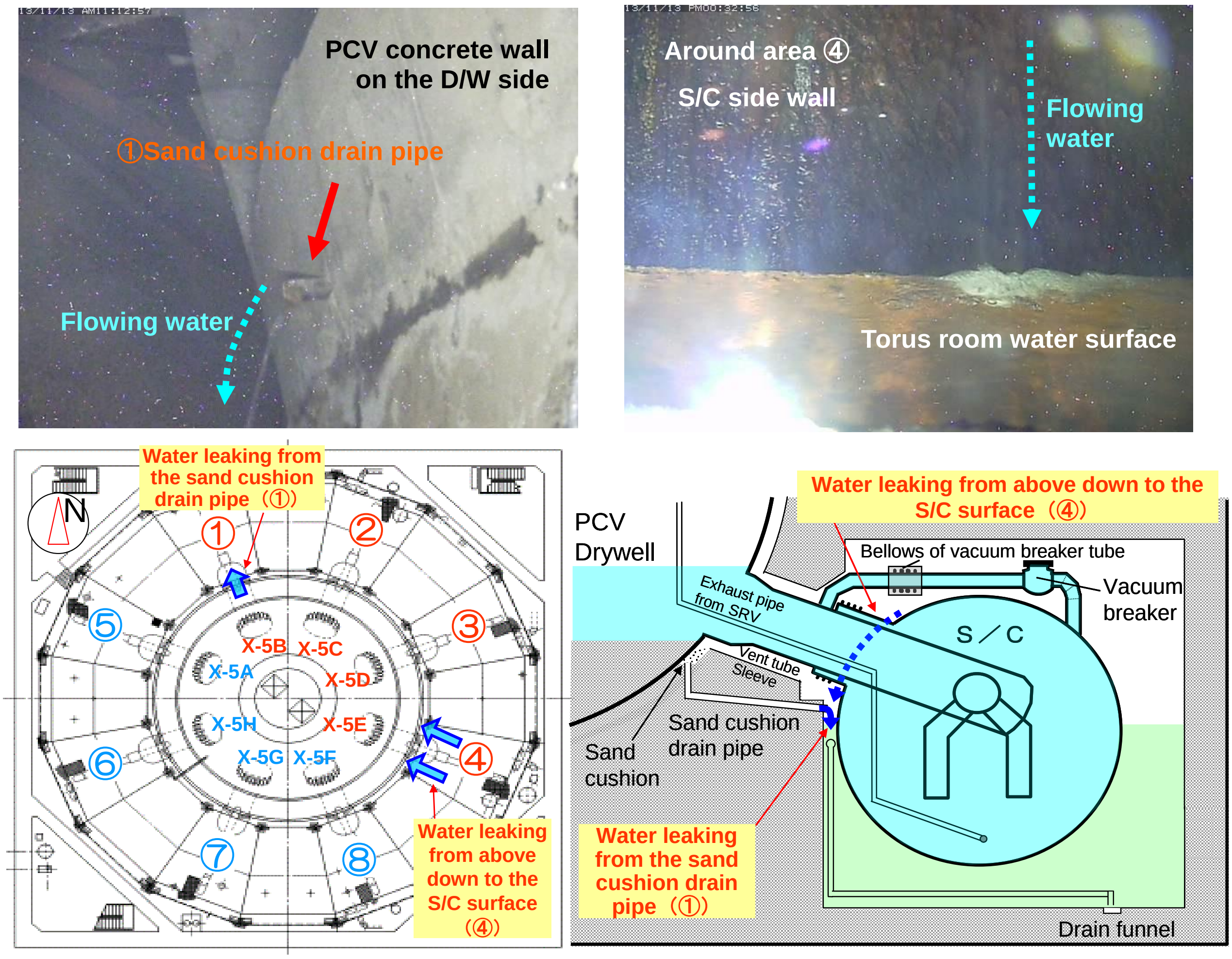


Figure 2-4 Camera images taken below the vent tubes in Unit-1 torus room (part)

[2-4] Handout document: Fukushima Daiichi Nuclear Power Station, Vent pipe lower area survey results of Fukushima Daiichi Nuclear Power Station Unit-1. About vent pipe lower area survey results of Fukushima Daiichi Nuclear Power Station Unit- 1, 10th Decommissioning Measures Steering Panel, November 28, 2013.

Water leaks into the sand cushion occur only when water leaks directly from the drywell. The leakage is probably from a low position of the drywell below the water level (for example, the containment vessel shell, pipe penetrations, etc.). The low location of the water leaks in the drywell would indicate the possible influence of molten fuel that fell to the PCV bottom. This information is of critical significance in estimating the conditions of the core and PCV.

Meanwhile, the water leakage down to the S/C surface around both sides of the vent tube X-5E indicates the possibility of water leaks from the vacuum breaker tube (its bellows, for example) immediately above the vent tube. This elevation of the vacuum breaker tube is about the same level as that of the upper limit of PCV water level which was reached in an attempt to flood the PCV by increasing the amount of water injected in May 2011; where the PCV water level was calculated from the injected nitrogen gas pressure. The nitrogen gas pressure was stopped from changing at a specified level, which means the PCV water level was leveled off, i.e., an indication of leak hole existence at the level (Figure 2-5).[2-5]

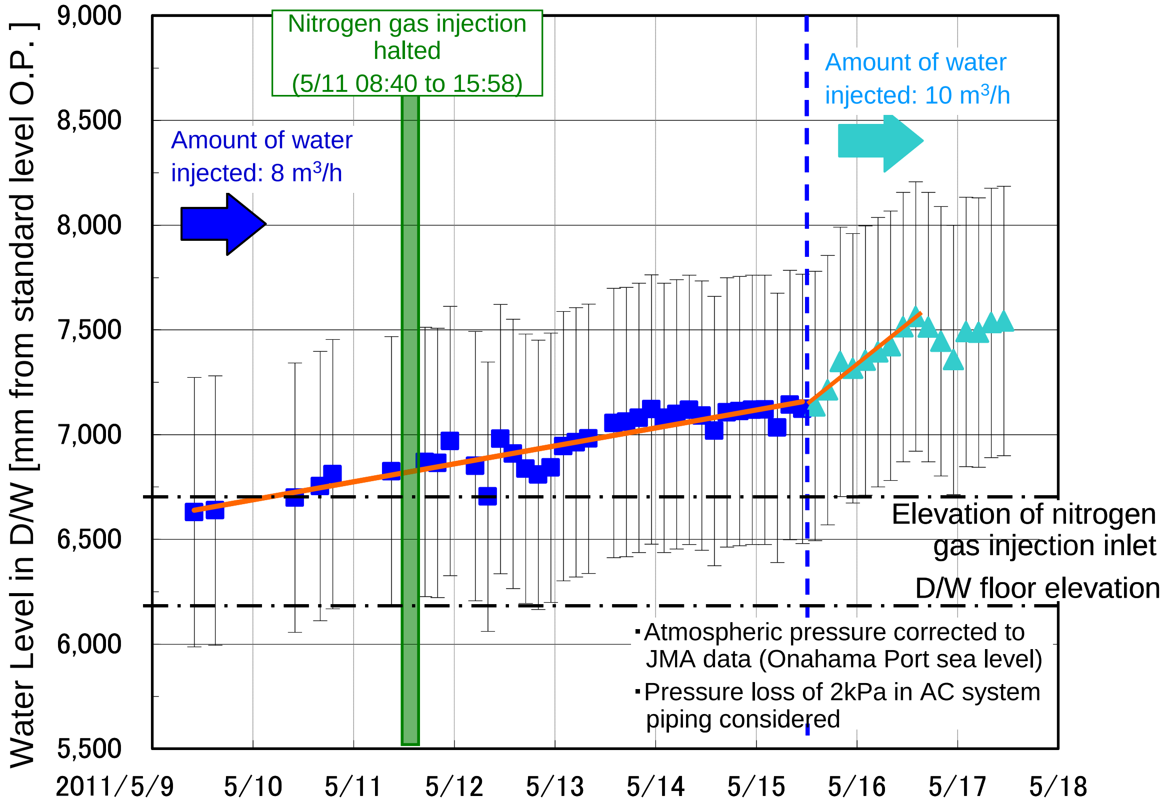


Figure 2-5 Estimated PCV water level changes during flooding operation of Unit-1 (May 2011)

The vertical distribution of radiation doses measured when lowering the instrumentation boat in November 2013 was similar to that measured in February 2013 (in the area surrounding the torus). Dose distribution along the boat traveling route was about 1 to 2 Sv/h, and the highest spots were in the southeast area (Figure 2-6). [2-5]

[2-5] Handout document: Long-term cooling strategy team, Special Project, May 19, 2011.

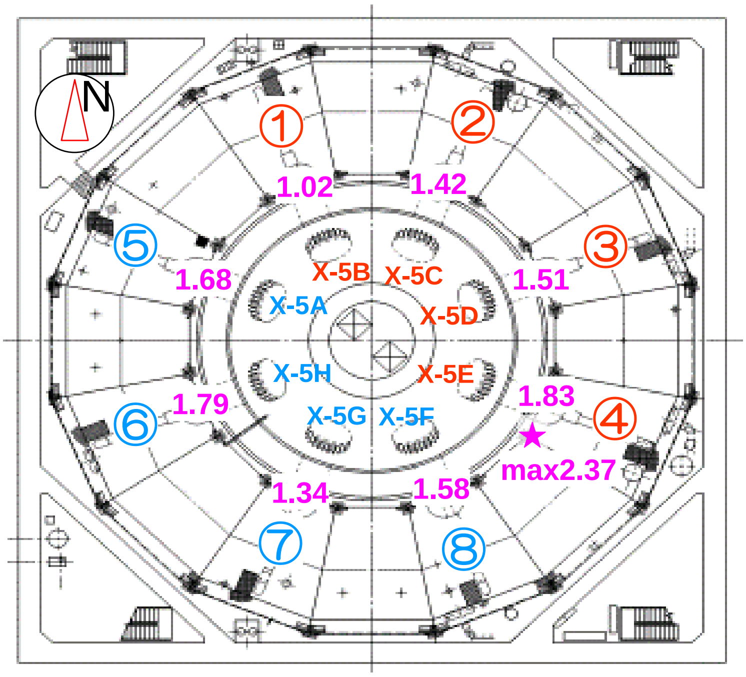


Figure 2-6 Dose rate distribution measured during the torus room investigation underneath vent tubes of Unit-1

In June 2011, steam blown from the pipe penetrations was witnessed at the southeast corner of the first floor of the reactor building. This would mean that radioactive materials carried by the steam were blown to the torus room after the accident and they deposited on the walls and structure surfaces there. The dose rate in the torus room is considered to be the sum of dose rates due to these contamination sources. The estimated dose rate on the water surface due to water–retained radionuclides (7.3×10^4 Bq/cm^3 of Cs-134 and 1.5×10^5 Bq/cm^3 of Cs-137, sampled on February 22, 2013) are about 100 mSv/h and not a dominant contributor to the 1 to 2 Sv/h dose rates measured. [2-6]

In May 2014, survey instrumentation robot was introduced through a drilled hole in the northwest area of the first floor of the Unit-1 reactor building to explore the S/C top area in order to locate the leak source near the vent tube X-5E, where leaking had been confirmed. 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 line. No leaks were noticed from the vacuum breaker valve, torus hatch, shutdown cooling system (SHC) piping or atmospheric control system (AC) piping (Figure 2-7).[2-7]

[2-6] Handout document: Consideration of dose measurement results in the torus room of Unit-1, 7th Meeting of Specific Nuclear Facilities Survey and Examination, March 29, 2013.
[2-7] Handout document: Conducting a demonstration test of the S/C (pressure suppression chamber) upper part inspection device currently under development as part of the research and development project "Development of Technology for Identifying and Repairing Leakage Points in Containment Vessels," 1st Meeting of the Decommissioning and Contaminated Water Response Team Joint Meeting, May 29, 2014.

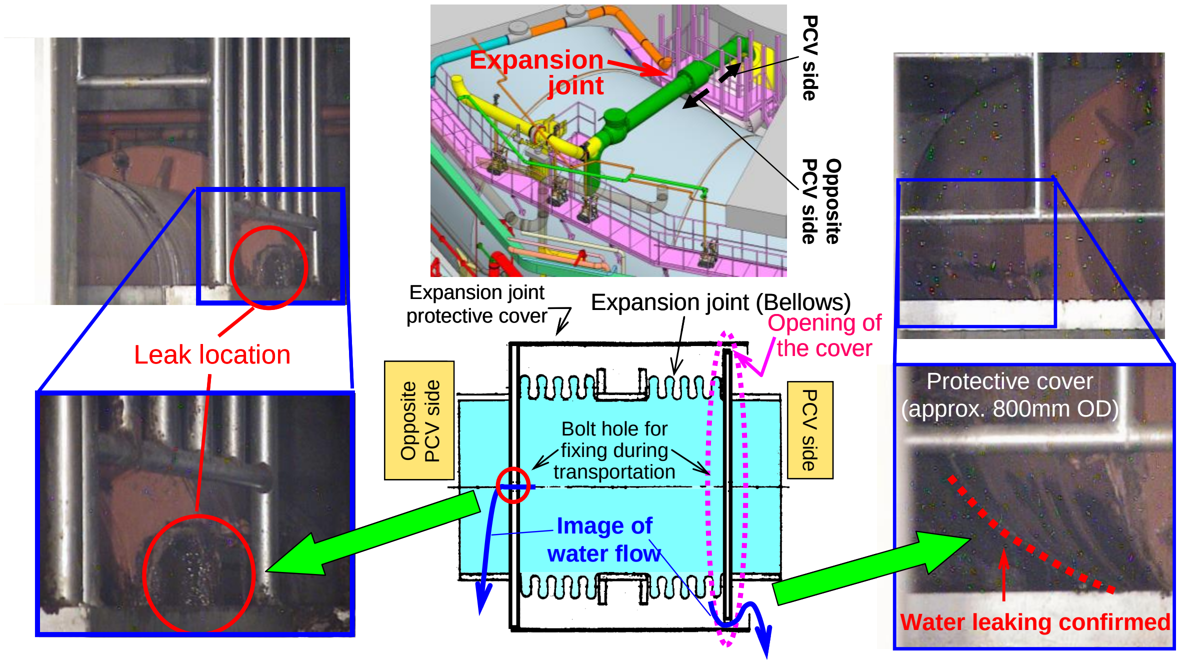


Illustration of Expansion joint (Bellows) for vacuum breaker tube

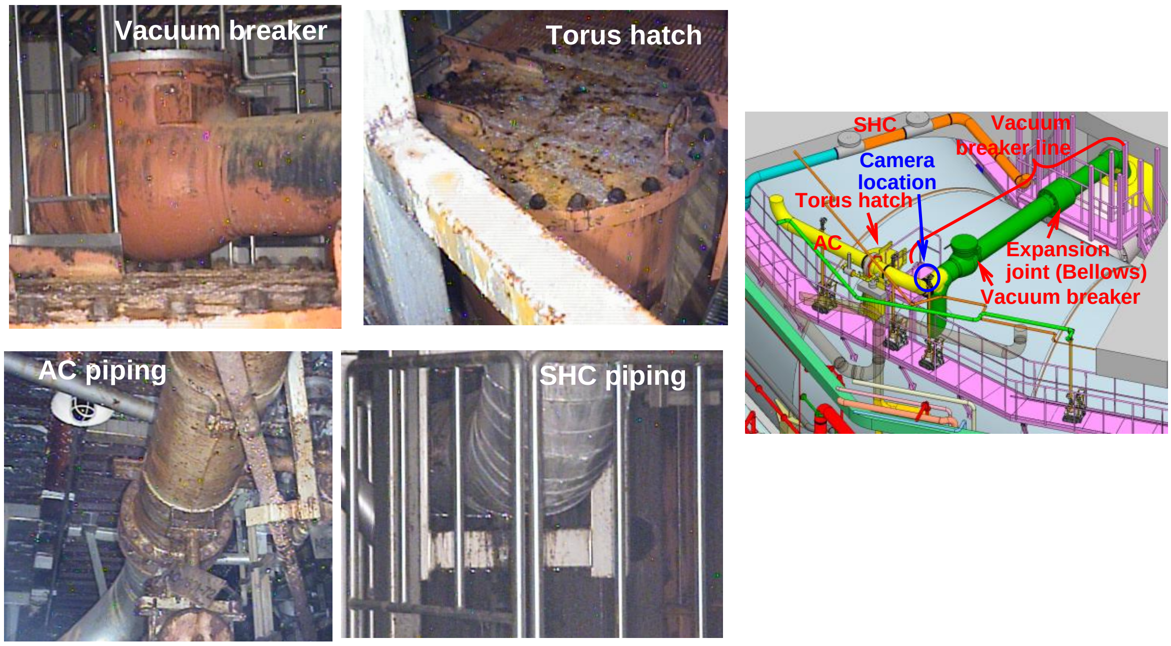


Figure 2-7 Camera images of Unit-1 S/C top area exploration (around vent tube X-5E) (part)

(5) Contamination survey on the Unit-1 reactor building first floor

In December 2013, contamination was surveyed on the first floor of the Unit-1 reactor building by dose rate measurements and gamma camera images taken using a robot. It was found that contamination was relatively high on the atmospheric control system (AC) piping and drywell humidity control system (DHC) piping (Figure 2-8). [2-8]

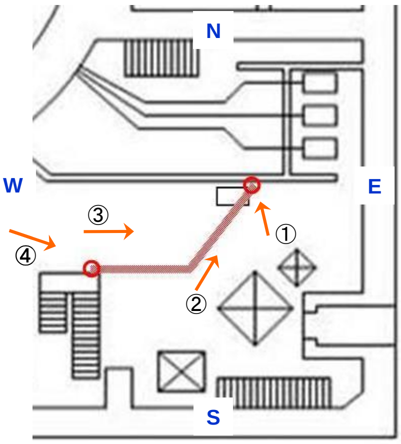


Figure 2. AC system piping route (about 2m elevation)

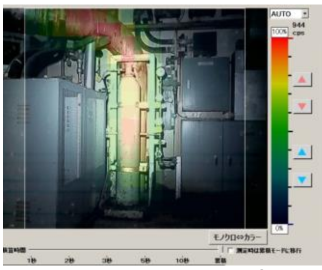


Figure 3. Photo from ①

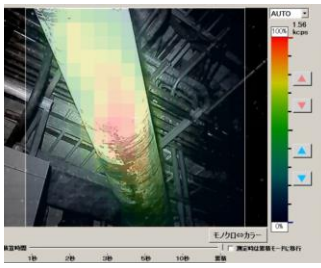


Figure 4. Photo from ②


Figure 5. Photo from ③


Figure 6. Photo from ④

(Gamma camera image taken near the AC system piping in the reactor building 1st floor)

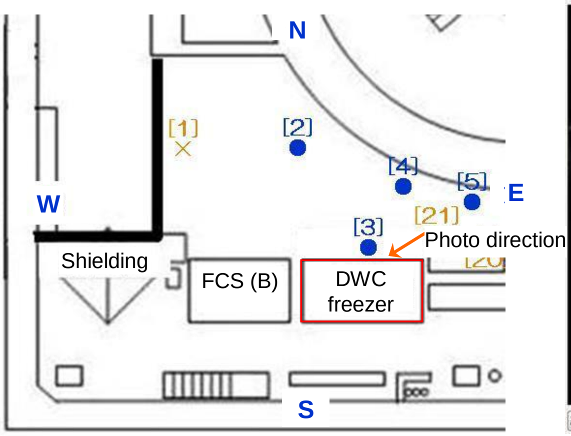


Figure 9. Layout

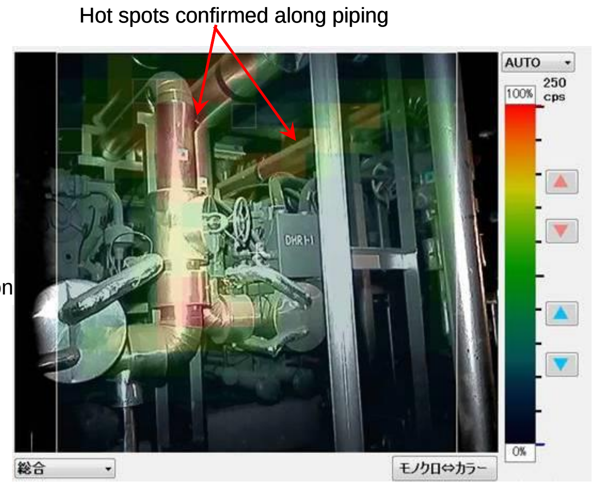


Figure 10. Gamma camera image taken near the DWC freezer

(Gamma camera image taken near the DWC system piping in the reactor building 1st floor)

Figure 2-8 Gamma camera images in the southern area of Unit-1 reactor building 1st floor (part)

The AC piping is where the steam passed through when the wetwell (W/W) venting was carried out during the accident. Its high contamination is considered to be due to venting flows, and the situation was similar to that for the area near the standby gas treatment system (SGTS) train

[2-8] Handout document: National project "Development of remote decontamination technology for reactor buildings" Fukushima Daiichi Nuclear Power Station Unit 1 reactor building 1st floor south side investigation results (preliminary report), 2nd Decommissioning and Contaminated Water Response Team Joint Meeting, January 30, 2014.

entrance room or near the SGTS piping connected to the main stack, where high dose rate had been already confirmed.

The DHC piping is connected to the reactor building closed cooling water system (RCW), and therefore its high dose rate is considered to be due to the same mechanism as that of RCW piping, where high dose rate had been already confirmed.

(6) Investigation of the grating floor on the ground level outside the pedestal of Unit-1

The grating floor on the ground level outside the pedestal of Unit-1 was investigated from April 10 to 18, 2015. A running robot machine was sent through the PCV penetration X-100B. The robot patrolled clock-wise and anti-clock-wise by about 180 degrees for investigating damaged conditions of existing structures and presence of obstacles. Figure 2-9 and Figure 2-10 present a part of the photos taken during the anti-clock-wise and clock-wise patrols, respectively. As seen in Figure 2-9, no big damage was recognized on the HVH (heating ventilating handling unit), PLR piping, pedestal walls, PCV inner walls, etc., although fallen objects were noticed on the patrol path.

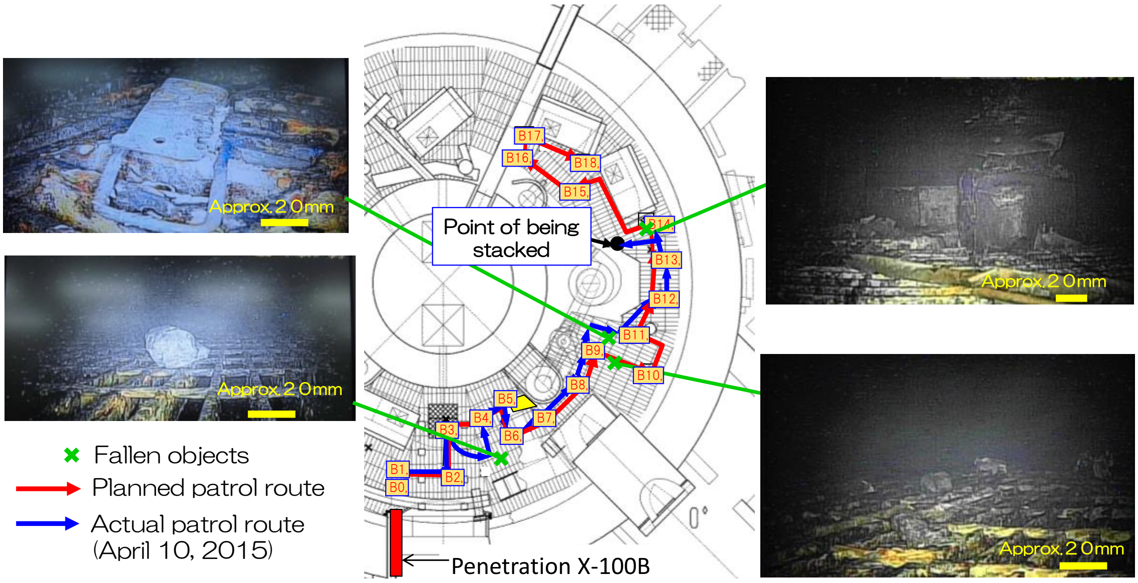


Figure 2-9 Photos taken during the anti-clock-wise patrol [2-9]

[2-9] Handout document: Results of the field demonstration test of the investigation on the grating on the first floor outside the pedestal (B1 survey) of the "Development of technology for investigating the inside of the reactor containment vessel", 17th Decommissioning and Contaminated Water Response Team Joint Meeting, April 30, 2015.

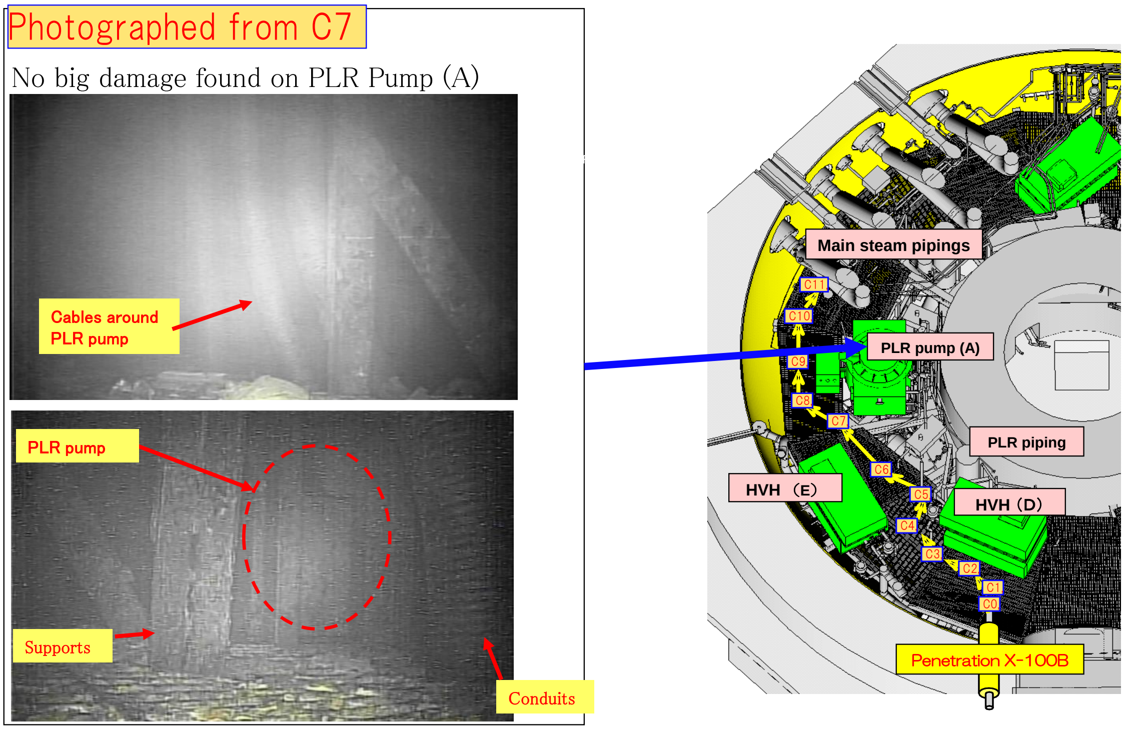


Figure 2-10 Photos taken during the clock-wise patrol [2-9]

(7) Investigation of Unit-1 using the muon tomography measurement device

A fluoroscope technology for nuclear reactors (transmission method) using muons is being developed jointly by the International Research Institute for Nuclear Decommissioning (IRID) and the High Energy Accelerator Research Organization (KEK) of Japan, as a project within the "Development of detecting technologies of fuel debris in a nuclear reactor" under the "Project of Decommissioning and Contaminated Water Management in the FY2013 Supplementary Budget" sponsored by the Agency for Natural Resources and Energy (ANRE). Figure 2-11 [1] shows the points of measurement on the Unit-1 reactor building Floor 1. At Point 1 and Point 2 the data were collected for 96 days from February 9 to May 21, 2015, and at Point 3 the data were collected for 106 days from May 25 to September 7, 2015. The conditions in the reactor were evaluated from the collected data.

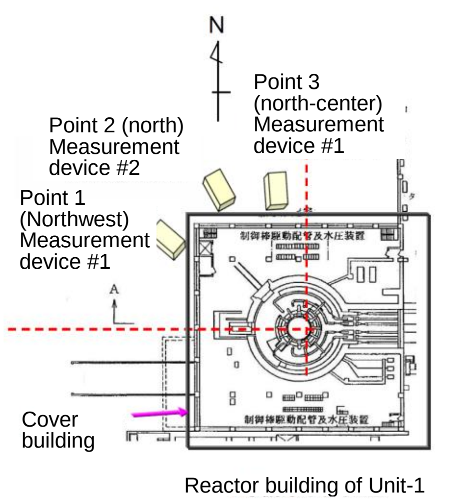


Figure 2-11 Arrangement of measurement devices [2-10]

Figure 2-12 shows the image of the reactor predicted from the design (left) and the image of the actual reactor obtained by muon measurements for 96 days (right), both at Point 1. The basic principle of measurements by the muon transmission method is the same with that of

[2-10] Final Report on Development of Fuel Debris Detection Technology under the government subsidized project "Decommissioning and Contaminated Water," December 2015.

X-ray (Roentgen) photography. High density objects absorb more muons and are photographed in black. In the image of the reactor as designed with intact fuel, a black portion is recognized corresponding to the position of the core. In the image of the actual reactor, on the other hand, some recognizable equipment such as the spent fuel pool and isolation condensers are found, but there is no high density object (fuel) at the core position.

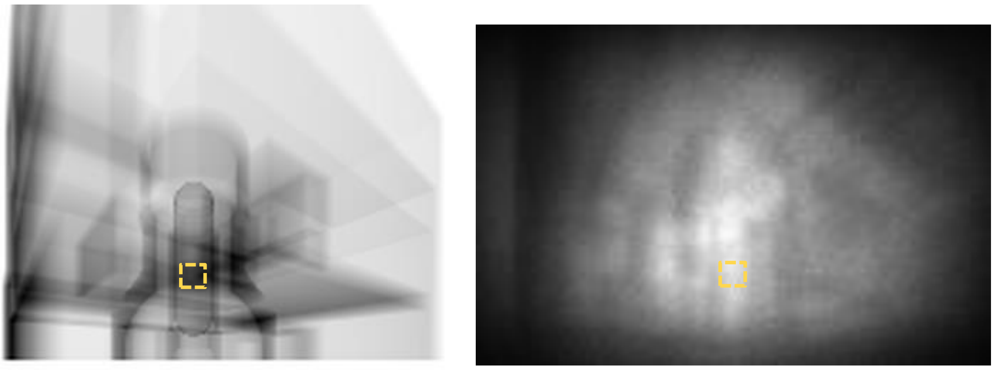


Figure 2-12 Muon image of the reactor predicted from the design (left) and that of the actual reactor based on measurements for 96 days (right) [2-11]
(dotted region corresponds to the original core position)

By combining data measured using three muon measurement devices, images can be restructured in three dimensions. Distribution maps of high density materials are shown in Figure 2-13 at different elevations of the reactor building (R/B). The red region on the distribution maps is where high density materials are detected by the two muon measurement devices at each elevation. High density materials are recognized at the spent fuel pool position, but not at the core position.

From these considerations, no fuel is thought to be left at the core position. This is consistent with the estimated conditions of the core and PCV previously announced by TEPCO.

[2-11] Handout document: Completion report on completion of development measurement work for in-reactor fuel debris detection technology (preliminary report), 18th Decommissioning and Contaminated Water Response Team Joint Meeting, May 28, 2017.

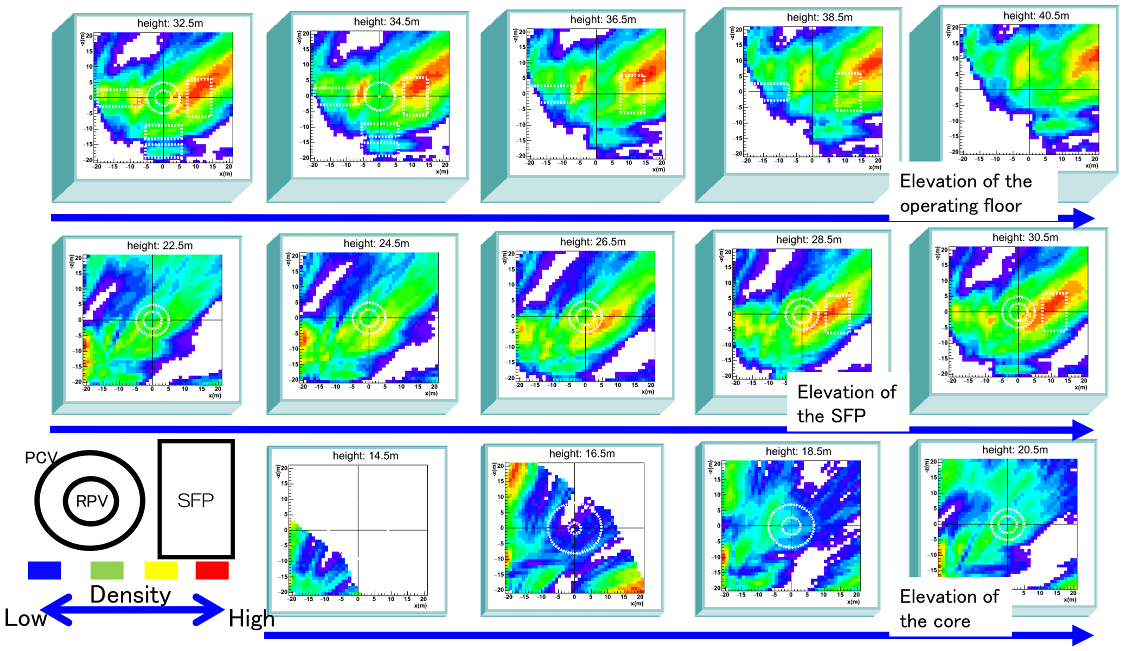


Figure 2-13 High density material distribution maps at three separate elevations [2-10]

(8) Investigation of the TIP room of Unit-1

The TIP (traversing in-core probe) room on the R/B ground floor was investigated from September 24 to October 2, 2015. This investigation was implemented to check feasibility of reducing the radiation dose near the PCV penetration X-6, stopping the water in the lower part of the PCV, repairing the PCV, etc.

Figure 2-14 presents the air dose rates measured in the TIP room and Figure 2-15 shows a γ-camera photo taken there. A high dose rate of above 100 mSv/h was observed near the PCV penetrations, especially around X-31, -32 and -33, while on the turbine side behind the chamber shield the dose rate was as low as below 2 mSv/h. The γ-camera photos located a radiation source near the penetrations X-31, -32 and -33 (Region 1 in Figure 2-15). No other outstanding radiation source was found in the room, including at the penetrations X-35A to -35D which were outside the view of the wide-angle lens camera (the area encircled by the broken line in Figure 2-15).

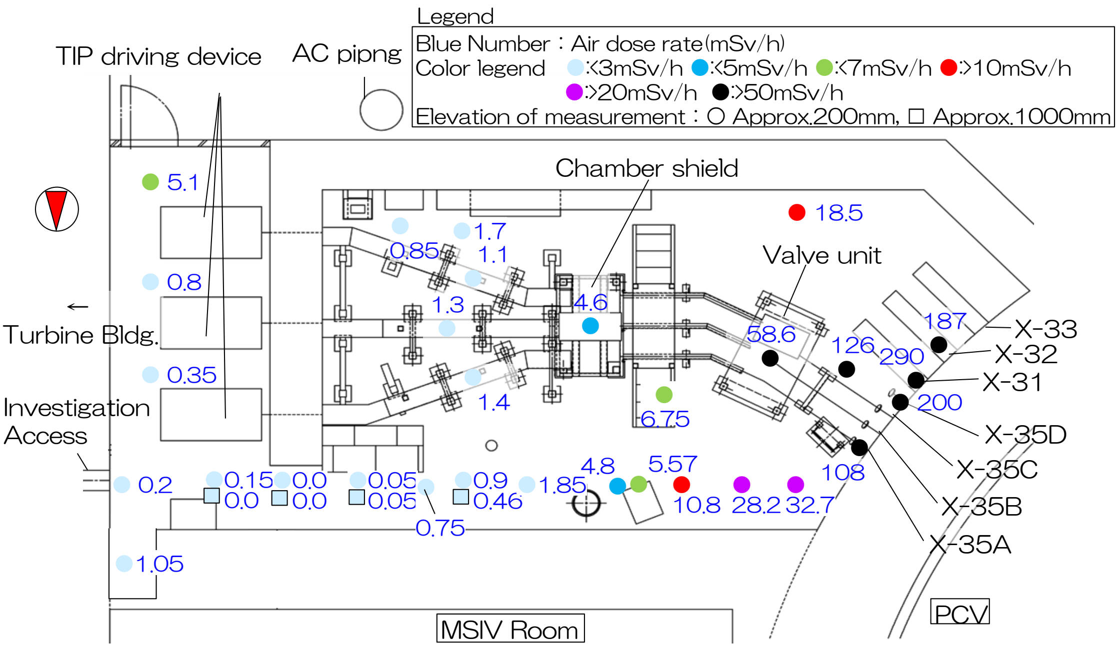


Figure 2-14 Air dose rates distribution measured in the TIP room [2-12]

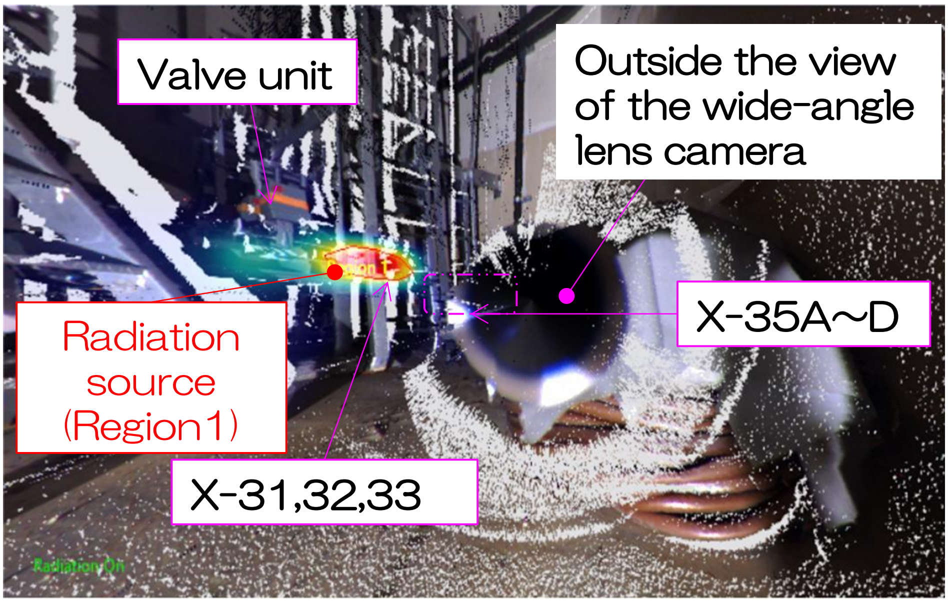


Figure 2-15 γ-camera photo taken in the TIP room [2-6]

Exterior appearances of PCV penetrations, piping and other objects in the TIP room were surveyed by an optical camera. Some brown-colored specks hinting at flow lines were noticed around PCV penetration X-35A (Figure 2-16). But, as mentioned above, no radiation source was noticed around the area of penetrations X-35A to X-35D. No other marks hinting at leaks were recognized at any places around the penetrations, piping and other objects in the TIP room including the penetrations X-31, -32 and -33.

[2-12] Handout document: Results of the TIP room survey among the small rooms on the first floor of the Unit-1 reactor building, 23rd Decommissioning and Contaminated Water Response Team Joint Meeting, October 29, 2015.

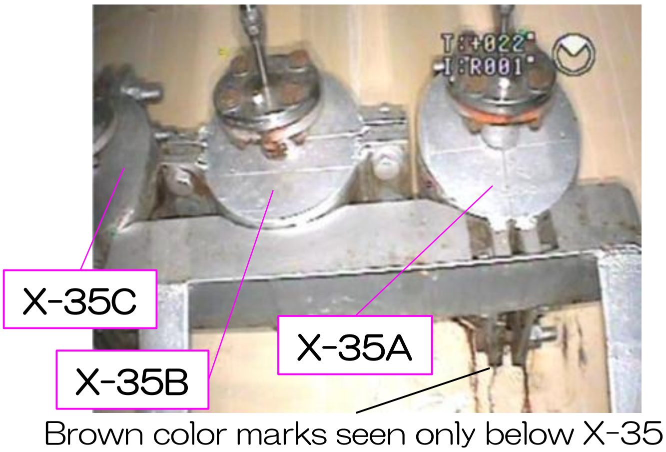


Figure 2-16 Optical camera photo taken at the penetration X-35 [2-6]

(9) Investigation inside the PCV (basement floor outside pedestal) of Unit-1

An investigation was conducted in March 2017 using a self-propelled investigation device to find the spreading area of fuel debris of the basement floor outside the pedestal and whether the debris had reached the PCV shell. Figure 2-17 shows the survey area and survey schematics. The self-propelled investigation device was operated on the Floor 1 grating floor, and video cameras and dosimeters were lowered to observe the PCV bottom and outside the pedestal at the survey points shown in Figure 2-18.

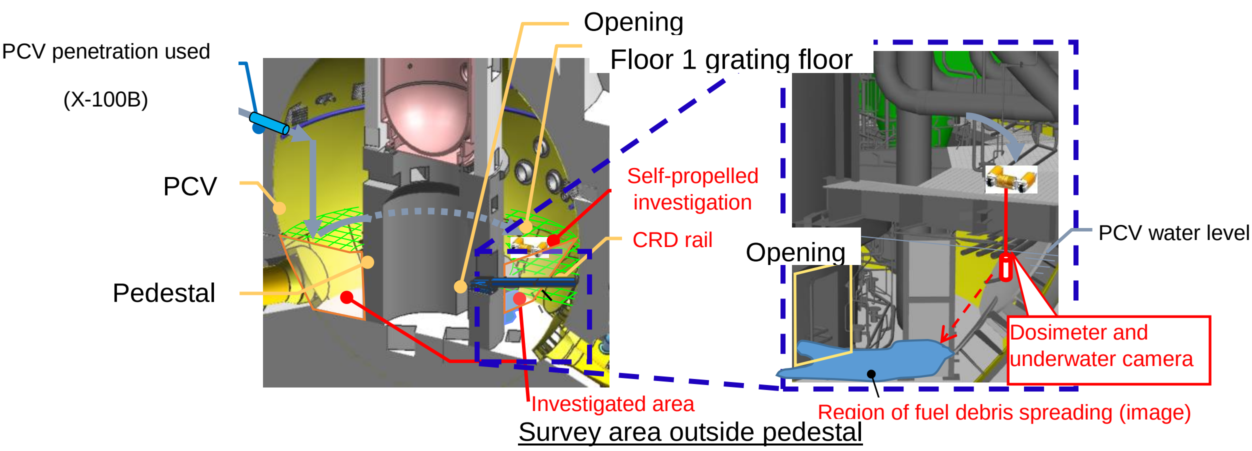
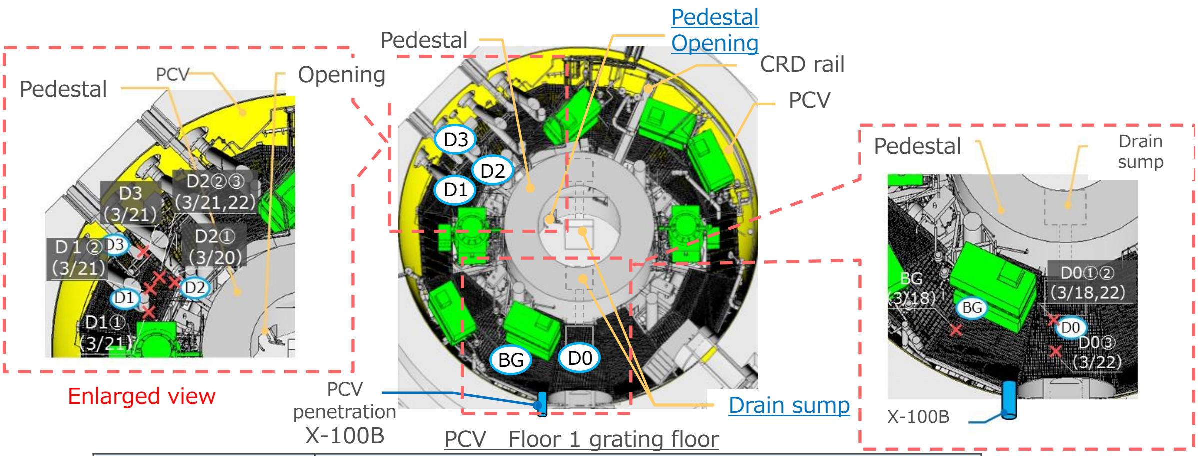


Figure 2-17 Survey area and survey schematics [2-13]

[2-13] Hand-out document: Investigation Plan of Inside Reactor Containment Vessel, 53rd Session of the Committee on Supervision and Evaluation of the Specific Nuclear Facility, May 22, 2017.



Point of measurement	Purpose of measurement
D0①～③	Observation of debris spreading from drain sump
BG	Background measurement for the data at D0 to D3
D1①②, D2①～③	Observation of debris spreading from pedestal opening
D3	Observation of debris spreading to pedestal shell

Figure 2-18 Survey positions [2-14]

Figure 2-19 presents the observation results by cameras from Point D0 near the drain sump outside the pedestal. No big damage or collapse of structures around the drain sump can be confirmed.

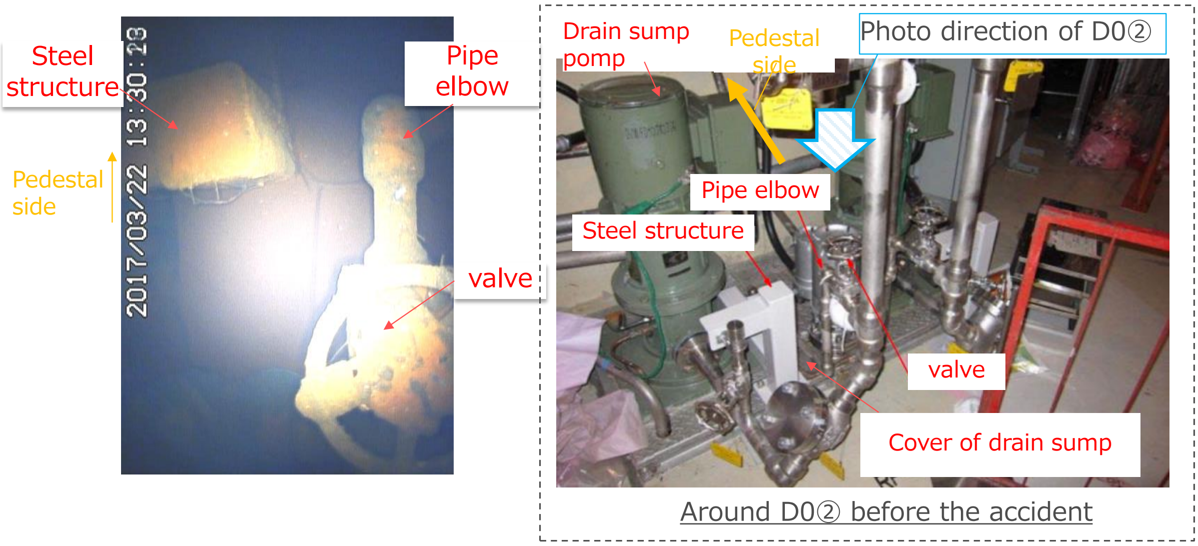
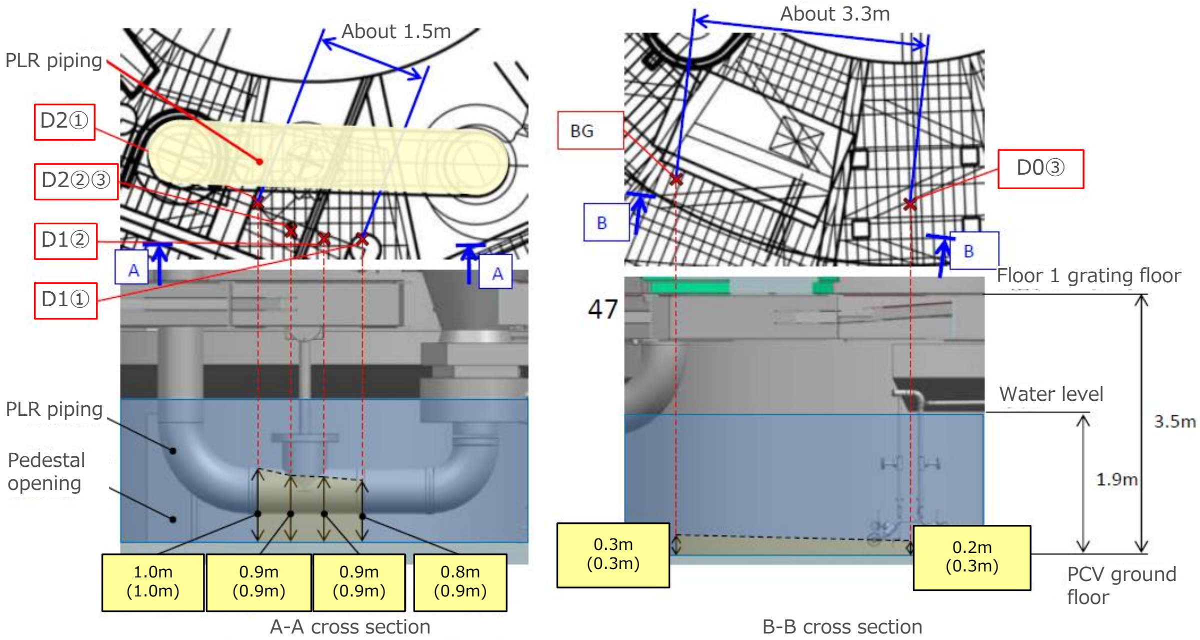


Figure 2-19 Observation from Point D0② [2-7]
(Left, photo from Point D0②; right, view around Point D0② before March 11, 2011)

The sediment height at the PCV bottom was estimated (Figure 2-20) using the distance that the sensor was lowered from the self-propelled investigation device, and the distance between the sensor and sediment obtained by analyzing the photo image. It was about 0.8 to 1.0 m at Points

[2-14] Handout document: Investigation inside the containment vessel of Unit-1 reactor - Analytical results of video data and dose data-, 44th Decommissioning and Contaminated Water Response Team Joint Meeting, July 27, 2015.

D1 and D2 near the pedestal opening, and 0.2 to 0.3 m at Points BG and D0 away from the pedestal opening. The sediment height was confirmed to be higher closer to the pedestal opening than at a distance from the pedestal opening.



Height calculated from sensor position

※1: Sediment heights estimated from the distance between the sensor and sediment using the structure from motion (SFM) technique and the distance the sensor was lowered

- Numbers in the yellow frames are the sediment heights estimated from photo image analysis
- Numbers in parentheses are the lowest elevation of the sensor
- Sediment thicknesses below the surface are not confirmed

Figure 2-20 Estimated sediment heights [2-7]

Dose measurements focused on the relationship between the distance from the sediment surface and its dependence on the dose rate attenuation to investigate the presence of fuel debris under the sediment. The results are seen in Figure 2-21. The blue points in Figure 2-21 (left), representing dose rate measurements at Point BG, confirmed the good agreement with the results of analysis (red line) of a situation when Cs-137 had existed in the sediment surface layer and no fuel debris was in underlayers. It was estimated, therefore, that no or very little fuel debris was present in the sediment underlayers. The same thing was also confirmed at Point D0③.

Figure 2-21 (right) compares, on the other hand, the dose rate measurements at Point D2③, and the results of analysis of a situation when 0.9 m, 0.3 m or 0.1 m thick sediments were assumed to cover the fuel debris. A big difference was evident in the dose rate attenuation behavior between the cases of 0.1 m or 0.3 m thick sediments, but the difference was very minor in the cases of 0.3 m or 0.9 m thick sediments. After all, it could not be confirmed whether fuel debris was present underneath the sediments. This was the case, too, at Points D1 and D2.

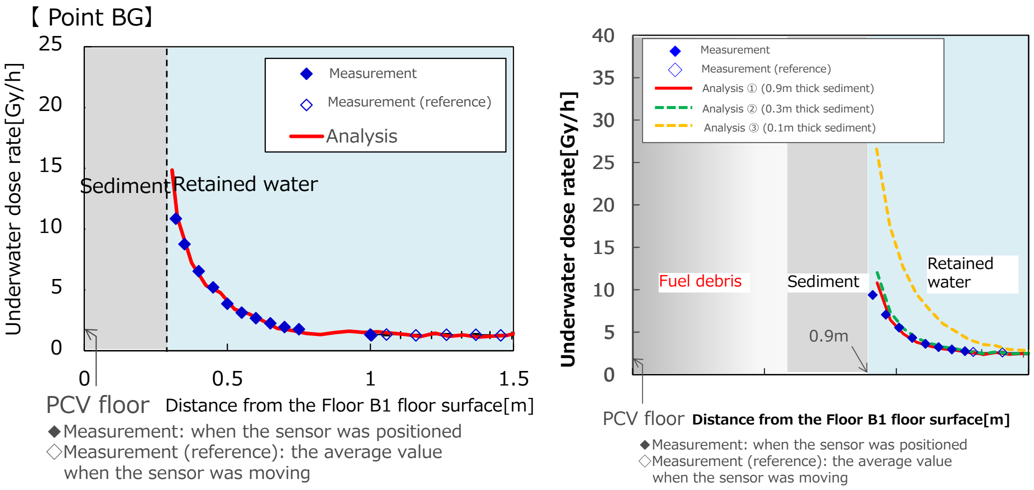


Figure 2-21 Dose rate measurements (Left: at Point BG, Right: Point D2③) [2-7]

The sediment at the PCV bottom was sampled (Figure 2-22). The γ-ray emitting nuclides were analyzed, and the results are given in Table 2-1. The simplified fluorescent X - ray analysis identified, in the sediment, Fe and Ni in stainless steel used in core structures, thermal insulators or other parts, Zn contained in painting materials, and Pb contained in shielding materials.

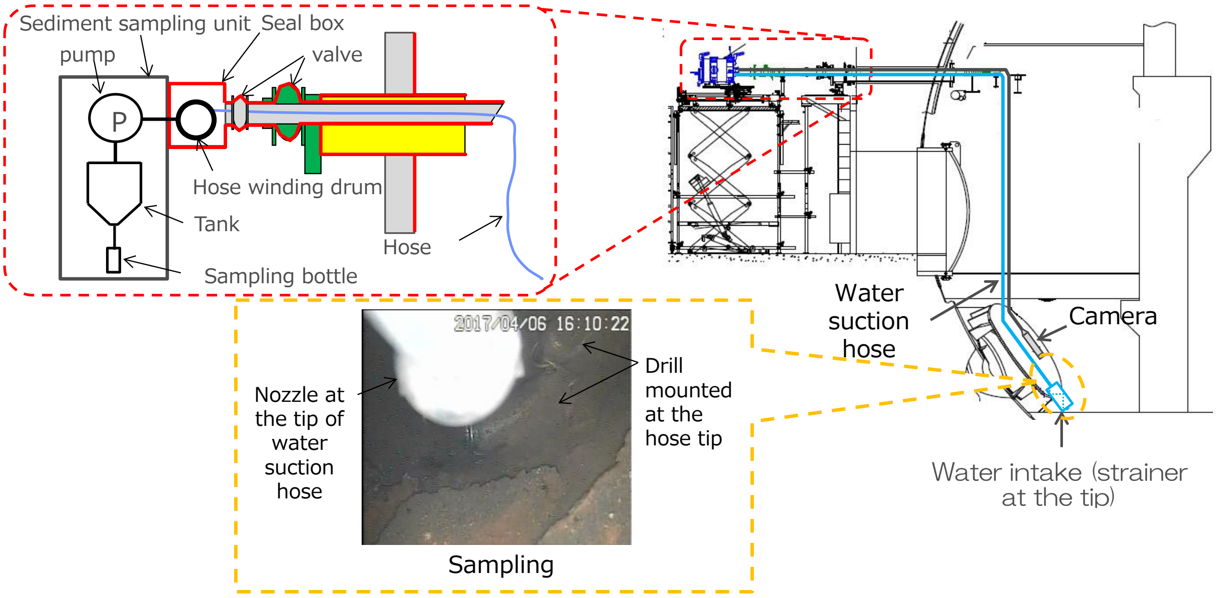


Figure 2-22 Schematic diagram of sediment sampling [2-15]

[2-15] Handout document: Investigation inside the containment vessel of Unit-1 - Results of analysis of sediments -, 42nd Decommissioning and Contaminated Water Response Team Joint Meeting Panel, May 25, 2017.

Table 2-1 γ -ray emitting nuclides analysis [2-15]

Gamma emitting nuclides detected	Radioactivity [Bq/g]
Cs-134	3.5E+06
Cs-137	2.7E+07
Co-60	1.1E+07
Sb-125	7.0E+05

(10) Operating floor survey of Unit-3 reactor building

The north side of the Unit-1 operating floor has been surveyed since November 2016. One of the survey items is the reactor well-plug. The well-plug was a three-layered structure of upper, middle and bottom layers and each layer was composed of three concrete slabs.

Figure 2-23 shows the well-plug damage conditions as observed on the operating floor. The position of each well-plug is estimated to have moved, as shown in Figure 2-24. The north piece of the upper well-plug was found to have moved to the west by about 720 mm. The center and north pieces of the upper well-plug were found to have been deformed downward by a maximum of 155 mm and 84 mm, respectively.

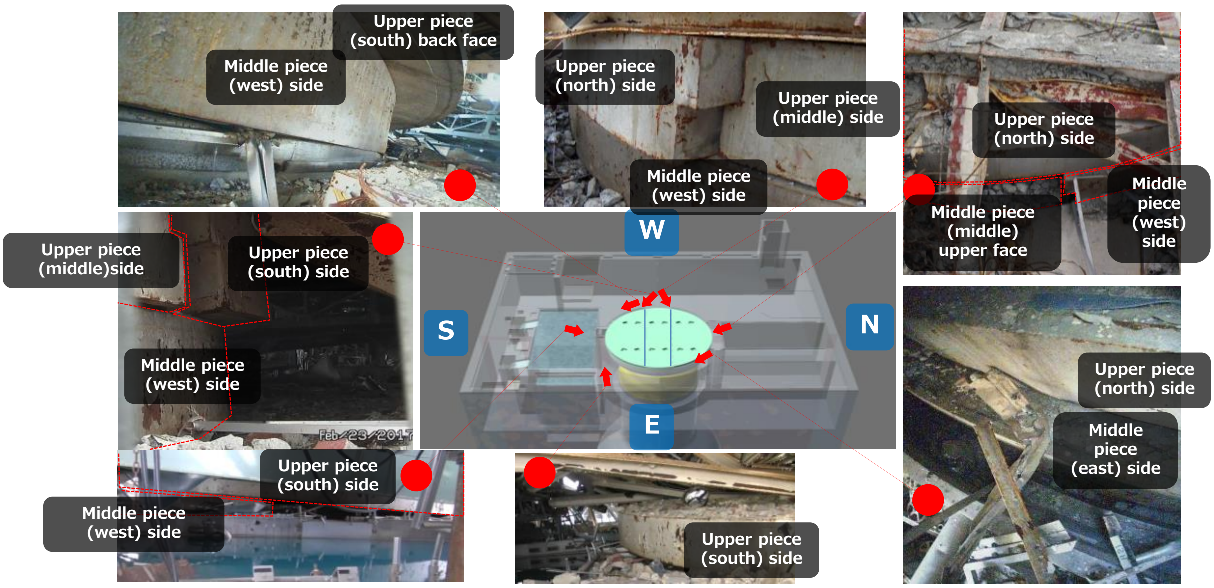
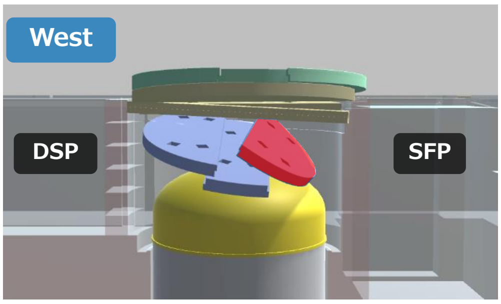


Figure 2-23 Well-plug pieces as they were observed on the operating floor [2-16]

[2-16] Handout document: Action progress towards decommissioning of Units 1 to 4 of Fukushima Daiichi Nuclear Power Station, 40th Decommissioning and Contaminated Water Response Team Joint Meeting, March 30, 2017.



SFP: Spent Fuel Pool
DSP: Device Storage Pool

Figure 2-24　An image layout of well-plugs estimated from investigation [2-17]

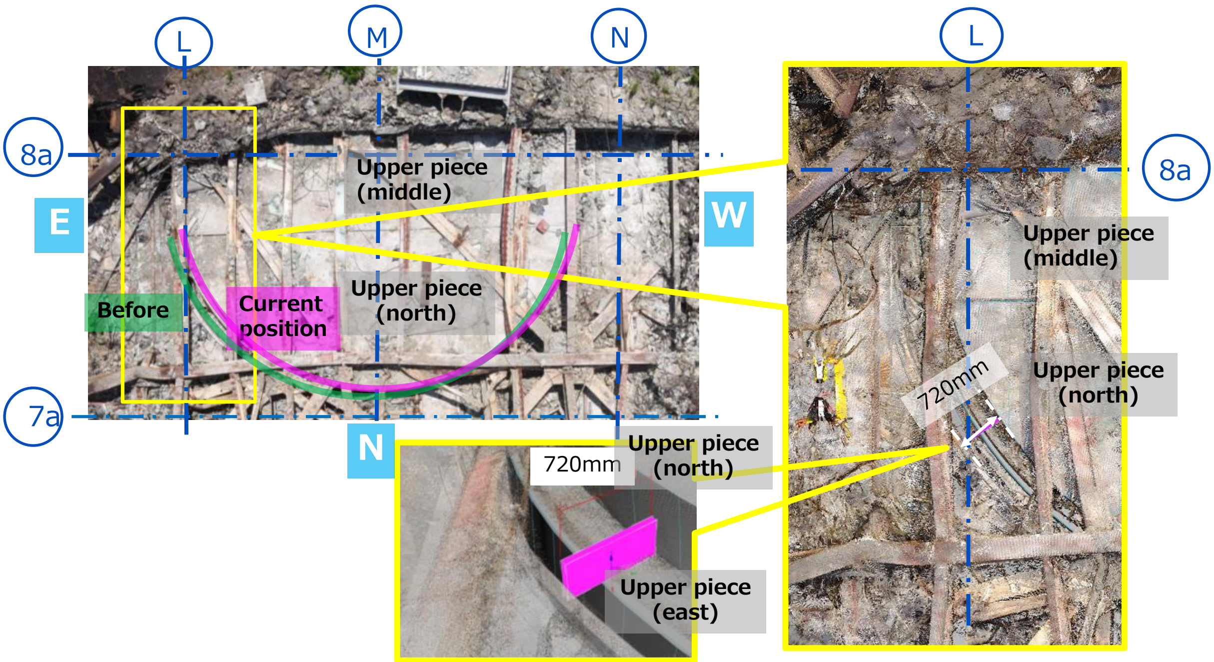
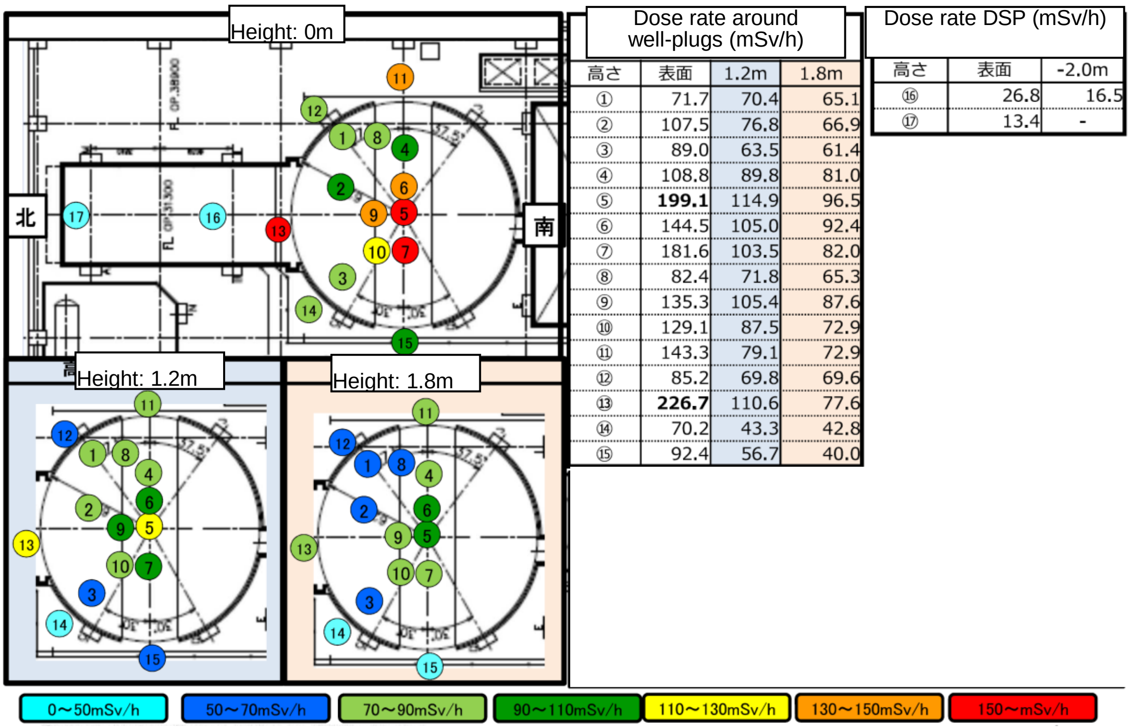


Figure 2-25 Well-plug pieces shifted from original position [2-17]

Figure 2-26 shows the results of dose measurement. Dose rates were found to be higher at the central part on the well-plugs and in the north side around the well-plugs.

[2-17] Handout document: Results of survey on the debris situation on the operating floor of the Unit-1 reactor building (additional survey), 46th Decommissioning and Contaminated Water Response Team Joint Meeting, September 28, 2017.



Dose rate around well-plugs (mSv/h)

高さ	表面	1.2m	1.8m
①	71.7	70.4	65.1
②	107.5	76.8	66.9
③	89.0	63.5	61.4
④	108.8	89.8	81.0
⑤	**199.1**	114.9	96.5
⑥	144.5	105.0	92.4
⑦	181.6	103.5	82.0
⑧	82.4	71.8	65.3
⑨	135.3	105.4	87.6
⑩	129.1	87.5	72.9
⑪	143.3	79.1	72.9
⑫	85.2	69.8	69.6
⑬	**226.7**	110.6	77.6
⑭	70.2	43.3	42.8
⑮	92.4	56.7	40.0

Dose rate DSP (mSv/h)

高さ	表面	-2.0m
⑯	26.8	16.5
⑰	13.4	-

Figure 2-26 Dose rates on the operating floor [2-18]

(11) Unit 1 reactor building operating floor survey (Part 2) [UPDATE]

A survey using a remotely operated robot was conducted on the reactor building operating floor from July to August 2019 to check the status of the reactor well-plug retained structure and the contamination (Figure 2-27).

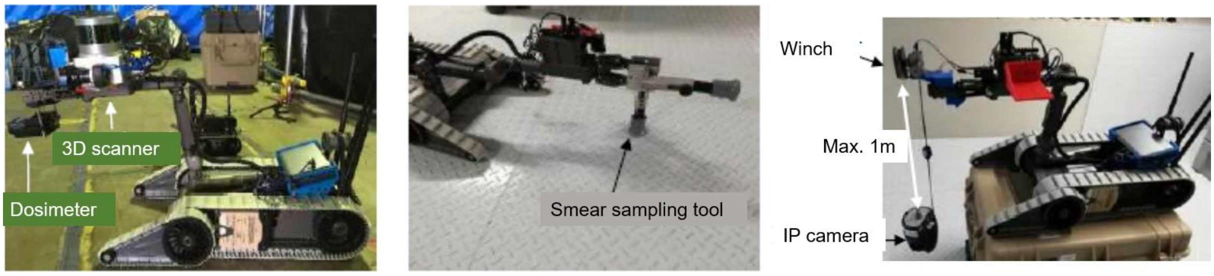


Figure 2-27 Remote-control robot [2-19]

The investigation consisted of suspending the camera from the point where the well-plug was tilted, 3D measurement, dose rate measurement, and smear sampling. As a result of the investigation, the status of the middle and lower plugs was confirmed. The results of Figures 2-28 and 2-29 confirmed the positional relationship between the upper and middle plugs, the inclination of the plugs, and the condition of the lower plug. 3D measurements of the lower surface of the upper plug, the upper surface of the middle plug, and a part of the west side of the lower plug were

[2-18] Handout document: Results of radiation measurement on the operating floor of the Unit-1 reactor building (additional survey), 44th Decommissioning and Contaminated Water Response Team joint Meeting, July 27, 2017.
[2-19] Handout document: Investigation of interferences inside reactor building SFP and well plug investigation for Unit-1, 44th Decommissioning and Contaminated Water Response Team Joint Meeting, August 29, 2019.

conducted to the extent possible, and it was confirmed that the plug had such deformations as a deflection (Figure 2-30).

A survey robot traveled along the upper surface of the middle plug and conducted smear sampling and dose rate measurement within the accessible area (Figures 2-31 to 2-33).

In the air dose rate measurements, the dose rate tended to be higher near the center of the middle plug and lower toward the outer periphery. On the other hand, surface dose rates varied widely. This is presumably due to the contribution of roof debris falling through gaps in the upper plug and runoff to the lower part of the plug due to rainwater intrusion (Figures 2-32 and 2-33).

In Figure 2-34, air dose rates were measured by hanging a dosimeter in the gap between the middle plugs. The dose rate tended to be higher below the middle plug at each measurement location.

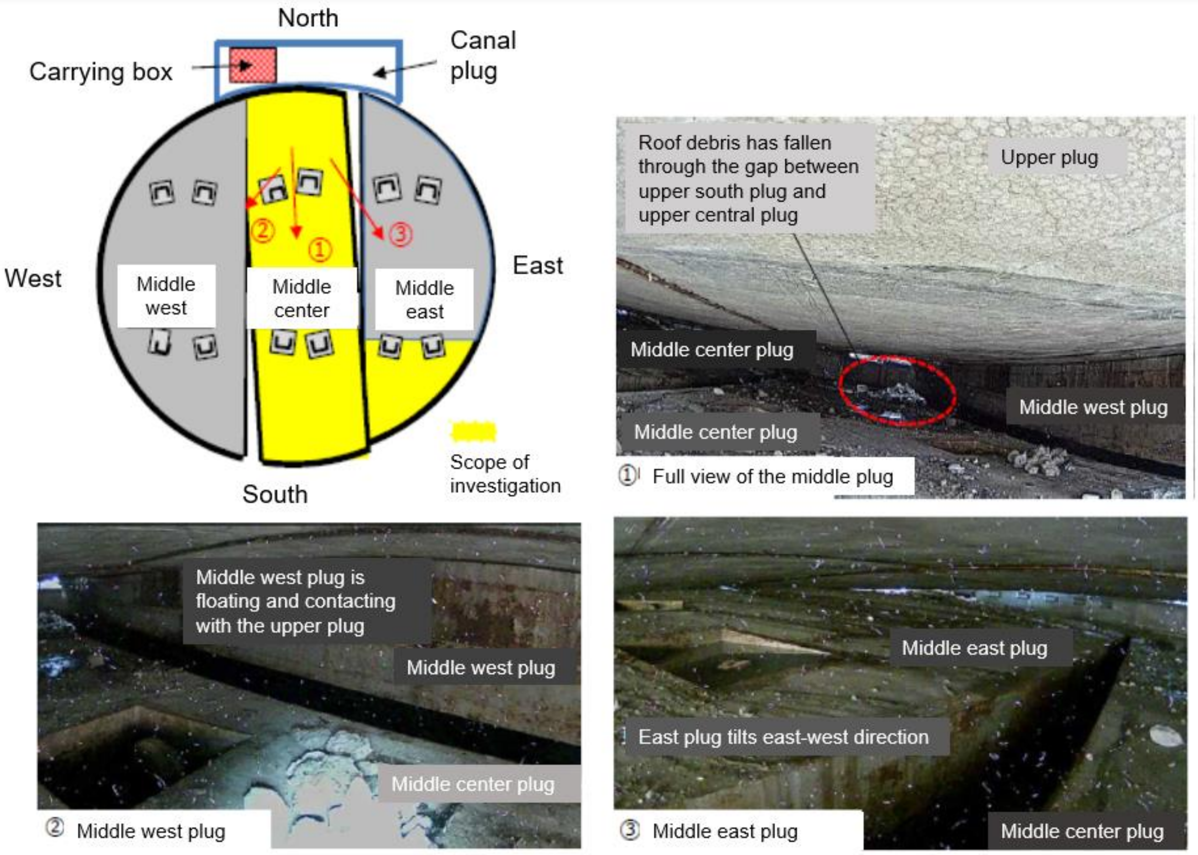


Figure 2-28 Confirmation of middle plug status [2-19]

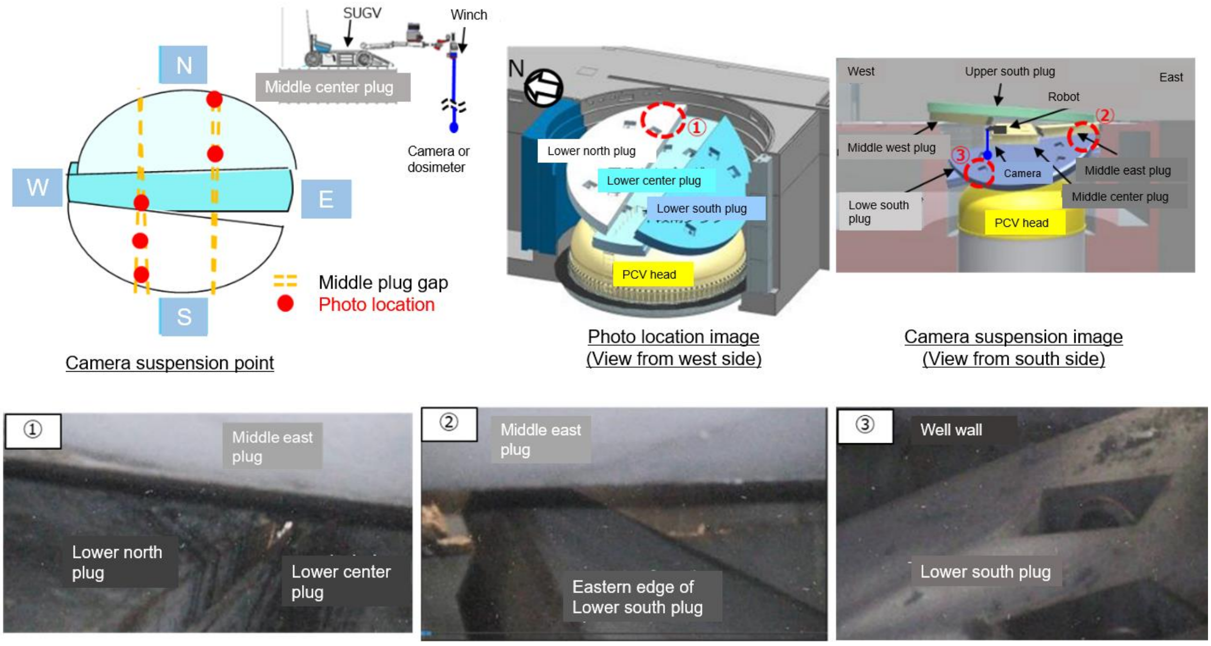


Figure 2-29　Confirmation of lower plug status [2-19]

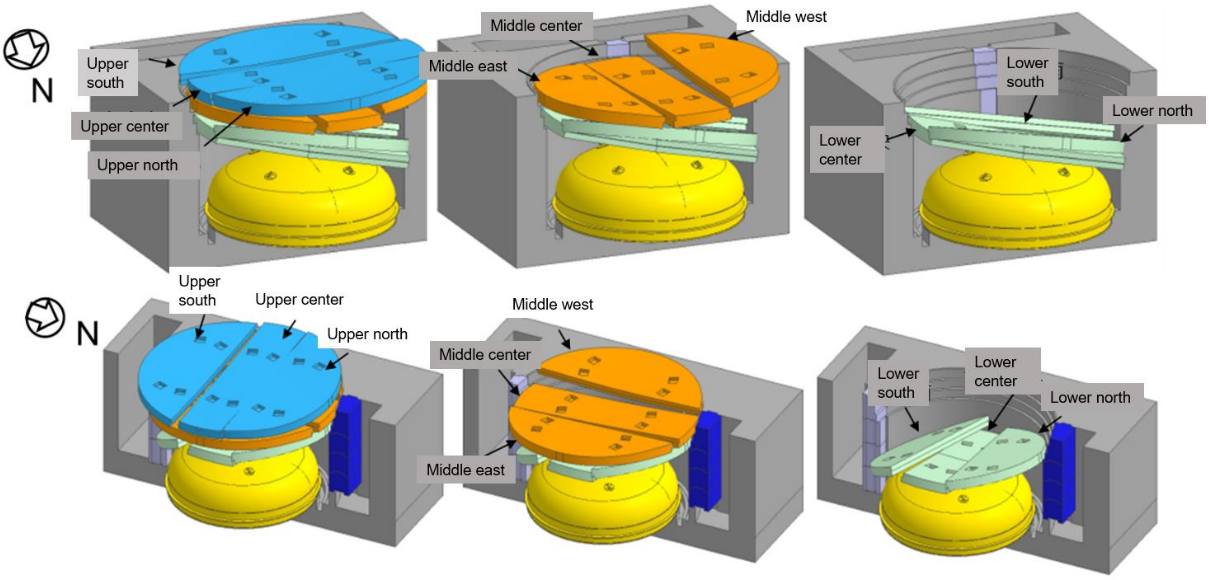


Figure 2-30　3D measurement results of well-plug [2-20]

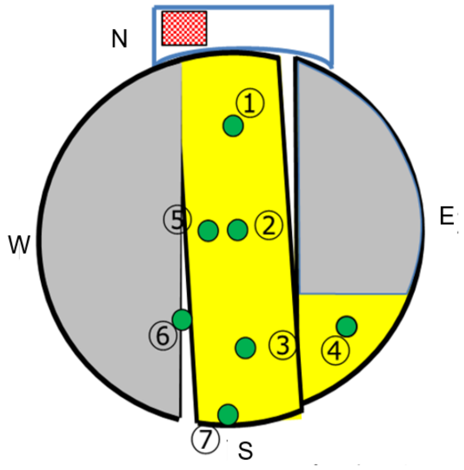
[2-20] Handout document: Results of survey related to debris removal from the Unit-1 reactor building and start of cutting the steel frame of the north side roof, 70th Decommissioning and Contaminated Water Response Team Joint Meeting, September 26, 2019.

Analytical result (Bq/cm²)

No	Location	γ-nuclides				α-nuclides ＊
		Cs-134	Cs-137	Co-60	Sb-125	
①	Upper plug, North lower surface	7.0E+3	1.0E+5	6.4E+1	4.8E+3	6.4E-1
	Middle plug, North upper surface	4.7E+3	6.9E+4	1.6E+1	8.4E+2	6.4E-1
②	Upper plug, Center lower surface	1.1E+4	1.6E+5	5.5E+1	4.4E+3	1.1E+0※3
	Middle plug, Center upper surface	－※2	2E+6	＜ 8E+1	－※2	6.4E-1※3
③	Upper plug, South lower surface	6.2E+3	9.2E+4	6.3E+1	5.7E+3	＜5.7E-1※3
	Middle plug, South upper surface	5.9E+3	8.7E+4	＜2.6E+1	7.2E+2	6.4E-1※3
④	Upper plug, East lower surface	1.3E+3	1.9E+4	2.7E+1	1.8E+3	8.5E-1※3
	Middle plug, East upper surface	1.3E+3	1.9E+4	4.8E+0	1.9E+2	＜5.7E-1※3
⑤	Upper plug, Center lower surface	1.5E+4	2.2E+5	8.7E+1	6.7E+3	2.7E+0※3
	Middle plug, Center upper surface	3.4E+3	5.3E+4	＜1.1E+1	＜3.2E+2	＜5.7E-1※3
⑥	Middle plug, West side surface	－※2	3E+6	＜ 1E+2	－※2	6.6E+0
⑦	South side, Well wall	2.7E+3	3.9E+4	＜1.0E+1	9.2E+2	1.3E+0

※3 : Correction was made because it was confirmed that the lower surface of the upper plug and the upper surface of the middle plug were misplaced. (2019.11.21)

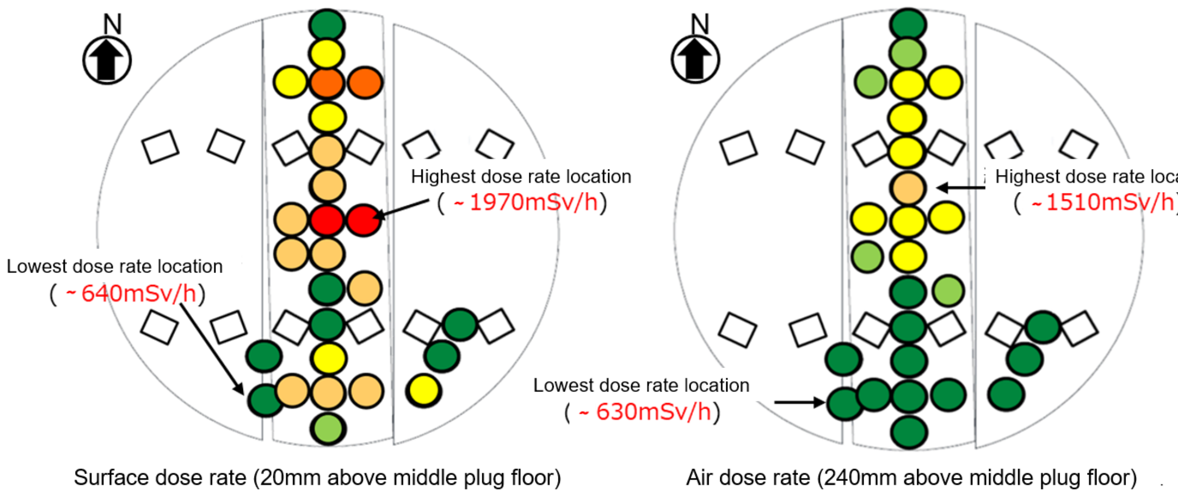
● : Smear sampling point
Yellow : Access is possible
Gray : Access is impossible (Because the upper and middle plugs are close or in contact)



Smear sampling points (Middle plug surface)

※1 : Evaluation results by ZnS Scintillation survey meter
※2 : Because the dose was too high for the Ge semiconductor spectrometer, resulting in underestimation (high dead time), measurements were made with a different spectrometer (CZT). Only nuclides possessing a standard source were quantified.

Figure 2-31 Smear measurement on the middle plug [2-20]



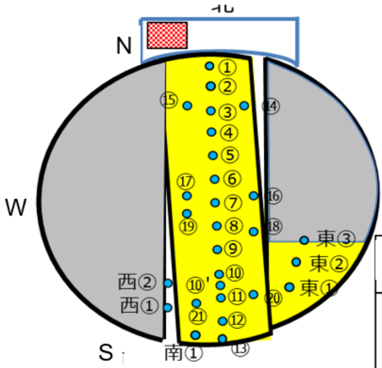
● : Less than 1000mSv/h
● : More than 1000mSv/h, less than 1200mSv/h
● : More than 1200mSv/h, less than 1400mSv/h
● : More than 1400mSv/h, less than 1600mSv/h
● : More than 1600mSv/h, less than 1700mSv/h
● : More than 1700mSv/h, ＊Data are γ-ray dose rate

Figure 2-32 Dose rate measurement on the middle plug [2-20]

Max. dose | >1500mSV/h | >1000mSV/h (Unit : mSv/h)

Location	Low		Middle		High	
	Downward	Upward	Downward	Upward	Downward	Upward
①	850	700	–	–	–	–
②	1390	1010	–	–	–	–
③	1640	1250	–	–	–	–
④	1290	1330	–	–	–	–
⑤	1560	1380	1530	1260	–	–
⑥	1560	1510	1550	1270	–	–
⑦	1720	1240	1560	1360	–	–
⑧	1570	1200	1260	1120	–	–
⑨	760	730	920	700	–	–
⑩	840	820	800	800	–	–
⑩'	1080	860	1000	760	–	–
⑪	1250	920	1010	790	940	820
⑫	1400	900	880	930	800	700
⑬	1090	700	840	690	600	460
⑭	1630	1210	–	–	–	–
⑮	1370	1000	–	–	–	–
⑯	1970	1330	1390	1170	–	–
⑰	1550	1200	1280	1040	–	–
⑱	1520	1140	1220	1020	–	–
⑲	1520	1070	1130	950	–	–
⑳	1350	860	870	860	840	700
㉑	1540	940	980	730	720	620

: Smear sampling point
: Access is possible
: Access is impossible (Because the upper and middle plugs are close or in contact)



Dose measurement location on middle plug (Middle plug surface)

Unit : mm)

Measurement height		Robot posture		
		Low	Middle	High
Dosimeter direction	Up-ward	240	470	690
	Down-ward	20	250	470

(Unit : mSv/h)

Location	Low			Middle			High		
	bottom ,	top ,	wall	bottom ,	top ,	wall	bottom ,	top ,	wall
西①	640	630	–	–	–	–	–	–	–
西②	690	660	–	–	–	–	–	–	–
東①	1350	930	–	900	950	–	–	–	–
東②	850	830	–	920	780	–	–	–	–
東③	960	770	–	730	690	–	–	–	–
南①	1240	920	920	850	710	700	650	690	660

Figure 2-33 Dose rate measurement results between the upper and middle plugs [2-20]

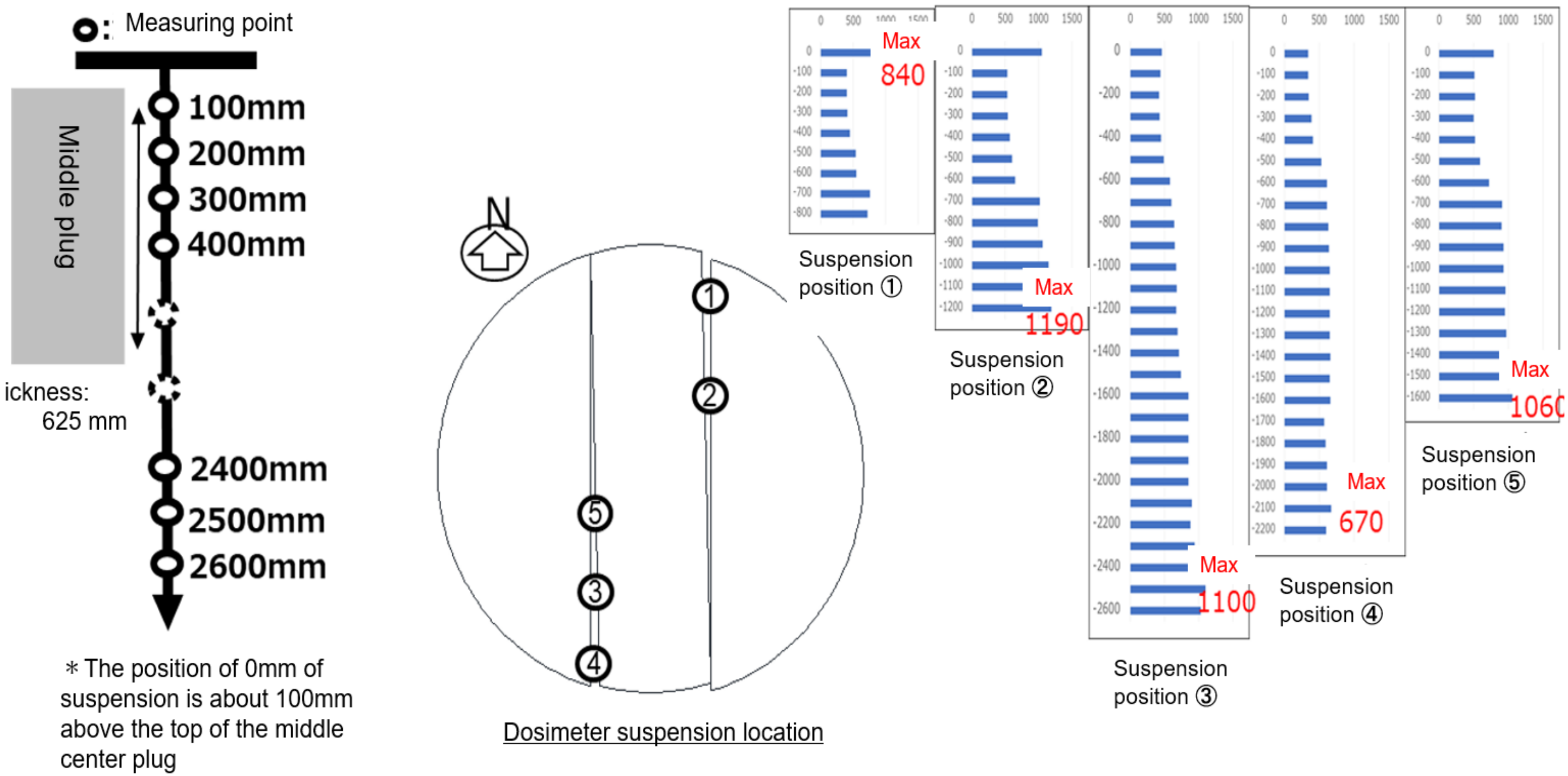
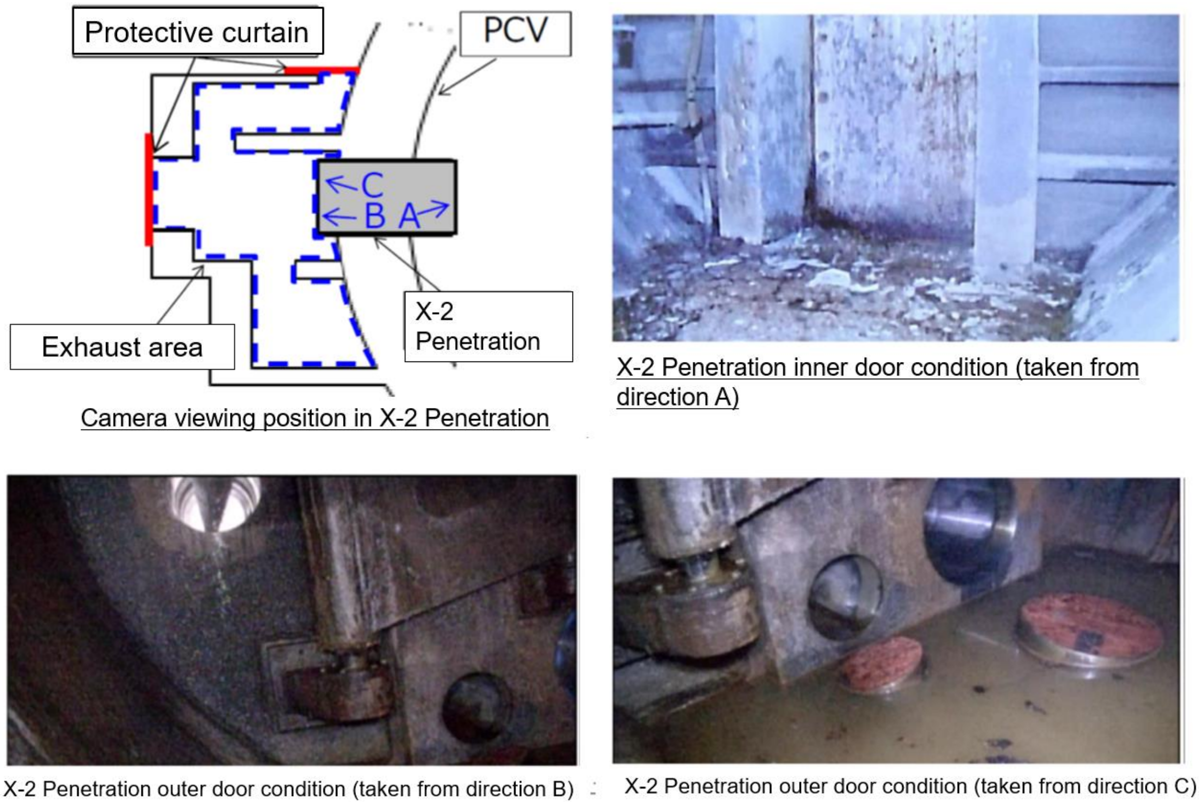


Figure 2-34 Result of dose rate measurement between middle plug and lower plug [2-20]

(12) Unit-1 reactor building X-2 penetration internal investigation results [UPDATE]

Drilling of the X-2 penetration outer door was carried out from April to May 2019 to confirm the internal conditions. As a result, matter presumed to be paint was peeled off and deposited in front of the X-2 penetration inner door (Figure 2-35).



Camera viewing position in X-2 Penetration

X-2 Penetration inner door condition (taken from direction A)

X-2 Penetration outer door condition (taken from direction B)

X-2 Penetration outer door condition (taken from direction C)

Figure 2- 35 Status in X-2 penetration [2-21]

(13) Results of the investigation of the Unit-1 containment top head [UPDATE]

In parallel with the work to construct an access route for the investigation inside the containment vessel of Unit-1, the status of the flange of the PCV top head (PCV upper lid), which is estimated to have been the main leakage path during the accident, was confirmed in November 2019. Video images were obtained of the situation inside the reactor cavity (well), and no significant damage to the PCV top head or other parts was observed (Figure 2- 36). No significant damage or major deformation was observed in the flange section as well, although there was some deterioration of the paint (Figure 2-37). Since high contamination was estimated from the white noise in the video, leakage from the flange was considered to have occurred under the high PCV pressure conditions at the time of the accident.

[2-21] Handout document: Status of the construction of the access route to investigate the inside of the containment vessel of the nuclear reactor from the X-2 Penetration of Unit-1, 66th Decommissioning and Contaminated Water Response Team Joint Meeting, May 30, 2019.

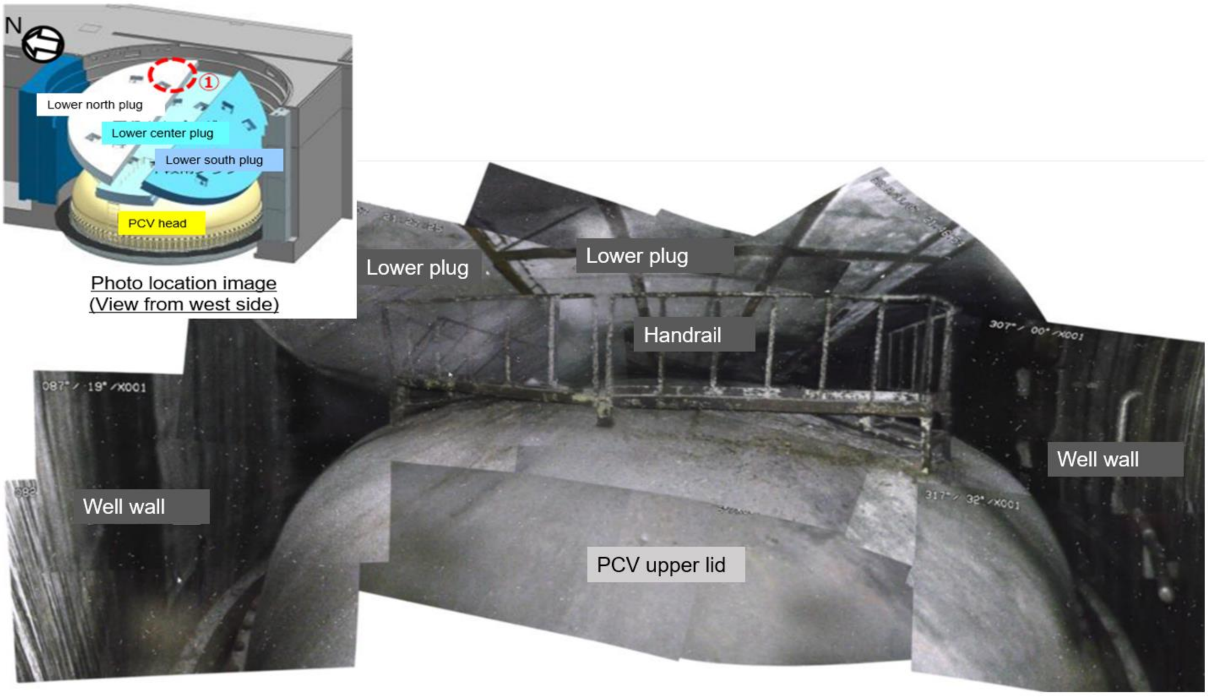


Figure 2-36 Status of the upper part of the PCV upper lid (multiple photos superimposed): View from the east side [2-22]

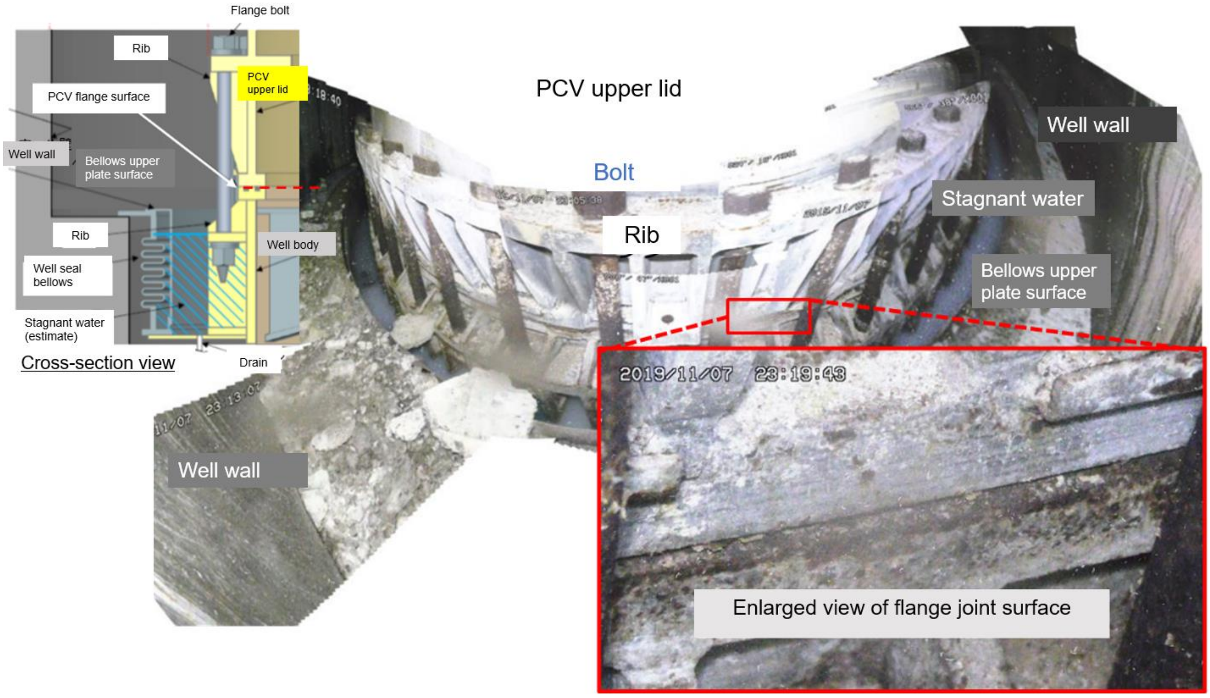


Figure 2-37 PCV flange status (multiple photos superimposed) [2-22]

[2-22] Handout document: Confirmation of the containment upper lid status of the Unit-1 reactor, 72nd Decommissioning and Contaminated Water Response Team Joint Meeting, November 28, 2019.

Figure 2-38 Flange of PCV lid before the accident [2-22]

(14) Results of the investigation of the Unit 1 SGTS room [UPDATE]

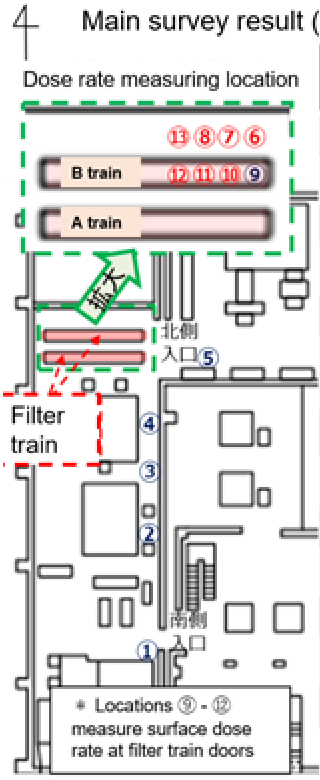
As a survey related to Unit-1/Issue-10, dose rate measurements of the Unit-1 SGTS room and filter train were conducted in December 2020 using a remotely operated robot. The results of the dose rate measurements in the room confirmed high air dose rates of about 1500 mSv/h in the vicinity of the filter train, and surface dose rates at the filter train door increased toward the downstream side of the train (Figure 2-39). In addition, γ imaging measurements confirmed contamination along the SGTS piping connected to the vent line (Figure 2- 40). These results clearly indicated that Unit-1 had a backflow of vent gas into its own unit.

In addition, a survey conducted by the Nuclear Regulation Authority in January 2022 confirmed dose rates of up to 3380 mSv/h in the SGTS room and 1140 mSv/h near the filter train. [2-23]

[2-23] Handout document: Status of investigations, etc. inside the Unit-4 and Unit-5 reactor buildings (Unit-4 reactor building fire, Unit-2 shield plug deformation investigation (including investigations of Fukushima Daiichi Unit-5 and Shimane Unit-1), PCV inner cable investigation (including investigations of Fukushima Daiichi Unit-5 and Shimane Unit-1), etc.), 28th meeting of the study group for analysis of the accident at TEPCO's Fukushima Daiichi Nuclear Power Station, February 28, 2022.

Main survey result (December 15th)

Dose rate measuring location

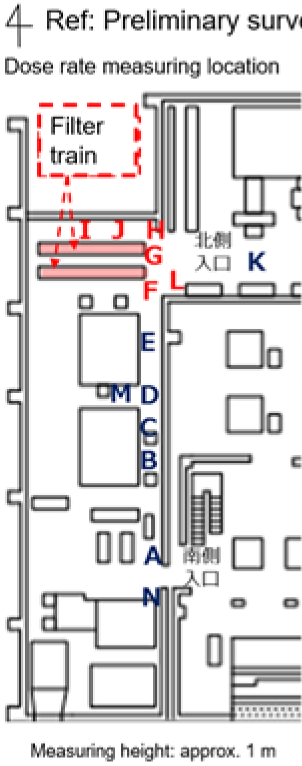


Location	Dose rate (mSv/h)
①	5.57
②	16.7
③	120
④	280
⑤	84.1
⑥	1330 2270 (floor)
⑦	1420
⑧	1510
⑨※	880
⑩※	1020
⑪※	1180
⑫※	1470
⑬	1580

Measuring height: approx. 1 m

Ref: Preliminary survey result (August 25th)

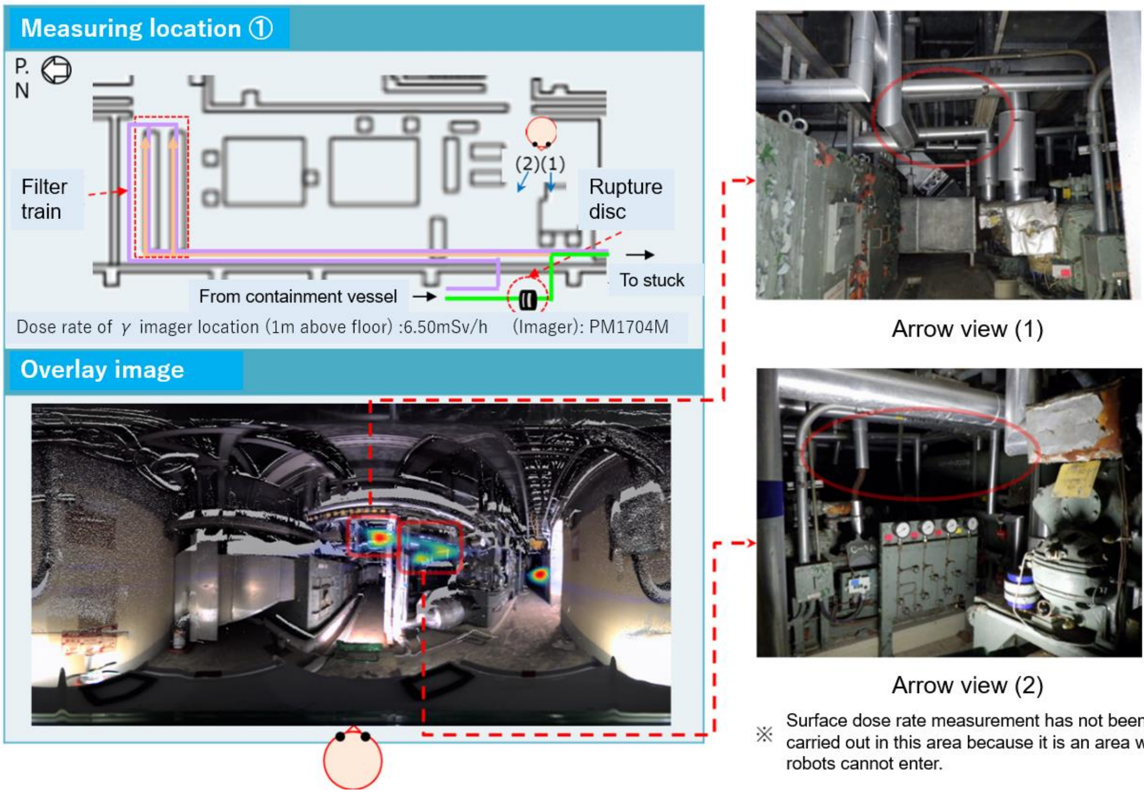
Dose rate measuring location



Location	Dose rate (mSv/h)
A	5.1
B	16.6
C	110
D	150
E	310
F	1050
G	2050 3050 (floor)
H	1270
I	1620
J	1040
K	50
L	1060
M	160
N	6.76

Measuring height: approx. 1 m

Figure 2-39 Results of air dose rate measurements in the SGTS room [2-24]



※ Surface dose rate measurement has not been carried out in this area because it is an area where robots cannot enter.

Figure 2-40 γ imaging measurement results in the SGTS room (Part 1) [2-24]

[2-24] Handout document: Results of the SGTS Room Survey for Units-1 to -4, 88th Decommissioning and Contaminated Water Response Team Joint Meeting Panel, March 25, 2021.

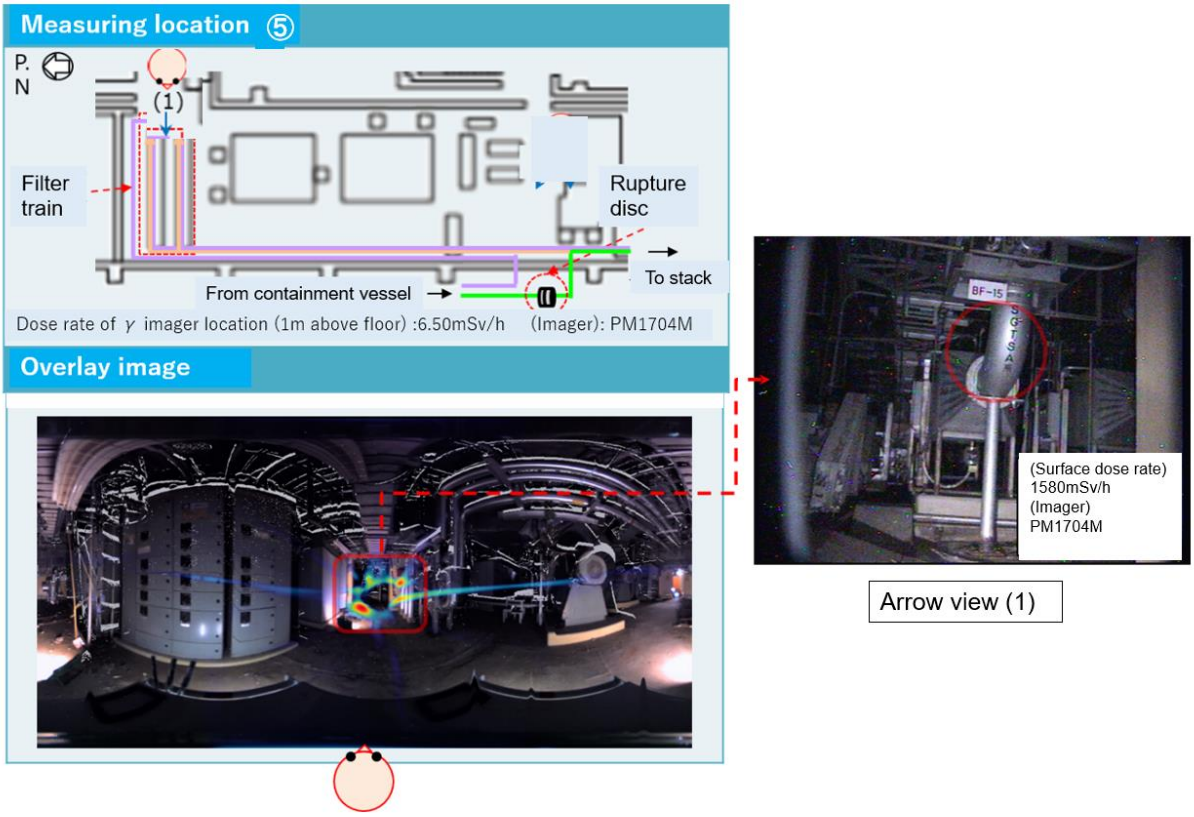


Figure 2-41 γ imaging measurement results in the SGTS room (Part 2) [2-24]

(15) Investigation results of the lower part of Units-1/2 exhaust stack and SGTS piping [UPDATE]

The SGTS piping connected to the lower part of the Units-1/2 exhaust stack and the lower part of the exhaust stack is being removed to avoid interference with the decommissioning work and to provide an environmental improvement (dose reduction) for the on-site work. The inspection results of the interior of the exhaust stack showed that the SGTS piping was not damaged.

As a result of checking the stack interior, deposits such as sludge were found at the bottom of the stack (Figure 2-42). These deposits were considered to be rust, sand, and gravel inside the exhaust stack and lining pieces that had deteriorated over time.

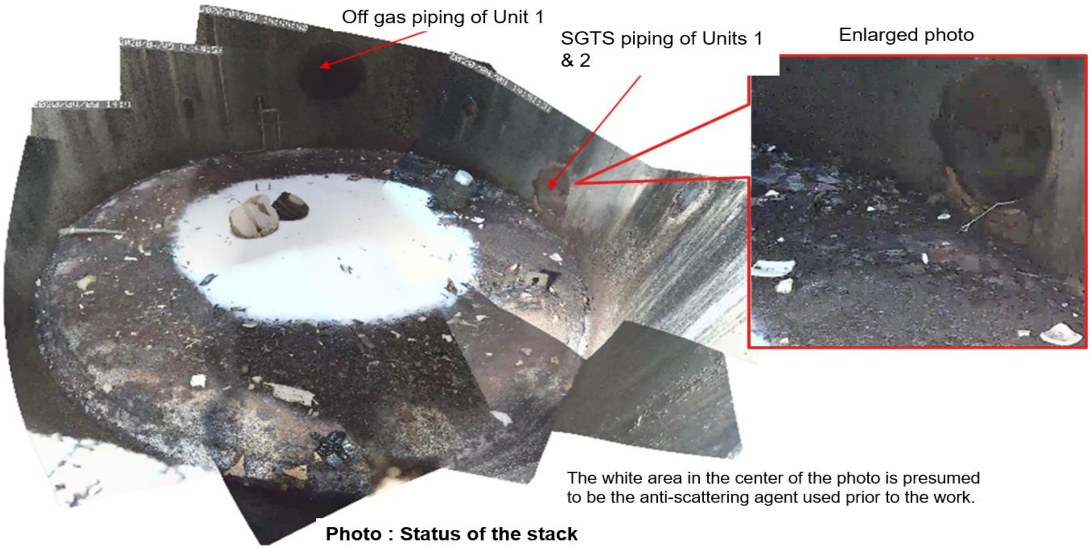
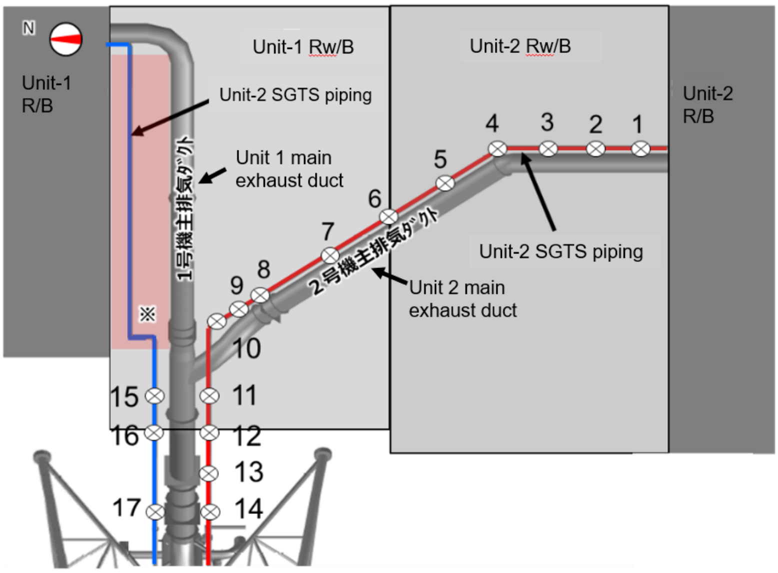


Figure 2-42 Condition at the bottom of the exhaust stack [2-25]

As a result of measuring the dose rate near the SGTS piping, the highest value of approximately 650 mSv/h was confirmed at a height of 0.1 m from the surface of the SGTS piping in Unit-2 (Figure 2-43, measurement point No. 13). Since butterfly valves are installed near measurement points No.13 and No.14, where high dose rates were confirmed, it is considered to be an environment where radioactive materials can easily concentrate. On the other hand, No. 8 and No. 9, where the next highest dose rates were observed, were in the horizontal piping area (Figure 2-44).



mSv/h

Location	SGTS piping Surface (0.1m)	SGTS piping Above (1m)
1	6.0	3.0
2	8.0	4.0
3	17.0	5.0
4	26.0	8.0
5	27.0	12.0
6	20.0	8.0
7	60.0	30.0
8	150.0	85.0
9	160.0	50.0
10	60.0	40.0
11	11.0	3.0
12	4.3	2.5
13	650.0	160.0
14	400.0	130.0
15	2.0	1.0
16	2.0	1.4
17	4.0	3.0

- Measurement points 1-10, 13, and 14 were measured by ICW.
- Measurement points 11-12 and 15-17 were measured by ICS.
- Measured by collimating to avoid influence from surroundings

(See P3 for measurement method)

※ Dose measurement by crane is not possible due to the lower part of the Unit 1 reactor building cover structure.

Figure 2-43 Results of dose rate survey near SGTS piping (conducted in 2020) [2-25]

[2-25] Handout document: Future investigation policy for the removal of SGTS piping in Units-1/2, 82nd meeting of the Specific Nuclear Facility Monitoring and Evaluation Study Group, July 20, 2020.

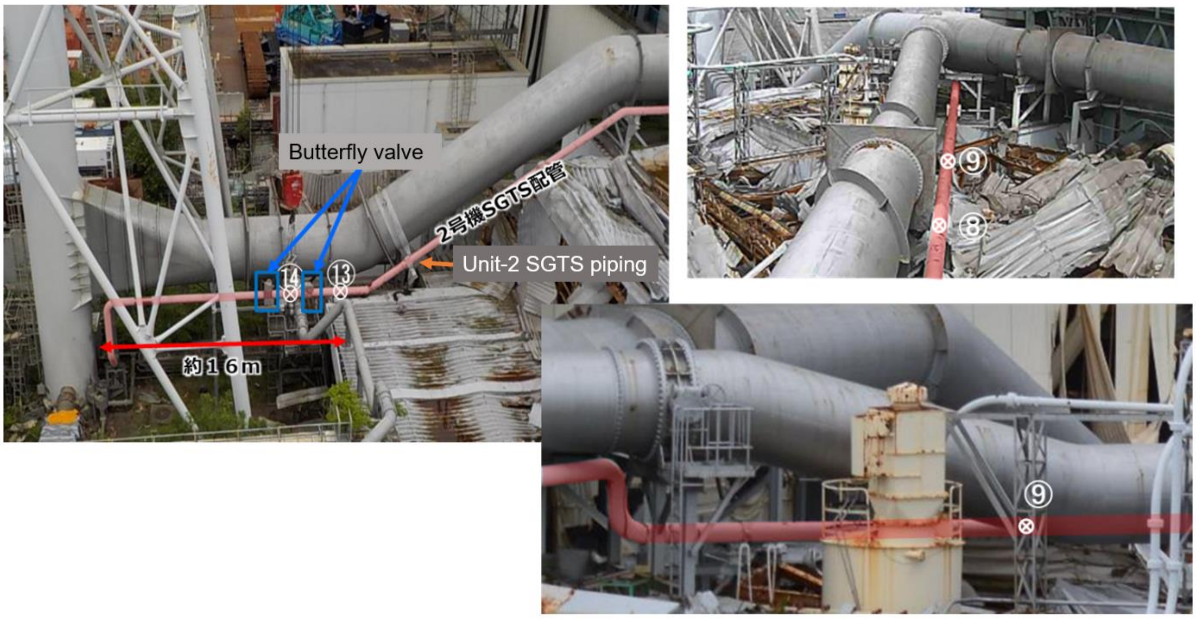
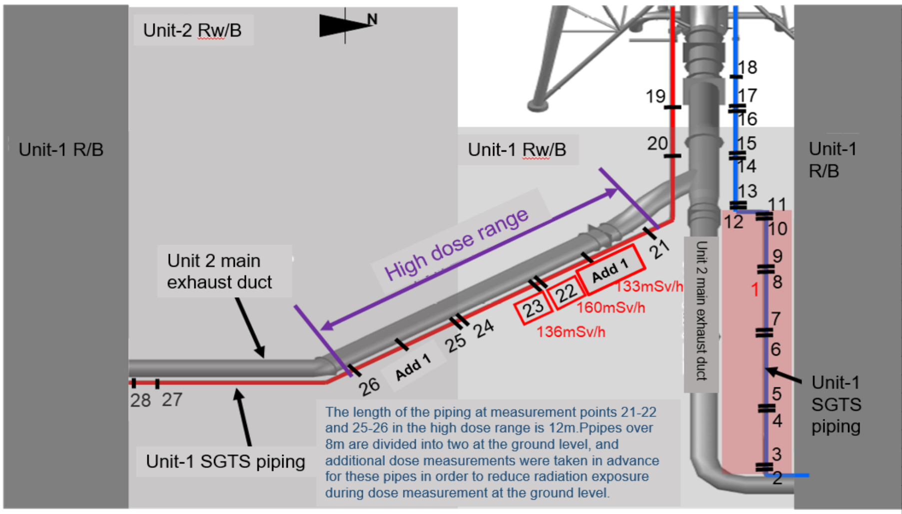


Figure 2-44 Locations where high dose rates were observed in the vicinity of SGTS piping (The numbers in the figure correspond to the measurement locations in Figure 2-43) [2-25]

In May 2021, another SGTS piping dose rate survey was conducted, and the results showed that the Unit-2 side was higher and the Unit-1 side was lower, similar to the results of the 2020 survey (Figure 2-45). This trend is assumed to be due to the fact that the SGTS system (filters, rupture disks, etc.) in the reactor building in Unit-2 acted as resistance to the Unit-1 piping, which had a faster venting flow velocity, suppressing the flow velocity and causing retention.



mSv/h

Location	SGTS Piping	
	0.1m	1.0m
1※	4.15	4.46
2	2.56	1.52
3	3.12	1.74
4	3.76	2.64
5	7.56	3.85
6	3.97	2.95
7	6.74	3.90
8	5.05	5.58
9	5.73	4.41
10	4.35	2.65
11	10.4	3.74
12	5.84	3.42
13	4.43	1.91
14	2.25	0.90
15	3.60	1.13
16	3.70	2.37
17	8.50	4.50
18	5.62	3.00
19	4.31	7.60
20	6.06	2.85
21	65.1	27.0
22	160.0	60.3
23	136.0	55.8
24	52.3	16.7
25	27.0	9.80
26	18.9	7.47
追1	16.5	6.53
追2	133.0	45.0

＊The area with the highest radiation dose is near the top 3.0m inside the red frame.

Figure 2-45 SGTS piping dose rate survey results (conducted in 2021) [2-26]

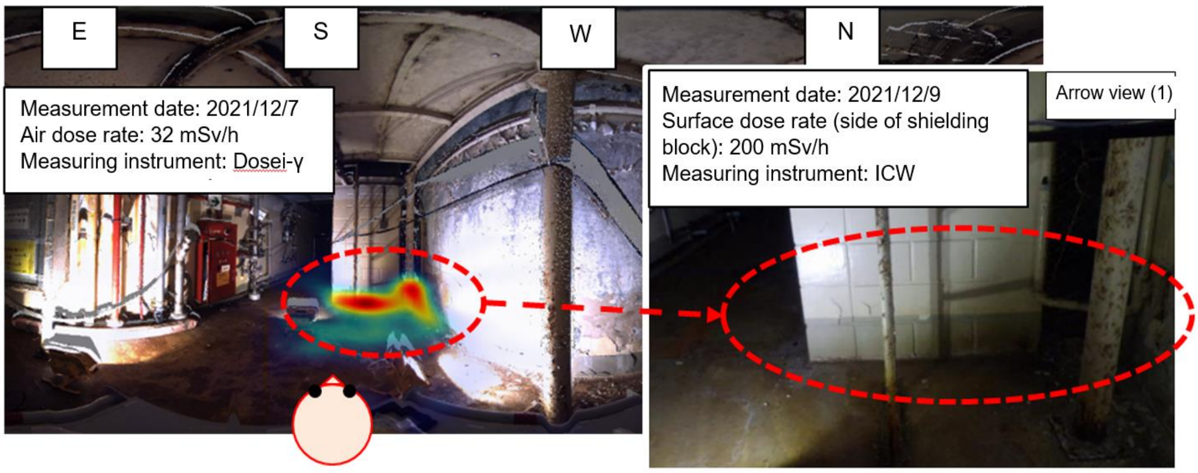
[2-26] Handout document: Partial Removal of Emergency Gas Treatment System Piping at Fukushima Daiichi Nuclear Power Station Units-1 and -2, 21st Meeting of the Study Group on Analysis of the Accident at TEPCO's Fukushima Daiichi Nuclear Power Station, July 8, 2021.

(16) Results of the survey inside the Unit-1 reactor building [UPDATE]

From November to December 2021, a survey was conducted using a γ imager, 3D scanning equipment, and dosimeters to obtain detailed spatial information (accessibility, etc.) and dose rate information inside the reactor building (ground floor) for the planning of future surveys inside this building.

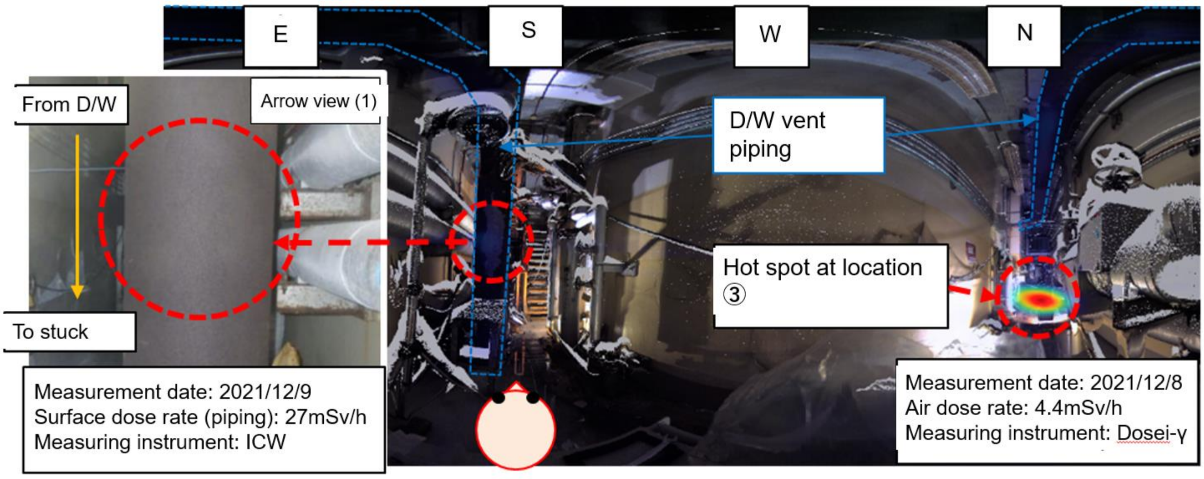
As a result of the γ imaging measurements, a hot spot was confirmed on the floor inside the shielding block where the AC piping (D/W vent piping) and D/W vent valve were installed on the east side of the third floor (Figure 2-46). As a possible cause of hot spots, gases in the D/W leaked from the PCV penetration or the D/W vent valve to the building side due to the D/W pressure increase and high temperature caused by hydrogen formation, and radioactive materials may have adhered to the floor surface after condensing inside the shielding block.

The D/W vent piping on the east side of the third floor is connected to the S/C vent piping used during the accident. Since rust was confirmed on the surface of the piping during a previous survey, the condition of the piping and its contamination were confirmed (Figure 2-47). It was further confirmed that the dose rate was high along the piping, and it was presumed that the inside of the piping was contaminated. The cause of the contamination was estimated to be that although the D/W vent valve was closed, some of the gas in the D/W might have leaked downstream from the valve or some of the S/C vent gas might have flowed into the piping due to the D/W pressure increase and high temperature. Unlike the surrounding piping, rust was observed on the entire surface of the D/W vent piping. As a possible cause of the rust, it was estimated that the rust may have occurred due to deterioration of the piping coating because of the passage of high temperature gas and the heat generated by radioactive materials adhering to the inner surface.



* The intensity distribution is shown relative to the maximum value of source intensity in the image (red) up to 10% of the maximum value (blue).

Fig. 2-46 γ imaging measurement results (inside the shielding block of the 3rd floor east side) [2-27]



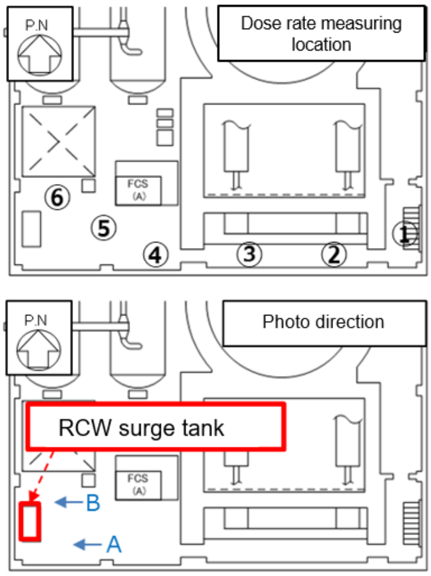
＊The intensity distribution is shown relative to the maximum value of source intensity in the image (red) up to 10% of the maximum value (blue).

Figure 2-47 γ imaging measurement results (3rd floor east side AC piping (D/W vent piping)) [2-27]

It is presumed that contamination spread to the RCW system in Unit-1 due to the accident (see Attachment 1-9), and a dose rate of 90 mSv/h was confirmed around the RCW surge tank located southwest of the 4th floor during a survey from April 2013 to February 2014. [2-28]

[2-27] Handout document: Results of the investigation of the top floor of the reactor building of Units 1-2, 99th Decommissioning, Contaminated Water and Treated Water Response Team Joint Meeting, February 24, 2022.
[2-28] Air dose rate inside the building, TEPCO, March 27, 2014.

This time, as a result of investigating the area around the RCW surge tank again, it was confirmed that the dose rate in the area was high (Figure 2-48).



Location No.	Air dose rate [mSv/h]
①	7.5
②	1.0
③	1.0
④	6.0
⑤	20
⑥	21

Date of measurement and photo: 2021/11/19

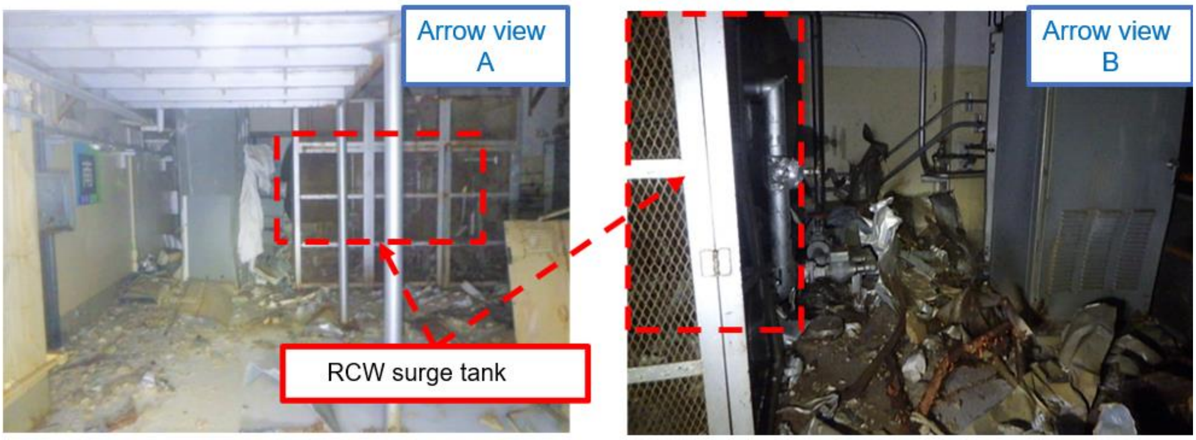
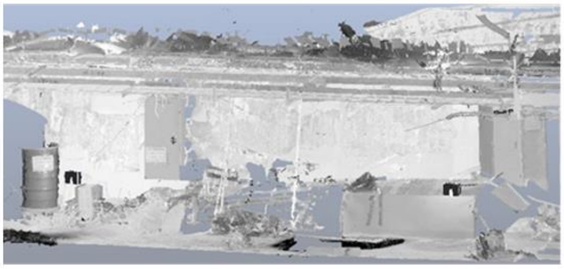
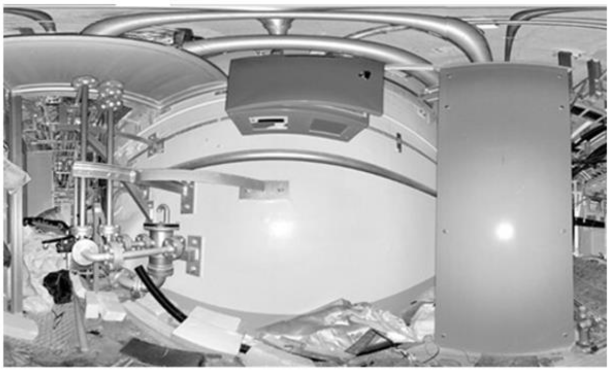


Figure 2-48　Status around the RCW surge tank on the west side of the 4th floor [2-27]

In addition, in order to obtain information that would contribute to the evaluation of the integrity of the reactor building, condition survey and point cloud data acquisition using a 3D scanning device were conducted on the reactor shell walls on the west, north, and east sides of the third floor and the pool wall on the southwest side of the fourth floor (Figure 2-49). As a result of the survey, peeling paint and cracks were observed in some areas, but no damage or signs of aging deterioration (e.g., peeling of surface concrete and rust) that could lead to a decrease in seismic performance were observed.



North face of shell wall on 3rd floor (point cloud data)



North face of shell wall on 3rd floor (photograph)

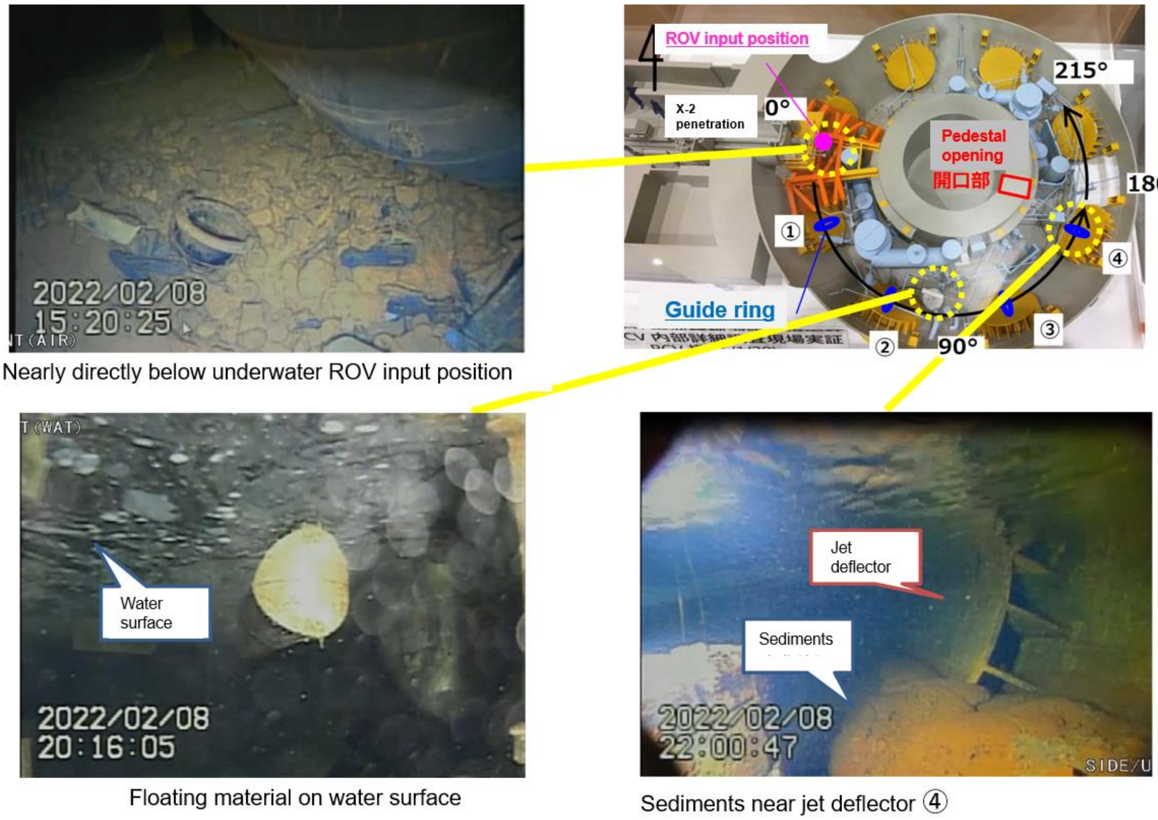
West face of pool wall on 4th floor (photograph)

Figure 2-49　Survey results of reactor shell wall and pool wall [2-27]

(17) Results of the investigation inside the containment vessel of Unit 1 [UPDATE]

In 2022, an investigation was started using a remotely operated vehicle (underwater ROV) to obtain information on the amount and origin of sediments in the PCV for the purpose of studying the means and equipment for recovering these sediments.

In February 2022, a guide ring (a ring to prevent cable entanglement of the underwater ROV) was installed using the underwater ROV-A prior to the survey. At that time, floating material on the water surface and sediments at the bottom of the PCV were identified (Figures 2-50 and 2-51). In addition, the condition inside the pedestal opening was checked from near or around the opening, and massive sediments and what appeared to be rebar were confirmed (Figure 2-52).

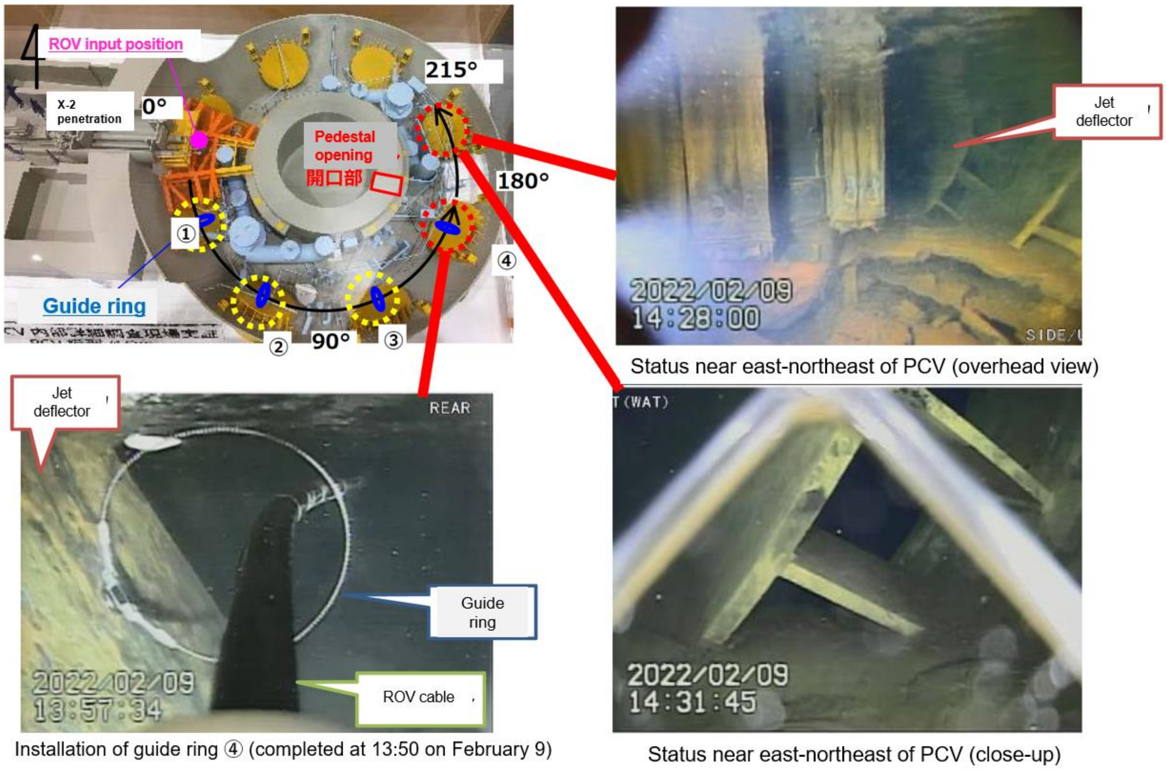


Nearly directly below underwater ROV input position

Floating material on water surface

Sediments near jet deflector ④

Figure 2-50 PCV interior as confirmed by the underwater ROV-A [2-29]



Status near east-northeast of PCV (overhead view)

Installation of guide ring ④ (completed at 13:50 on February 9)

Status near east-northeast of PCV (close-up)

Figure 2-51 Installation status of guide ring (4) and PCV east-northeast (215°) [2-30]

[2-29] Fukushima Daiichi Nuclear Power Station Unit-1 Reactor Containment Vessel Internal Investigation Status (as of February 9), Tokyo Electric Power Company Holdings, February 9, 2022.
[2-30] Fukushima Daiichi Nuclear Power Station Unit-1 Reactor Containment Vessel Internal Investigation Status (as of February 10), Tokyo Electric Power Company Holdings, February 10, 2022.

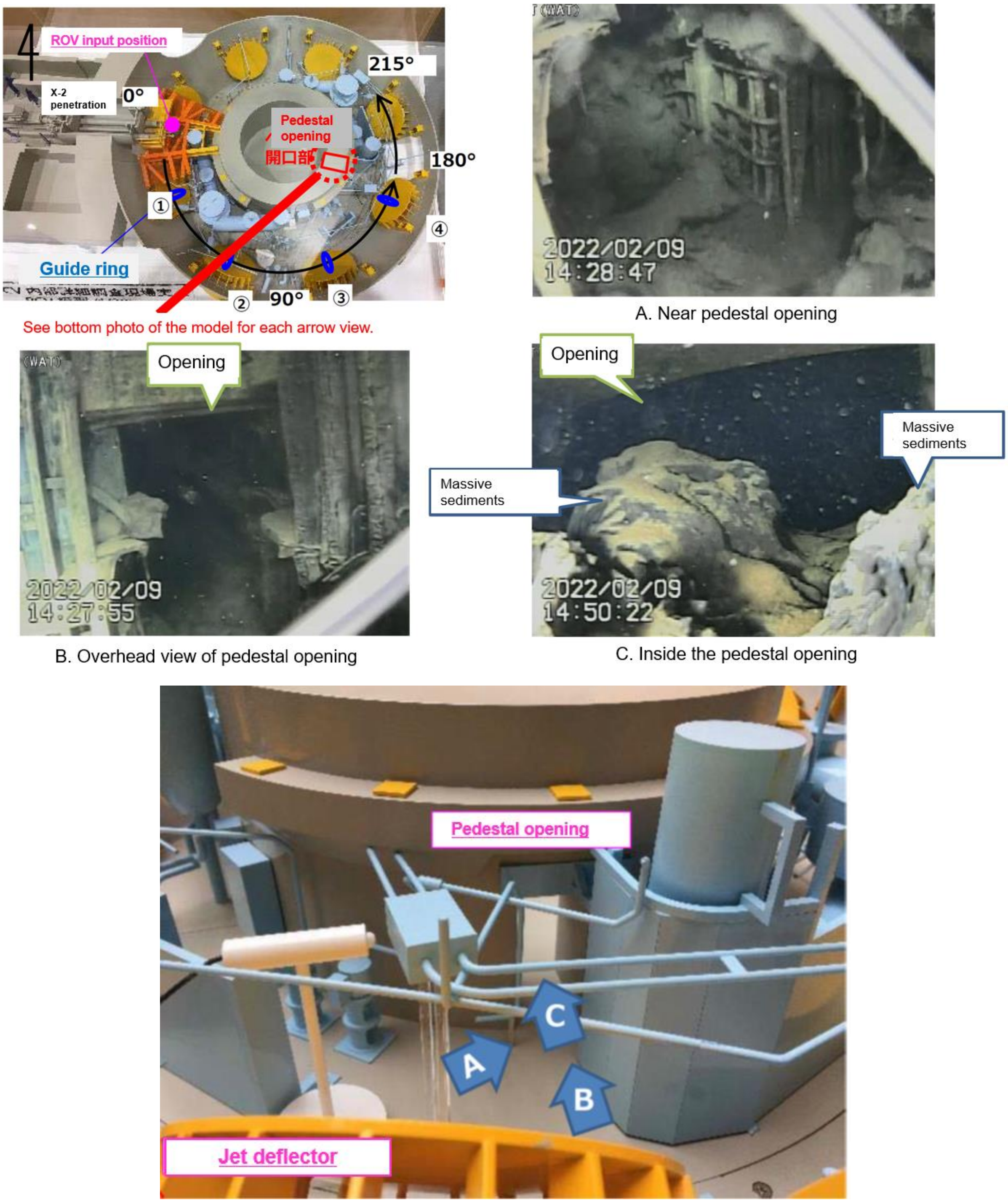


Figure 2-52　Survey status near the pedestal opening [2-30]

From March to May 2022, the underwater ROV-A2 was used to investigate the condition of the existing structures around the pedestal, the spread of sediment, etc., and make neutron flux measurements of the sediment. The survey area is shown in Figure 2-53.

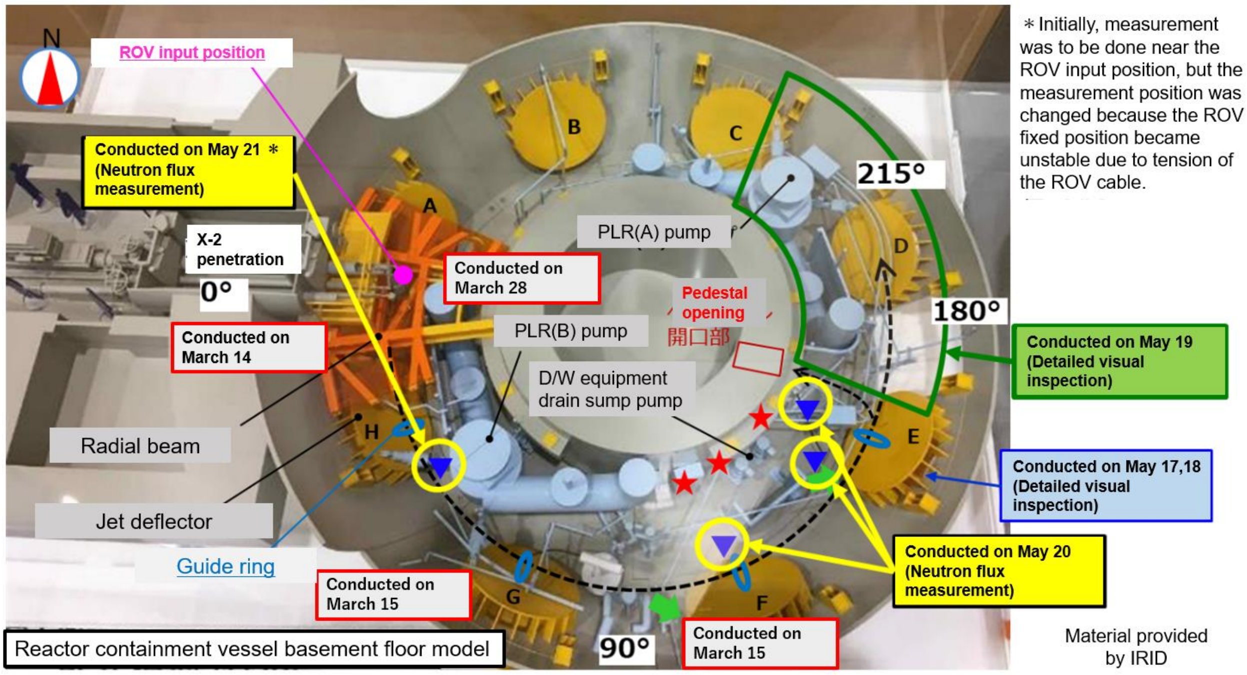


Figure 2-53 Survey area by underwater ROV-A2 [2-31]

The following information was obtained from the results of the underwater ROV-A2 survey.

- No major external damage was observed in the existing structures of the primary loop recirculation system (PLR) (B) piping, pumps, jet deflector, pedestal foundation, etc. (Figures 2-54 to 2-61).
- Sediments were observed below the lead wool mats near the inlet valve of the PLR (B) (bottom of the PCV) and near the jet deflector (F) (Figure 2-54, Photo 3; Figure 2-59, Photo 2).
- Sediments were observed behind the jet deflectors (H, G, F, E, D) (on the side of the pressure suppression chamber) (Figures 2-55, 2-56, 2-58, 2-65, and 2-70).
- On the back side of the jet deflector (F) (suppression chamber side), bubbles were observed to be coming out continuously from the vent pipe connected to the pressure suppression chamber (Figure 2-58, Photo 2).
- An interfering object was observed near the jet deflector (G) (Figures 2-56 and 2-57).
- The bottom of the jet deflector (A) was visible, and the height of sediments near the jet deflector (A) was lower than that near the worker access (Figure 2-60, Photo 2).
- The piping was missing near the pedestal opening where the RCW system piping was presumed to have been installed (Figure 2- 62).
- It was confirmed that the sediments at the bottom of the PCV near the equipment drain sump formed upper and lower layers, leaving the interior hollow (Figures 2-63 and 2-64, and Figures 2-66 to 2-68).

[2-31] Handout document: Unit-1 PCV Internal Investigation Status, 102nd Decommissioning, Contaminated Water and Treated Water Response Team Joint Meeting, February 24, 2022.

- The concrete near the pedestal opening (base) was found to have exposed pedestal rebar and inner skirt (Figure 2-64 and Figures 2-66 to 2-69).
- The sediments at the pedestal opening (base) had upper and lower layers with a hollow interior, and the rebar of the pedestal base was exposed in the hollow. The pedestal base remained above the upper layer (Figures 2-59, 2-64,and 2-68).
- Multiple bulky deposits were observed at the pedestal opening (in front of the interior) (Figure 2-69).
- In the vicinity of the jet deflector (C), sediments were observed (Figure 2-70).

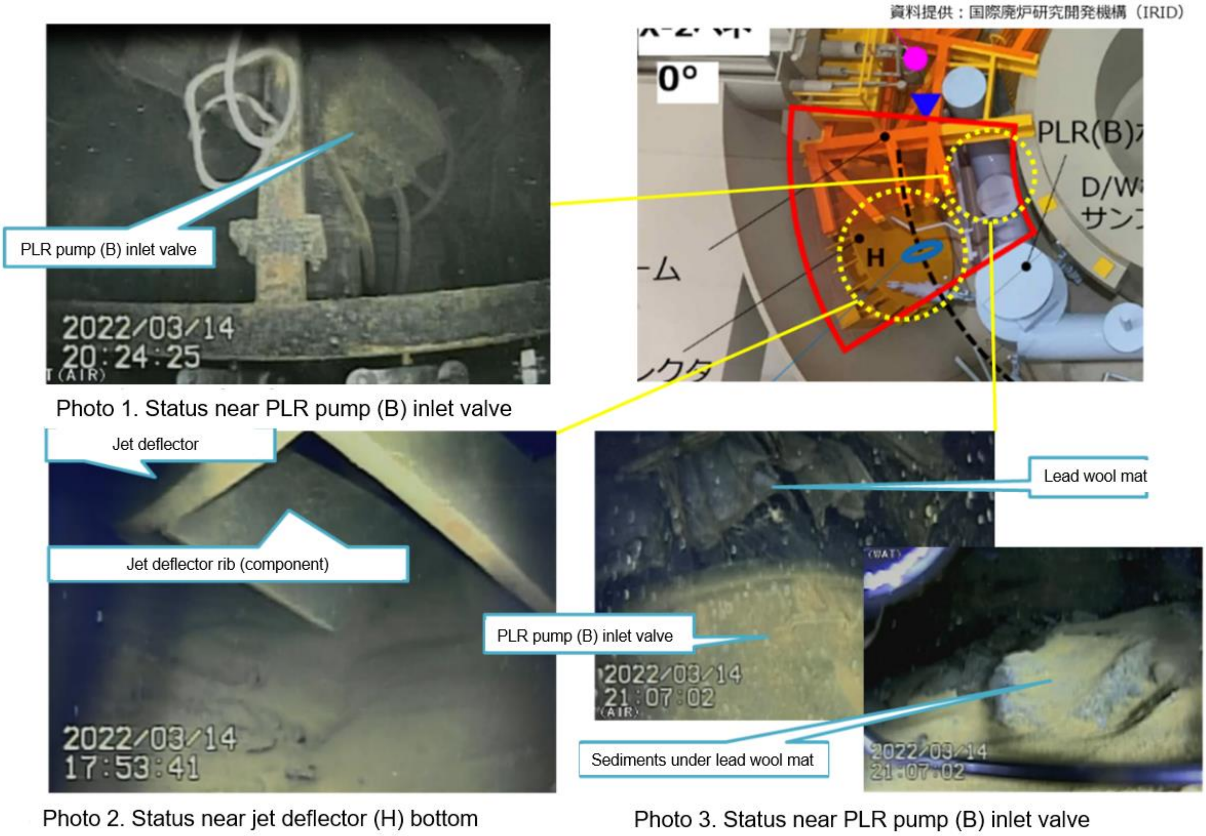


Figure 2-54　Status near the PLR pump (B) and near the jet deflector (H) [2-32]

[2-32] Fukushima Daiichi Nuclear Power Station Unit-1 Reactor Containment Vessel Internal Investigation (ROV-A2) (March 14-16, 2011), Tokyo Electric Power Company Holdings, March 24, 2022.

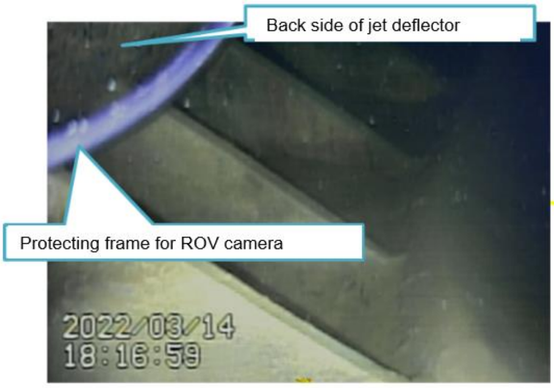


Photo 1. Status near jet deflector (H) back side ①

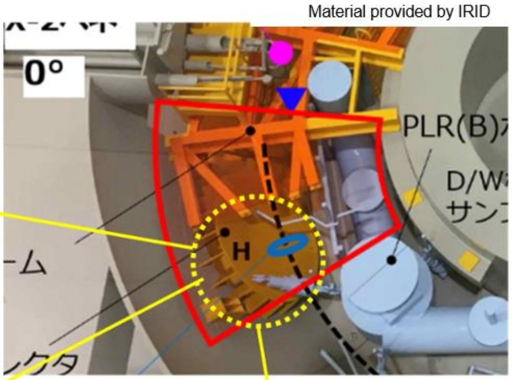


Photo 2. Status near jet deflector (H) back side ②

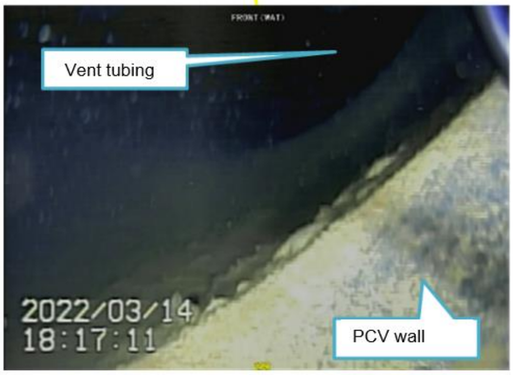


Photo 3. Status near jet deflector (H) back side ③

Figure 2-55　Status near the jet deflector (H) [2-32]

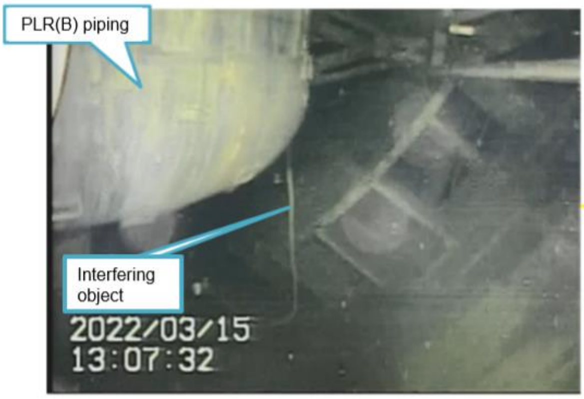


Photo 1. Overhead view of jet deflector (G)

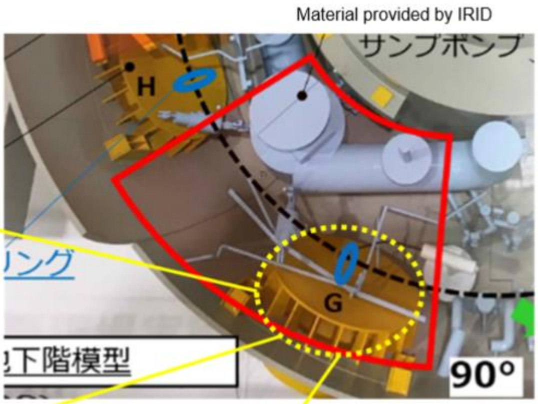


Photo 2. Status near jet deflector (G)

Photo 3. Status of jet deflector (G) back side

Figure 2-56　Status near the jet deflector (G) [2-32]

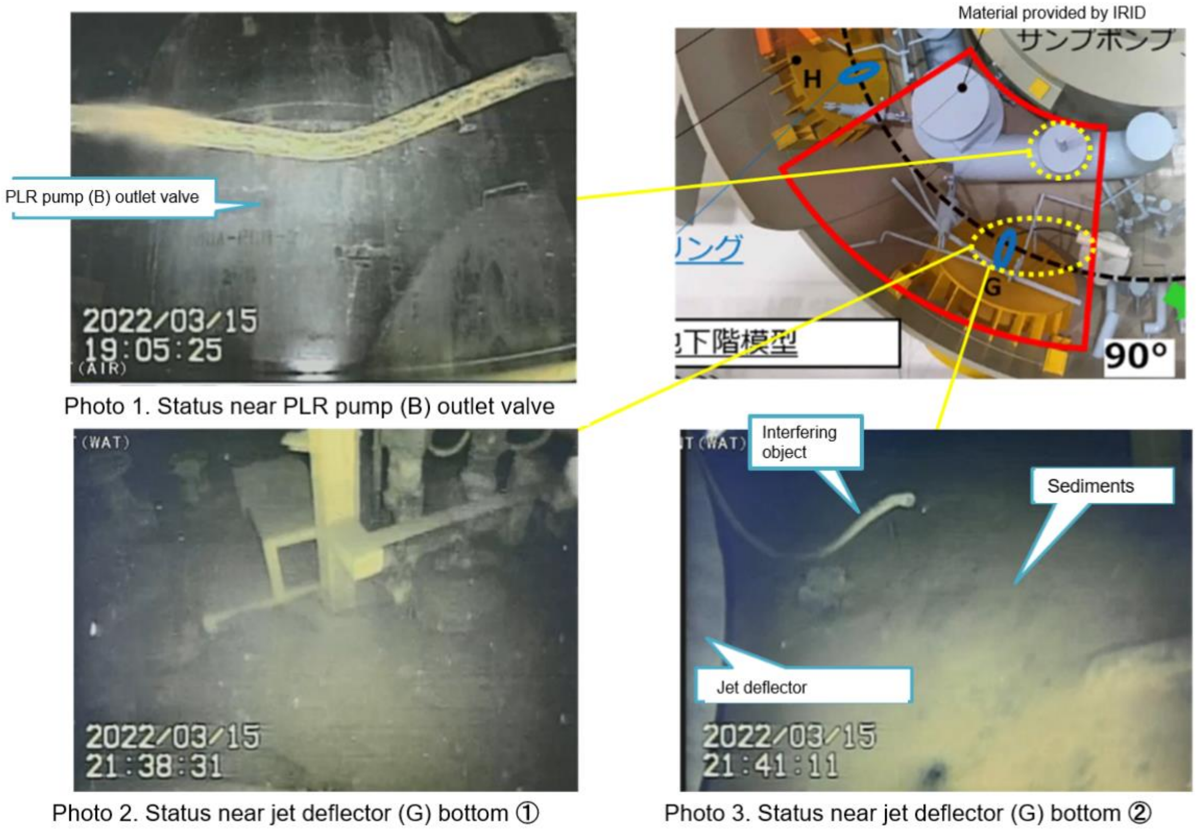


Figure 2-57　Status near the PLR pump (B) and near the jet deflector (G) [2-32]

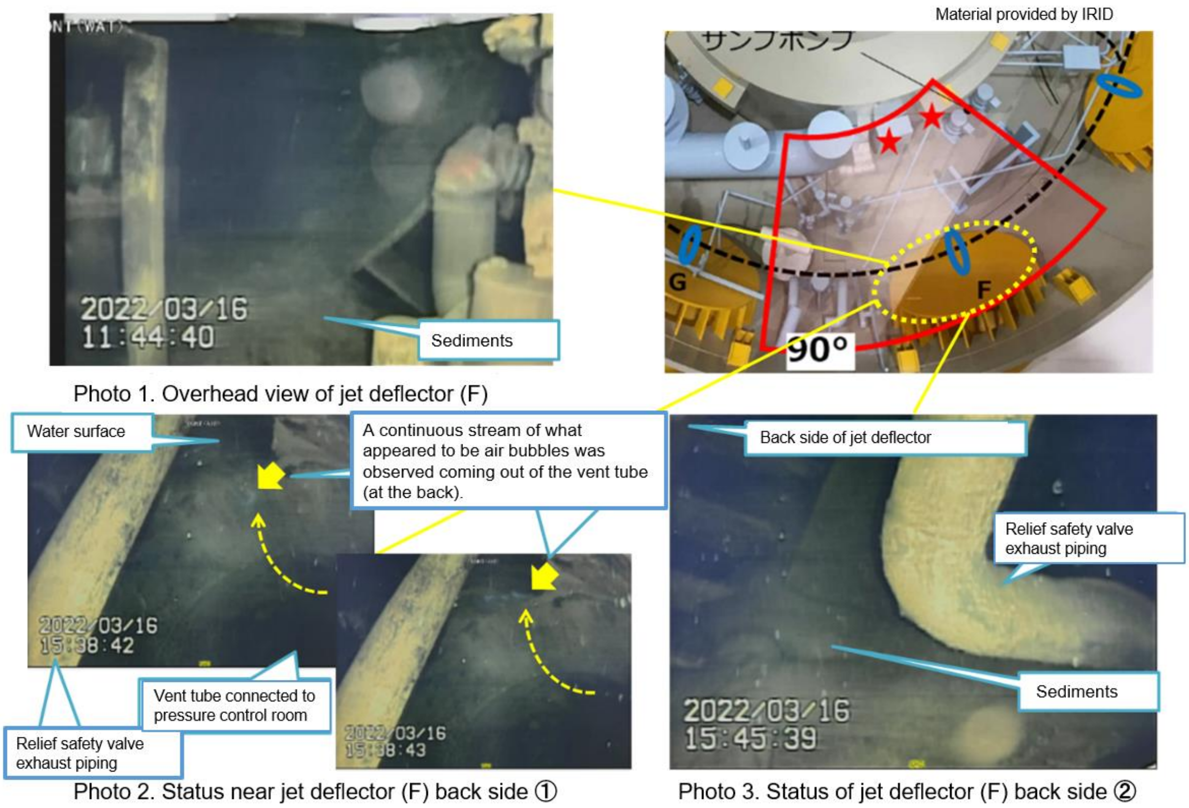
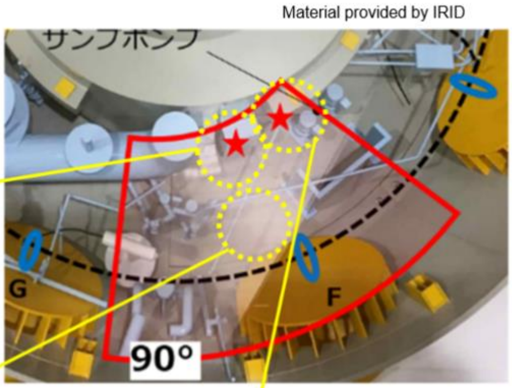


Figure 2-58　Status near the jet deflector (F) [2-32]

Photo 1. Status near pedestal base ①



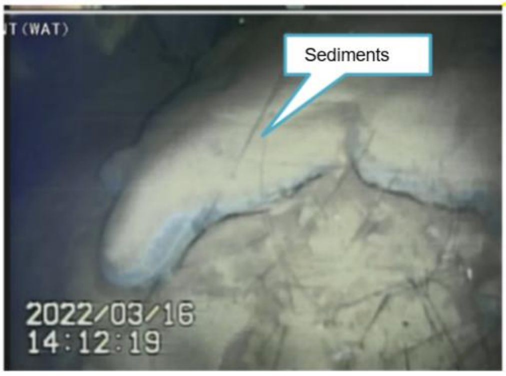


Photo 2. Status near jet deflector (F)

Photo 3. Status near pedestal base ③

Figure 2-59 Status near the pedestal and near the jet deflector (F) [2-32]

Photo 1. Overhead view of jet deflector (A)

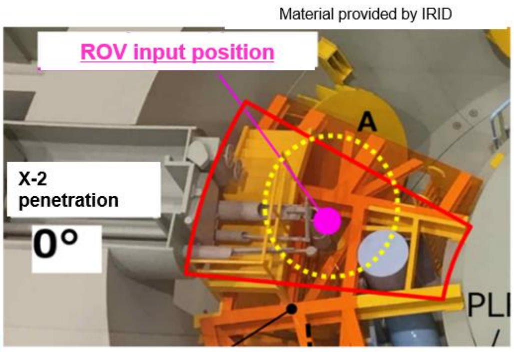


Photo 2. Status near jet deflector (A)

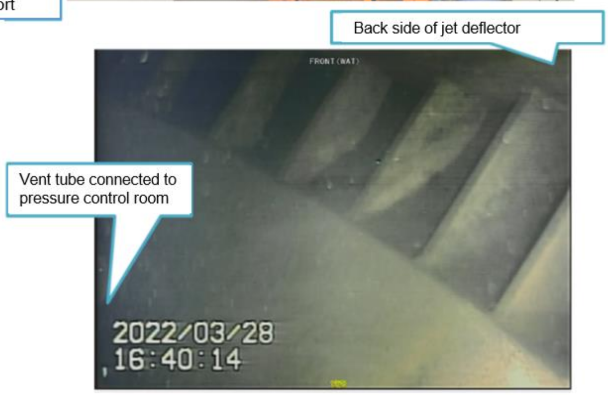


Photo 3. Status of jet deflector (A) back side

Figure 2-60 Status near the jet deflector (A) [2-33]

[2-33] Handout document: Unit-1 PCV Internal Investigation Status, 100th Decommissioning, Contaminated Water and Treated Water Response Team Joint Meeting, March 31, 2022.

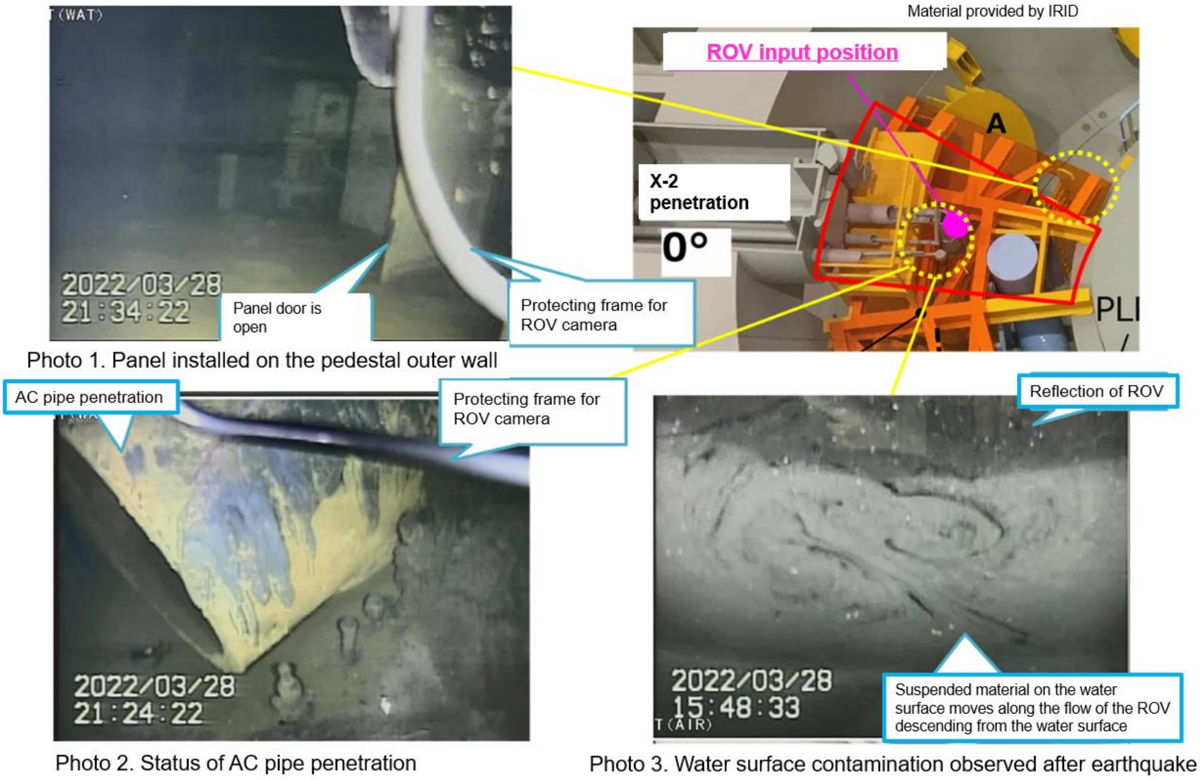


Figure 2-61 Existing structures, etc. near the jet deflector (A) [2-33]

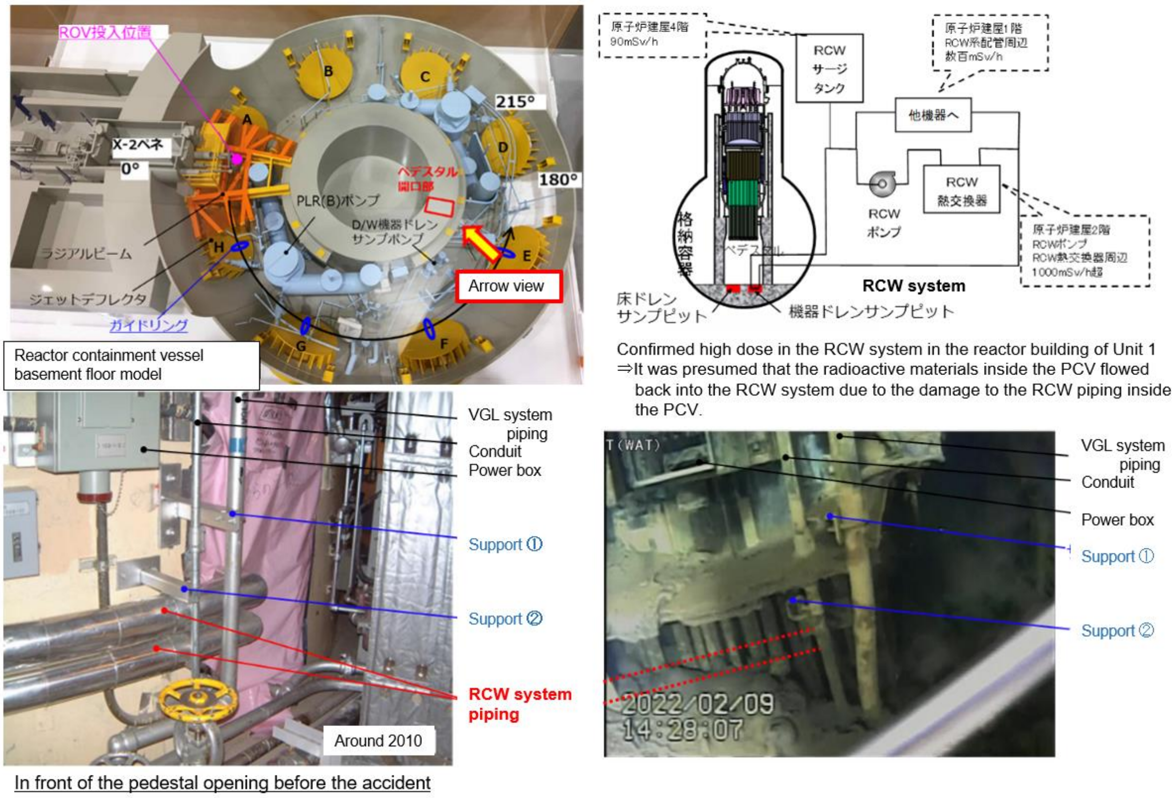


Figure 2-62 RCW system piping near pedestal opening [2-34]

[2-34] Handout document: Unit-1 PCV internal investigation status, 29th meeting of the study group for analysis of the accident at TEPCO's Fukushima Daiichi Nuclear Power Station, April 26, 2022.

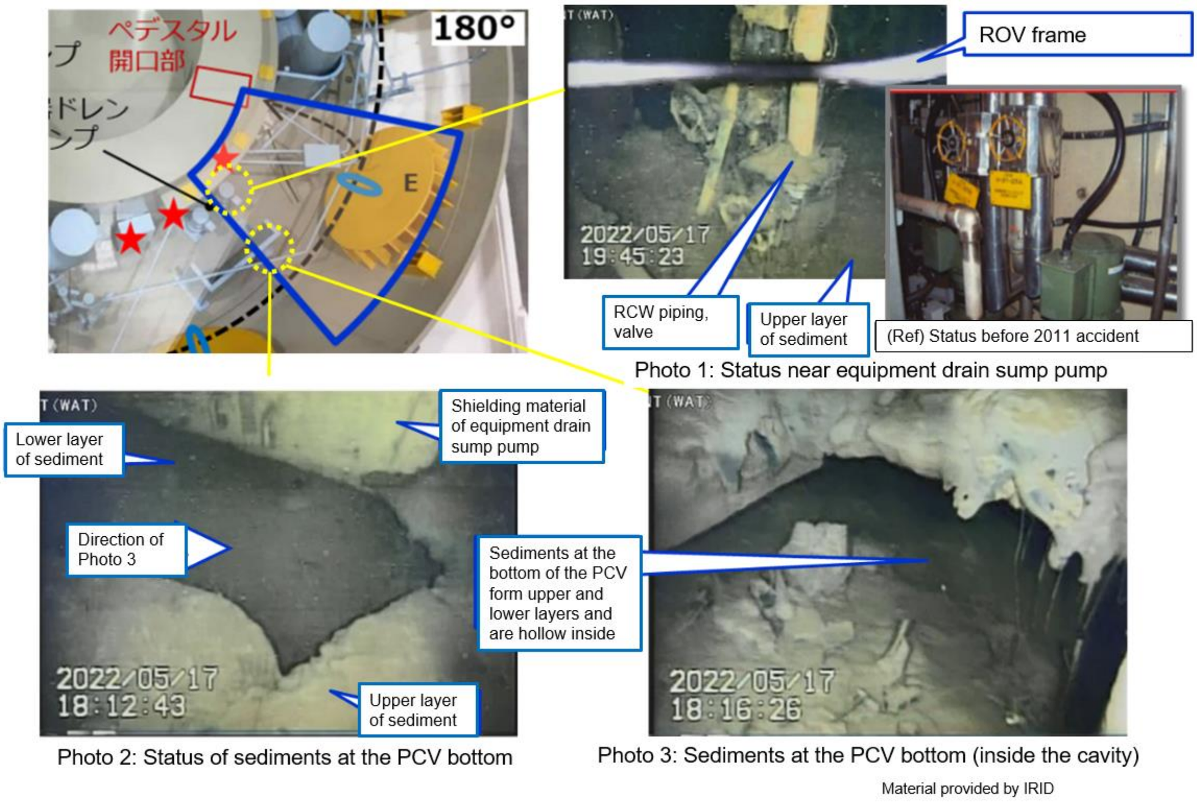


Figure 2-63 Status near equipment drain sump and bottom of PCV [2-31]

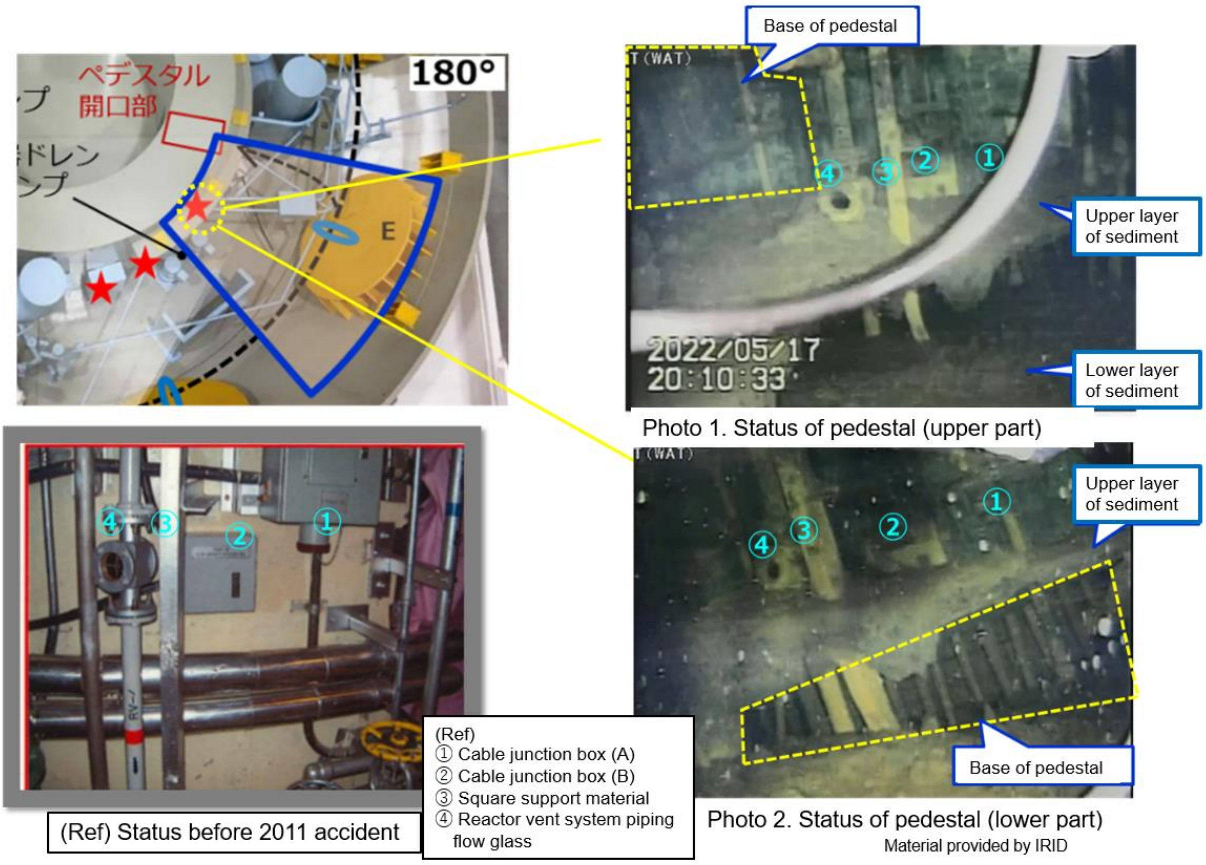


Figure 2-64 Status near the pedestal [2-31]

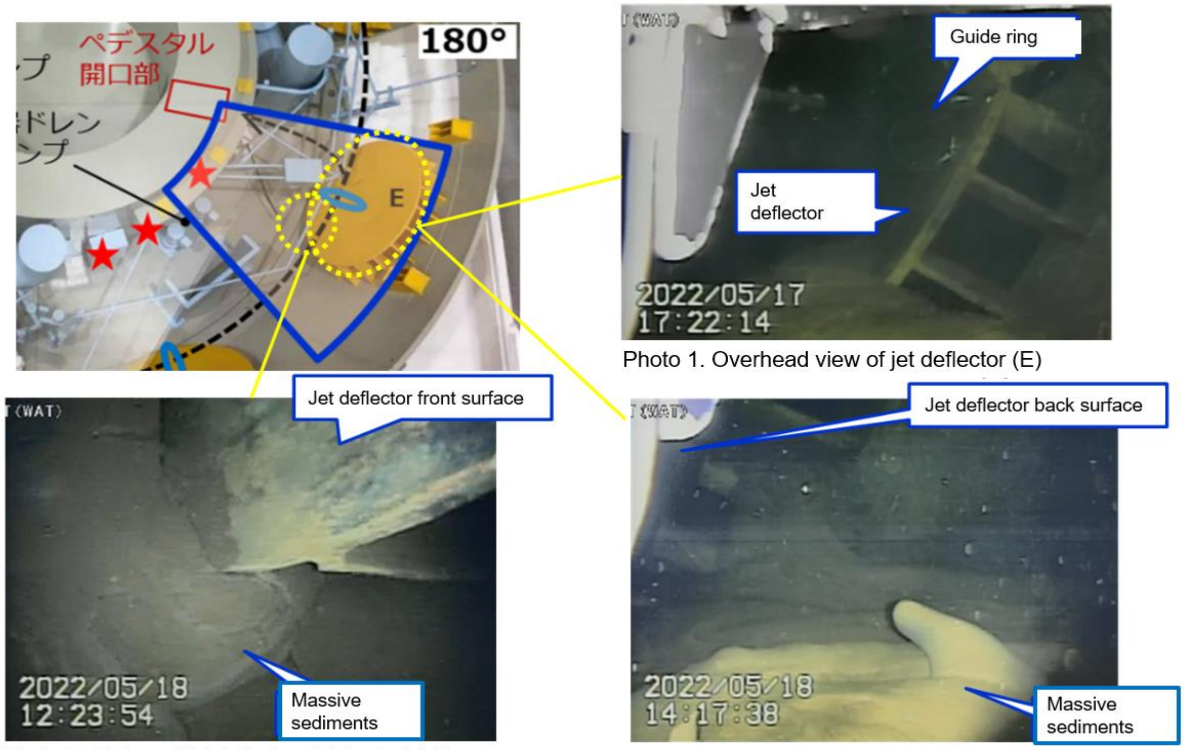


Photo 1. Overhead view of jet deflector (E)

Photo 2. Status of jet deflector (E) front side lower part

Photo 3. Status of jet deflector (E) back side

Material provided by IRID

Figure 2-65 Status near the jet deflector (E) [2-31]

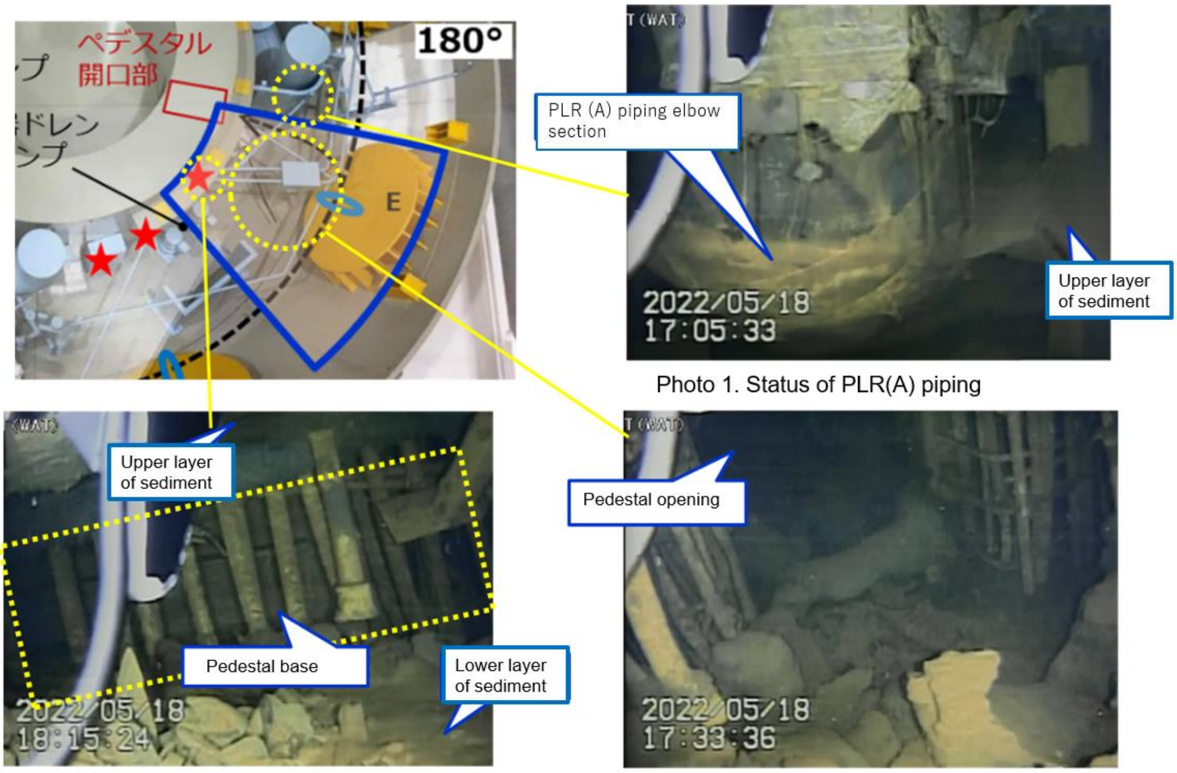


Photo 1. Status of PLR(A) piping

Photo 2. Status near pedestal base

Photo 3. Status of sediment in front of pedestal opening

Material provided by IRID

Figure 2-66 Status near PLR(A) piping and pedestal [2-31]

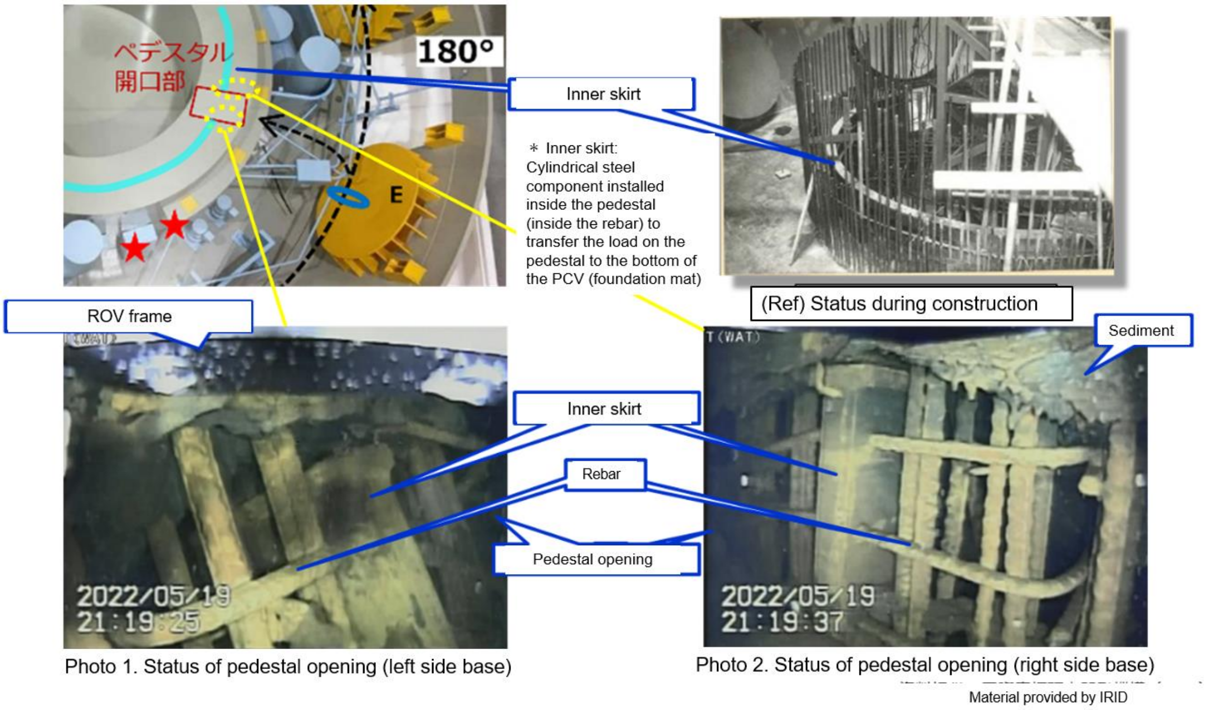


Figure 2-67 Pedestal opening (base) [2-31]

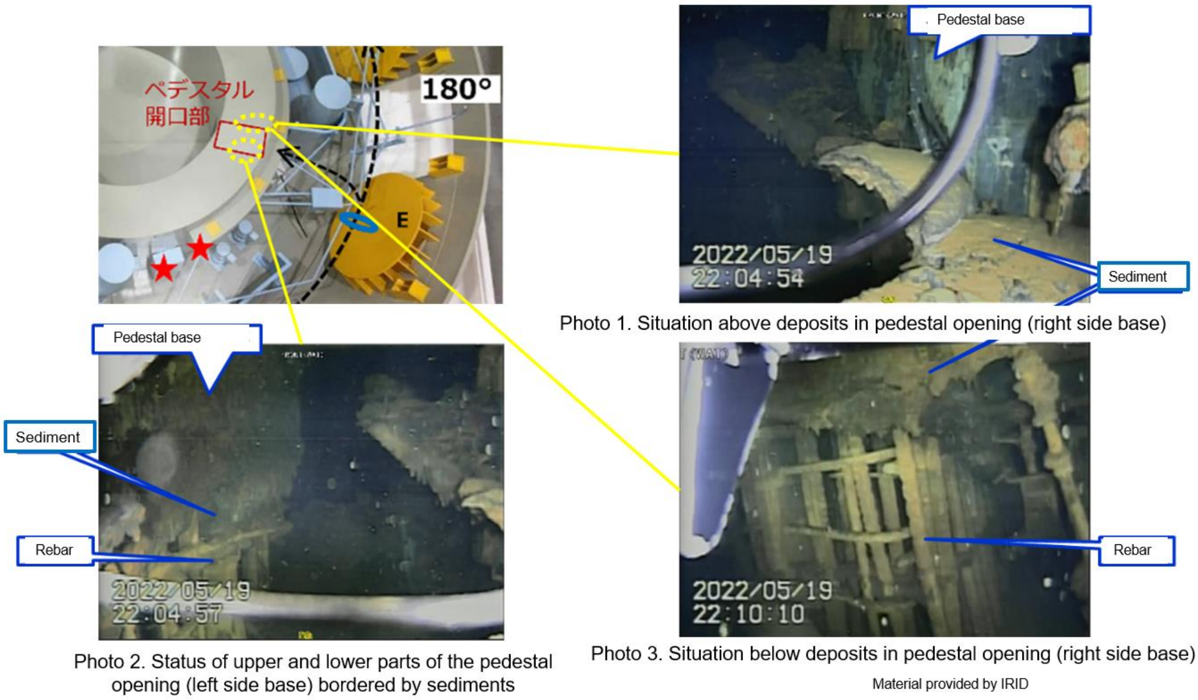


Figure 2-68 Status near pedestal opening (base) [2-31]

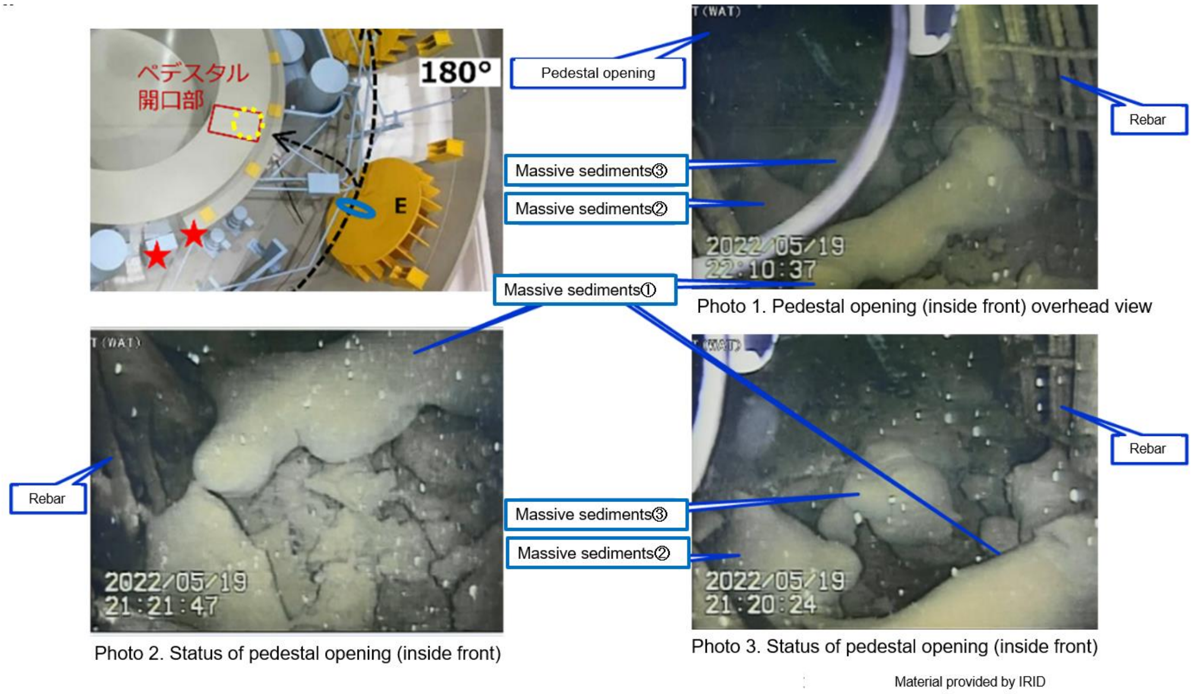


Figure 2-69　Status near pedestal opening (front inside) [2-31]

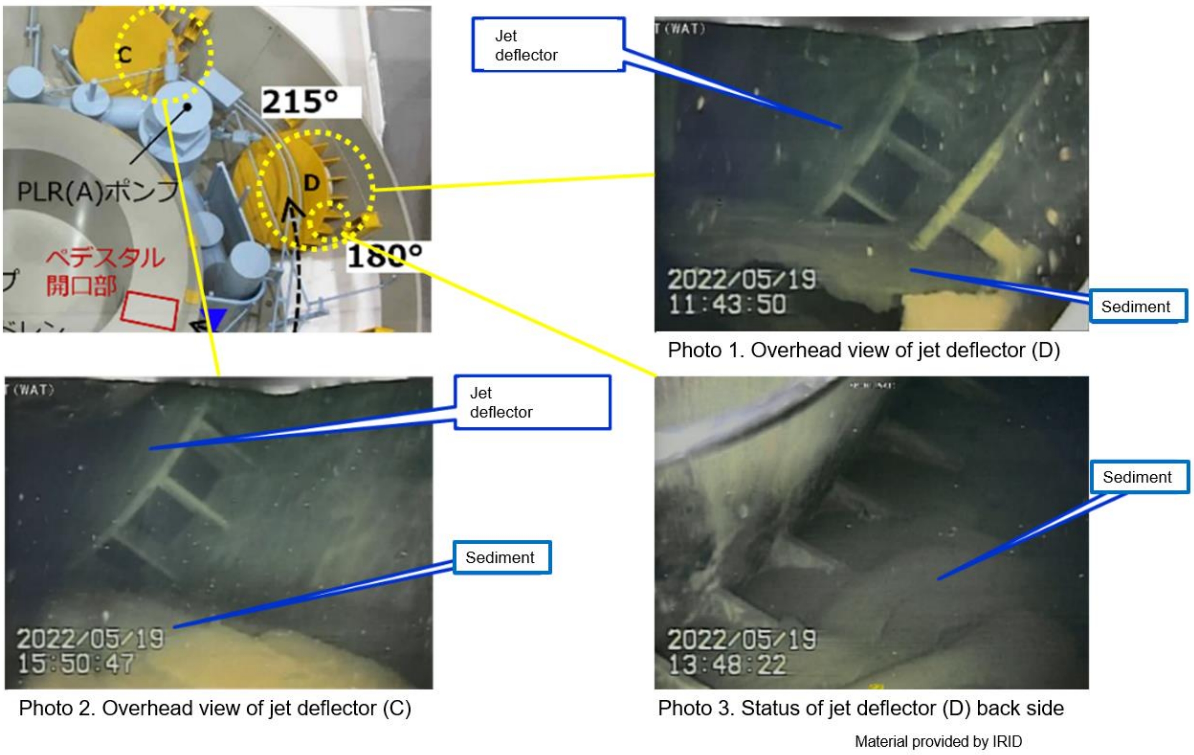
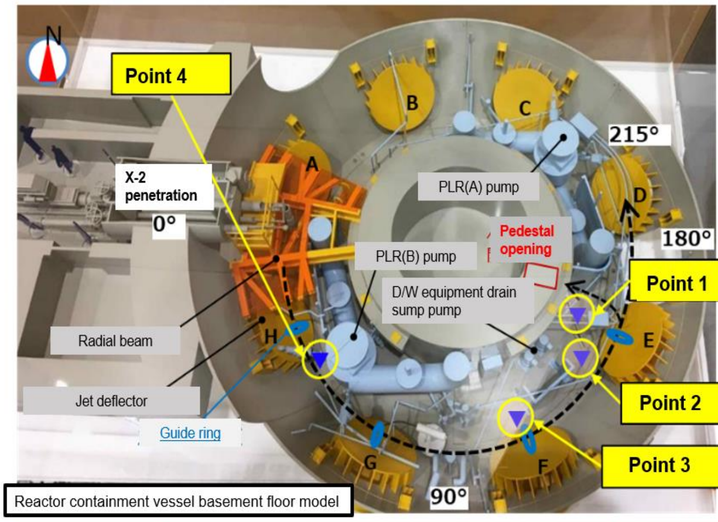


Figure 2-70　Status near jet deflectors (C, D) [2-31]

Neutron flux measurements on the sediment were conducted in May 2022 using a neutron detector mounted on the underwater ROV-A2, and thermal neutron flux was confirmed at all measurement points (Figure 2- 71).

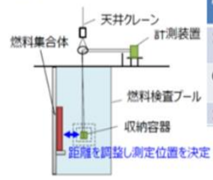


Location	Point 1	Point 2	Point 3	Point 4
Thermal neutron flux [/cm²/s]	48.0	29.1	50.2	5.8

- Thermal neutron flux is the total distance traveled by thermal neutrons in a unit volume per unit time.
- 60 minutes for each measurement
- The measurement results are expressed in terms of neutron flux evaluated from the number of counts for 60 minutes.

(Ref) Fuel assembly measurement result by B10 detector mounted on ROV-A2 @NFD

■測定方法
・燃料軸方向の中心部に設置
・燃料最寄位置を含め３つの位置で測定
(線量率：14.4, 6.5, 1.5 Gy/h)
・測定時間：3分



線量率	線源-検出器距離	熱中性子束評価値※
14.4 Gy/h	約16 cm	8.8×10^1 /cm²/s
6.5 Gy/h	約33 cm	1.1×10^1 /cm²/s
1.5 Gy/h	約78 cm	0 /cm²/s

Material provided by IRID

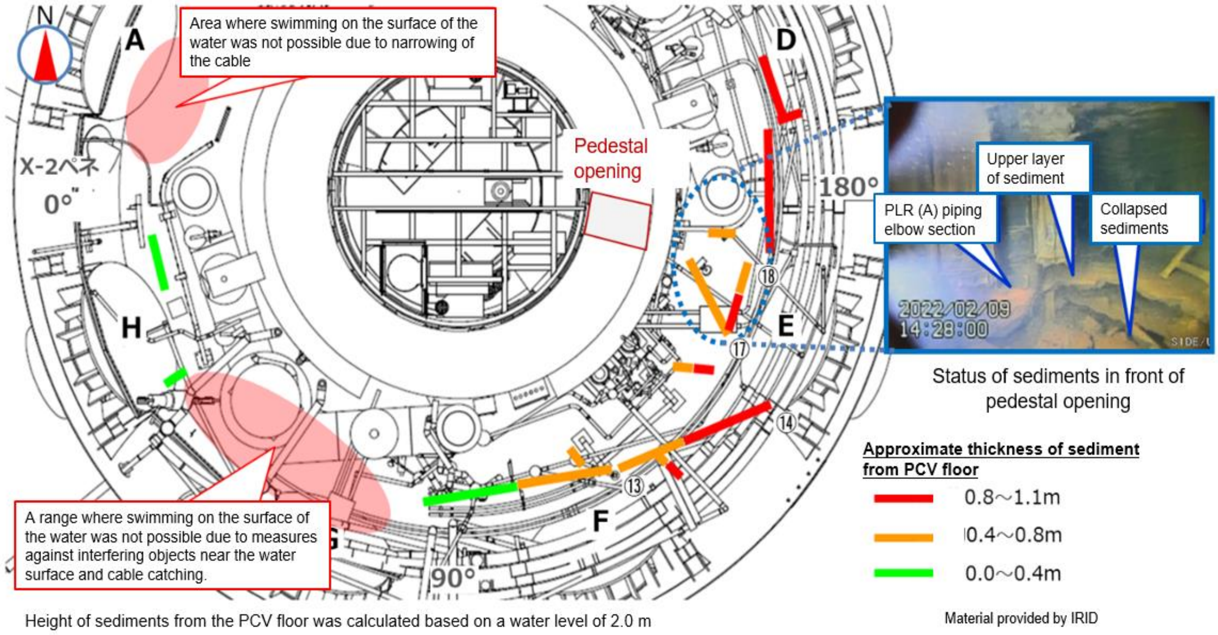
Figure 2-71　Neutron flux measurement results [2-31]

In June 2022, the distance from the water surface to the sediments and the height (thickness) of the sediments were estimated by comparing ultrasonic measurement data acquired using an underwater ROV-C with video images and location information of existing structures at the measurement location. The estimation results are shown in Figure 2-72.

From the ultrasonic measurement data and the underwater ROV-C/A2 survey videos, it was found that the thickness of the powdery and muddy sediments was less than expected. In addition, the internal conditions (existence of cavities, etc.) of sediments (including powdery/muddy and plate-like/massive sediments) could not be evaluated from the results of this survey. It was confirmed that the thickness of deposits from the bottom of the PCV was relatively high near the pedestal opening and gradually decreased as the X-2 penetration was approached, where the underwater ROV was put into the PCV.

The sediments in front of the pedestal opening were partially lowered (dotted blue box in Figure 2-72), which was presumed to be due to collapsed sediments based on video images obtained during the survey.

As a survey point that characterizes the state of the deposits, a cavity was confirmed from the survey video at ⑬-⑭ in Figure 2-72, and the measurement results also confirmed a difference in level in the cavity (Figure 2-73). At ⑰−⑱ in Figure 2-72, the collapsed deposits were confirmed from the survey videos, and the unevenness of the collapsed deposits was also confirmed in the measurement results (Figure 2-74).



r

Figure 2-72 Results of sediment thickness estimation by ROV-C [2-35]

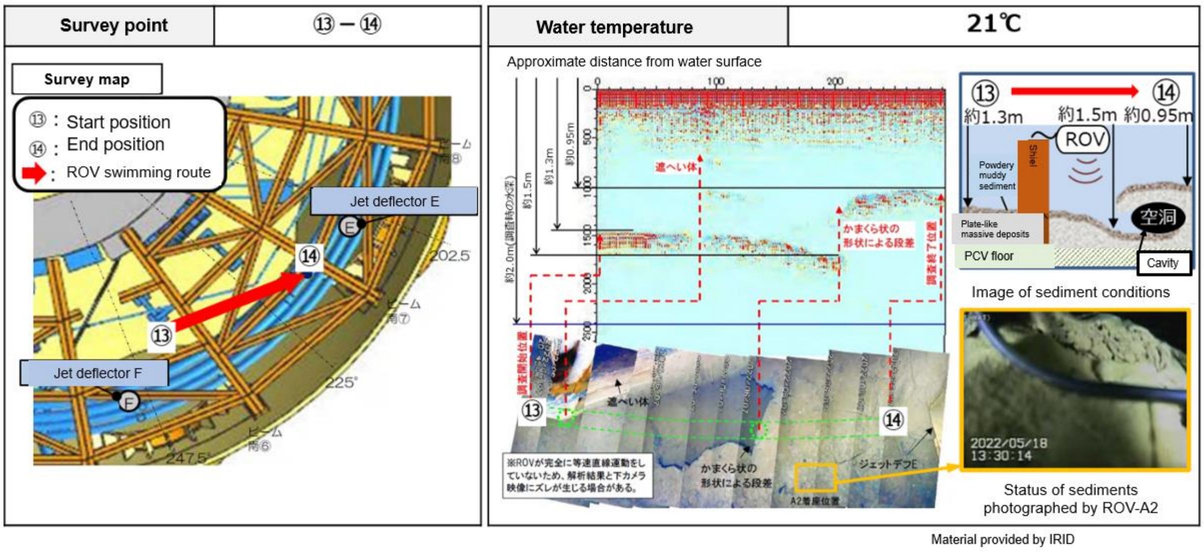
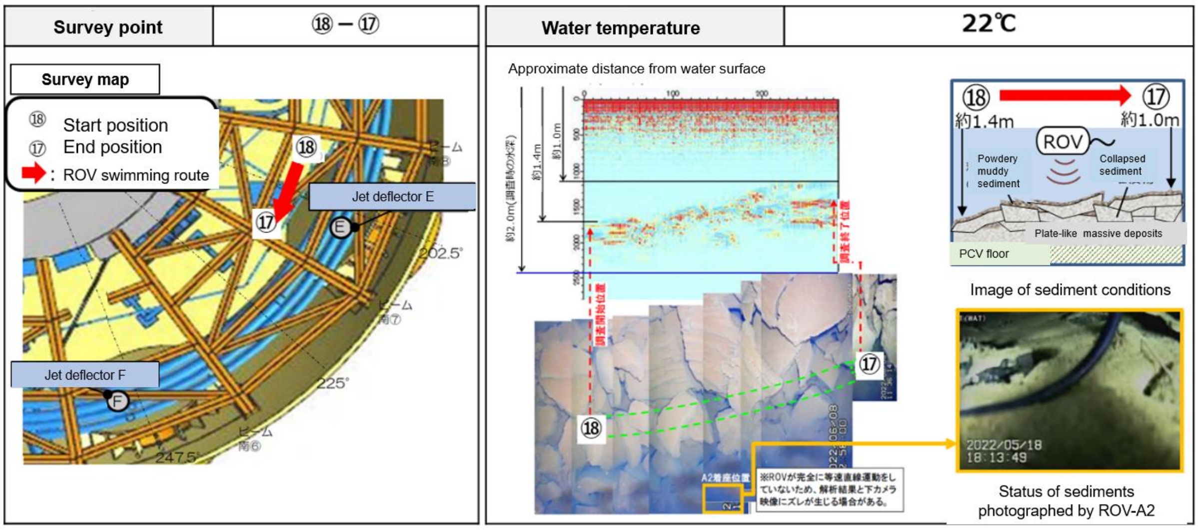


Figure 2-73 Evaluation results for survey points ⑬-⑭ [2-35]

[2-35] Handout document: Unit-1 PCV Internal Investigation Status, 100th Decommissioning, Contaminated Water and Treated Water Response Team Joint Meeting, July 28, 2022.



Material provided by IRID

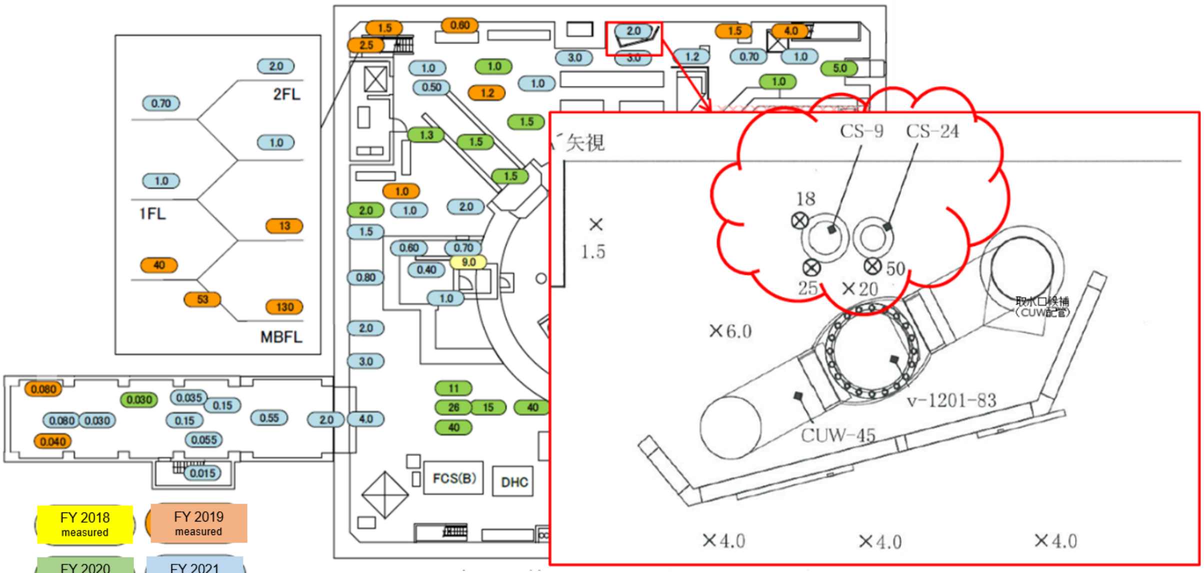
Figure 2-74 Evaluation results for survey points ⑱-⑰ [2-35]

(18) Measurement results of surface dose rate of piping in the CS (A) system test line and others in Unit-1 [UPDATE]

In June 2022, a site survey was conducted in order to sample the water contained in the S/C to measure the radiation dose rate of the CUW piping and the surrounding area, which is a candidate for the installation of the Unit-1 S/C water intake facility. When the surface dose rate of the CS piping adjacent to the CUW piping was measured, it was about 50 mSv/h for the CS (A) system test line (CS-24) and about 25 mSv/h for the CS (A) system pump outlet line (CS-9) (Figure 2-75).

The reason for the high dose rate in the abovementioned piping is assumed to be the migration of fission products (FPs) from the RPV or S/C. Since CS-24 had a higher dose rate than CS-9*, where FPs could easily migrate from the RPV, FP migration from S/C is considered (Figure 2- 76).

*: This was because there is no sluice valve from the branch of CS-9/CS-24 to the dose rate measurement point of CS-9.



From “Air dose rate inside buildings May 20, 2022”

Figure 2-75 Dose rate measurement results of CUW piping and surrounding [2-36]

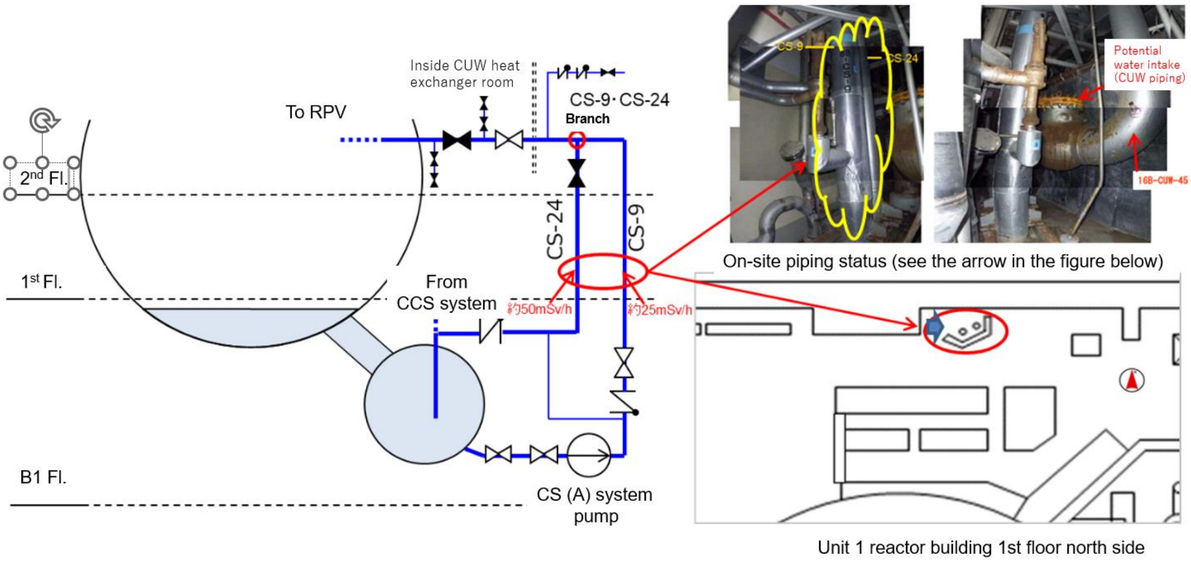


Figure 2-76 System status of CS (A) system piping [2-36]

[2-36] Handout document: Measurement Results of Piping Surface Dose for CS(A) System Test Line and Others, 30th meeting of the study group for analysis of the accident at TEPCO's Fukushima Daiichi Nuclear Power Station, March 30, 2022.

3. Conditions of Unit-2 core and the PCV

(1) In-containment water level measured

In March 2012, investigation was conducted into the PCV of Unit-2, when photos were taken by cameras, the level of water retained in the PCV was confirmed, and dose rates and temperatures were measured [1] by inserting survey devices into the PCV through a hole dug at the PCV penetration (X-53, on the first floor of the reactor building).[3-1]

The level of water retained was confirmed to be about 60cm above the D/W floor by the video image scope (as of March 26, 2012) (Figure 3-1).

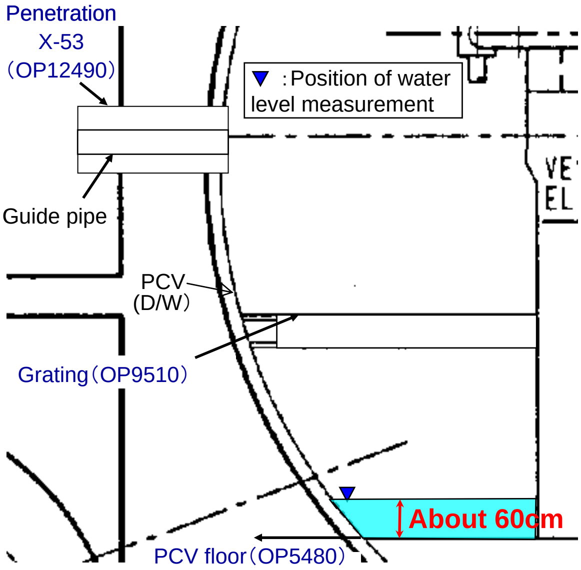


Figure 3-1 Measured level of retained water in Unit-2 PCV

(2) Survey results near the PCV pedestal opening of Unit-2

In July and August 2013, a survey was conducted inside the PCV of Unit-2, when instrumentation was introduced through the PCV piping penetration X-53 (reactor building first floor) to take camera images and make dose and temperature measurements in the vicinity of the control rod drive mechanism (CRD) replacement rail and pedestal opening (Figure 3-2). [3-2]

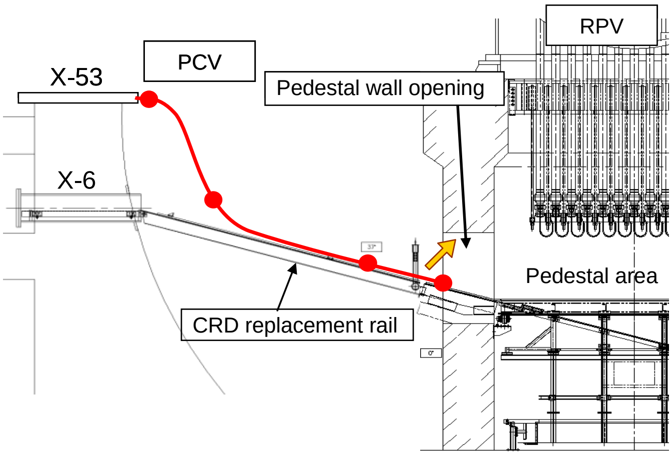


Figure 3-2 Survey areas inside Unit-2

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 (Figure 3-3).

Dosimeters measured the dose rates as far as the top of the CRD replacement rails. The values were about 45 to 80Sv/h. As supplementary information, dose rates were estimated from the camera image noises; they were about 30Sv/h near the landing point in the replacement rail and

[3-1] Results of PCV investigation and plan for identifying leak path, Technical workshop for Fukushima Daiichi accident, July 24, 2012.
[3-2] Handout document: Unit-2 PCV internal re-examination results, 7th Decommissioning Measures Steering Committee, August 28, 2013.

about 36Sv/h near the pedestal opening. Even when approaching the pedestal opening on the CRD replacement rail, no rapid increase in dose rate was observed that would suggest fuel debris was being approached.

(Reference information)

Pedestal opening of Unit-5

Figure 3-3 Photos taken inside the PCV pedestal at the pedestal opening (processed image)

(3) Test results of injecting nitrogen gas into the S/C of Unit-2

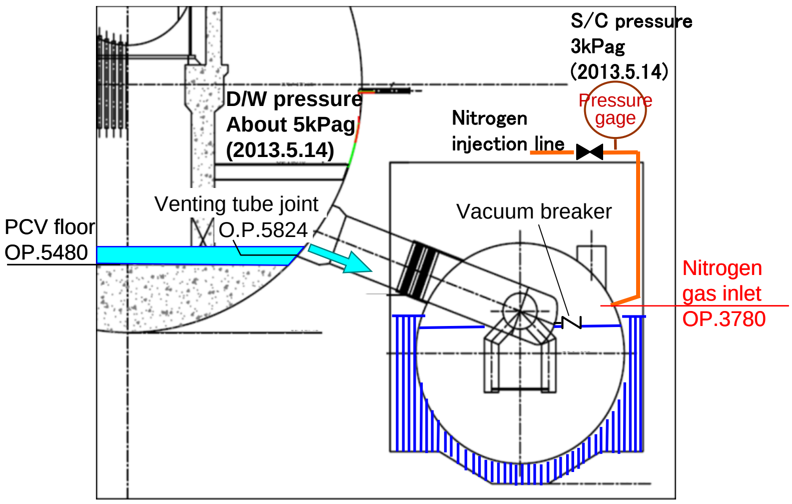


Figure 3-4 Closed space assumed in Unit-2 S/C

The S/C pressure was confirmed to be 3kPag (as of May 14, 2013) in a nitrogen gas injection test into the S/C done in May 2013. The absolute water level in the S/C was not accurately known, but it was confirmed to be approximately on the level of the nitrogen gas inlet (O.P. 3780), because some reasonable pressure due to the water head should exist at the inlet if the S/C were almost filled with water. If the low water level in the D/W is considered in combination, the water injected into the reactor vessel is considered to have reached the S/C via the D/W and venting tubes. If this hypothesis is correct, the current S/C water level will be on the same level as that of water retained in the torus room (Figure 3-4). [3-3]

Since December 2011, the hydrogen gas concentration and Kr-85 radioactivity measured by the containment gas control system of Unit-2 increased as a consequence of D/W pressure decreasing operations. This test was conducted to check if hydrogen and Kr-85 remained that had originated in the early phase of the accident as in the Unit-1 S/C.

The gradual pressure increases of the S/C from 3 kPag to 7 kPag before and after the injection confirmed that nitrogen gas had been injected into the S/C. But no change was observed in the hydrogen gas concentration and Kr-85 radioactivity measured by the PCV gas control system. Further tests were conducted to check if this was because there was no flow path from the S/C to the D/W or the hydrogen gas concentration in the S/C was already too low to send response signals.

In July 2013, upon injecting nitrogen gas into the D/W, a D/W pressure increase and an accordingly slight increase of S/C pressure were confirmed. Also, in October 2013, upon injecting nitrogen gas into the S/C, the S/C pressure increased to the level of the D/W pressure, after that, both pressures showed similar increasing trends in conjunction. When the nitrogen gas injection to S/C was terminated, the S/C pressure decreased concomitantly with the D/W pressure. [3-4]

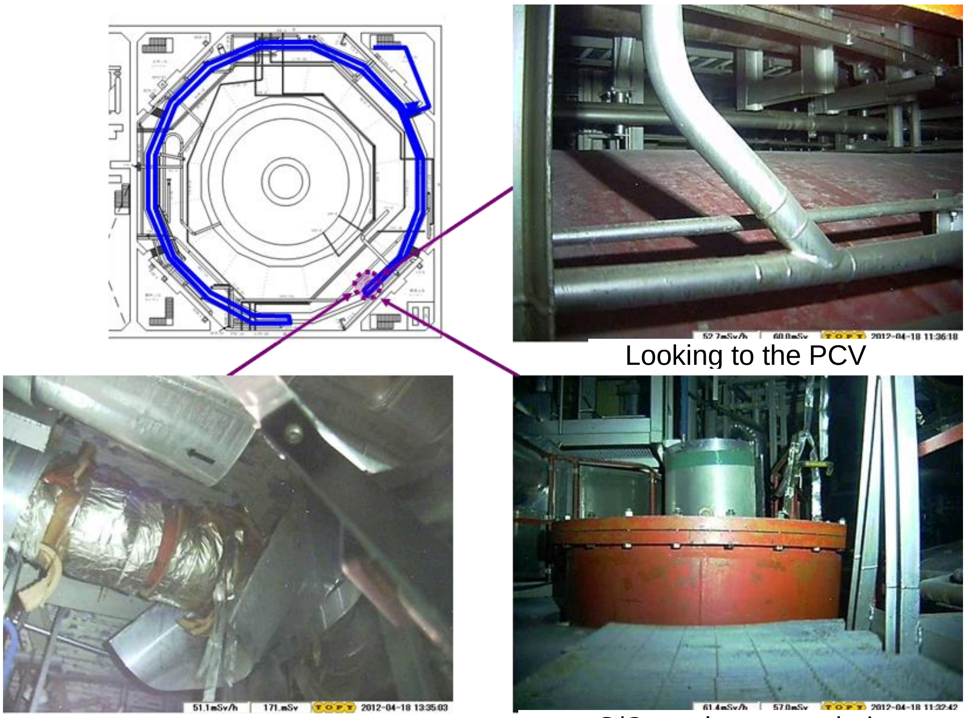
From these findings, it was confirmed that nitrogen gas injected into the S/C was flowing to the D/W. And also, from findings of nitrogen gas injection into the S/C flowing to the D/W with no change in hydrogen gas concentrations observed in the PCV gas control system, it was concluded that no more hydrogen gas remained in the S/C. It is considered, in this situation, that the vacuum breaker valve (OP. 3305) was not flooded and the nitrogen gas was flowing through this valve, because the water level in the reactor building was below about OP. 3400 during the tests and the S/C water level would follow the torus room water level (torus room water level minus level decrease due to internal pressure)

(4) Investigation of the torus room of Unit-2

In the Unit-2 torus room investigation in April 2013, a robot accessed the gallery inside. Videotaping, dose rates measurement, acoustic checks, etc. were carried out to the extent possible. [3-5]

No water leaking position in the S/C has been located yet. At least, no leak was confirmed on the flange, etc. of the S/C manholes, as far as the camera photos show (Figure 3-5).

[3-3] Handout document: [Reference 3] Status of review and implementation for each individual plan, 3rd Decommissioning Measures Steering Panel, May 30, 2013.
[3-4] Handout document: Nitrogen Injection Test for S/C Hydrogen Purge at Fukushima Daiichi Unit-2 (2nd test) （result), 1st Decommissioning and Contaminated Water Response Team Joint Meeting, December 26, 2013.
[3-5] Handout document: Investigation of the torus room on the basement floor of the Unit-2 reactor building, 5th Steering Committee, Government – TEPCO Joint Board on Mid- and Long-term Response Policy, April 23, 2012.



Looking to the PCV

Looking up from the S/C southeast manhole

S/C southeast manhole

Figure 3-5　Photos in the torus room of Unit-2 (part)

(5)　Investigation of the situation at the bottom of the vent tubes in the torus room of Unit-2

In the Unit-2 torus room, 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. [3-6]

No water leaking position in the S/C has been located yet. At least, no leak was confirmed from the lower end of venting tubes within the visible range (Figure 3-6).

[3-6] Handout document: Investigation results around the lower part of Unit-2 vent piping, 1st Decommissioning Measures Steering Panel, March 28, 2013.

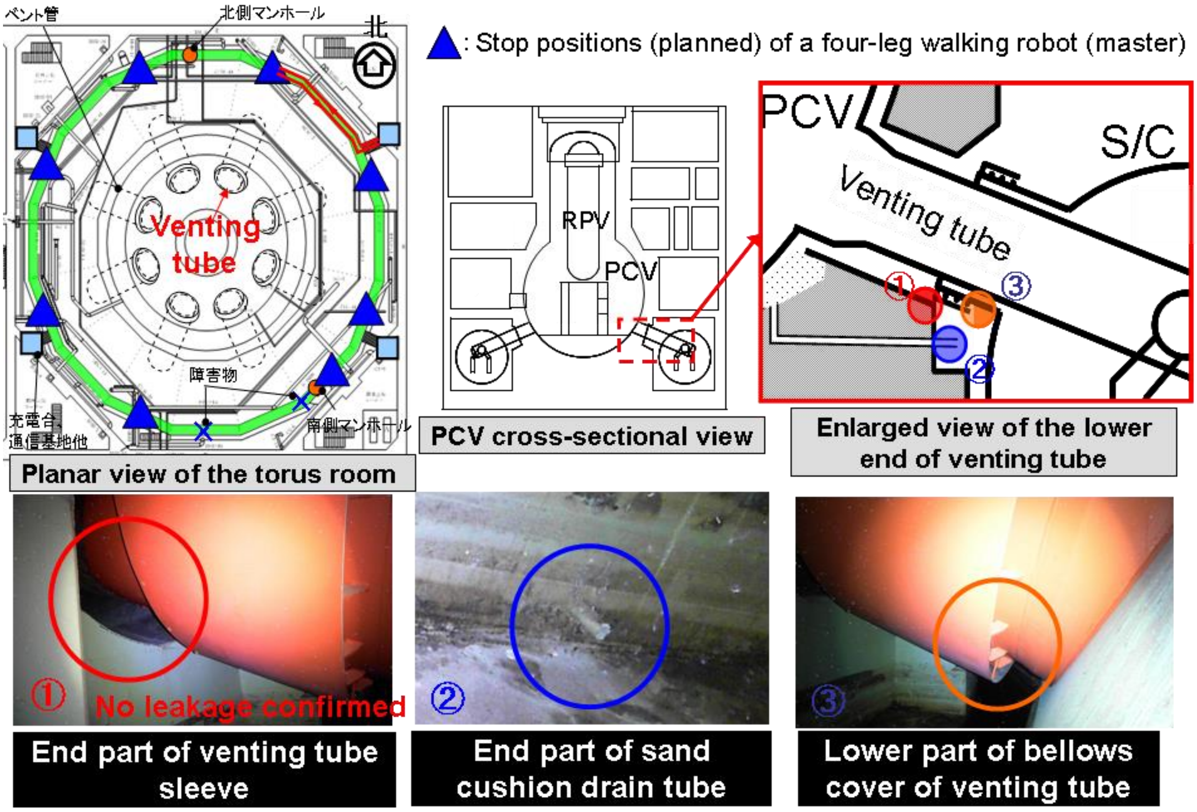


Figure 3-6　Photos around the lower end of the venting tubes in the torus room of Unit-2 (part)

(6)　S/C water level measurements of Unit-2

In January 2014, the S/C water level was remotely measured using ultrasonic techniques from the chamber outer surface. 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 estimated by observing where the reflective waves disappeared (Figure 3-7). [3-7]

The S/C water level is in correspondence with the water level retained in the torus room. This is consistent with the water level estimated earlier by the nitrogen gas injection tests. This information confirms that water leaks occurred at the S/C lower position (including piping).

[3-7] Handout document: Result of water level measurement in S/C of Unit 2, 2nd Decommissioning and Contaminated Water Response Team Joint Meeting, January 30, 2014.

Measurement date	Jan. 14, 2014	Jan.15, 2014	Jan. 16, 2014
S/C water level	About OP. 3210	About OP. 3160	About OP. 3150
Water level retained in the torus room (reference info.)	About OP. 3230	About OP. 3190	About OP. 3160
Level difference	About 20mm	About 30mm	About 10mm
Method of measurement	Direct distance measurement between underwater structures		

(Note) S/C water level seems to be affected by water level retained in the torus room

Measurement scene

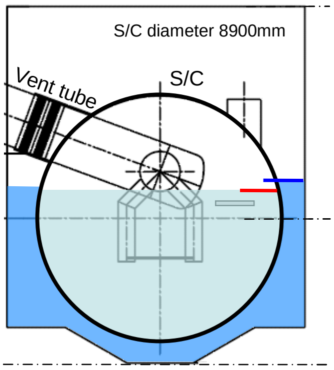


Figure 3-7 S/C water level measurements of Unit-2

(7) Investigations relevant to the rupture disk in the Unit-2 SGTS room (UPDATE)

Dose rates were measured in November 2014, around the rupture disk and the standby gas treatment system (SGTS) filters mounted in the Unit-2 SGTS room as a step to solve the Unit-2/Issue 9 “Rupture disc actuated at Unit-2.”

Figure 3-8 illustrates the system configuration for venting from the primary containment vessel (PCV) to the Unit-1 and -2 stack. The green line is the venting line to release the PCV pressure when it exceeds its design pressure. This venting line bypasses the SGTS filters mounted, as early as the very beginning of the construction phase, on the emergency heating and ventilation air conditioning system. This venting line is also connected to the purge line as well as the reactor building heating and ventilation air conditioning (R/B HVAC) system line. Valves in the figure are shown in black when fully closed, or when the opening aperture is unknown, and white when fully opened. The opening aperture of the valve (MO-271) located immediately upstream from the rupture disk was recorded as being operated to 25% mid-open position as of March 13, 2011. The valve continues to hold the state even now. On the other hand, the valve immediately downstream on the S/C side of the PCV was operated to open its large and small vent valves until March 14, 2011, but their real states when the venting line pressure reached the rupture disk working pressure remain unknown.

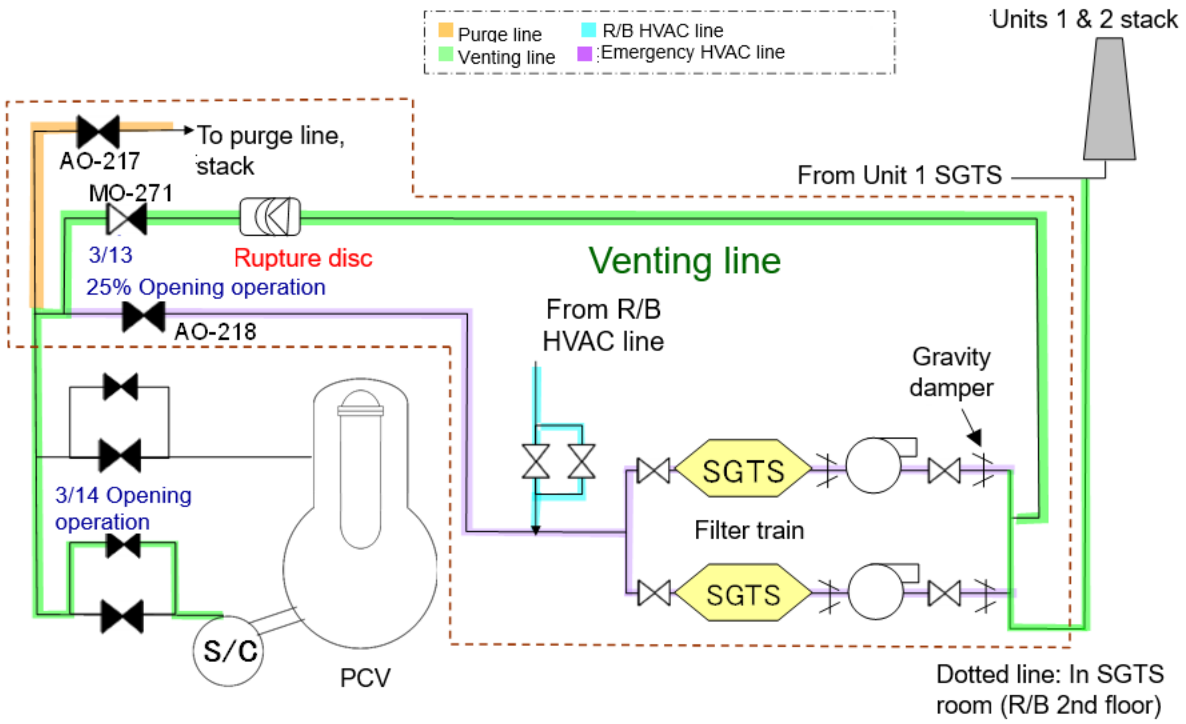


Figure 3-8 System configuration relevant to the rupture disk

Figure 3-9 shows the results of dose rate measurement around the rupture disk. The measurement was done on October 8, 2014. The dose rate measured was 0.30mSv/h from the north face, while it was 0.08mSv/h from the south face. Both were at about the same level of 0.30mSv/h (north face) and 0.12mSv/h (south face) before the rupture disk on the venting line, or 0.30mSv/h (north face) and 0.16mSv/h (south face) after the disk. These values indicate that the area is not contaminated to the extent predictable for such lines as the Unit-1 venting line, through which gas containing a large amount of radioactive materials went.

Furthermore, there exists a consistent tendency in the surrounding area that the dose rate on the north face are higher than those on the south face. This may indicate that the observed radiation levels were influenced by some unknown high level radioactive source existing on the north side of the rupture disk area, i.e., the radiation levels observed on the north face of the venting line and the rupture disk were actually measurements of radiation coming from that unknown high level radioactive source on the north side without the shielding effect of the piping, while the radiation levels obtained on the south face were actually measured after being shielded by the piping. Therefore, it is highly likely that the rupture disk and the piping around the disk are the least contaminated.

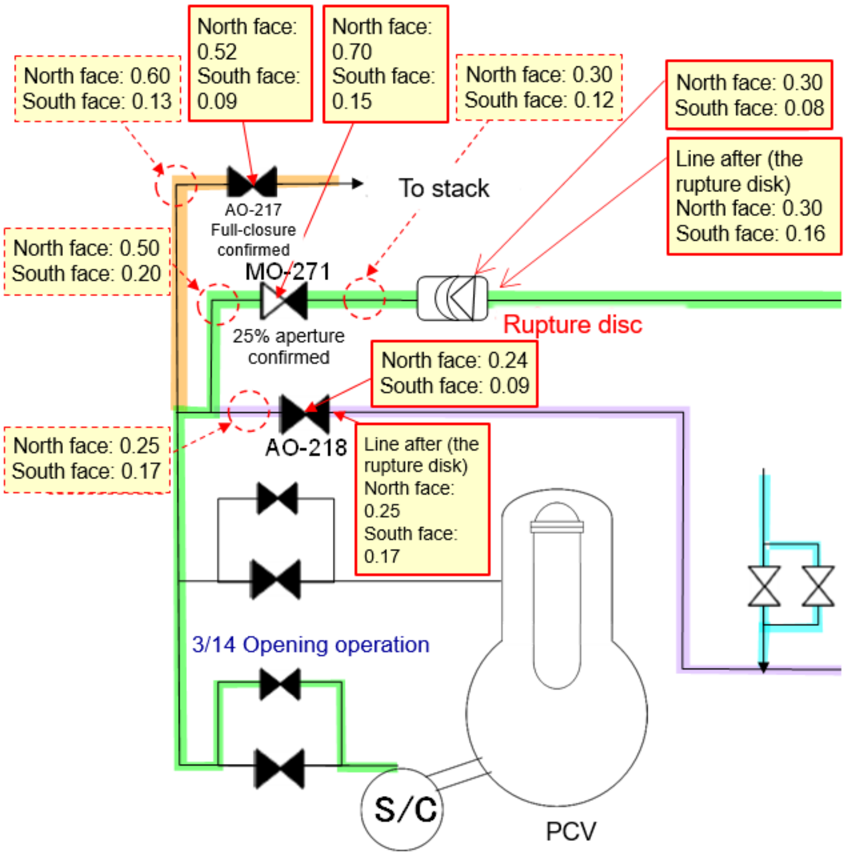
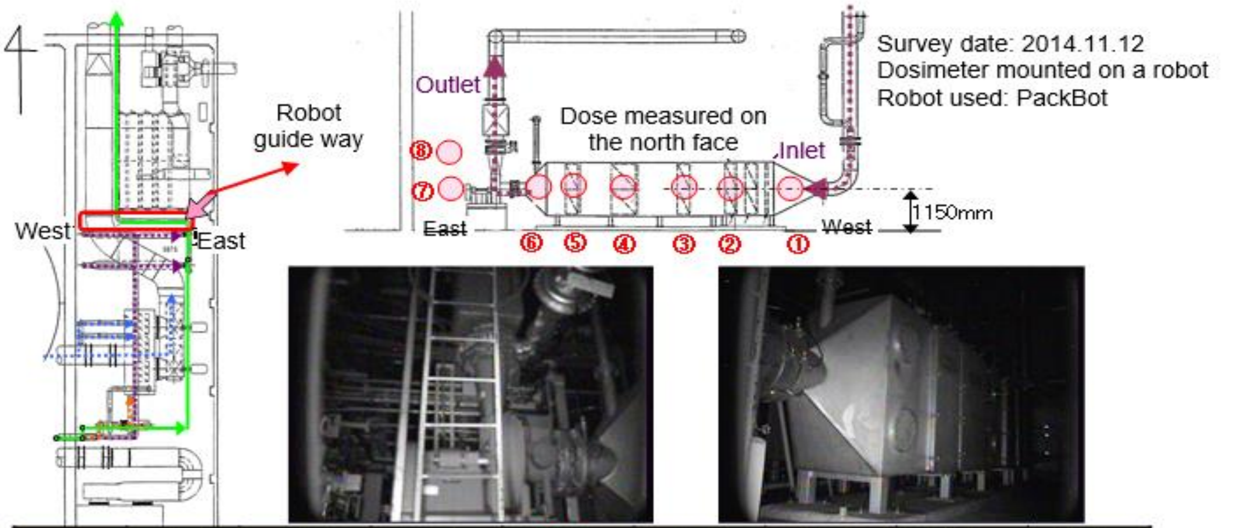


Figure 3-9　Dose rate observed around the rupture disk (mSv/h)

As described above, the unknown high level radioactive source on the north side of the rupture disk area was assumed to be fairly high dose rate. For this reason, the dose rate measurement on the north side was conducted using a robot (November 12, 2014).

Figures 3-10 and 3-11 show the radiation measurement results around the SGTS filters (A) and (B), respectively. For both filters (A) and (B), radiation levels as high as about 1Sv/h were obtained. The maximum contamination has been observed on the HEPA filter at the SGTS filter outlet. Generally, the SGTS filter captures radioactive materials from its inlet side, which means that the results observed may indicate the gas containing the radioactive materials flowed into the SGTS filter from the opposite direction (backward flow). As clearly recognizable from Figure 3-8, there are two possible paths for this backward flow: from the Unit-2 venting line and the Unit-1 venting line (this is the same situation as having occurred when hydrogen gas flowed backward from Unit-3 to Unit-4).

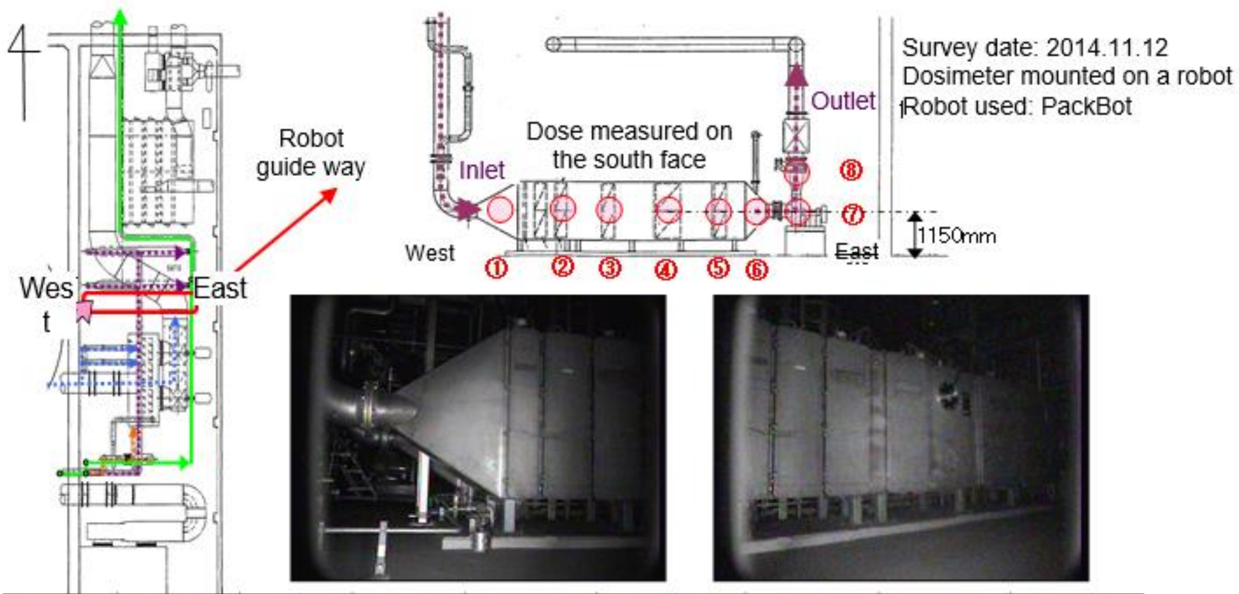
Although the survey did not identify any contamination in the vicinity of the rupture disk, it was not possible to determine whether the Unit-2 rupture disk was actuated.



Measure-ment point	⑧-A Outlet pipe	⑦-A Outlet pipe	⑥-A Outlet	⑤-A HEPA filter	④-A Charcoal filter	③-A HEPA filter	②-A Prefilter	①-A Inlet
Elevation	2170mm	1150mm	1150mm	1150mm	1150mm	1150mm	1150mm	1150mm
Dose rate	79mSv/h	85mSv/h	400mSv/h	1Sv/h *	460mSv/h	220mSv/h	140mSv/h	69mSv/h

* Measured at the point about 20cm from the filter train surface (about 65cm from filter center)

Figure 3-10 Radiation level observed (SGTS filter (A))



Measure-ment point	①-B Inlet	②-B Prefilter	③-B HEPA filter	④-B Charcoal filter	⑤-B HEPA filter	⑥-B Outlet	⑦-B Outlet pipe	⑧-B Outlet pipe
Elevation	1150mm	1150mm	1150mm	1150mm	1150mm	1150mm	1150mm	2170mm
Dose rate	15mSv/h	29mSv/h	44mSv/h	160mSv/h	850mSv/h *	500mSv/h	210mSv/h	120mSv/h

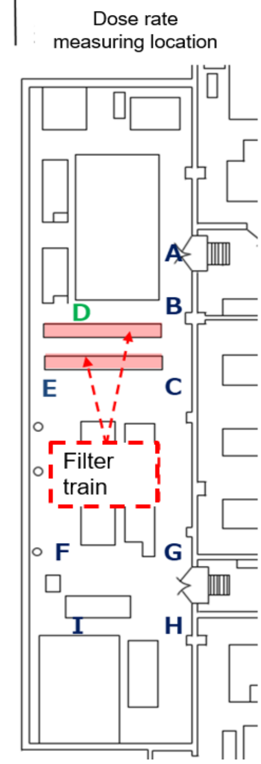
* Measured at the point about 20cm from the filter train surface (about 65cm from filter center)

Figure 3-11 Radiation level observed (SGTS filter (B))

Therefore, from December 2020 to January 2021, dose rate measurements were conducted in the Unit-2 SGTS room, around the filter train, and around the rupture disk using a remotely operated robot. The results of the dose rate measurements in the room showed that the air dose rate was higher near the downstream side of the filter train, with a maximum of 640mSv/h (Figure 3-12). The γ imaging measurements confirmed the presence of contamination downstream from the filter train (Figures 3-13 and 3-14). The results of dose rate measurements around the rupture disk confirmed that there was little contamination around the rupture disk (Figure 3- 15). Based on these results, it was concluded that the rupture disk was not activated and venting was not successful in Unit-2. It was also determined that the contamination in the filter train was due to backflow from Unit-1, with which it shared an exhaust stack.

A spot that appeared to be a leak was confirmed on the downstream side of the filter train B system (Figure 3-16). At the time of confirmation, no leak had occurred, and the leak trace itself was not wet. In Units-3 and -4, accumulated water was confirmed in the filter train, so it is thought that in Unit-2 as well, water remained in the filter train and leaked into the passage. The Nuclear Regulation Authority also conducted an investigation in the same room from July to August 2021, and since the location of the leak traces and the drain pipe of the filter train are different, the possibility is high that the leak is not from the filter train. [3-8]

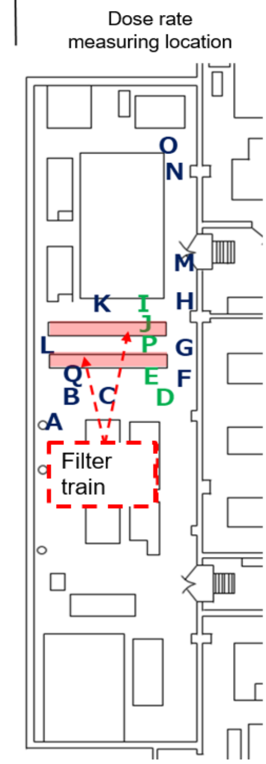
Main survey results (January 20)



Location	Dose rate (mSv/h)
A	17.0
B	71.5
C	61.8
D	140
E	12.6
F	0.52
G	0.18
H	0.094
I	0.17

Measuring height: about 1m

Ref: Preliminary survey results (August 7)



Location	Dose rate (mSv/h)
A	8.9
B	12.4
C	36.5
D	170
E	640
F	28.3
G	56.9
H	63.5
I	410
J	560
K	96.6
L	26.0
M	8.01
N	5.1
O	4.66
P	300
Q	13.5

Measuring height: about 1m

Figure 3-12 Results of air dose rate measurements in SGTS room [3-9]

[3-8] Handout document: Status of Field Investigation, 22nd Meeting of the Study Group on Analysis of Accident at TEPCO's Fukushima Daiichi Nuclear Power Station, September 14, 2021.
[3-9] Handout document: Results of the SGTS room investigation for Units-1 to -4 , 88th Decommissioning and Contaminated Water Response Team Joint Meeting, March 25, 2021.

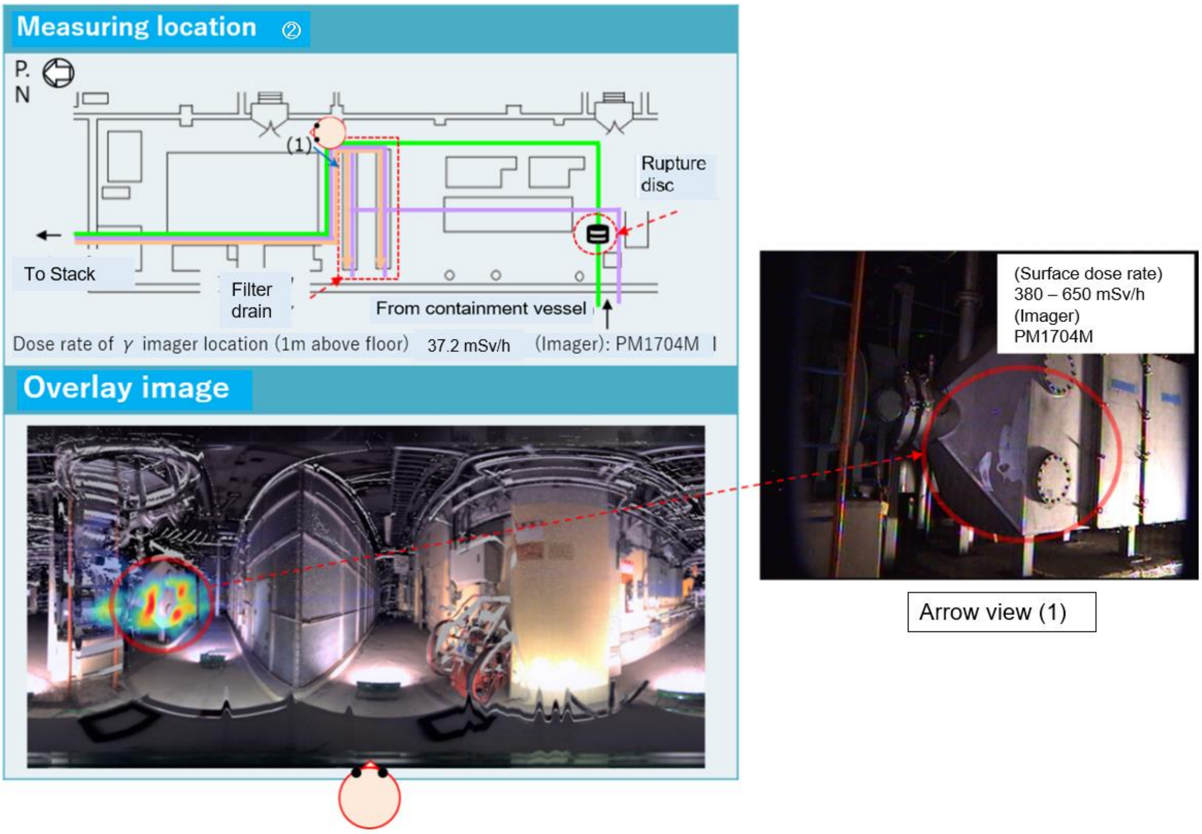


Figure 3-13 Results of γ imaging measurement in SGTS room (part 1) [3-9]

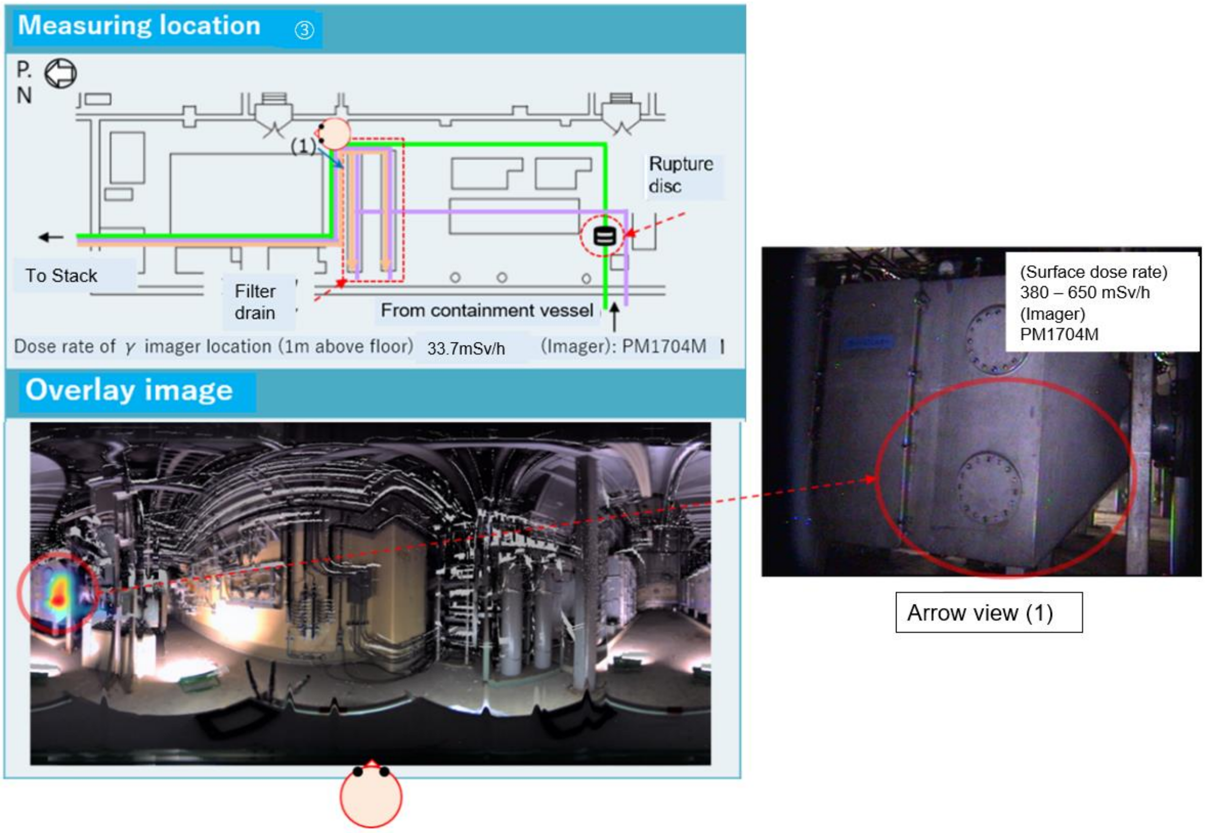
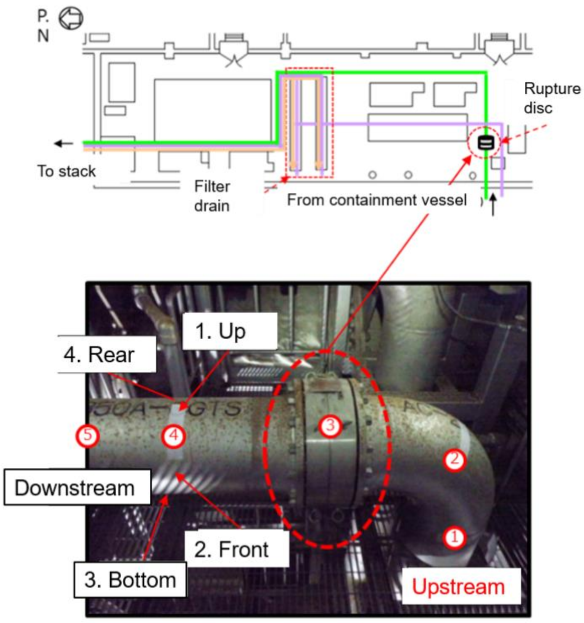


Figure 3-14 Results of γ imaging measurement in SGTS room (part 2) [3-9]



No.	Surface dose equivalent rate (mSv/h)	No.	Surface dose equivalent rate (mSv/h)
	γ ray		γ ray
①-1	0.10	④-1	0.20
①-2	0.10	④-2	0.15
①-3	0.15	④-3	0.10
①-4	0.050	④-4	0.10
②-1	0.10	⑤-1	0.15
②-2	0.15	⑤-2	0.20
②-3	0.15	⑤-3	0.15
②-4	0.050	⑤-4	0.10
③-1	0.20		
③-2	0.25		
③-3	0.15		
③-4	0.050		

Instrument: ICW

*Position of ①-⑤ is set at 50 mm intervals.

Figure 3-15 Dose rate measurement results around the rupture disk [3-9]

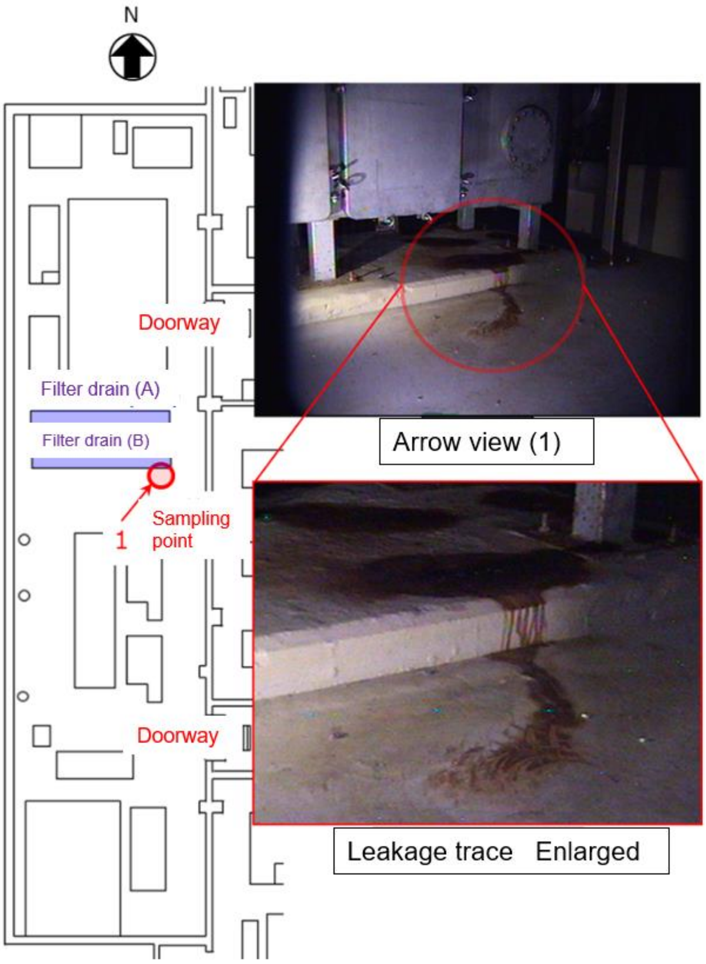


Figure 3-16 Leakage traces observed downstream from filter train B [3-9]

(8) Investigation of the area around PCV penetration X-6 [UPDATE]

In preparation for the internal investigation of the PCV and pedestal, shield blocks and iron plates were removed (between June 11 to October 1, 2015) from the front of the PCV penetration X-6 (see Figure 3-17 Building layout), which had been selected as the access route. When the area around the penetration was investigated, during the removal work, some melted matter was found and a high dose rate of more than 1,000mSv/h was noticed on the penetration flange and on the floor.

Figure 3-18 shows a photo of the melted matter. The melt hung from the penetration flange and lay spread on the floor. It is thought to be materials that had covered cables of the CRD replacement machine or O-ring materials of the penetration flange seals. The melt on the floor was solidified and confirmed to be easily removable by spatulas or similar tools.

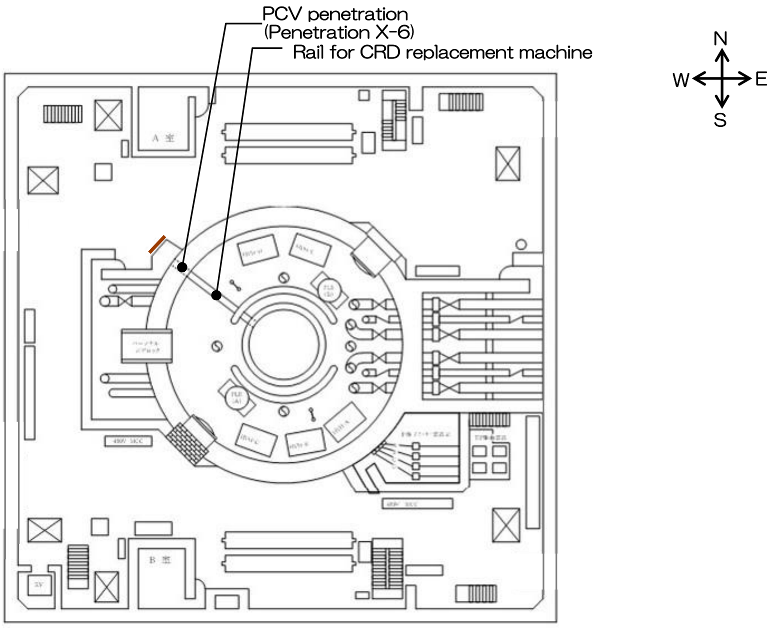


Figure 3-17 Layout on the ground floor of Unit-2 R/B [3-10]

[3-10] Handout document: Development of Technology for Investigating the Inside of Reactor Containment Vessel" Progress of the removal of the shielding block for the X-6 penetration from the inside of the containment vessel of the Unit-2 reactor (A2 investigation), 19th Decommissioning and Contaminated Water Response Team Joint Meeting, June 25, 2015.

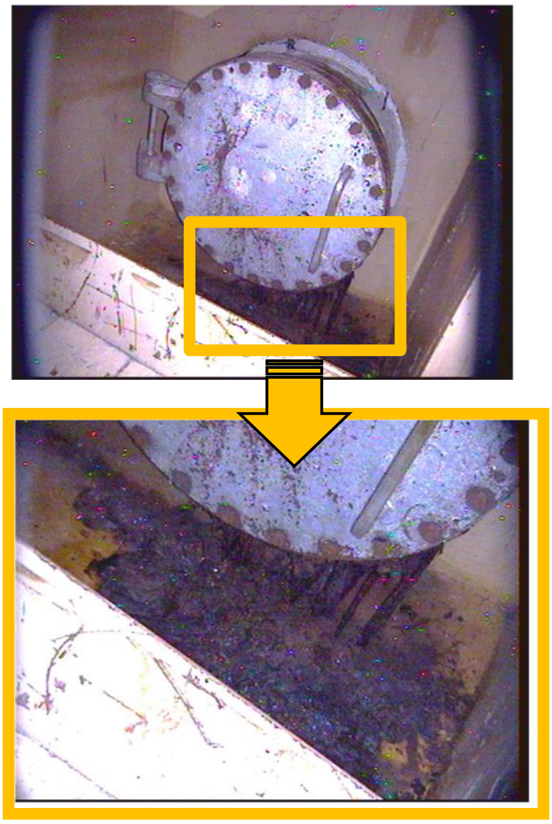
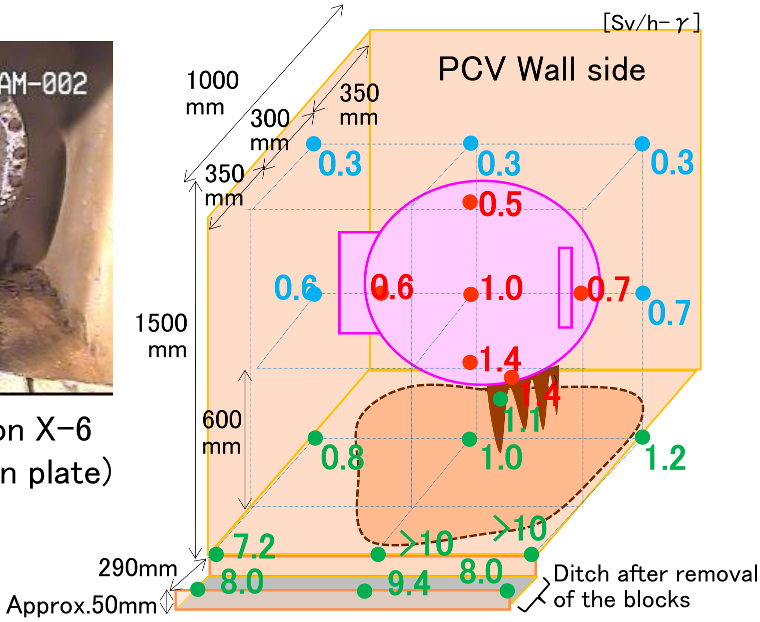


Figure 3-18 Melted matter at the penetration flange [3-11]

Figure 3-19 gives the radiation dose rate measured on the surface. An increasing tendency for the surface dose is noticed in the order: on the ceiling < in the middle < on the floor; and especially a high dose rate is observed in the ditch after the shield blocks had been removed. The contamination might have spread from the area where the melt lay into the ditch. If the radiation dose rate difference between the locations at the penetration and on the wall comes from the contribution of a radiation source inside the penetration X-6, it is estimated to be approximately 1Sv/h at the maximum.

[3-11] Handout document 3: Investigation inside Unit-2 reactor containment vessel on platform inside pedestal (A2 investigation). Study status of X-6 shielding block removal and implementation of investigation around X-6. 21st Decommissioning and Contaminated Water Response Team Joint Meeting, August 27, 2015.

Flange of Penetration X-6
(after removal the iron plate)



Red: Surface dose rate on the penetration surface

Blue: Surface dose rate on the ceiling/wall

Green: Surface dose rate on the floor and the molten object

Figure 3-19 Measurement results of surface radiation dose rate [3-12]

In addition, to obtain detailed information on the deposit status for the removal of sediments in the X-6 penetration, which would interfere in the PCV internal investigation and trial debris retrieval operations, an investigation device was inserted using a through-hole in the X-6 penetration hatch in October 2020 to conduct a contact investigation and 3D scan of the sediments. A schematic drawing of the investigating equipment is shown in Figure 3-20.

As a result of the contact survey, it was confirmed that the sediments changed shape upon contact and that the cables were not adhered to anything and could be lifted (Figure 3-21). In addition, deposits, cable status, and the X-6 penetration hatch hole were confirmed by video (Figure 3-22). The 3D scan results showed that the sediments in the X-6 penetration were deposited on the slope from the reactor building side toward the pedestal (Figure 3-23).

[3-12] Handout document: Results of the X-6 penetration contamination investigation at Unit-2 and future actions, 23rd Decommissioning and Contaminated Water Response Team Joint Meeting, October 29, 2015.

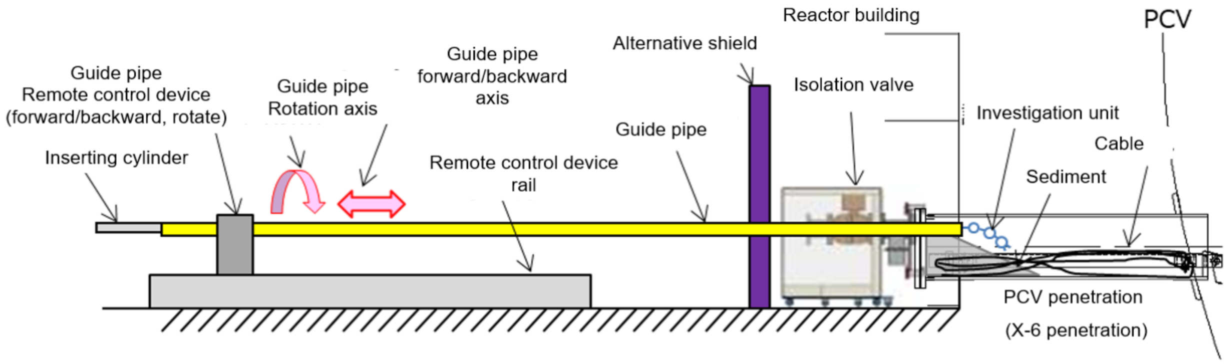


Figure 3-20　Schematic drawing of the set-up for the PCV internal investigation [3-13]

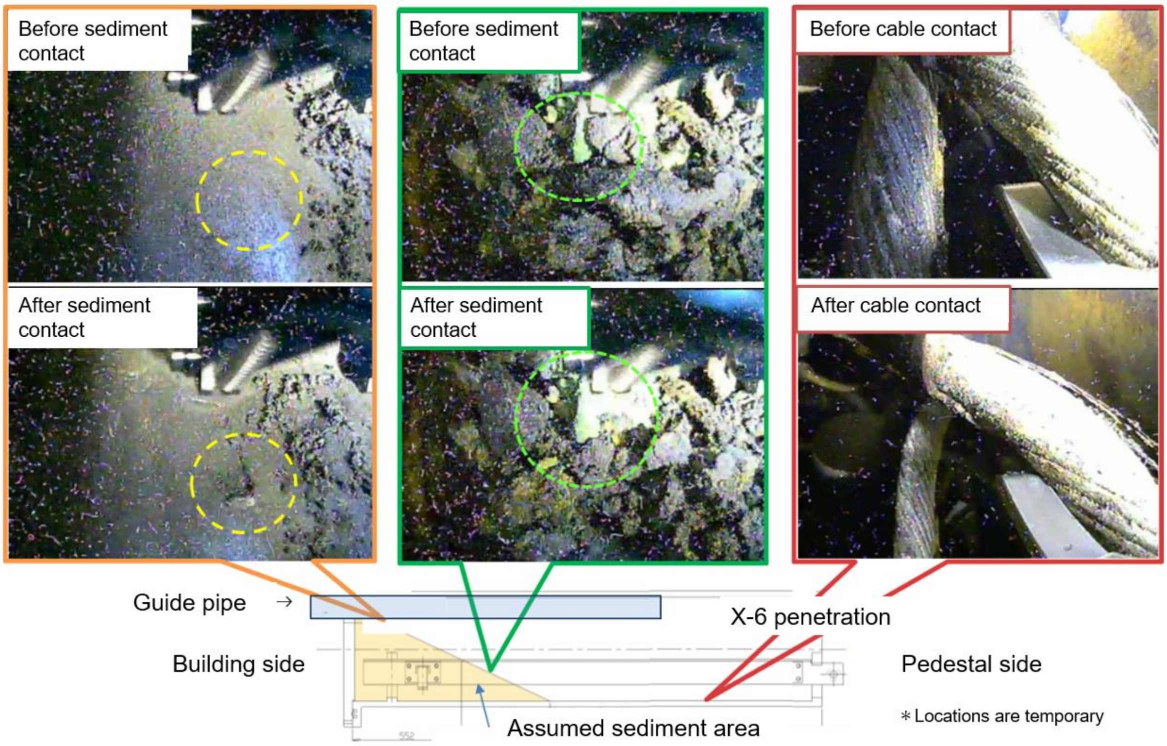


Figure 3-21 Contact survey of sediments in X-6 penetration result (1) [3-13]

[3-13] Handout document: Unit-2 PCV internal investigation and preparations for trial retrieval, Results of X-6 penetration sediment investigation, 84th Decommissioning and Contaminated Water Response Team Joint Meeting, November 26, 2020.

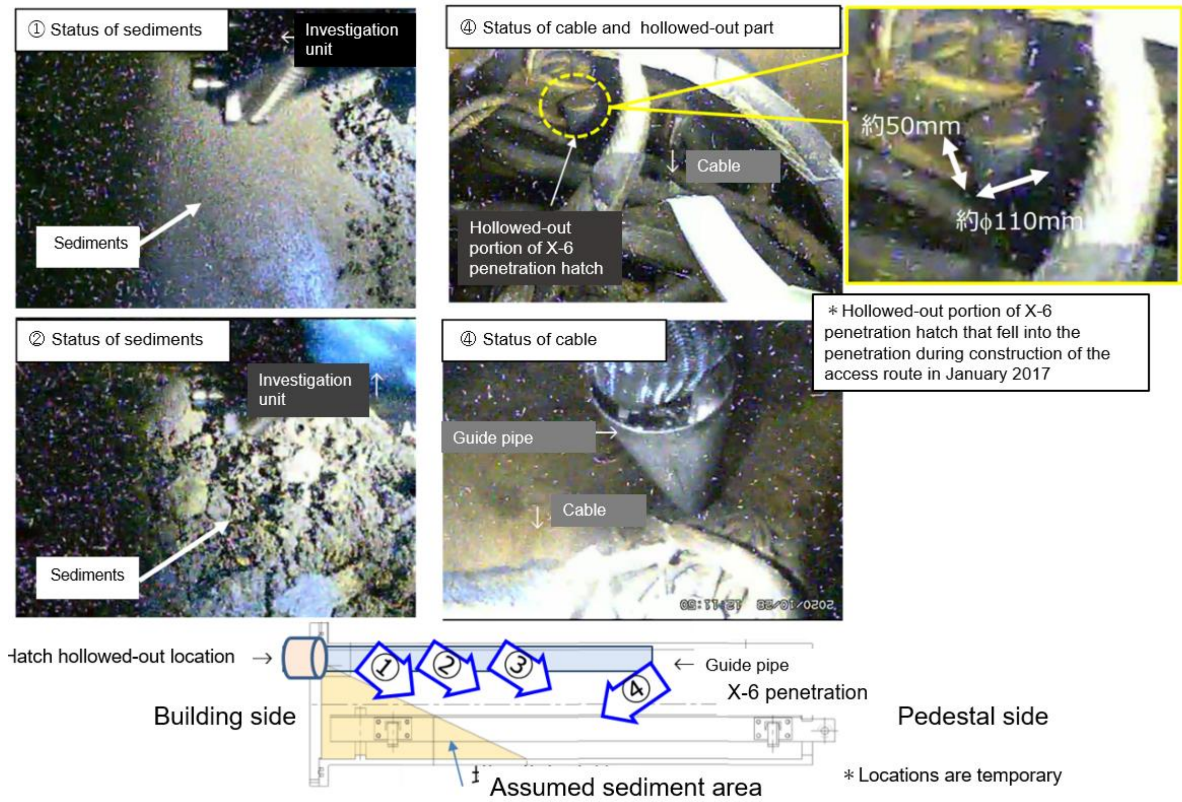


Figure 3-22　Contact survey of sediments in X-6 penetration result (2) [3-13]

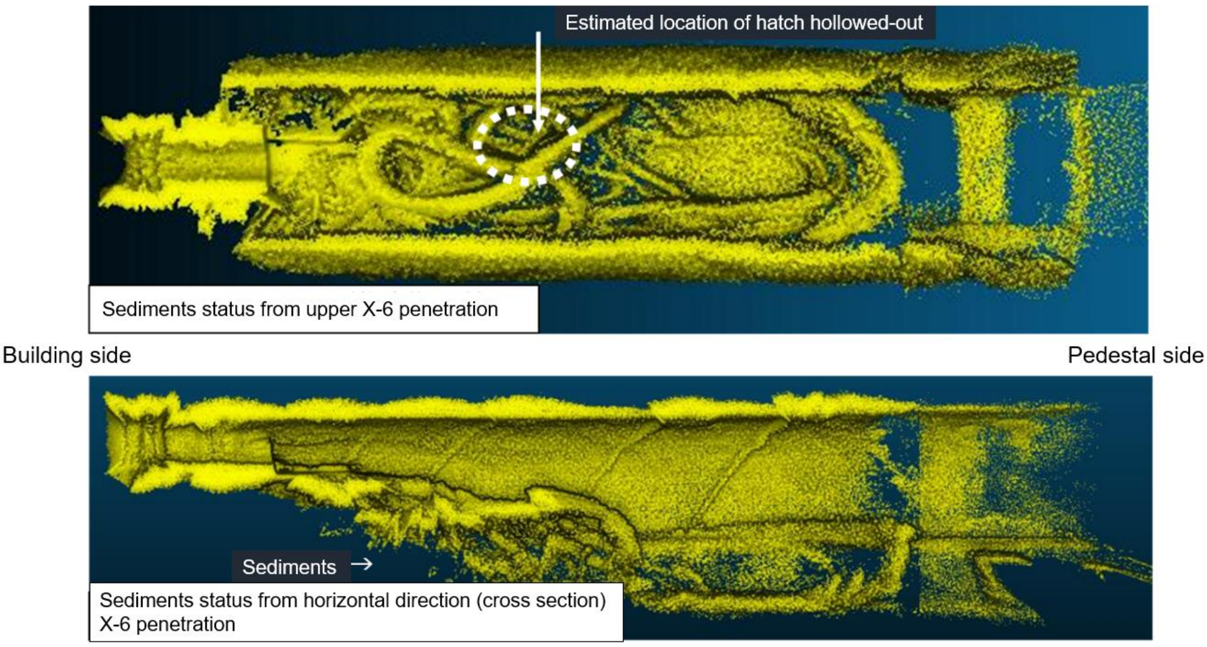


Figure 3-23 3D scan survey of sediments in X-6 penetration result [3-13]

(9) Investigation of Unit-2 using the muon device

Muon investigations were conducted in March to July 2016 using a muon tomography measurement device and the conditions inside the PCV were evaluated. A small device was used which was based on the same principles as the device used for Unit-1 investigations. Figure 3-24 shows the muon device used, and Figure 3-25 shows the measurement location.

Muon measurement device
(small type: about 1m x 1m x 1.3mH)

Figure 3-24 Muon measurement device [3-14]

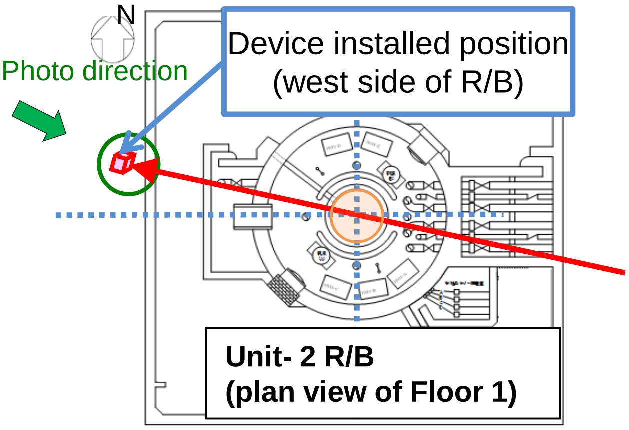


Figure 3-25 Measurement device location [3-14]

Figure 3-26 shows the analysis results of material quantities (density length) obtained from the muon measurements (left) and the enlarged portion of the RPV lower part (right). Besides the PCV and fuel in the spent fuel pool, dark tones were identifiable in the RPV lower part, too, indicating the presence of high density materials. Evaluation by muon technologies becomes difficult below about O.P. 15m due to reduced muon population detectable, but the dark tones at the RPV lower part can be believed to be significant, since those in the PCV at a similar elevation are identifiable.

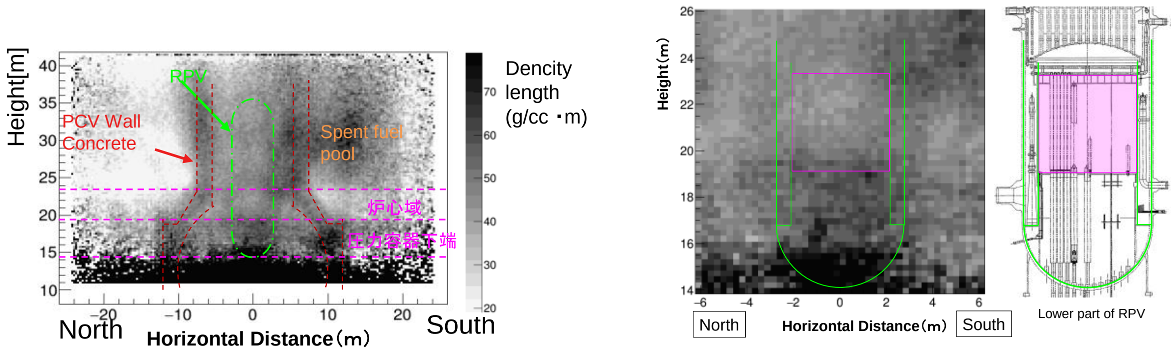


Figure 3-26 Distribution map of material quantities by muon measurements [3-14]
(Left, overall view; right, lower part of reactor pressure vessel)

Figure 3-27 compares the results of measurements and simulation. In the simulation, two cases were assumed: one was the case in which high density materials (2g/cc or 6g/cc) simulating nuclear fuel were present in each evaluation region; and the other was the case in which they were not present. From the comparison, it was estimated that nuclear fuel had been present in the

[3-14] Handout document: Fukushima Daiichi Nuclear Power Station Unit-2 in-core fuel debris location determined by muon measurement, 32nd Decommissioning and Contaminated Water Response Team Joint Meeting, July 28, 2016.

peripheral region of the core lower part and in the RPV bottom part, where the measured results had been closer to the results simulating the presence of high density materials there.

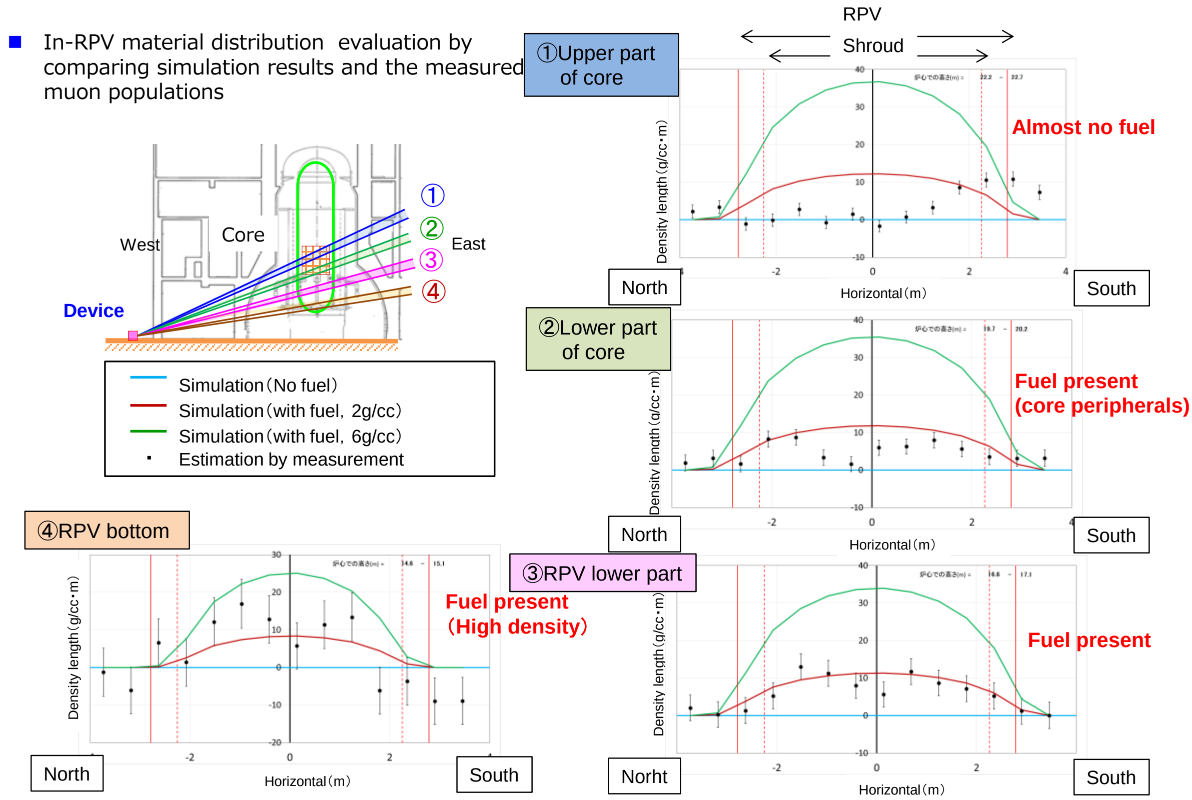
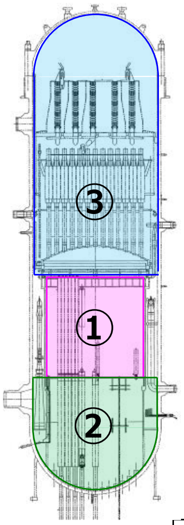


Figure 3-27 Distributions of material quantities in RPV [3-14]

The results of quantitative evaluation of material quantities in the RPV are compared in Figure 3-28 with the material quantities before the accident. Most of the fuel debris was estimated to be present at the RPV bottom.



＜Quantitative Evaluation Results＞

<table>
<tr><th></th><th colspan="2">Evaluation results [ton]</th><th>(Reference) quantities before the accident※ [ton]</th></tr>
<tr><td>① Core region (Inside shroud)</td><td>Approx. 20～50</td><td rowspan="4">Uncertainties in evaluation results: About several tens of tons</td><td>Approx. 160 (Fuel assemblies)
Approx. 15 (Control rods)</td></tr>
<tr><td>② Bottom of RPV</td><td>Approx. 160</td><td>Approx. 35 (Structures)
Effect of water is non-consideration</td></tr>
<tr><td>①＋②</td><td>Approx. 180～210</td><td>Approx. 210</td></tr>
<tr><td>(Reference)
③ Upper part of RPV</td><td>Approx. 70～100</td><td>Approx. 80 (Structures)</td></tr>
</table>

*Weights in design. They do not necessarily agree with the muon results because part of the structures was ignored for simplicity and the muon measurements were performed looking upward and in the tilt direction.

Figure 3-28 Results of quantitative evaluation of material quantities in RPV [3-14]

(10) Investigation results obtained inside the containment vessel of Unit-2 (Part 1)

The inside of the pedestal was observed on January 30, 2015 with a pan-tilt camera mounted on a guide pipe, as a preparatory survey for the investigation using the self-propelled investigation device. Figure 3-29 presents an integrated photo combining parts of images taken by the camera after image processing for clarification. When compared to the similar arrangement at Unit-5 shown in the figure (left), the grating of Unit-3 can be found to be missing. At and above this place, cable-shaped fallen objects, and other fallen objects looking like TIP guide-tubes have been identified.

Figure 3-30 is a photo of the upper part inside the pedestal. Damaged portions of the local power range monitoring system (LPRM) and control rod position indicator probe (PIP) cables can be identified. PIP cables and LPRM cables can be located at some places but cannot be located at other places. Figure 3-31 summarizes damage conditions in the pedestal, investigation results of cables in the upper part of inside the pedestal and other findings.

From February 7 to 9, 2017, attempts were made to remove sediments on the CRD replacement rail using the sediment removing equipment. Figure 3-32 is the photo taken by the camera mounted on the equipment. The sediments were mixtures of black paste-like materials, thin fragmented objects, and pebble-shaped objects. The sediments became more sticky and difficult to remove when the equipment approached the pedestal from the PCV penetration. On February 16, 2017, the self-propelled investigation device was sent in for investigation. The equipment failed to reach inside the pedestal, but could collect data on temperatures, dose rates and conditions of structures around it.

Figure 3-33 shows the dose rates at four points obtained in this investigation. Around the pedestal opening, the dose rates inside the pedestal were lower than outside.

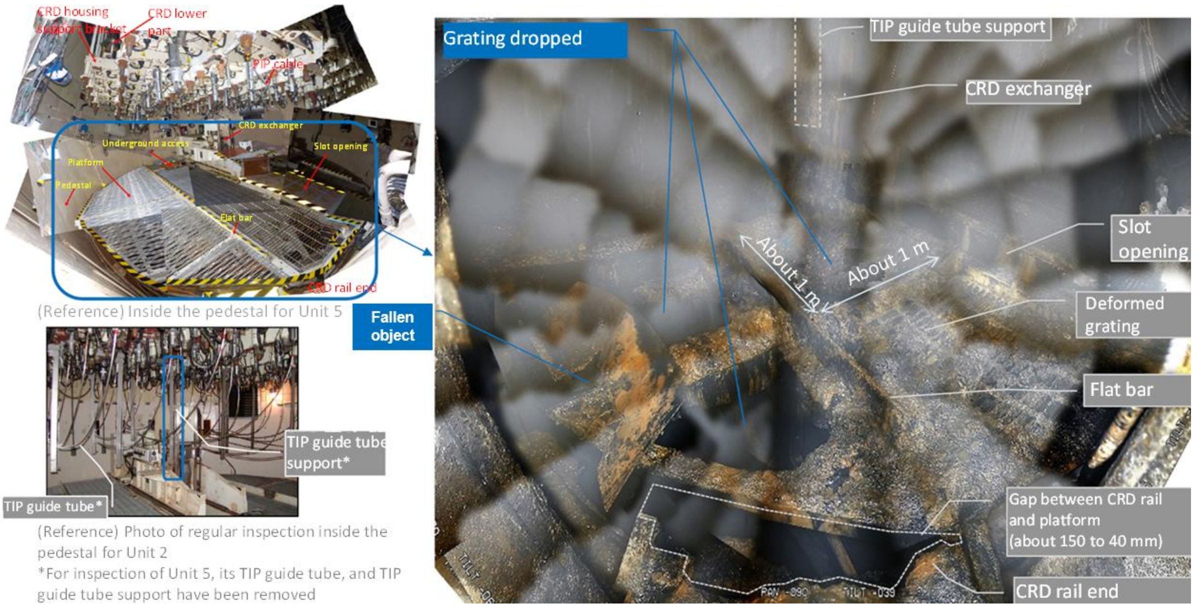


Figure 3-29 Status inside the pedestal [3-15]

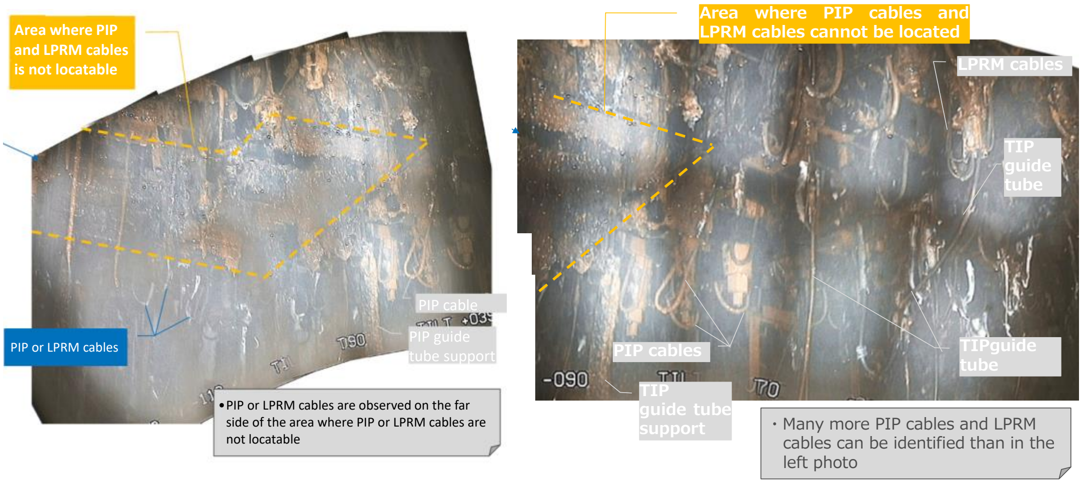


Figure 3-30 Status inside the upper pedestal [3-15]

[3-15] Handout document: Investigation of the inside of the containment vessel of Unit-2 - Additional report based on image analysis - , 40th Decommissioning and Contaminated Water Response Team Joint Meeting, March 30, 2017.

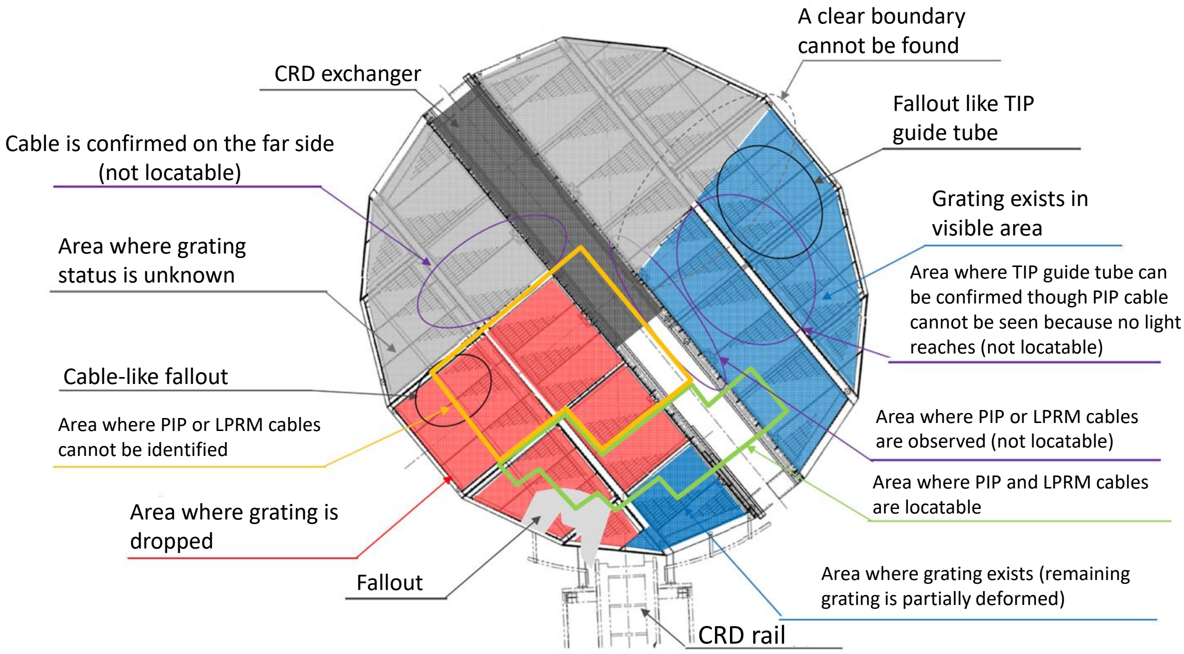


Figure 3-31 Status inside the pedestal (summary) [3-15]

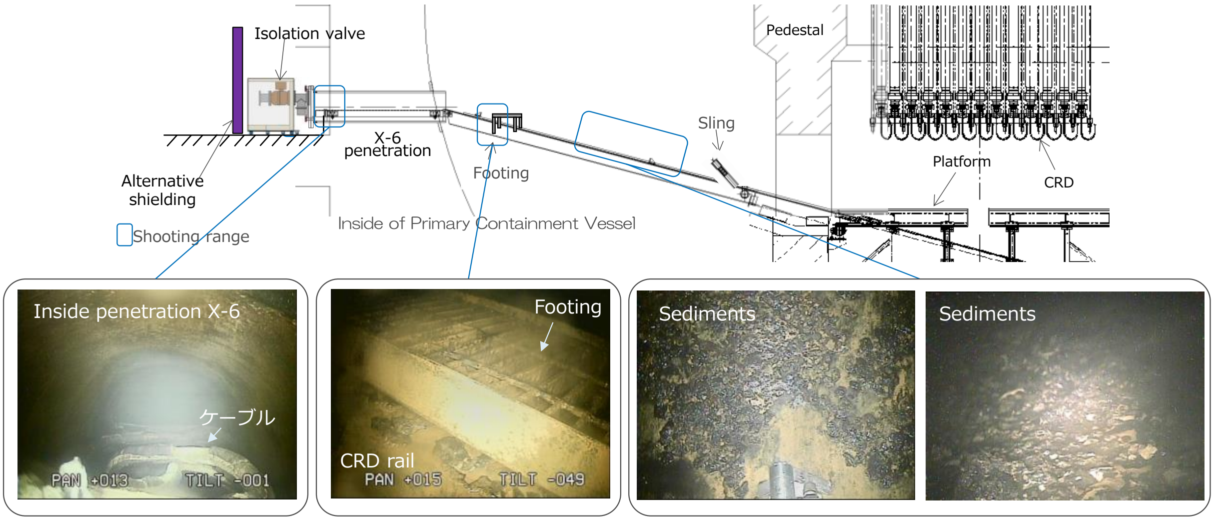


Figure 3-32 Images on the CRD rails starting from the X-6 penetration [3-16]

[3-16] Handout document: Investigation of the inside of the containment vessel of Unit-2, 39th Decommissioning and Contaminated Water Response Team Joint Meeting, February 23, 2017.

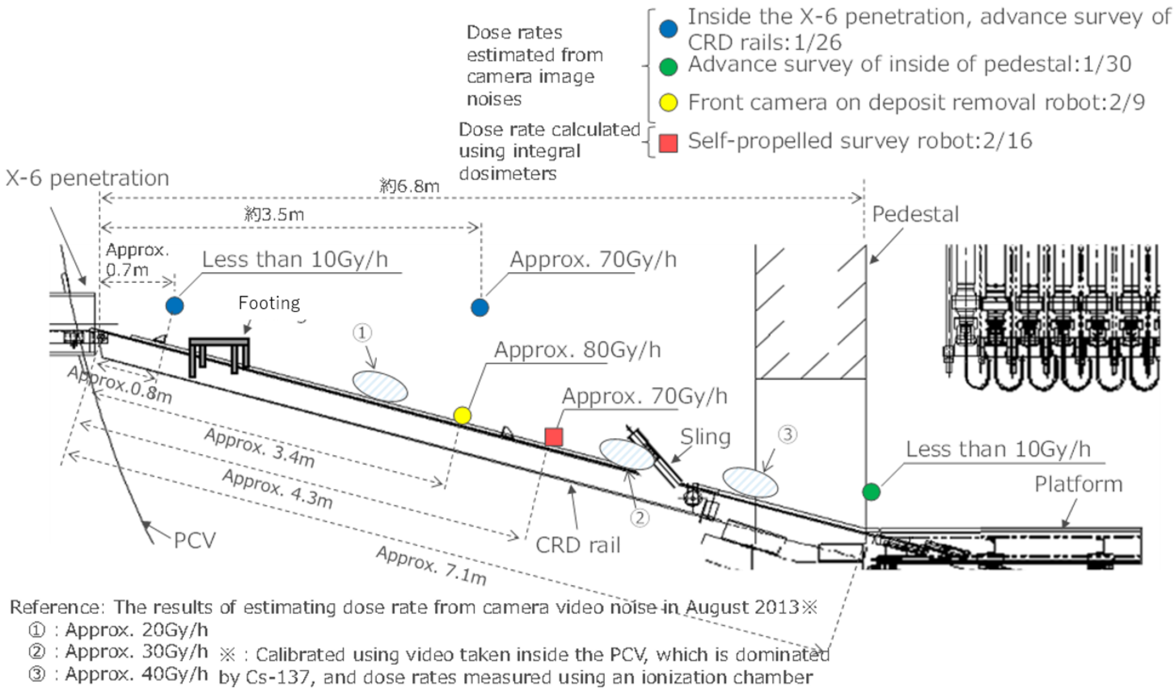
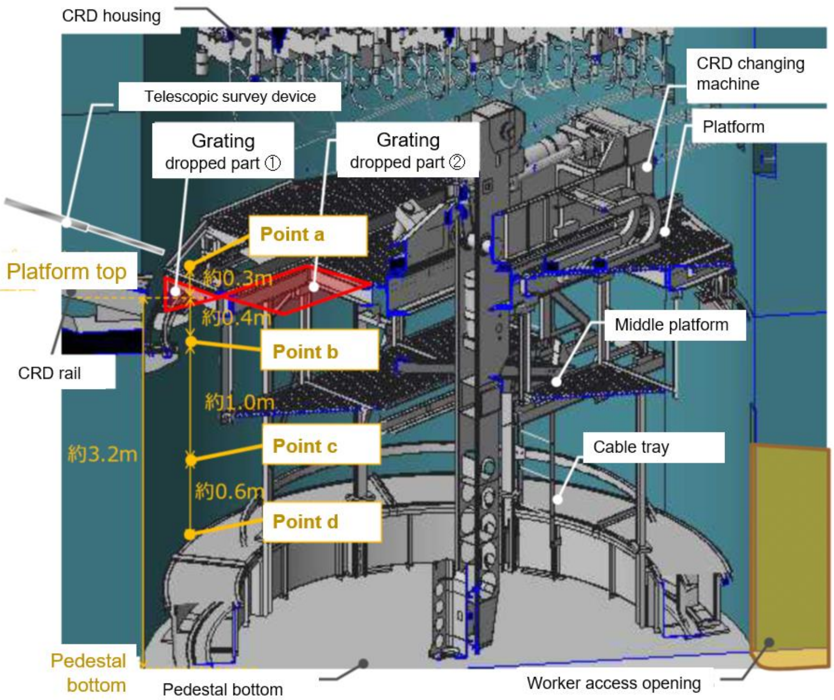


Figure 3-33 Dose rates measurement [3-17]

(11) Investigation results obtained inside the containment vessel of Unit-2 (Part 2) [UPDATE]

On January 19, 2018, an investigation inside the PCV was conducted by accessing the inside of the pedestal from the containment vessel penetration X-6 and suspending a camera, a dosimeter, and a thermometer from the dropped part of the grating to the bottom of the pedestal. The values measured with the dosimeters and thermometers suspended from the grating were almost constant regardless of the height of the measurement. The dose rate tended to be lower inside the pedestal than outside it (Figures 3-34 and 3-35). Figure 3-36 is a schematic drawing of the survey equipment used.

[3-17] Handout document: Investigation of the inside of the containment vessel of Unit-2 – dose rate measurement result -, 44th Decommissioning and Contaminated Water Response Team Joint Meeting, February 23, 2017.



Measuring point	Dose rate [*1,2] [Gy/h]	Temp.[*2] [℃]
a	7	21.0
b	8	21.0
c	8	21.0
d	8	21.0

Ref: Outside the pedestal[*3]

Dose rate : Max. 42 [Gy/h]

Temp. : Max 21.1 [℃]

＊[1]: Calibrated by Cs-137 source

＊[2]: Errors: dosimeter ±7%; thermometer ±0.5℃

＊[3]: Reference values as measurements were taken with the instrument housed in the survey equipment.

Figure 3-34 PCV inside investigation area [3-18]

※1： Reference values as measurements were taken with the instrument housed in the survey equipment.

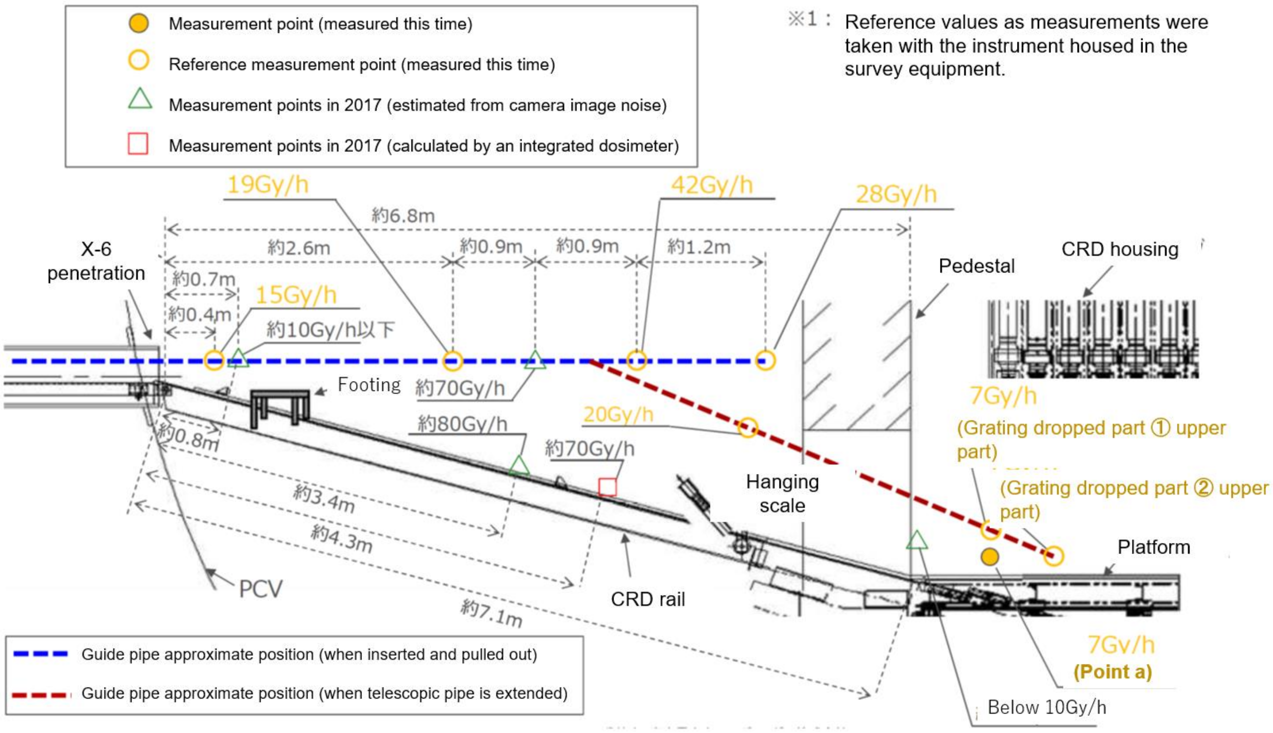


Figure 3-35 Dose rate before insertion into the pedestal [3-18]

[3-18] Handout document: Investigation result of the inside of the reactor containment vessel of Unit-2, 50th Decommissioning and Contaminated Water Response Team Joint Meeting, February 1, 2018.

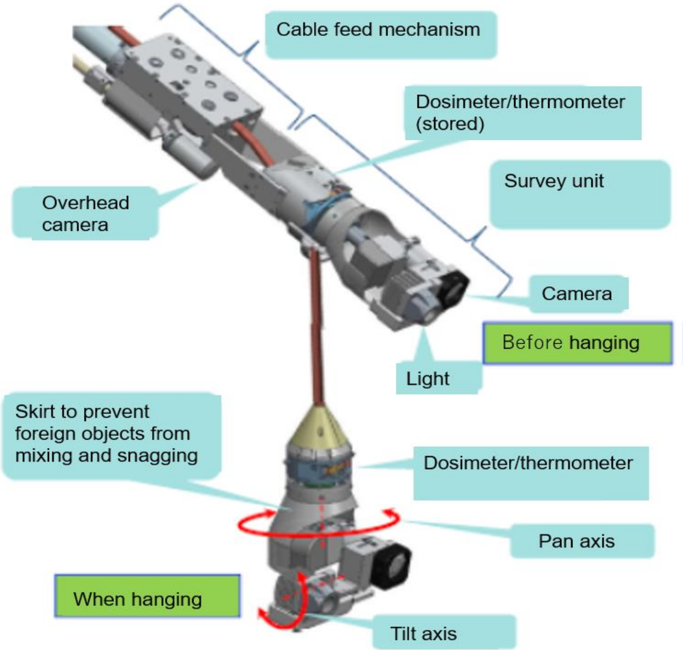


Figure 3-36　Schematic drawing of the survey equipment used [3-18]

① Investigation results on the platform

Similar to the January 2017 investigation, no major deformation or damage was observed for the CRD exchanger, platform frame, or other structures. In addition, no damage to the interior wall surface of the pedestal was observed. Damage or falling down of the TIP guide tube, PIP cable, and grating on the front side of the camera was confirmed, and there was also a relatively large amount of adhered material on the platform frame, which may have been a path by which the fuel debris dropped down. The processed images for the investigation results obtained on the platform are shown in Figure 3-37.

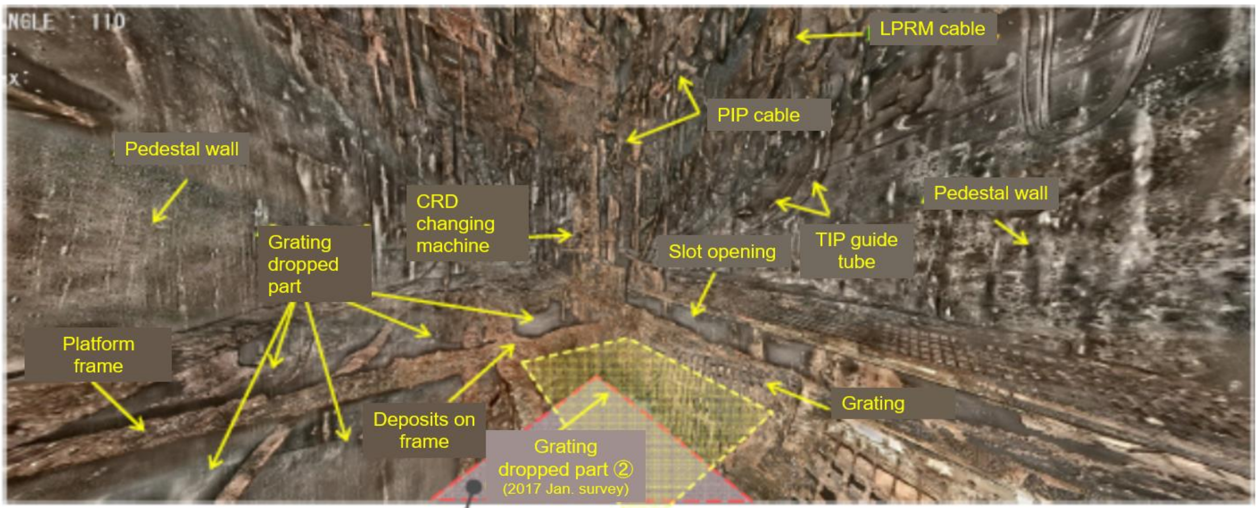


Figure 3-37 Investigation results on the platform [3-19]

[3-19] Handout document: Investigation result of the inside of the reactor containment vessel of Unit-2, 53rd Decommissioning and Contaminated Water Response Team Joint Meeting, April 26, 2018.

② Investigation results around the CRD housings

As far as could be confirmed, there was no dropping of the CRD housing support. However, there were areas where the TIP guide tube, PIP cable, and LPRM cable could not be confirmed due to deposits, and the grating directly below them was confirmed to have dropped down.

A processed image of the acquired data is shown in Figure 3-38.

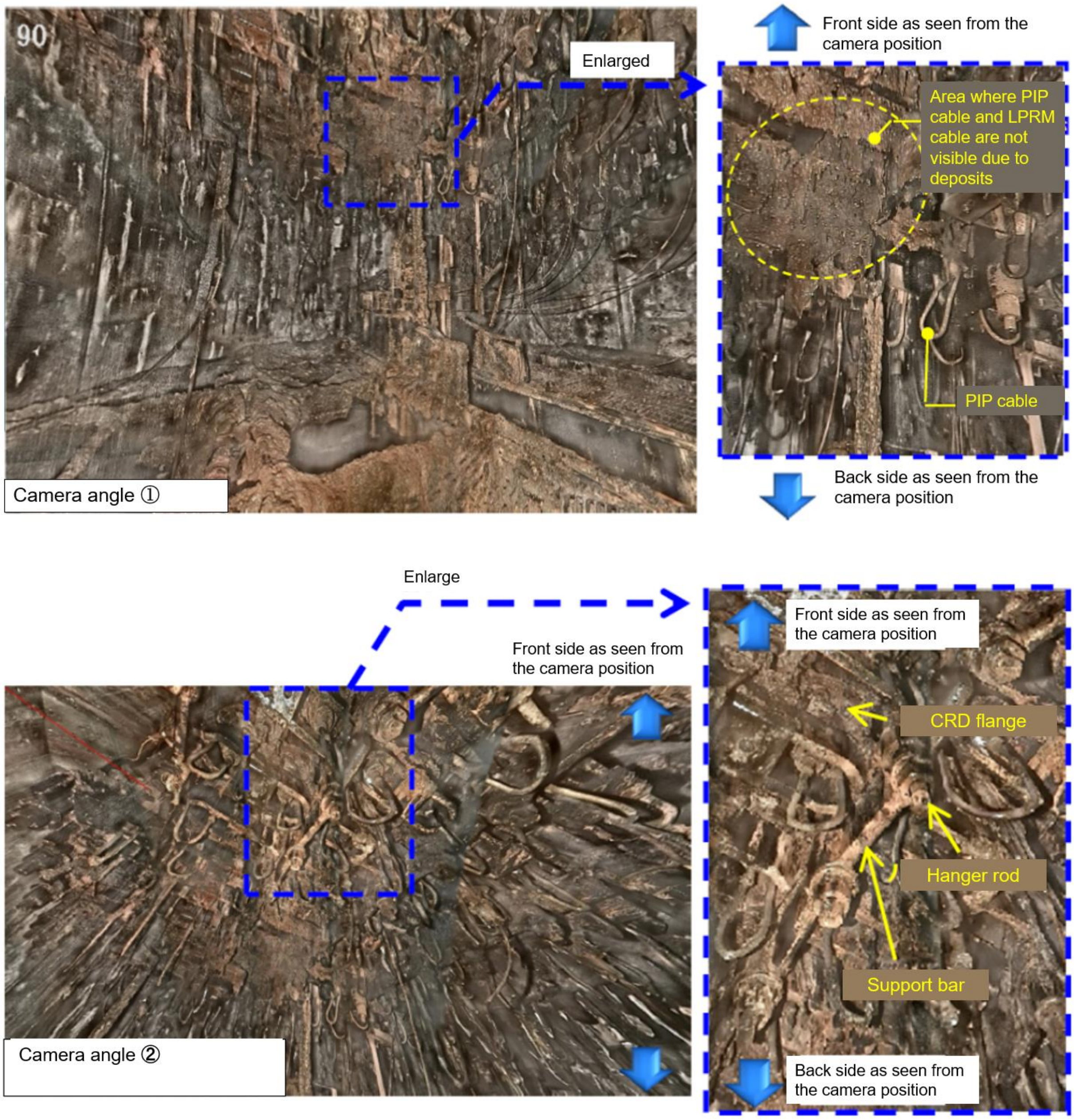


Figure 3-38 Image of the investigation around the CRD housings [3-19]

③Investigation results of the middle platform

As viewed from the camera suspension position, the grating was found to have dropped down at the location of the intermediate work platform, but no major damage to the platform frame was observed (Figure 3-39).

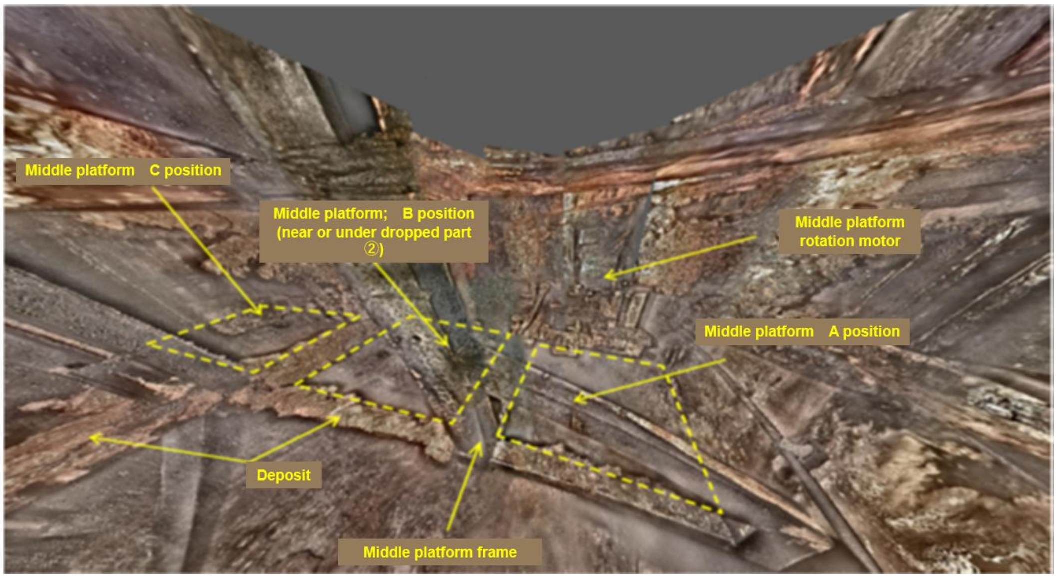


Figure 3-39 Middle platform investigation image [3-19]

④ Investigation results of the pedestal bottom

From the investigation of the pedestal bottom using a camera, it was confirmed that there was no major deformation or damage to the structures such as the CRD changing machine rotation frame, middle platform frame, support columns, and cable tray (Figure 3-40). In addition, pebbly or rocky sediments were found to be deposited all over the pedestal bottom. Since no deformation of the cable tray (4 mm thick stainless steel) was observed, it is possible that the temperature of the sediments when they began to accumulate on the tray was not high enough to cause its thermal deformation. The height of the sediments was considered to be over 70cm near the cable tray on the left side from the camera, and 40 to 50cm near the CRD changing machine elevator trolley.

In addition, upper tie plate and other fallen objects were observed in the sediments (Figure 3-41). Since structures inside the pressure vessel had fallen down, the bottom of the pressure vessel was damaged, and it was considered that a hole had opened that was large enough for the upper tie plate to fall through.

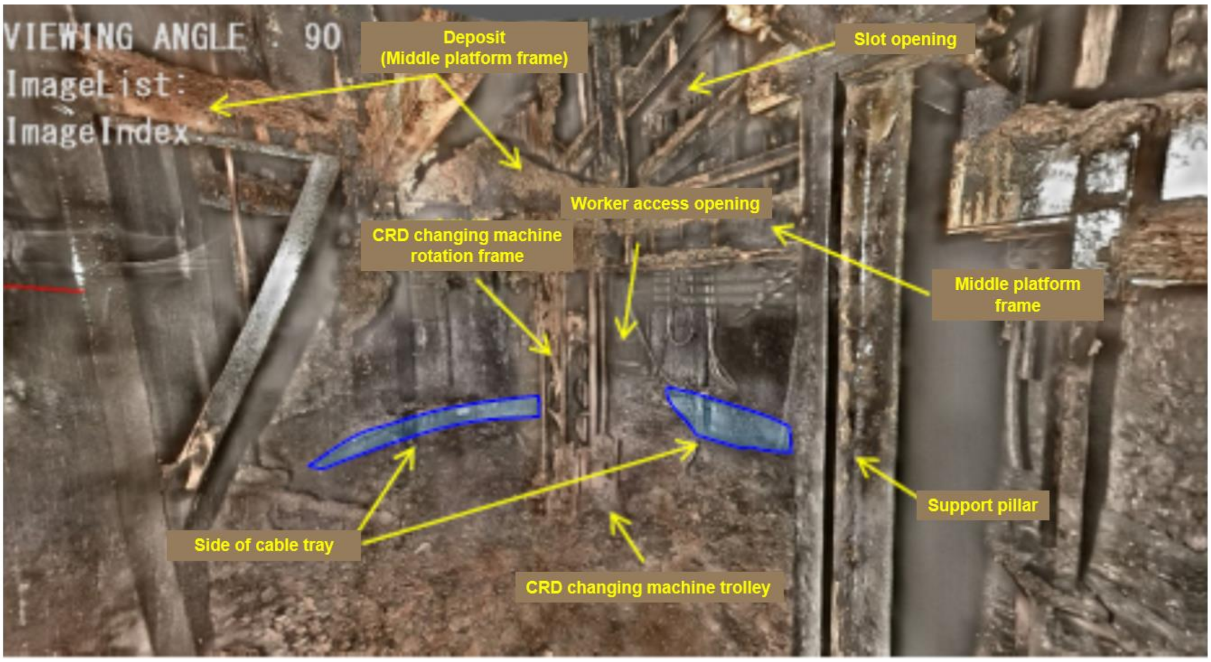


Figure 3-40 Pedestal bottom sediments [3-19]

Figure 3-41 Sediments and dropped upper tie plate [3-19]

A panoramic composite image was prepared from the acquired images to provide an overall view of the inside of the pedestal of Unit-2. The composite image on the grating is shown in Figure 3-42, and the composite image of the pedestal bottom is shown in Figure 3-43.

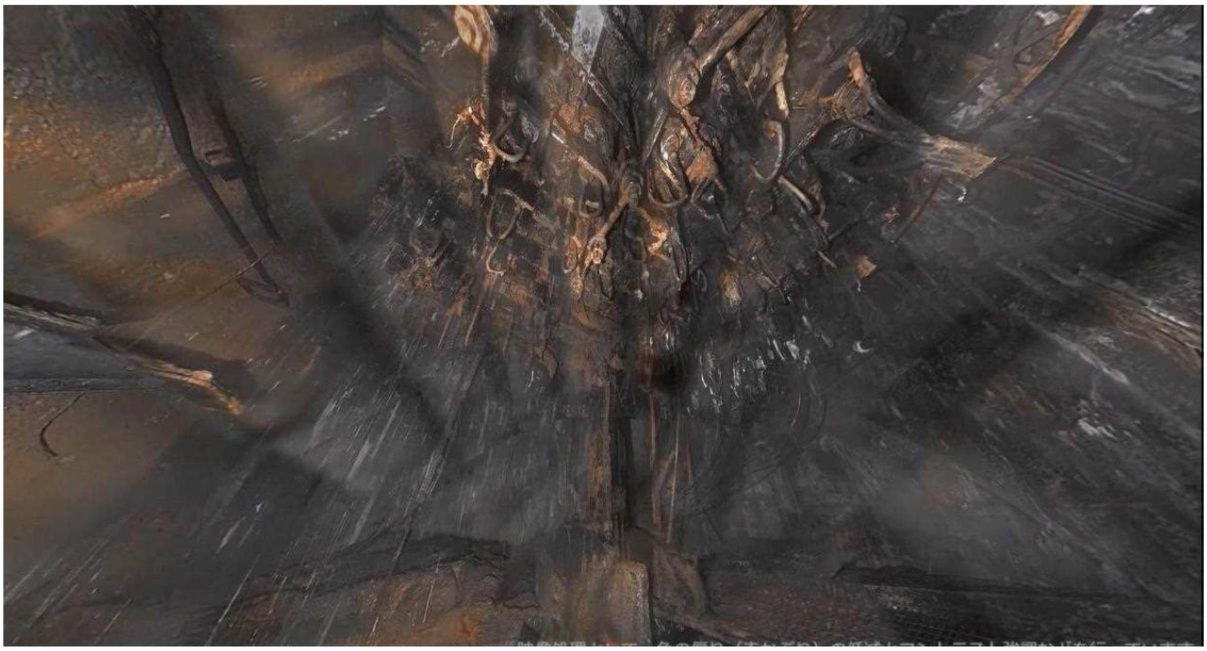


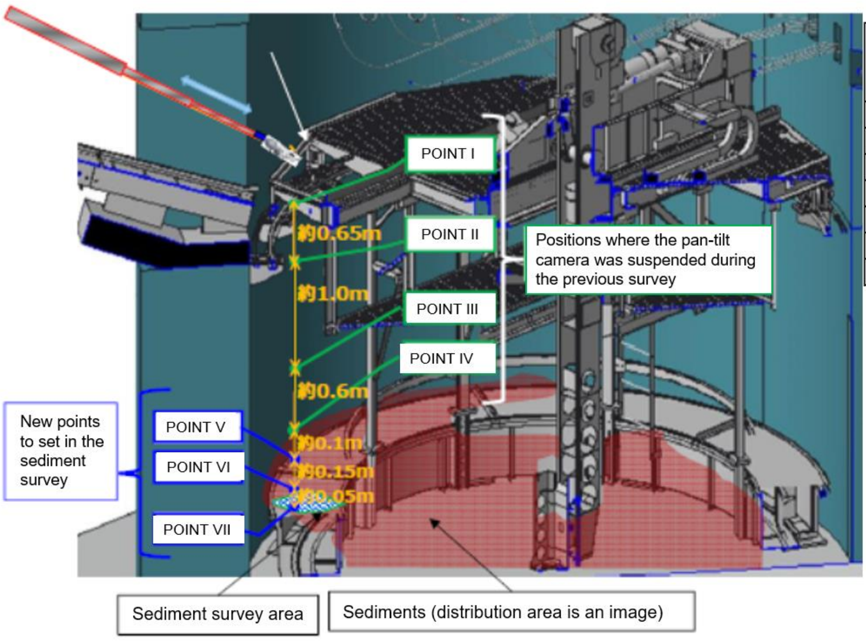
Figure 3-42 Composite image above grating [3-20]

Figure 3-43　Composite image of pedestal bottom [3-20]

[3-20] Handout document: Investigation of the inside of the reactor containment vessel of Unit-2, (January 2018) Image processing of acquired images, 63rd Decommissioning and Contaminated Water Response Team Joint Meeting, February 1, 2018.

(12) Investigation results obtained inside the containment vessel of Unit-2 (Part 3) [UPDATE]

In February 2019, survey equipment was lowered to the same location as the January 2018 survey to make contact with the sediments at the pedestal bottom. The temperature and dose rate at that time are shown in Figure 3-44, and the area where the contact survey was conducted is shown in Figure 3-45. The temperature was almost constant regardless of the measurement height, while the dose rate tended to increase as the pedestal bottom was approached. The sediments were classified into three types: pebbly, rocky, and apparent parts of structural materials, and the contact investigation results for each are shown in Figures 3-46 to 3-48. This investigation confirmed that the pebbly and structure-related sediments could be picked up and moved, and that there might be hard and rocky sediments that could not be picked up. By bringing the camera closer to the sediments, it was possible to obtain images that contributed to the estimation of the contours and size of the sediments.



Measuring point	Dose rate *1,2 [Gy/h]	Temp.*2 [℃]
I	6.4	23.2
II	6.8	23.1
III	6.5	23.1
IV	7.0	22.9
V	7.2	22.8
VI	7.5	22.9
VII	7.6	22.9

(Ref: Outside the pedestal*3])
Dose rate : Max. 43 [Gy/h]
Temp. : Max 23.7 [℃]

＊1: Calibrated by Cs-137 source
＊2: Errors: dosimeter ±7%; thermometer ±0.5℃
＊3: Reference values as measurements were taken with the instrument housed in the survey equipment.

Figure 3-44 Dose rate and temperature during survey of sediments [3-21]

[3-21] Handout document: Investigation result of the inside of the reactor containment vessel of Unit-2, 63rd Decommissioning and Contaminated Water Response Team Joint Meeting, February 28, 2019.

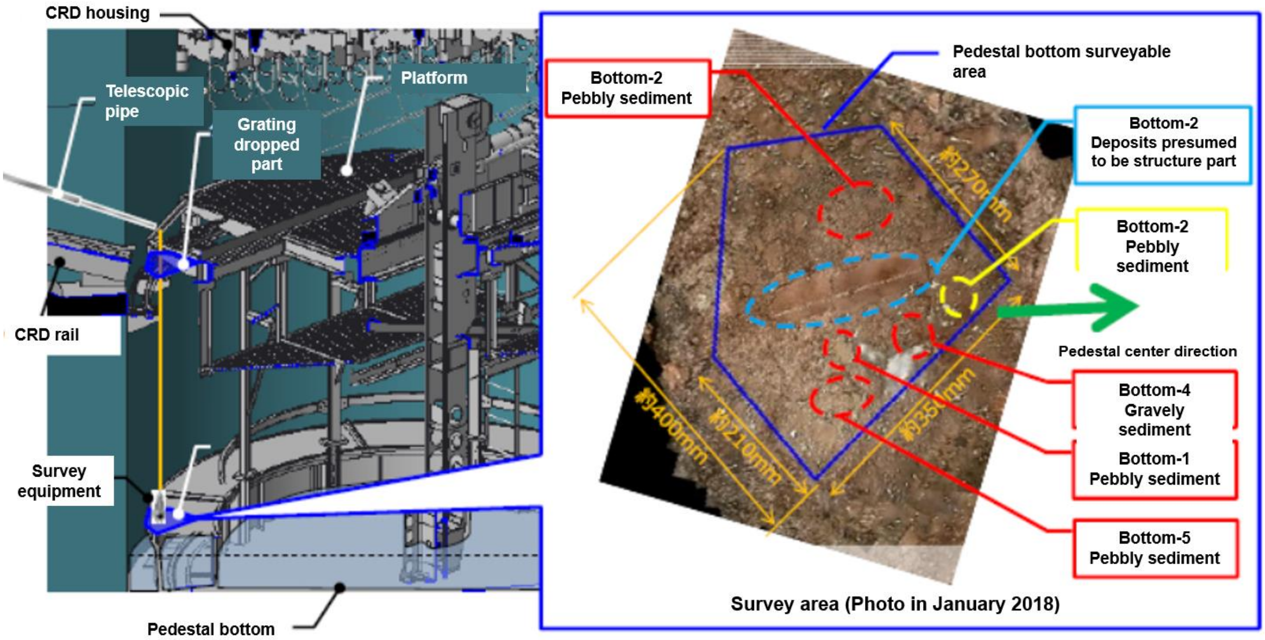


Figure 3-45 Pedestal bottom sediment contact investigation area [3-21]

■ Pebbly sediments were observed to move.

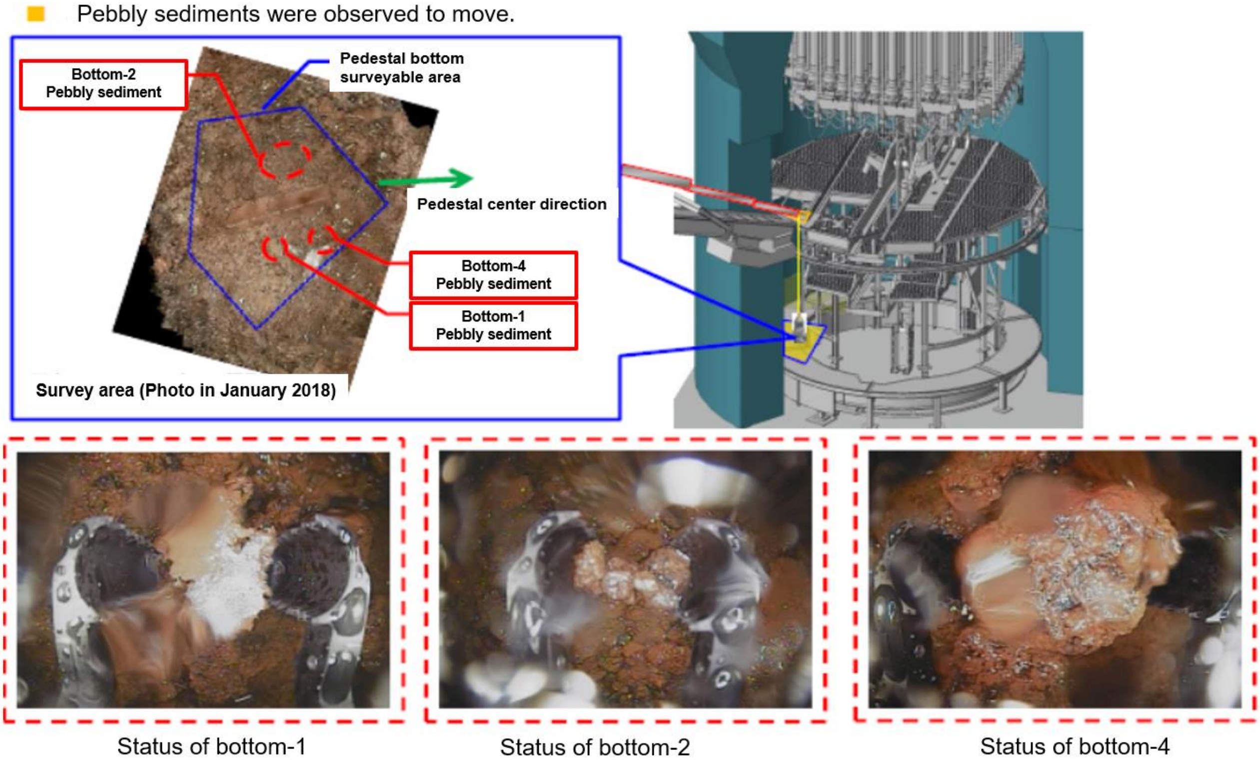
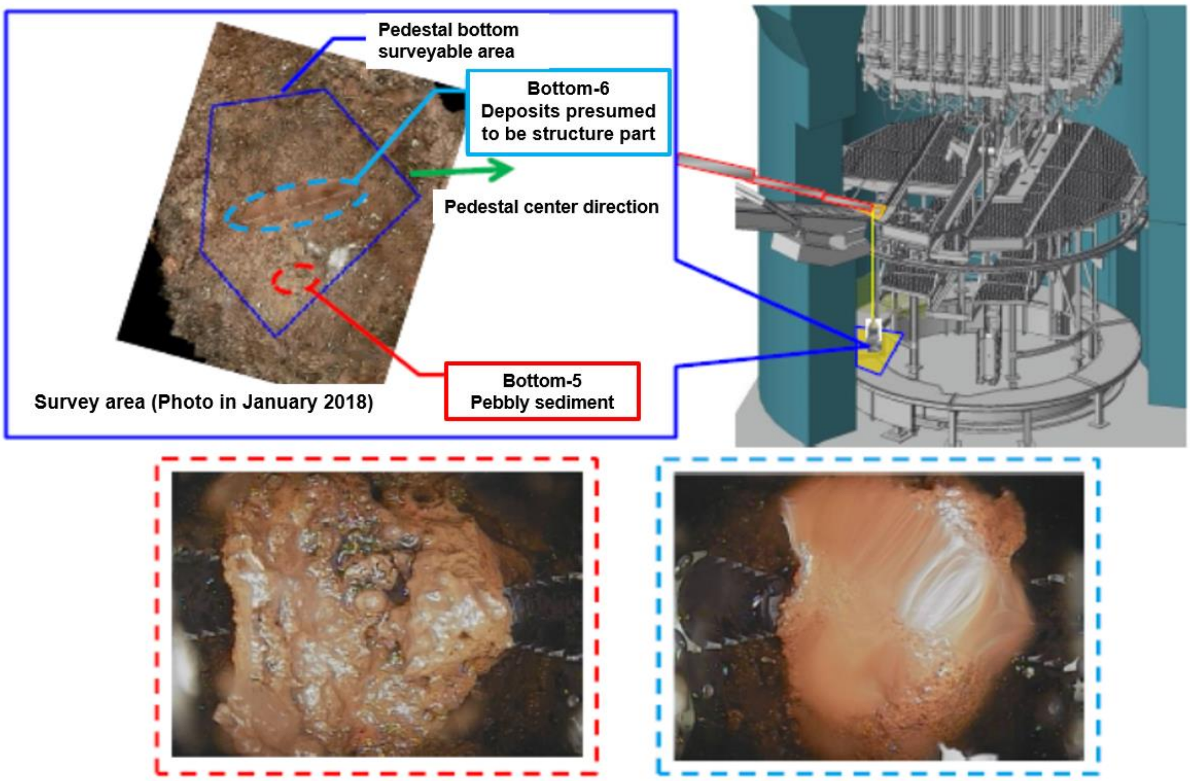


Figure 3-46 Pedestal bottom sediment contact investigation result (1) [3-21]

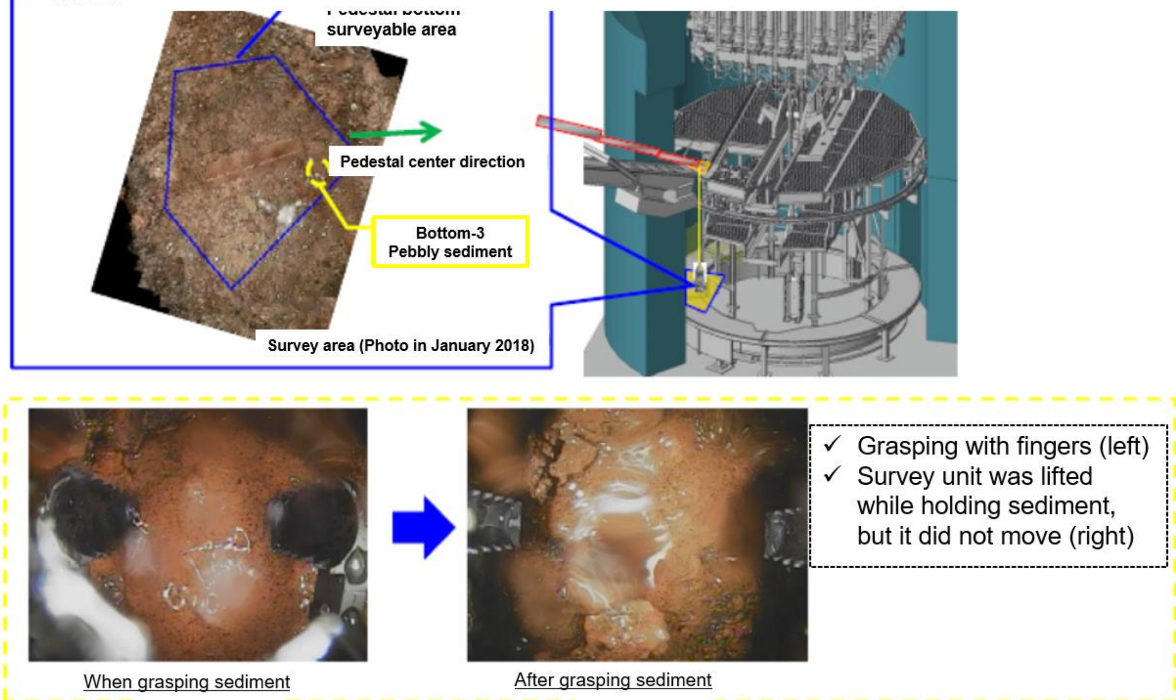
■ Pebbly sediments and structure-shaped sediments were observed to move.



Status of bottom-5 Status of bottom-6

Figure 3-47 Pedestal bottom sediment contact investigation result (2) [3-21]

■ The pebbly sediments were observed to be movable. No contact marks could be observed from video.



Survey at bottom-3 Survey date: February 13, 2019

Figure 3-48 Pedestal bottom sediment contact investigation result (3) [3-21]

Contact surveys were also conducted for sediments (pebbly, structure-shaped, and gravel-like) on the platform in a similar manner to the pedestal bottom surveys (Figures 3-49 to 3-51).

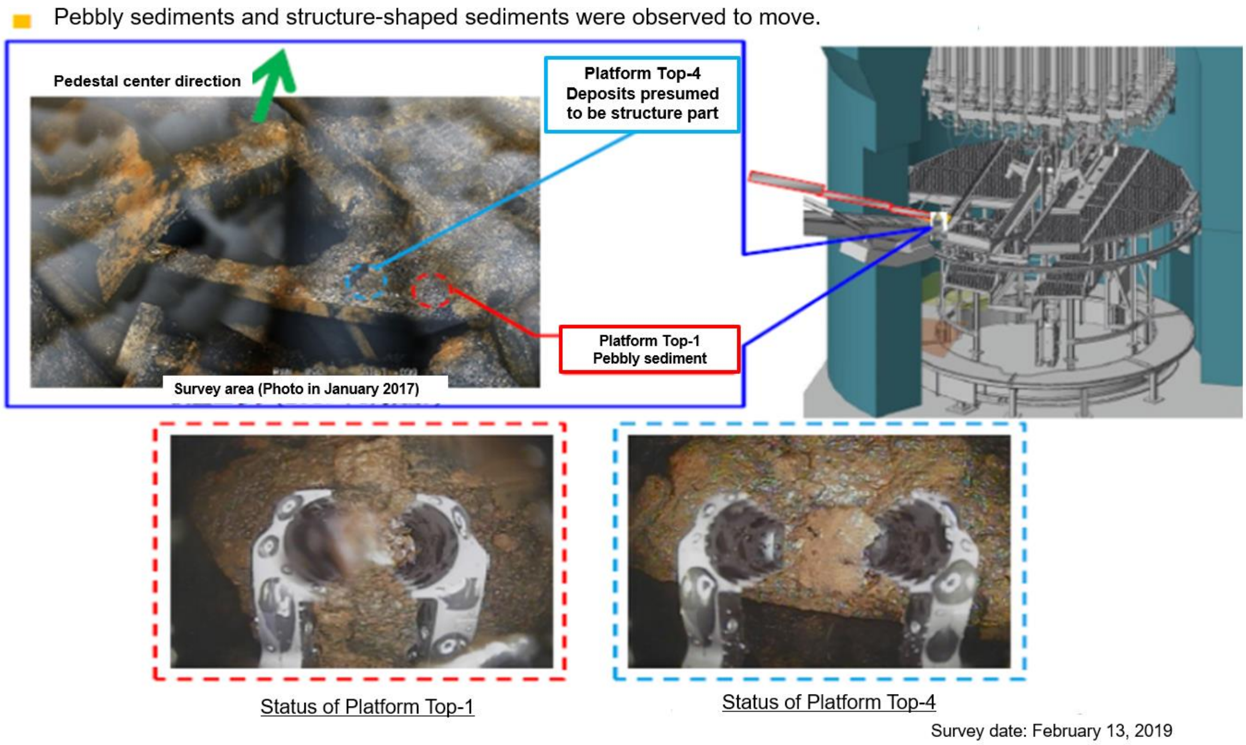


Figure 3-49 Platform sediment contact investigation result (1) [3-21]

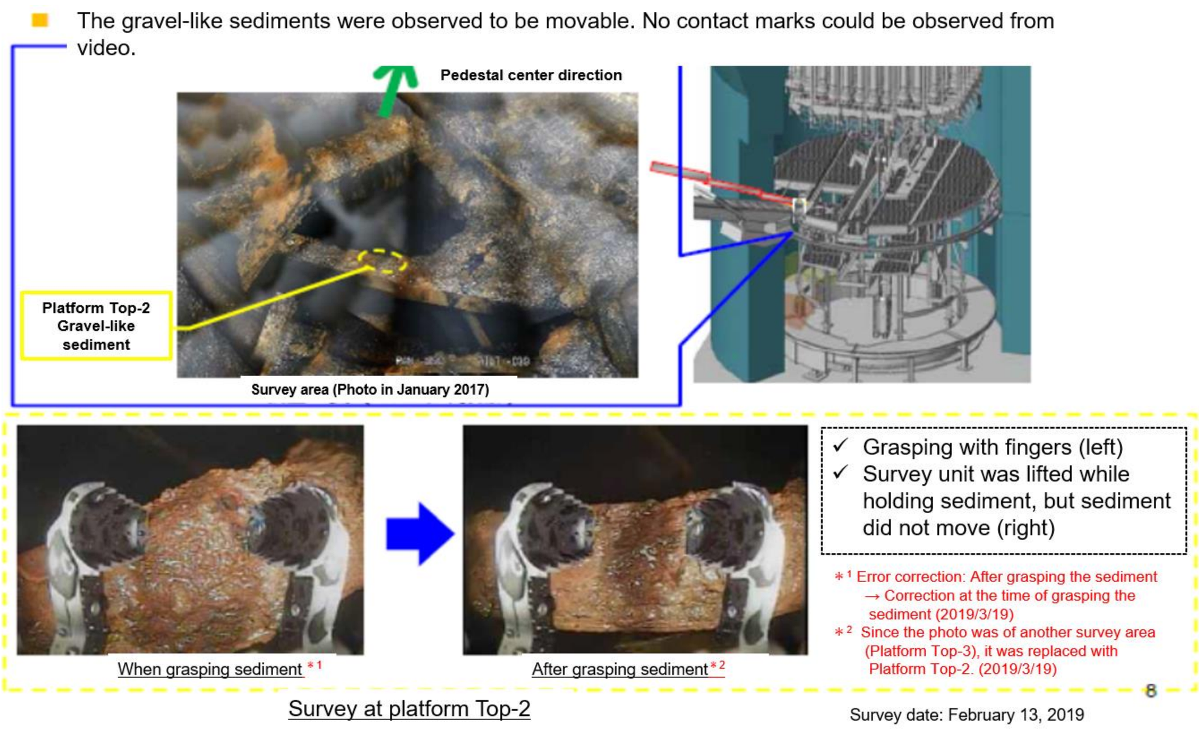


Figure 3-50 Platform sediment contact investigation result (2) [3-21]

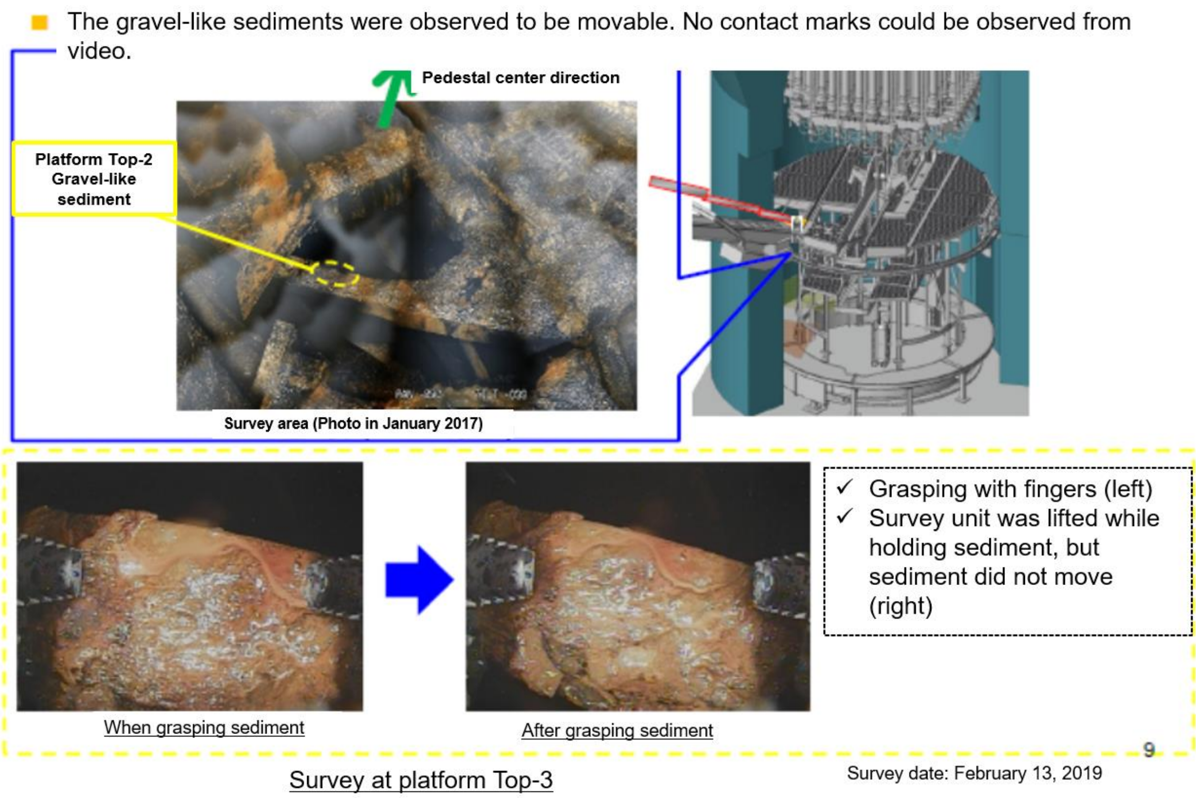
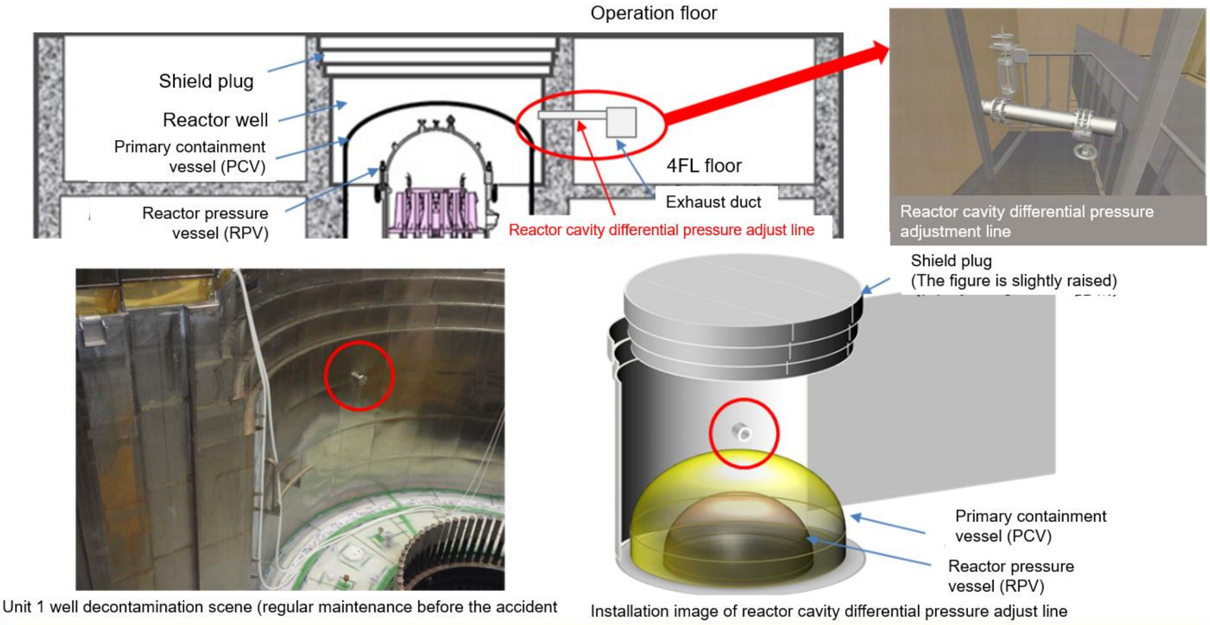


Figure 3-51 Platform sediment contact investigation result (3) [3-21]

(13) Inspection results of the Unit-2 reactor cavity differential pressure adjustment line and inside reactor well [UPDATE]

On-site inspection was conducted from January to March 2021 for the purpose of planning an examination using the reactor cavity differential pressure adjustment line on the west side to observe the inside of the reactor well below the Unit-2 shield plug. Figure 3-52 shows a schematic diagram of the reactor cavity differential pressure adjustment line.

From the on-site inspection it was found that the valve installed in the west reactor cavity differential pressure adjustment line was open, and the section on the straight line to the exhaust duct was deteriorated (Figure 3-53). Yellow deposits were observed on the bottom and sides of the duct inside (Figure 3-54), and sediments were observed inside the piping (Figure 3-55). The dose rate measurements showed one high-dose rate area was located at the bottom of the duct (Figure 3-56), and another high-dose rate area was near the floor at the lower part of the duct (4m below) (Figure 3-57).



＊Reactor cavity differential pressure adjust line: This line adjusts the differential pressure between the reactor cavity (reactor well) and the operation floor during operation and is connected to the exhaust duct of the reactor building ventilation and air conditioning system.

Figure 3-52 Schematic of reactor cavity differential pressure adjustment line [3-22]

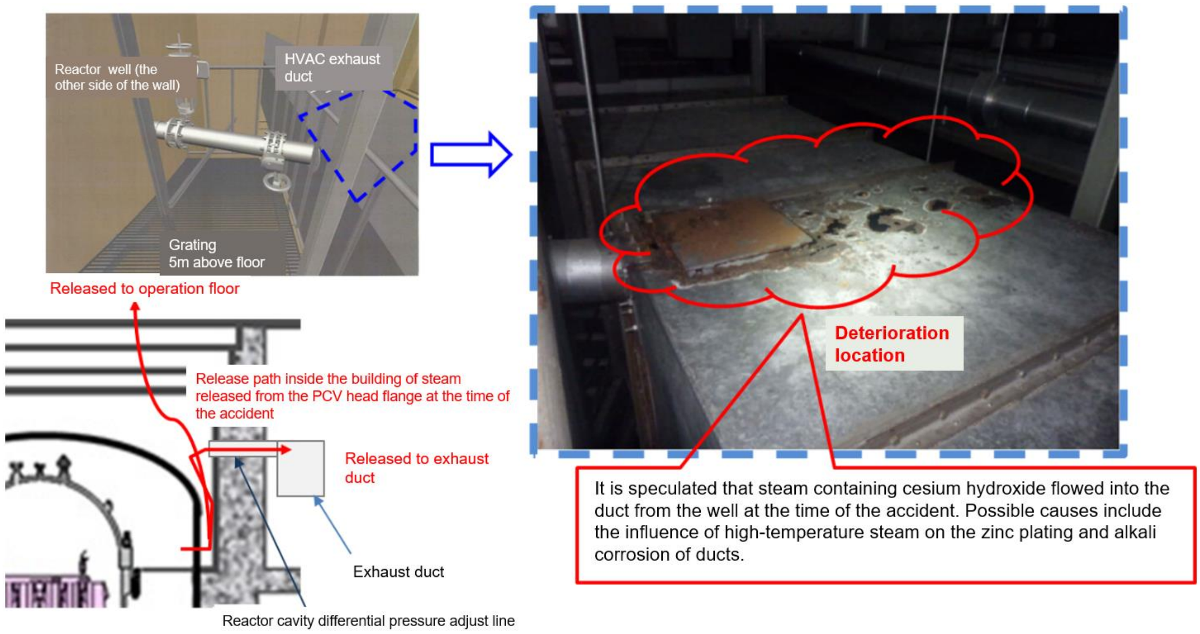


Figure 3-53 Deteriorated areas of exhaust duct [3-22]

[3-22] Handout document: Investigation of high contamination of shield plugs in Unit 2 (Preliminary report on the investigation results of dosimetry at work sites, etc.), 88th Decommissioning and Contaminated Water Response Team Joint Meeting, March 25, 2021.

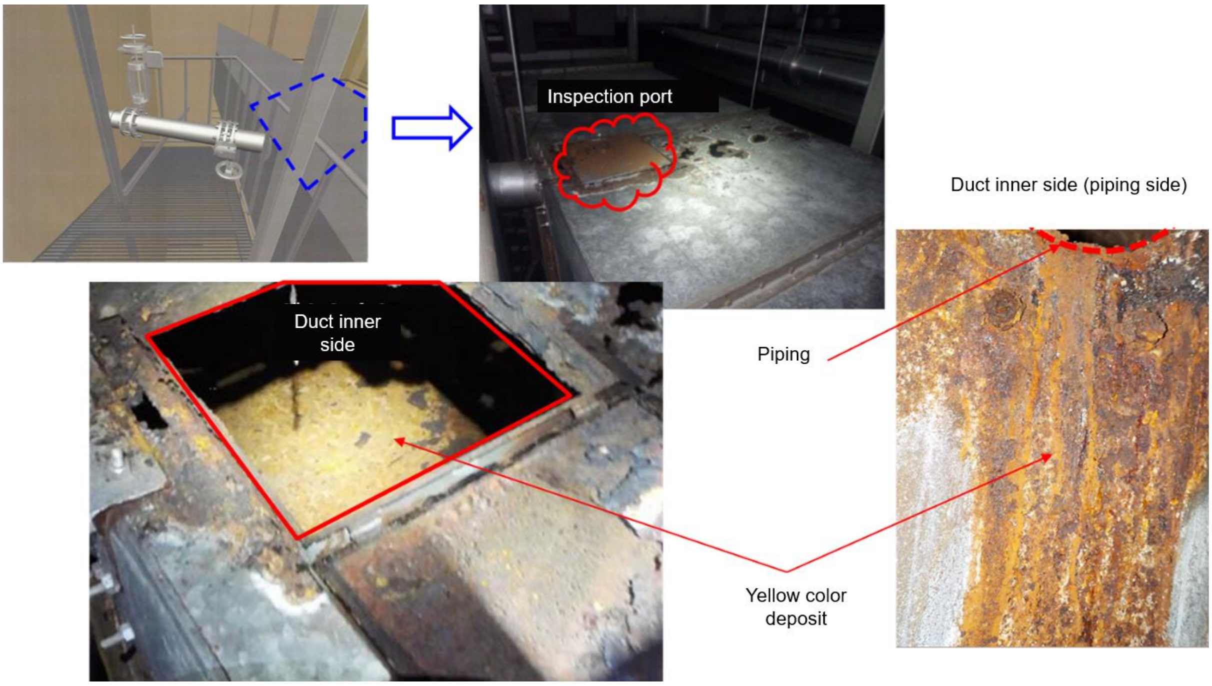


Figure 3-54 Deteriorated areas of exhaust duct [3-23]

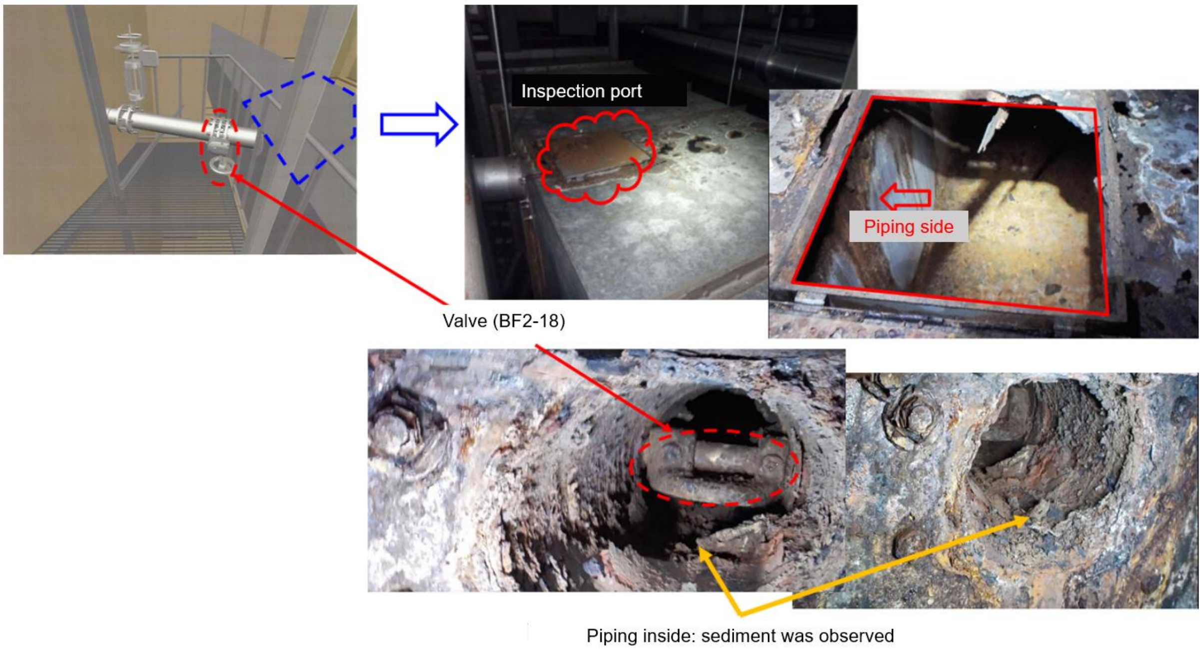
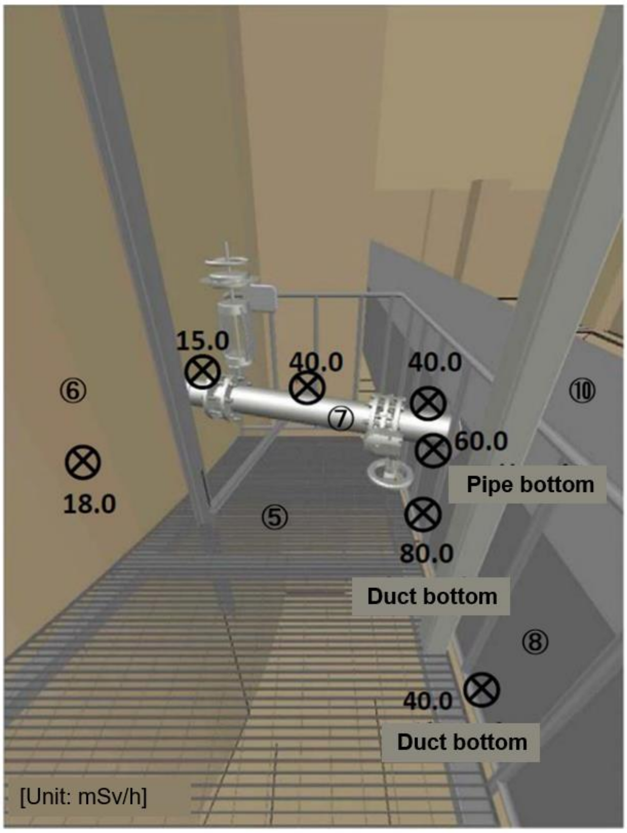


Figure 3-55 Sediments in piping [3-23]

[3-23] Handout document: Status of response to the high contamination of shield plugs in Unit-2 and future plans, Specific Nuclear Facility Supervising and Evaluation Committee Meeting, April 19, 2021.

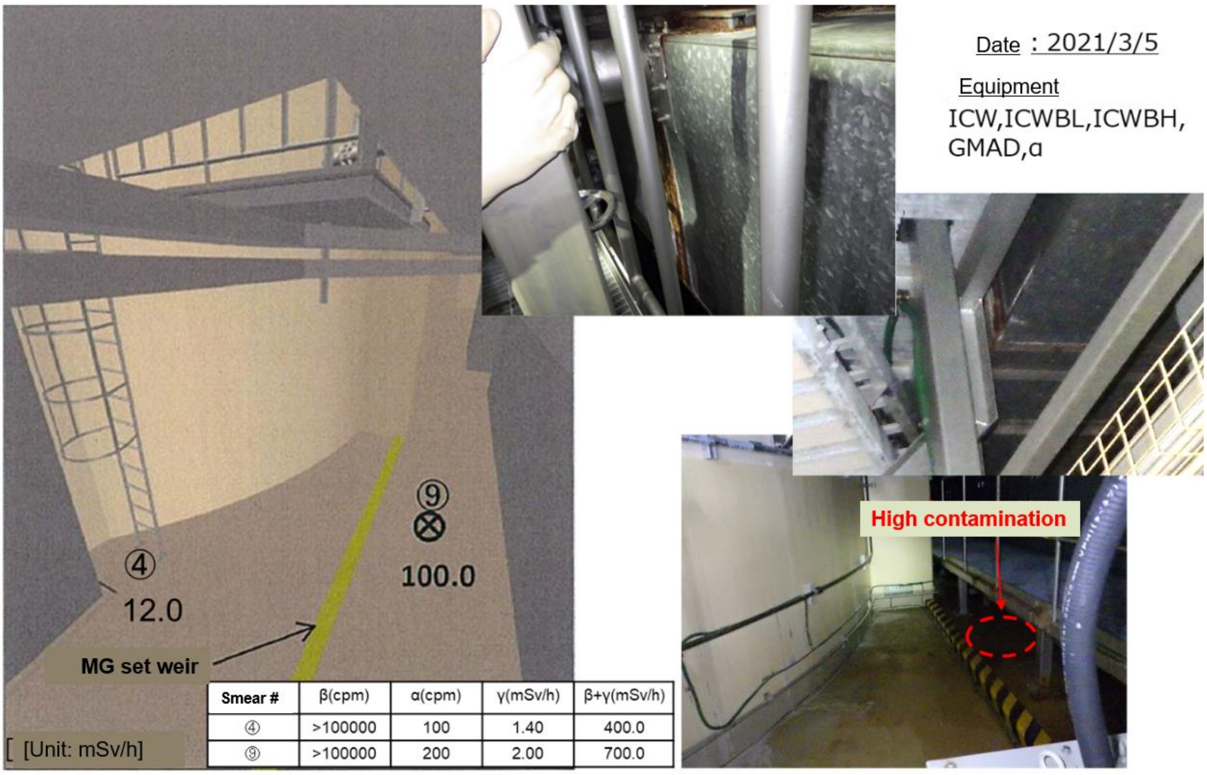


Date : 2021/3/5

Equipment
ICW,ICWBL,ICWBH,GMAD,α

Smear #	β(cpm)	α(cpm)	γ(mSv/h)	β+γ(mSv/h)
⑤	>100000	0	0.15	10.0
⑥	>100000	30	0.14	5.0
⑦	>100000	50	0.16	12.0
⑧	>100000	0	0.15	8.0
⑩	>100000	0	0.14	7.0

Figure 3-56 Results of dose rate measurement around the reactor cavity differential pressure adjustment line [3-23]



Date : 2021/3/5

Equipment
ICW,ICWBL,ICWBH,GMAD,α

Smear #	β(cpm)	α(cpm)	γ(mSv/h)	β+γ(mSv/h)
④	>100000	100	1.40	400.0
⑨	>100000	200	2.00	700.0

Figure 3-57 Dose rate measurements on the floor below the duct [3-23]

Subsequently, in May 2021, dosimeters and cameras were placed inside the reactor well via the west reactor cavity differential pressure adjustment line to conduct an investigation of dose rates and conditions inside the well. Figure 3-58 shows an outline of the investigation.

By observing the situation in the well with the camera, it was confirmed that there was no major damage to the PCV top head and other parts (Figures 3-59 and 3-60). The maximum measured dose rate was 530mSv/h obtained near the PCV flange (Figure 3-61). An investigation inside the piping of the reactor cavity differential pressure adjustment line was also conducted, and it was confirmed that the BF-2-12 valve was in the open status and that the piping upstream from the valve (SUS) had no rough surfaces and no sediments such as those observed in the piping and valve box (carbon steel) (Figure 3-62).

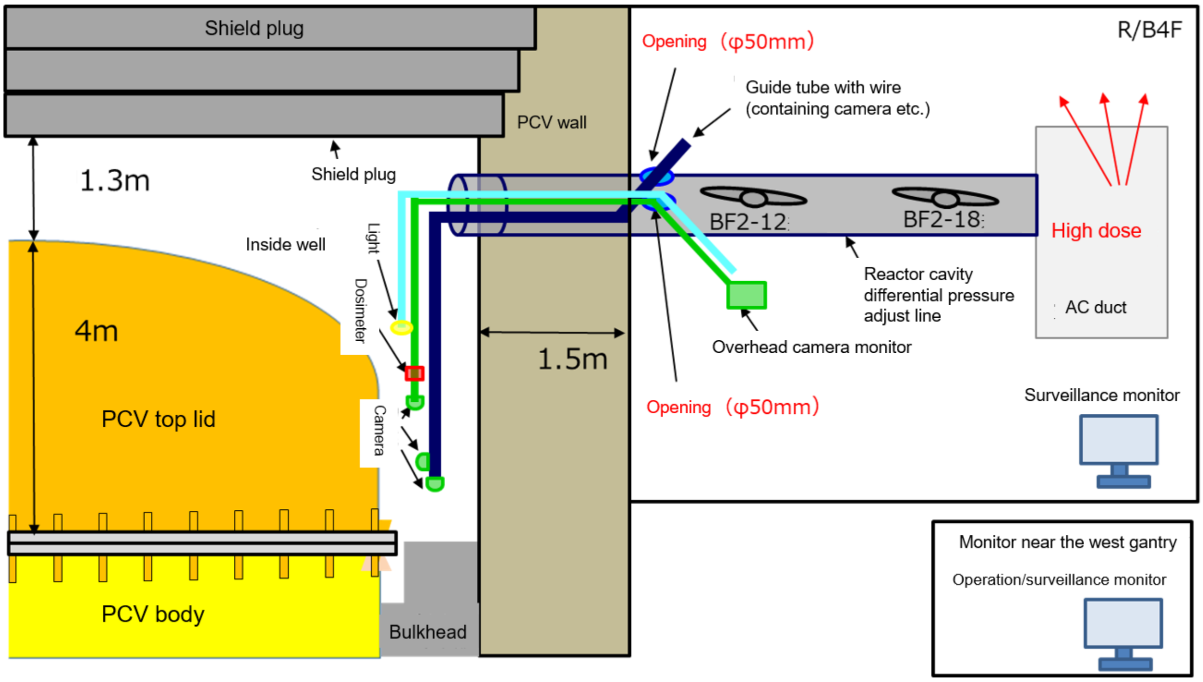


Figure 3-58 Outline of the investigation inside the reactor well [3-24]

[3-24] Handout document: [Preliminary report] Status of response to high contamination of shield plugs at Unit 2, Decommissioning, Contaminated and Treated Water Response Team Joint Meeting, May 27, 2021.

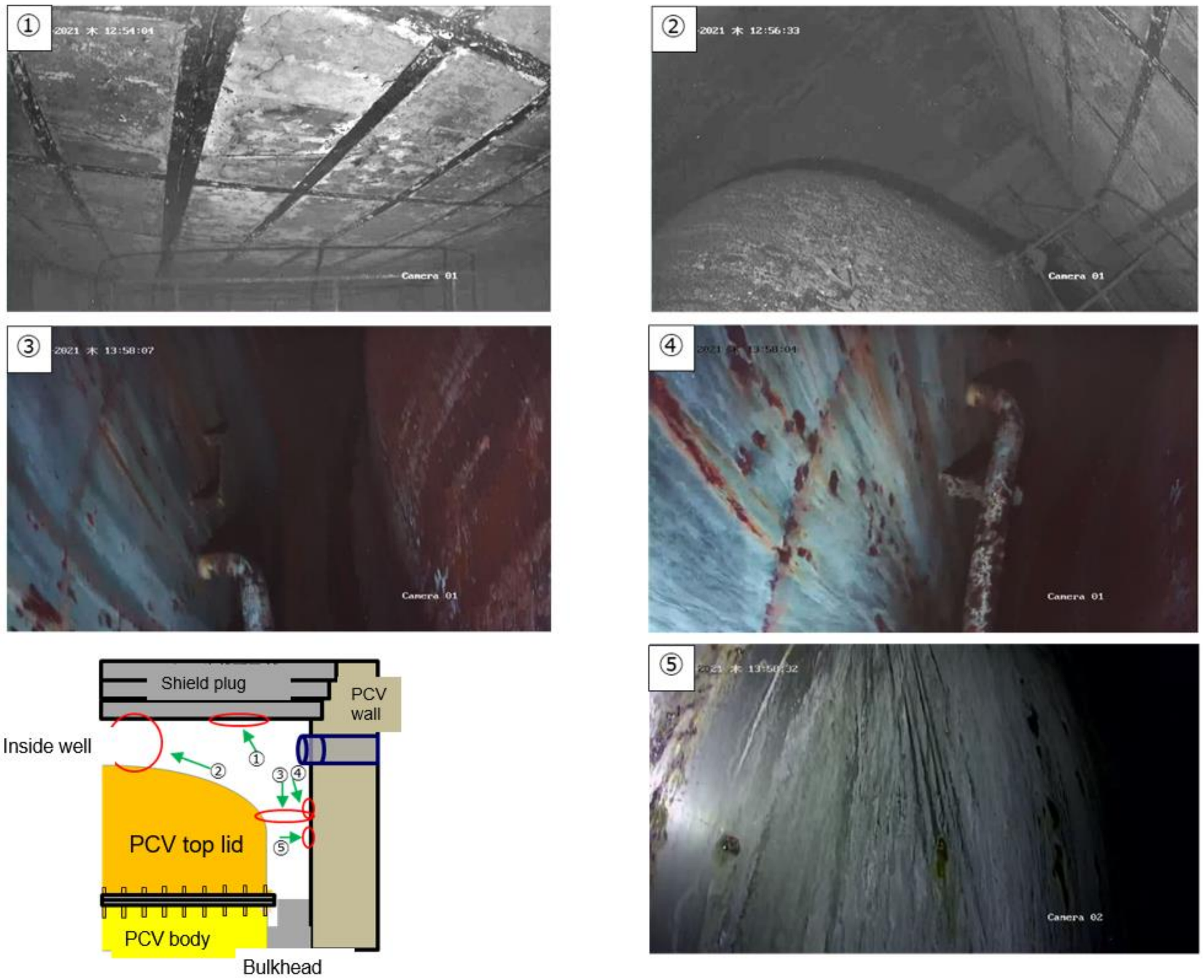


Figure 3-59 Status inside reactor well ① [3-24]

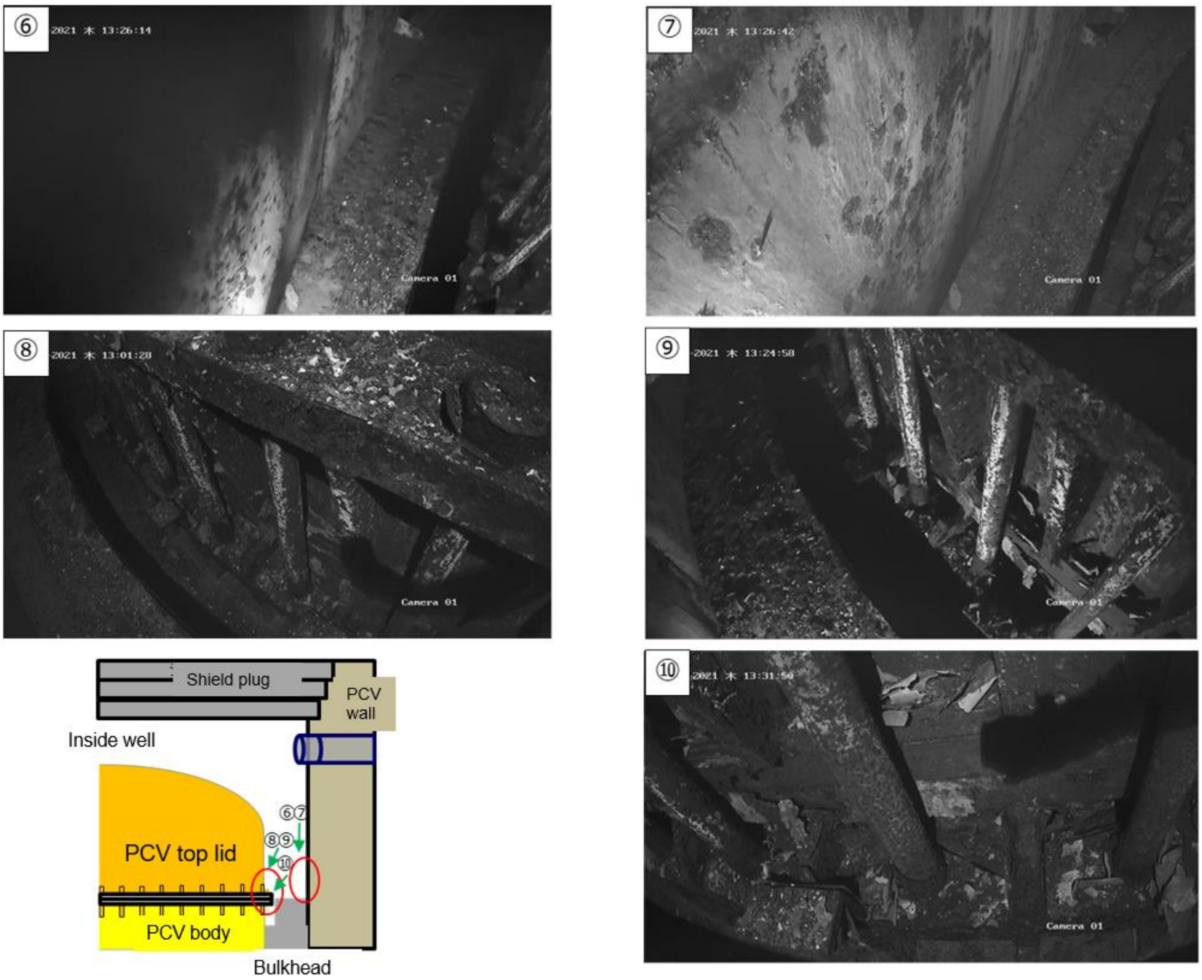
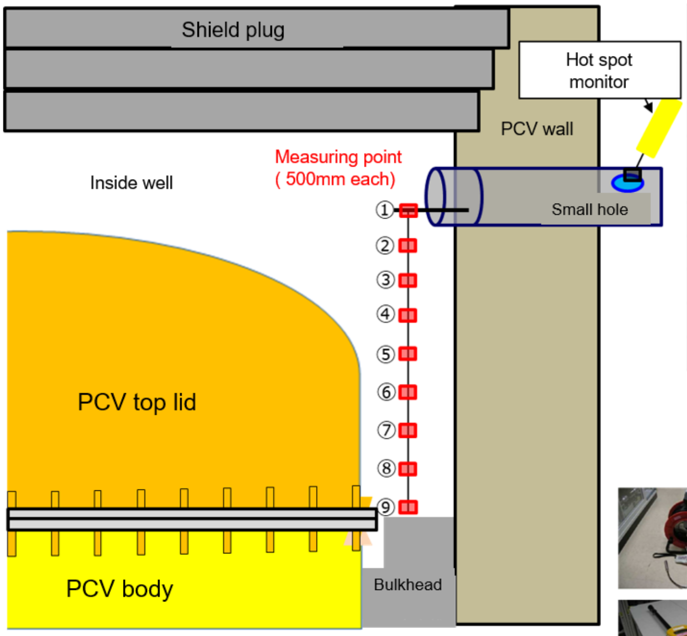


Figure 3-60 Status inside reactor well ② [3-24]



＊Based on the height of the pipe, measurements are taken downward every 500 mm.

Measuring point	Distance (mm)	Dose Equivalent Rate (mSv/h)
①	0	74.6
②	500	150
③	1000	330
④	1500	300
⑤	2000	310
⑥	2500	380
⑦	3000	440
⑧	3500	**530 (Max)**
⑨	4000	350

＊Hot spot monitor measurement
Measured value: 23.5 mSv/h(12.5 mSv/h (underwater survey meter))
Location: small hole, inside of piping

■ Measuring equipment
➢ Underwater survey meter
- ✓ Calibration date: February 10, 2021
- ✓ Measuring range: 1mSv/h – 1000Sv/h
- ✓ Cable length: about 50m

➢ Hot spot monitor
- ✓ Calibration date: December 21, 2020
- ✓ Measuring range: 0.01μSv/h – 9.9Sv/h
- ✓ Extension length: about 4m

Figure 3-61 Results of dose rate measurement inside reactor well [3-24]

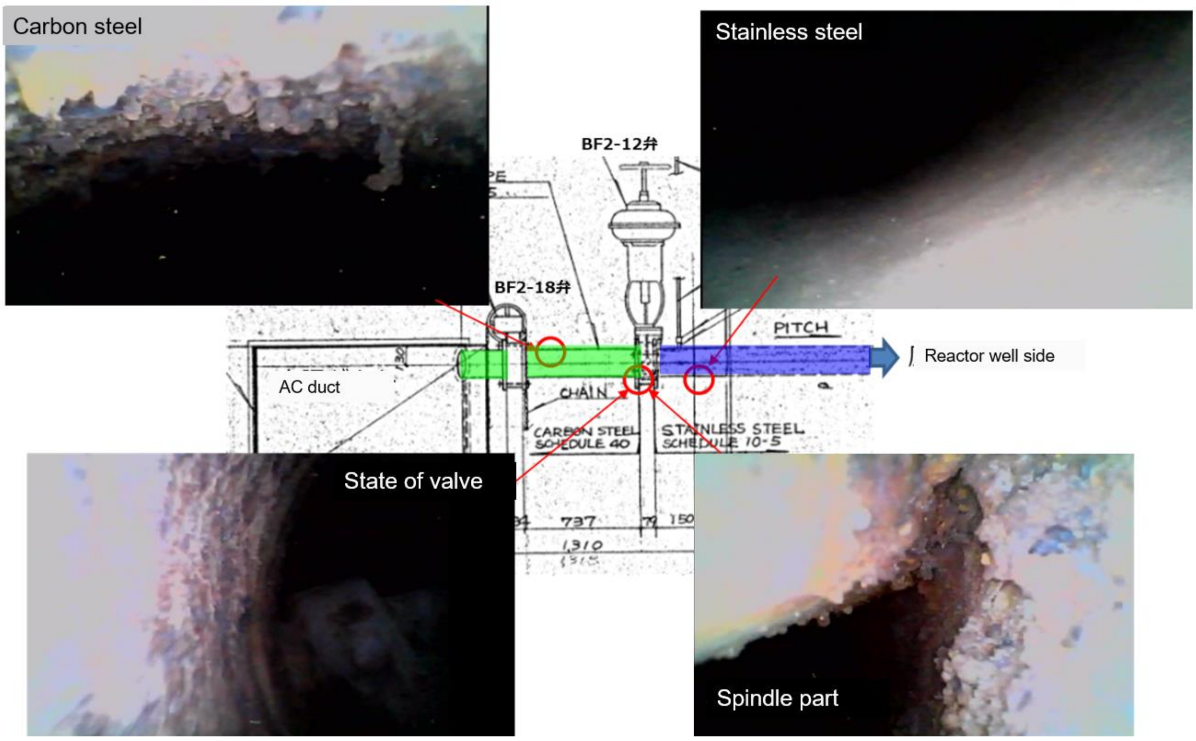


Figure 3-62 Inside views of the reactor cavity differential pressure adjustment line piping [3-24]

In addition, on-site inspection of the reactor cavity differential pressure adjustment line laid on the east side was conducted. The BF-2-13 valve was in the open status similar to that on the west side, and no significant deterioration was observed on the side, bottom, and inspection port of the duct (Figure 3-63). The dose rate measurement around the piping had a maximum value of 51 mSv/h (Figure 3-64).

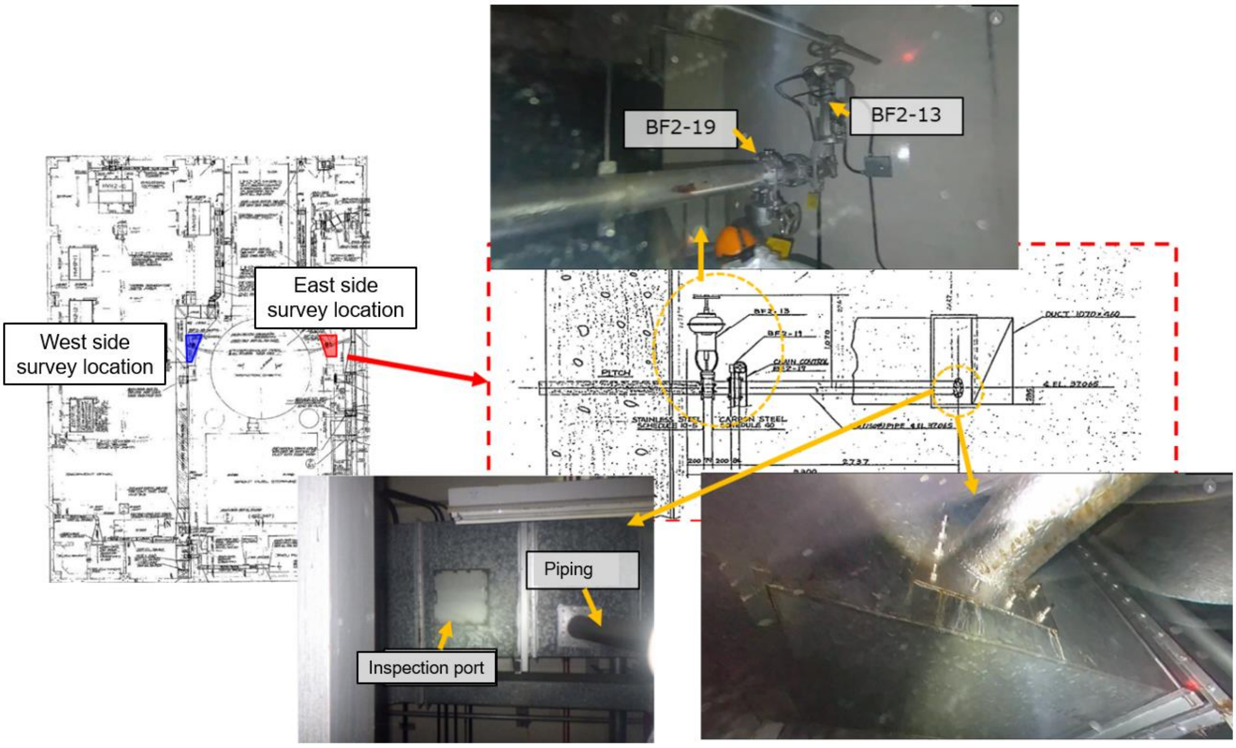
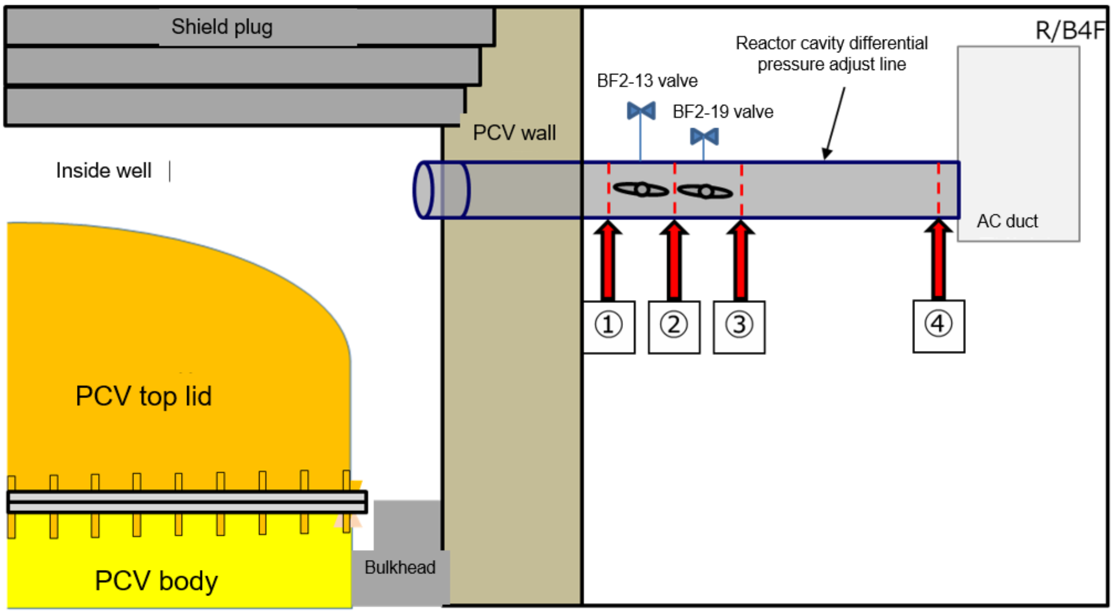


Figure 3-63 External investigation of the reactor cavity differential pressure adjustment line (east side) [3-24]



Measuring location		①	②	③	④
		PCV wall – BF2-13 valve	BF2-13 valve – BF2-19 valve	BF2-19 valve – duct	Near duct
Piping	Top (mSv/h)	13	41	25	18
	Bottom (mSv/h)	13	51	37	20

Figure 3-64　Dose rate measurement results around the reactor cavity differential pressure adjustment line (east side) [3-24]

(14)　Investigation results for the Unit-2 operation floor [UPDATE]

After the construction of an outer wall opening on the west side of the reactor building completed in June 2018, an investigation of the dose rates, etc. in the vicinity of the west wall opening of the operation floor was conducted in July 2018 to allow future inside work for the operation floor to proceed smoothly. From measurement of air dose rates, it was estimated that the main radiation source was the well-plug because the dose rate near the well-plug was high and the rate tended to decrease on moving away from it (Figure 3-65).

① Remote unmanned robot measurement situation (ceiling camera photography set-up)

② Remote unmanned robot measurement situation (robot photography set-up)

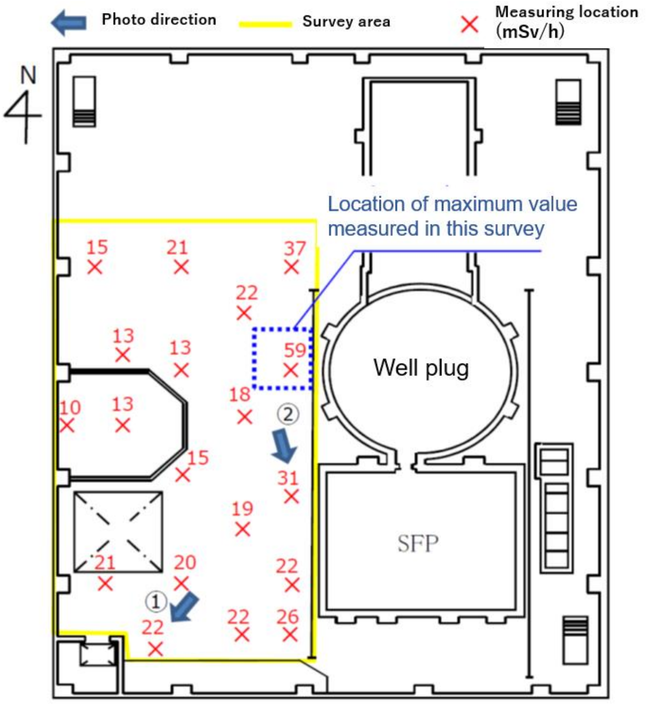


Figure 3-65 Dose rate measurements on the west side of the operating floor (obtained in June 2018) [3-25]

From November 2018 to February 2019, an investigation of the contamination and equipment conditions throughout the operation floor was conducted. Based on the air dose rate measurements (Figure 3-66), the dose rate on the well-plug was high and this plug was estimated to be the main contamination source. In addition, the dose rate decreased compared to the past investigation results, and it was estimated that natural attenuation, the effect of rainwater that flowed into the building, and the effect of moving and cleaning up materials were contributing factors to the drop. The surface dose rate on the well-plug was high (Figure 3-67), and it was estimated that the contamination on its top surface was caused by steam that had been retained between the well and the protecting sheet, and then dried up. Since the β+γ/γ ratio on the well was similar to that on the floor, it was estimated that the effect of surface contamination was significant and the effect of γ radiation from inside the reactor was small.

[3-25] Handout document: Investigation of the operating floor after the opening of the outer wall on the west side of the reactor building of Unit 2, 56th Decommissioning and Contaminated Water Response Team Joint Meeting, July 26, 2018.

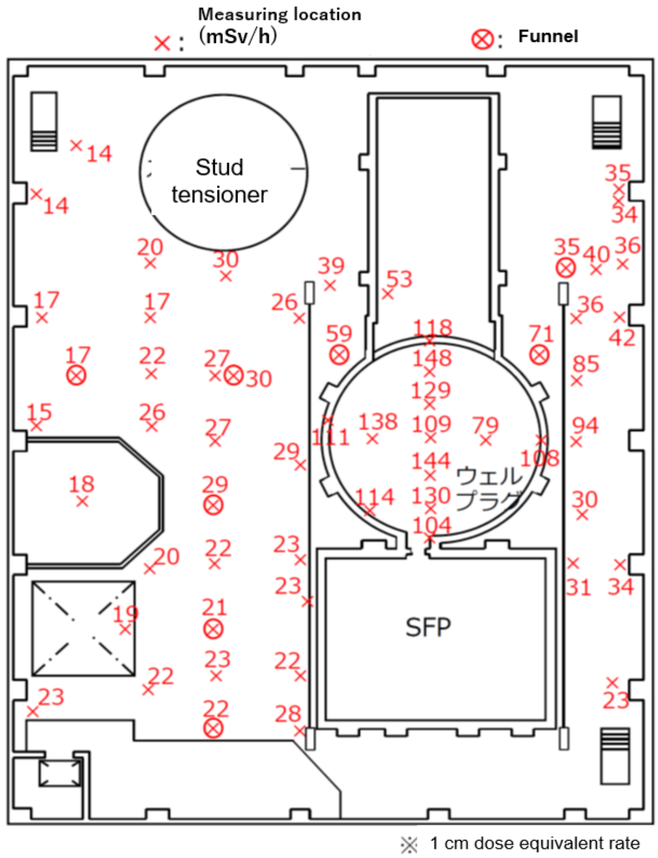
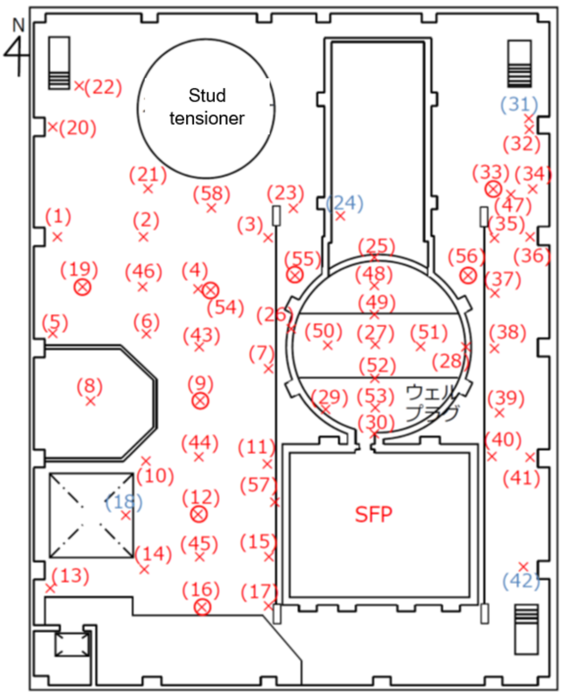


Figure 3-66 Air dose rates measured on the operation floor
(from November 2018 to February 2019) [3-26]

[3-26] Handout document: Results of the investigation after moving and cleaning up the remaining items inside reactor operating floor of Unit 2, 63rd Decommissioning and Contaminated Water Response Team Joint Meeting, February 28, 2019.

X : Measuring location ⊗: Funnel



(mSv/h)

Location	γ※1	β+γ※2	Note
(1)	0.2	40	
(2)	0.2	17	
(3)	0.3	57	
(4)	0.2	86	
(5)	0.4	79	
(6)	0.3	74	
(7)	1.9	46	
(8)	0.1	18	
(9)	1.8	306	Funnel
(10)	0.5	74	
(11)	0.3	58	
(12)	3.6	312	Funnel
(13)	0.5	62	
(14)	0.6	135	
(15)	0.3	77	
(16)	0.4	24	Funnel
(17)	0.9	11	
(18)	1.0	45	
(19)	1.9	126	Funnel
(20)	0.3	40	
(21)	0.3	184	
(22)	0.3	74	
(23)	1.8	139	
(24)	2.5	165	
(25)	6.4	645	Crevice
(26)	8.0	1030	Crevice
(27)	12	1410	
(28)	1.0	317	Crevice
(29)	2.0	926	
(30)	3.7	625	Crevice

Location	γ※1	β+γ※2	Note
(31)	1.0	73	
(32)	3.9	343	
(33)	4.5	156	Funnel
(34)	3.2	102	
(35)	0.6	58	
(36)	0.6	106	
(37)	2.3	133	
(38)	5.2	302	
(39)	0.3	42	
(40)	1.1	98	
(41)	2.2	105	
(42)	0.5	42	
(43)	0.5	130	
(44)	0.2	45	
(45)	0.2	62	
(46)	0.1	68	
(47)	1.5	204	
(48)	12	930	
(49)	13	3060	Crevice
(50)	7.1	1220	
(51)	8.0	247	
(52)	15	2720	Crevice
(53)	5.0	508	
(54)	6.0	769	Funnel
(55)	3.0	281	Funnel
(56)	8.6	503	Funnel
(57)	1.1	79	
(58)	0.6	173	

※1 : 1 cm dose equivalent rate, measured with a collimated dosimeter @ 30.5 cm above the floor
※2 : 70μm dose equivalent rate, measured with a collimated dosimeter @ 0.5cm above the floor

Note: Blue characters indicate locations measured away from the measurement location due to interference with existing equipment.

Figure 3-67 Surface dose rates measured on the operation floor (from November 2018 to February 2019) [3-26]

The air dose rate for the operation floor was measured again in December 2020 after work on the floor to clean up and remove remaining objects was completed. The overall dose rate was confirmed to have been reduced by about 20% compared to the results of measurements in 2018 (Figure 3-68). As factors contributing to the dose rate reduction, it was estimated that about 10% of the dose rate reduction was contributed by the cleanup and removal of objects, and about 10% by natural attenuation.

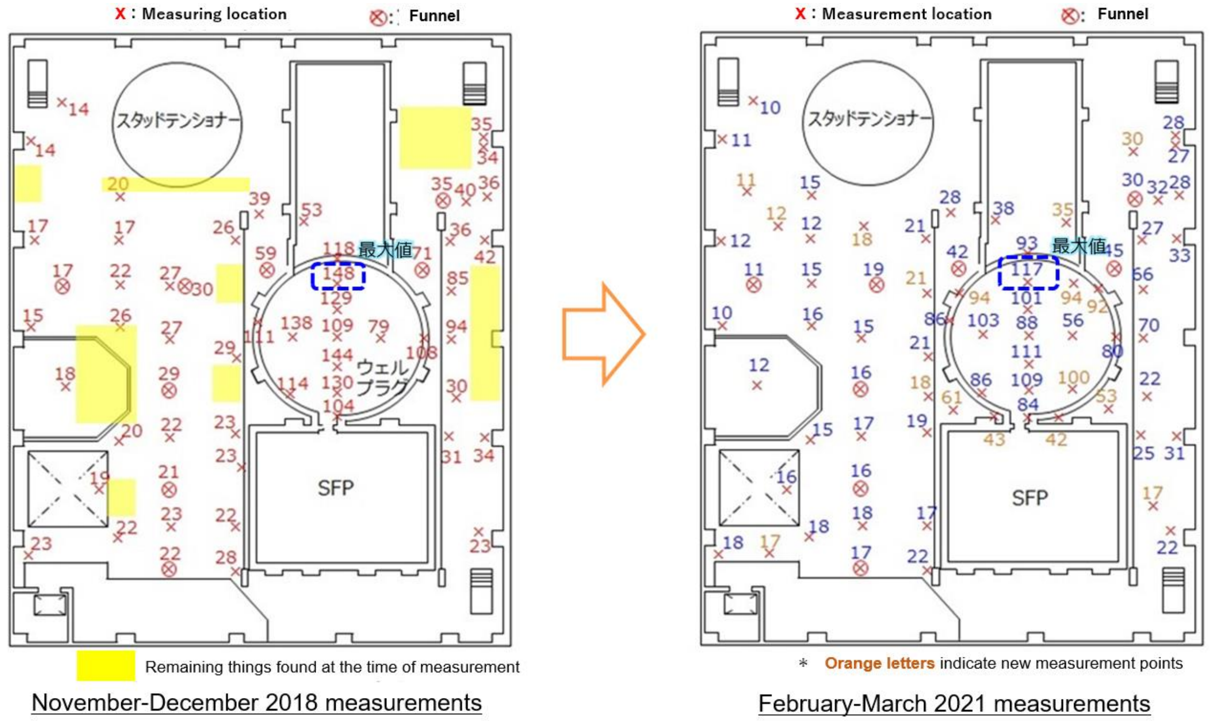
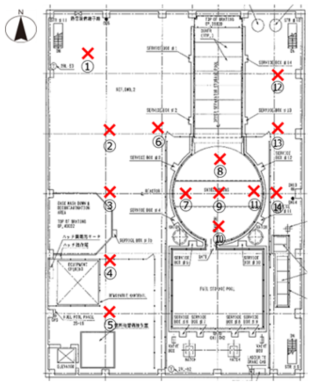


Figure 3-68 Comparison of measured air dose rates on the operation floor (obtained in December 2020) [3-27]

In April 2021, a dose rate investigation of the floor and ceiling surfaces of the operation floor was conducted using a remotely operated robot in cooperation with the Nuclear Regulation Authority. The results confirmed that the contamination densities of the floor surfaces for the operation floor (east side, west side, and on the shield plug) were similar to each other (Figure 3-69). The reason for the higher air dose rate above the shield plug in the operation floor air dose rate measurements conducted in March 2021 than in other areas was attributed to the effect of scattered radiation from cesium accumulated in the gaps and lower part of the shield plug. The dose rate contribution from the ceiling surface at a floor height of 1m was evaluated as about 0.9mSv/h when contamination of the ceiling surface was uniformly present (Figure 3-70).

[3-27] Handout document: Preliminary report on the investigation of the operating floor of the Unit-2 reactor building, 88th Decommissioning and Contaminated Water Response Team Joint Meeting, March 25, 2021.



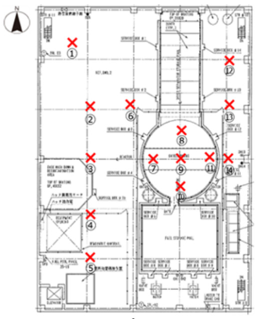
X : Measuring location

＊1 Only the yellow area is valid.
The measurement locations with a difference of more than 10% between the 1 cm dose equivalence rate with and without acrylic shielding were excluded from the evaluation because of the possibility of local high concentration of contamination near the dosimeter.

＊2 Surface contamination density conversion formula
Surface contamination density = (without acrylic shield (70 μm) - with acrylic shield (70 μm)) / conversion constant
・Conversion constant: 7.2E-04[(mSv/h)/BQ/cm2)
(Evaluated by KEK based on the measured values)

Location	1 cm dose equivalent rate mSv/h		70 μm dose equivalent rate mSv/h		Surface contamination (Evaluated value) ＊2 Bq/cm²
	w/ shield	w/o shield	w/ shield	w/o shield	
①	6.72	6.76	32.3	8.58	3.3E+04
②	14.2	25.7	29.8	40.3	－※1
③	5.92	5.84	15.1	6.80	1.2E+04
④	8.26	7.78	36.3	9.42	3.7E+04
⑤	19.2	14.2	42.7	16.8	－※1
⑥	17.5	16.3	65.0	20.6	6.2E+04
⑦	38.0	36.3	107	46.8	8.3E+04
⑧	229	254	362	353	1.2E+04
⑨	265	365	567	485	－※1
⑩	147	123	472	156	－※1
⑪	22.2	23.3	142	30.5	1.6E+05
⑫	50.3	49.1	132	60.5	1.0E+05
⑬	113	85.8	189	102	－※1
⑭	50.3	49.1	92.0	66.4	3.6E+04

Figure 3-69 Measurement and evaluation results for the operation floor (obtained in April 2021) [3-28]



X : Measuring location

＊1 Dose contribution from the ceiling at 1m above floor level when the average surface contamination density (2.3E+05 Bq/cm²) of the ceiling at 14 locations is uniformly present on the ceiling (evaluated by KEK based on measurements)

＊2 surface dose rate conversion formula
Surface dose rate = collimator value X conversion constant - air dose rate X lead attenuation rate
・Conversion constant: 8.2E-004[(mSv/h)/cps]
・Lead attenuation rate: 1.81E-3

＊3 Surface contamination density conversion formula
Surface contamination density = Surface dose rate X conversion constant
・Conversion constant: 1.38E+06 Bq/cm²/(mSv/h)
(Evaluated by KEK based on measured values)

Location	Measuring start time	Measuring end time	Collimator value (measured) (cps)	Air dose rate (measured) (mSv/h)	Surface dose rate (evaluated) ＊2 (mSv/h)	Surface contamination (evaluated) ＊3 (Bq/cm²)
①	11:03:00	11:04:00	113	8.78	0.08	1.1E+05
②	11:07:00	11:08:00	410	13.60	0.31	4.3E+05
③	11:10:00	11:11:00	263	11.08	0.20	2.7E+05
④	11:13:15	11:14:15	126	11.52	0.08	1.1E+05
⑤	11:15:35	11:16:35	155	13.68	0.10	1.4E+05
⑥	11:20:00	11:21:00	229	20.88	0.15	2.1E+05
⑦	11:27:45	11:28:45	299	61.27	0.13	1.8E+05
⑧	11:31:15	11:32:15	293	102.2	0.06	7.5E+04
⑨	11:34:30	11:35:30	379	117.2	0.10	1.4E+05
⑩	11:37:10	11:38:10	262	70.34	0.09	1.2E+05
⑪	11:41:20	11:42:20	346	61.27	0.17	2.4E+05
⑫	11:48:40	11:49:40	147	33.62	0.06	8.2E+04
⑬	11:52:20	11:53:20	343	31.32	0.22	3.1E+05
⑭	11:58:20	11:59:20	865	53.56	0.61	8.4E+05

Figure 3-70 Measurements and evaluation results for the operation floor ceiling surface (obtained in April 2021) [3-28]

[3-28] Handout document: Results of the Unit-2 operating floor investigation conducted in cooperation with the Nuclear Regulation Authority, 90th Decommissioning and Contaminated Water Response Team Joint Meeting, May 27, 2021.

Based on the above results, a survey was conducted from August to September 2021 using the existing shield plug drilling locations (center and east), which are less susceptible to surface contamination from the operation floor, for the purpose of improving the evaluation accuracy of the amount of radioactivity that was estimated to have accumulated in the gap between the upper and middle sections of the shield plug. An outline of the survey is shown in Figure 3-71. Based on the results (Figure 3-72), the dose equivalent rate of Cs-137 and Cs-134 accumulated in the gaps of the shield plug was calculated, and the possibility of radioactive materials including cesium adhering to and depositing in the gaps was high, and from the results of the center and east side, the overall contamination status of the shield plug was evaluated as widely variable throughout the shield plug (Figure 3-73).

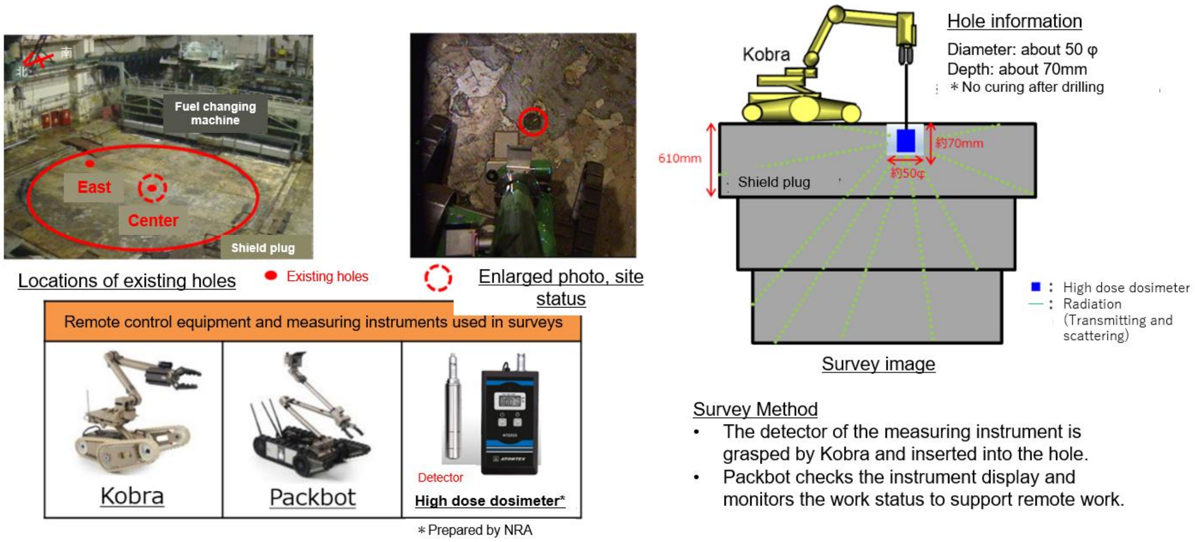
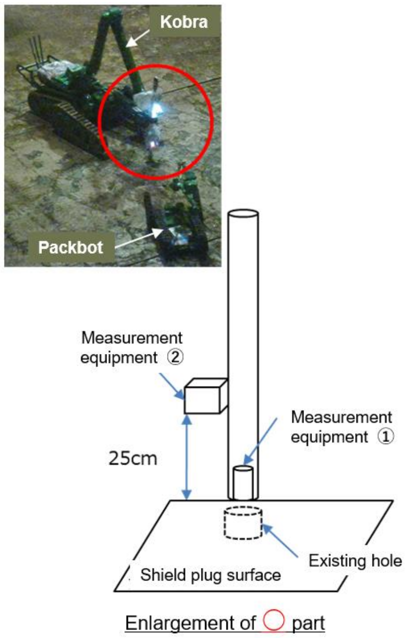


Figure 3-71 Outline of survey [3-29]

[3-29] Handout document: Investigation of the shield plug perforation in the Unit 2 operating floor, 94th Decommissioning and Contaminated Water Response Team Joint Meeting, September 30, 2021.

Unit: mSv/h

Location	Distance from floor to tube bottom [cm]	Equipment ①	Equipment ②
East	7.0	255	52.5
	6.0	277	51.5
	5.0	290 – 300	52.1
	4.0	292	50.9
	3.0	255	50.7
	2.0	225	51.9
	1.0	172	51.9
	7.0	255	51.5
Center	6.0	1169	230
	5.0	1070	236
	4.0	944	235
	3.0	825	225
	2.0	682 – 690	226
	1.0	600	225
	0.0	532	225



Measurement date: August 26, 2021

Figure 3-72 Measurement results (obtained in August 2021) [3-29]

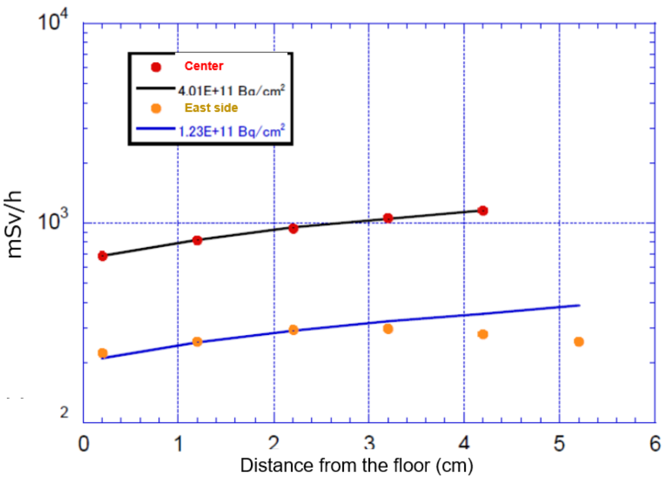
[Comparison with measurement results]

✓ Hole of reactor well center
Calculated value is consistent with contamination density of 4.01E+11Bq/cm² for Cs-137.

✓ Hole of reactor well east side
Calculated value is consistent with contamination density of 1.23E+11Bq/cm² for Cs-137, except for 4.2 cm and 5.2 cm. ＊1
＊1: Possibly due to the shape of the bottom of the hole.

➢ There is a high possibility that radioactive materials including Cs are adhering to and depositing in the crevice.

➢ The measurement results at the center and the east side indicate that there may be a large variation in the contamination status throughout the shield plug.



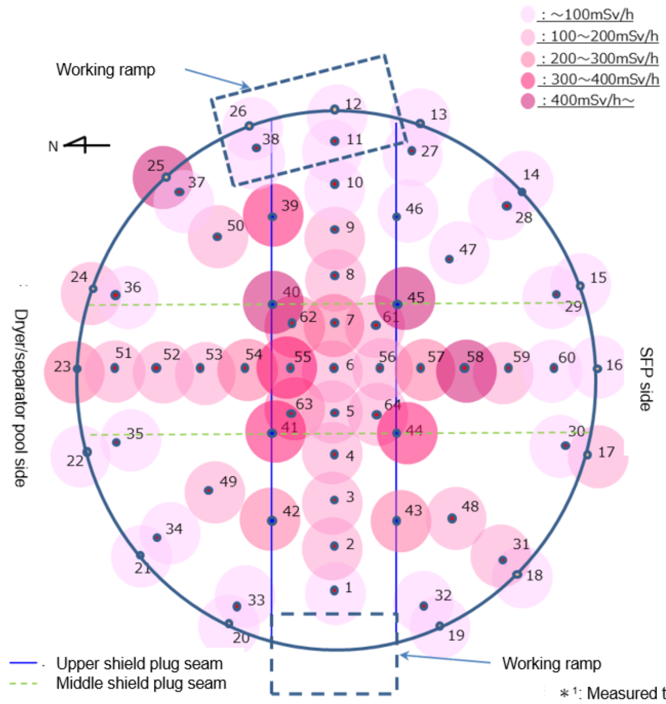
Comparison between measurement and calculation*

＊The calculated values were evaluated at 1.8cm above the tube bottom, which is the effective center position of the dosimeter.

Figure 3-73 Comparison between measured and calculated values [3-29]

To further understand the contamination status of the shield plug, an investigation using newly drilled locations was planned. First, a dose rate investigation on the shield plug was conducted in October 2021, and it was confirmed that the dose rate was high in the center and at the joints, and that the dose rate varied above the shield plug (Figure 3-74). Based on the dose rate above the shield plug, 13 new locations were selected for drilling, and a dose rate investigation for them was conducted from November to December 2021. The status of the dose rate investigation is shown in Figure 3-75. These measured dose rates were lower than those of the previously existing holes. The dose rates at measurement points Nos. ①, ②, ⑩, and ⑪, which are close to the outer circumference of the shield plug, were low, while those at Nos. ④, ⑨, ⑫, and ⑬ exceeded 100mSv/h (Figure 3-75). In order to confirm the influence of the rebar arrangement in the shield plug, the measurement of perforation points and the dose rate on the floor surface around the perforation points was conducted again (Figure 3-76). Based on these results, the Nuclear Regulation Authority conducted an evaluation of the total amount of contamination accumulated in the gap between the upper and middle sections of the shield plug and concluded that the amount of contamination was at the same level as the previous evaluation results (tens of PBq of Cs-137 were present) (Figure 3-77).

➢ It was confirmed that the radiation dose was high in the central part and the joint part, and that there was variation in the radiation dose at the top of the shield plug.



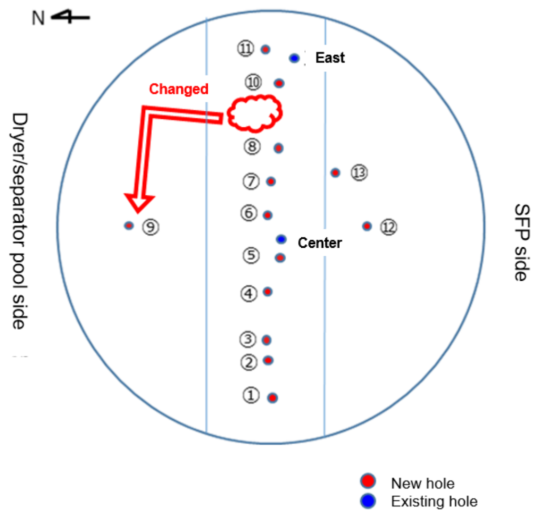
Uni t: mSv/h

No.	Value	No.	Value	No.	Value
1	44.8	23	270	45	524
2	107	24	126	46	93.2
3	175	25	521※2	47	59.3
4	136	26	23.0	48	143
5	152	27	17.0	49	105
6	104,144※1	28	96.5	50	105
7	294	29	73.0	51	139
8	117	30	78.0	52	157
9	134	31	105	53	138
10	76.2	32	23.0	54	259
11	14.4	33	51.5	55	390
12	37.5	34	36.0	56	123
13	58.8	35	36.3	57	228
14	50.0	36	73.5	58	420,385※3
15	62.6	37	59.6	59	113
16	82.0	38	12.9	60	61.5
17	103	39	340	61	165
18	92.1	40	560	62	270
19	40.0	41	391	63	262
20	72.4	42	266	64	148
21	70.3	43	241	–	–
22	71.0	44	343	–	–

*1: Measured twice. *2: There is material remaining nearby. About 70msv/h where remaining materials were avoided. *3: Measured twice.

Figure 3-74 Dose rate results above the shield plug (obtained in October 2021) [3-30]

[3-30] Handout document: Investigation of the shield plug drilling part in the operating floor of Unit No. 2, 97th Decommissioning and Contaminated Water Response Team Joint Meeting, December 23, 2021.



Status of drilled hole location

Measured on December 6th

No.	Depth *1 cm	Result mSv/h
①	9.5	11
	8	14
	6	18
②	8	11
	6	16
④	7	82(139)
	6	82(156)
⑤	7.5	34
	6	37
⑥	7	58
	6	58
⑦	9	67-69
	8	68-70
	6	66-69
⑫	8	97(117)
	6	112-120
⑬	10	97(135)
	8	97(105)
	6	112(120)

Measured on December 7th

No.	Depth *1 cm	Result mSv/h
③	8.5	50
	8	51
	6	52
⑧	10	41
	8	43
	6	44
⑨	8.5	112
	8	109
	6	114
⑩	7.5	8
	6	10
⑪	8	11
	6	14

Instrument: AT2533 (loaned from NRA)

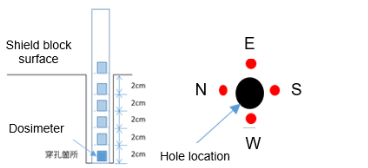
*1 Depth from floor to tube bottom. Drill work was carried out with the goal of 10 cm from the floor, but the depth varies due to the influence of the core cut surface.
( ) is the maximum value

Figure 3-75 Dose rate results measured at newly drilled locations (Part 1) (obtained in December 2021) [3-30]

Location No.	Maximum insertion depth *	Dose at insertion depth from hole floor (mSv/h) Insertion depth 8cm	6cm	4cm	2cm	0cm	Floor surface dose around the hole (mSv/h) East	West	South	North
①	(9.5) 7.37	7.35	10.2	12.0	24.1	35.3	35.7	35.9	33.5	41.6
②	(9.0) 8.65	8.15	8.23	10.4	18.5	41.4	52.0	41.1	44.2	43.9
③	(8.5) 32.3	34.7	43.3	43.7	65.5	101	176	104	105	99.4
④	(7.0) 72.2	–	66.3	86.4	110	147	157	207	161	159
⑤	(7.5) 24.5	–	25.1	26.2	70.1	125	132	169	107	142
⑥	(7.0) 42.8	–	44.4	45.9	78.1	169	145	196	191	169
⑦	(9.0) 52.0	51.5	52.7	53.9	72.9	112	243	95.8	147	154
⑧	(10.0) 36.5	40.4	45.6	40.5	65.5	137	176	119	138	135
⑨	(8.5) 70.2	69.5	91.5	93.6	97.4	176	157	314	222	183
⑩	(7.0) 4.83	–	5.34	6.37	12.7	24.0	22.9	30.5	25.9	30.0
⑪	(8.0) 5.90	–	8.41	10.6	15.6	26.8	26.4	26.5	26.0	26.6
⑫	(8.0) 87.3	–	92.4	95.6	111	228	440	138	222	213
⑬	(10.0) 75.4	76.2	77.6	88.4	91.3	182	264	175	182	278
Existing hole (Center)	(5.0) 950	–	–	807	773	529	304	512	307	302
Existing hole (East side)	(5.0) 293	–	–	289	221	136	126	101	74.7	102

* ( ) is the maximum insertion depth (cm)

Instrument: AT2533 (loaned from NRA)



(1): Inside the hole (2): Floor surface around the hole

(Change points)
(1) Added measurement of shield plug hole floor surface (0 cm)
(2) (2) Added 4-point measurement of the floor around the hole

Instrument : Polimaster PM1703MO-1

NRA attached detector to the instrument

Figure 3-76 Dose rate results measured at newly drilled locations (Part 2) (obtained in December 2021) [3-30]

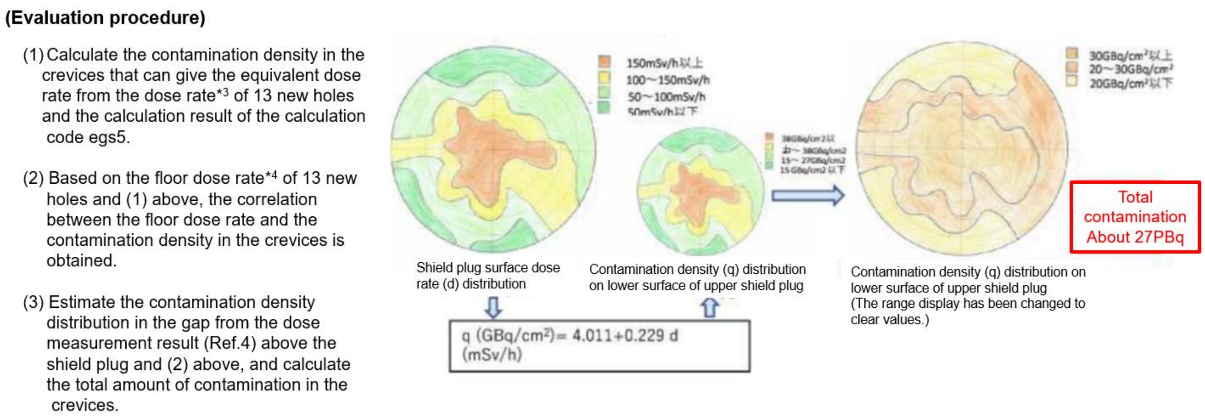


Figure 3-77 Evaluation results of total pollution [3-30]

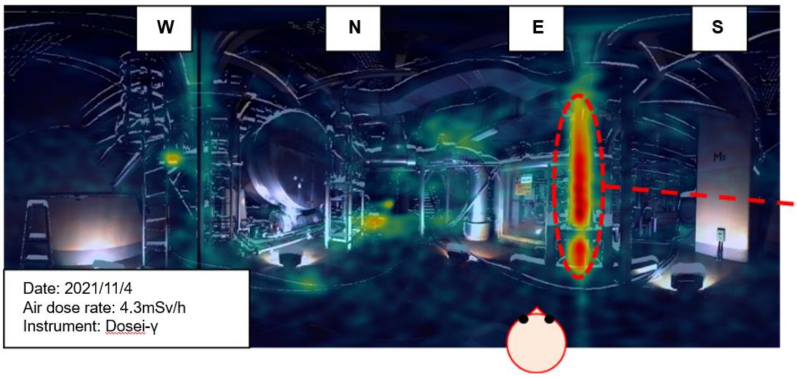
(15) Investigation results inside reactor building of Unit-2 [UPDATE]

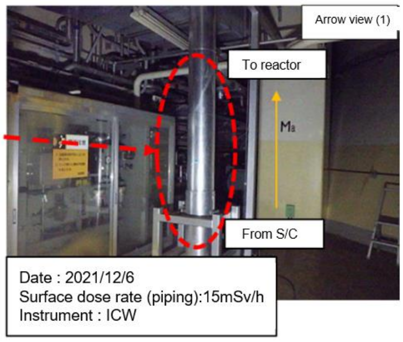
From November to December 2021, an investigation was conducted using a γ imager, 3D scanning equipment, and dosimeters to obtain detailed spatial information (accessibility, etc.) and dose rate information inside the reactor building (ground floor) for the planning of future investigations inside this building.

The γ imager measurements revealed a hot spot in the CS (A system) piping on the northeast side of the second floor (Figure 3-78). The inner surface of the piping seemed to be contaminated. The piping has not been used for water injection from the time of the accident to the present, and the cause of the contamination is unknown at this time.

Hot spots were also identified in the reactor instrumentation piping (reactor pressure and reactor water level) on the east side of the second floor (Figure 3-79). It seemed that the inner surface of the piping was contaminated. As to the cause of the contamination, since both the pressure gauges and the water level gauges measure water inside the reactor, it is possible that water inside the reactor that came into contact with molten fuel or gas produced inside the reactor flowed into the piping and radioactive materials were deposited.

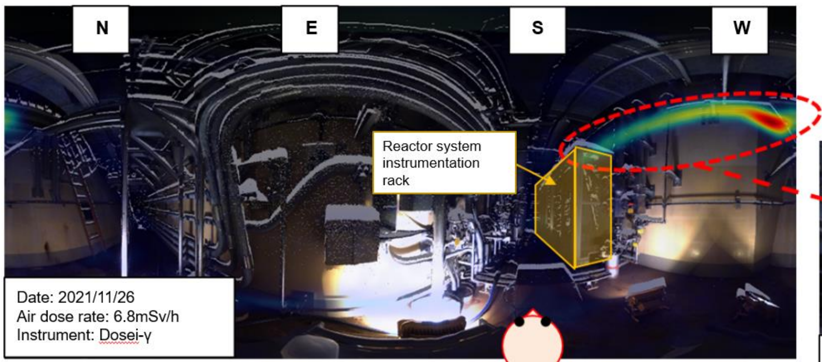
Furthermore, to obtain information that would contribute to the evaluation of the integrity of the reactor building, investigations of the conditions of the reactor shell walls and pool walls on the first (northwest and southwest), second (north, east, and southwest), and third (northeast and southwest) floors were conducted, and point cloud data was obtained using 3D scanning equipment (Figure 3-80). From these investigations, peeling paint and cracks were observed in some areas, but no damage or signs of aging deterioration (e.g., peeling surface concrete and rust) that would lead to a decrease in seismic performance were confirmed.

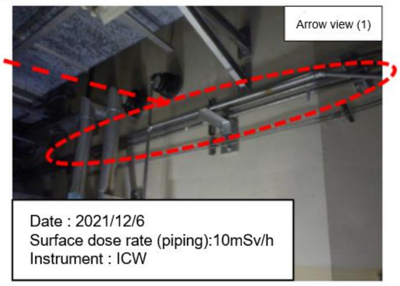




＊ Relative display of temperature distribution up to 10% of the maximum value (blue) based on the maximum value (red) of the radiation source intensity in the image

Figure 3-78 Measurement results obtained with the γ imager (CS (A system) piping on the northeast side of the second floor) [3-31]



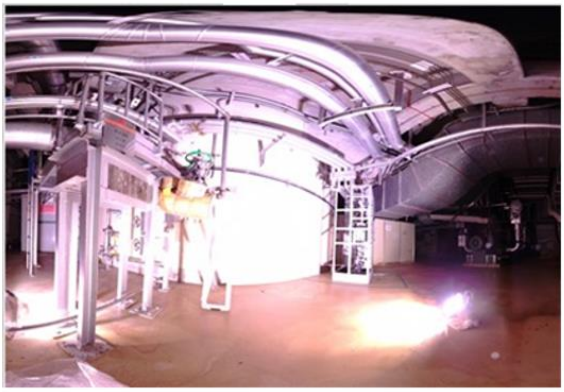


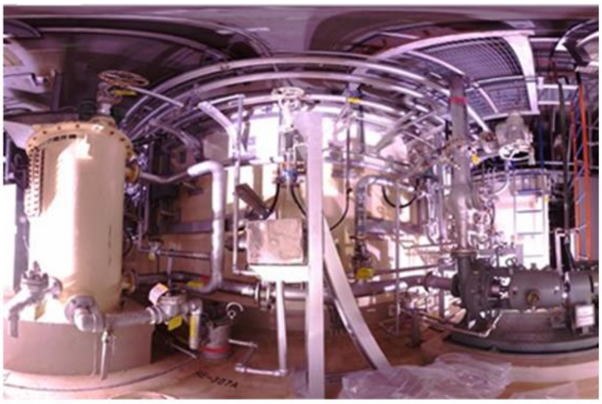
＊ Relative display of temperature distribution up to 10% of the maximum value (blue) based on the maximum value (red) of the radiation source intensity in the image

Figure 3-79 Measurement results obtained with the γ imager (reactor instrumentation piping on east side of the second floor) [3-31]

[3-31] Handout document: Investigation results of the upper floors of the reactor building of Units 1-2, 99th Decommissioning and Contaminated Water Response Team Joint Meeting, February 24, 2022.

2nd floor shell wall east face (point cloud data)





3rd floor shell wall east face (photo)

3rd floor pool wall west side (photo)

Figure 3-80 Investigation results of reactor shell wall and pool wall [3-31]

4. Conditions of Unit-3 core and PCV

(1) Investigation of torus room

In the Unit-3 torus room investigation in July 2012, a robot accessed the gallery inside. Videotaping, dose rates measurements, acoustic checks, etc. were also carried out to the extent possible.[4-1]

No water leaking position in the S/C has been located yet. At least, no leak was confirmed on the flange, etc. of the S/C manholes, as far as the photos show (Figure 4-1).

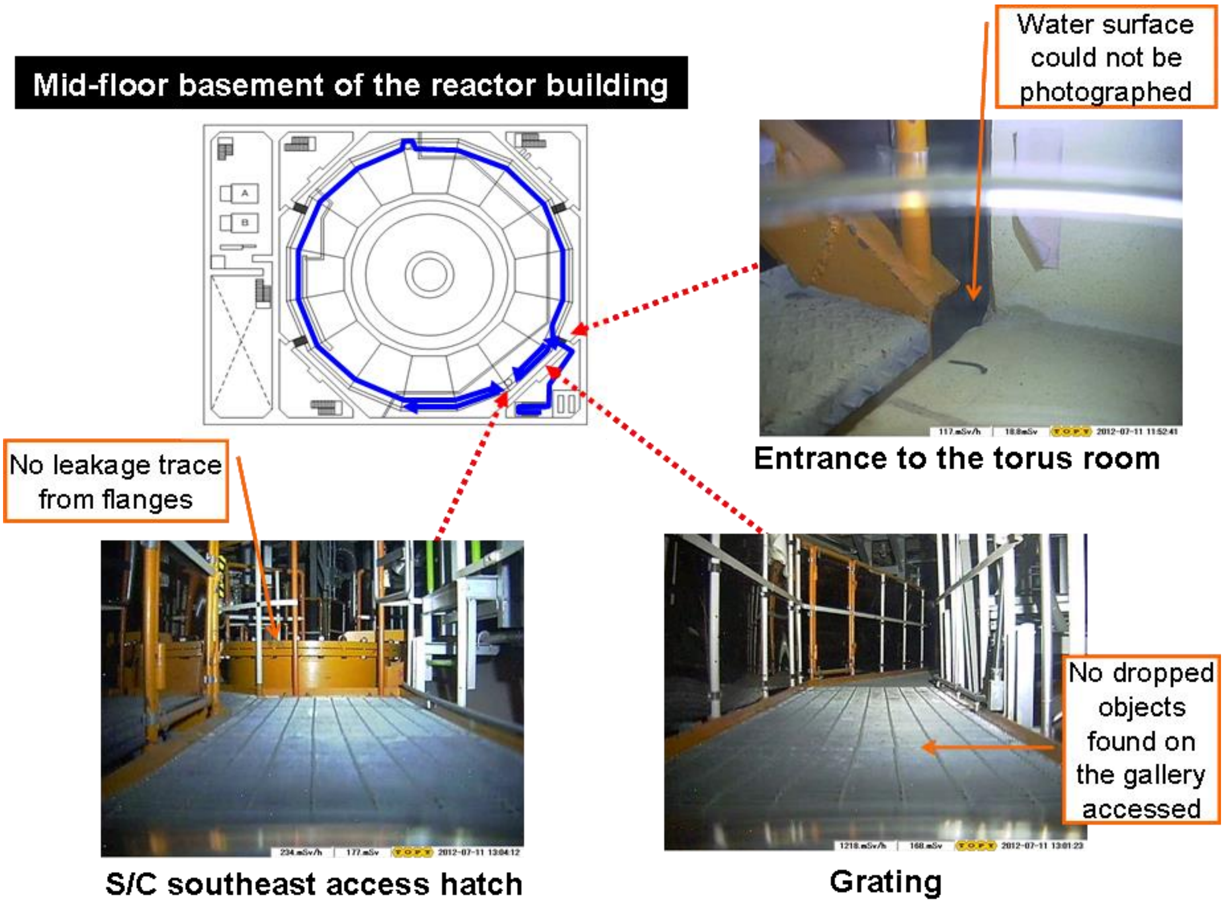


Figure 4-1 Photos in the torus room of Unit-3 (part)

(2) Oxygen concentrations in the PCV

Nitrogen is being sent to the PCV in order to maintain an inert atmosphere, while the containment gas control system discharges the same amount of gas from the PCV. It was confirmed through analyzing the discharged gas that the oxygen concentrations in the PCVs of Unit-1 and Unit-2 were nearly zero, while that in Unit-3 was about 8% (July 2012 [4-2]), analyzed again in March and April of 2013). Containment pressures of Unit-1 and Unit-2 PCVs are at

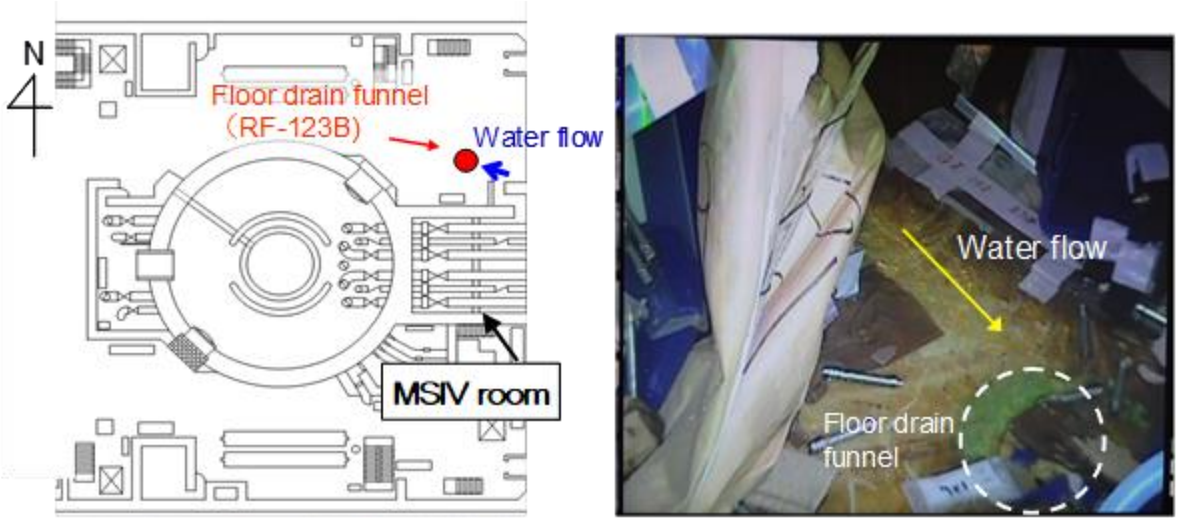
[4-1] Handout document: Investigation inside the reactor containment vessel of Unit-3, 48th Decommissioning and Contaminated Water Response Team Joint Meeting, November 30, 2017.

[4-2] Status inside reactor containment vessel based on the results of atmospheric gas measurements, Technical Workshop for the Accident of the Fukushima Daiichi Nuclear Power Station, July 23, 2012.

several kPa, and remaining positive, while the pressure of the Unit-3 PCV is almost constantly at the level of the atmospheric pressure. Consequently, the gas leak rate of the Unit-3 PCV was confirmed to be the highest.

(3) Survey results of leaked water in the MSIV room of Unit-3 (UPDATE)

In January 2014, while photos taken by the rubble and wreckage removal robot in the Unit-3 reactor building 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 first floor. The water was flowing towards a nearby floor drain funnel (Figure 4-2). [4-3]



Simplified plan view of Unit 3 R/B 1st floor (Photo on Jan. 18, 2014)

Figure 4-2 Water leak near the MSIV room door of Unit-3

The water level in the PCV is estimated as about OP.12 m (about 2m above the reactor building first floor) by converting the S/C pressure obtained by the existing pressure indicators to water head. This elevation is on the level of PCV penetrations for main steam lines, thus indicating the possibility of water leaks from the PCV penetration in the MSIV room as the source of the water flow. In consideration of this possibility, instrumentation was inserted into the MSIV room from the HVAC system room on the floor above, in April and May 2014, and photos were taken and dose rates were measured in the room, in order to locate the water flows in the room. Water leaks were from near the expansion joint of main steam line D. It was concluded that the leakage had occurred only from the main steam line D, based on: (1) confirmation of no leaks from the main steam lines A, B and C, and their main steam drainpipes; and (2) the flow directions of leaked water on the floor (Figure 4-3). [4-4]

[4-3] Handout document: Unit-3, Water flow from the vicinity of the main steam isolation valve room on the first floor of the reactor building to the floor drain funnel, 2nd Decommissioning and Contaminated Water Response Team Joint Meeting, January 30, 2014.
[4-4] Handout document: Unit 3 main steam isolation valve (MSIV) room investigation results, 6th Decommissioning and Contaminated Water Response Team Joint Meeting, May 29, 2014.

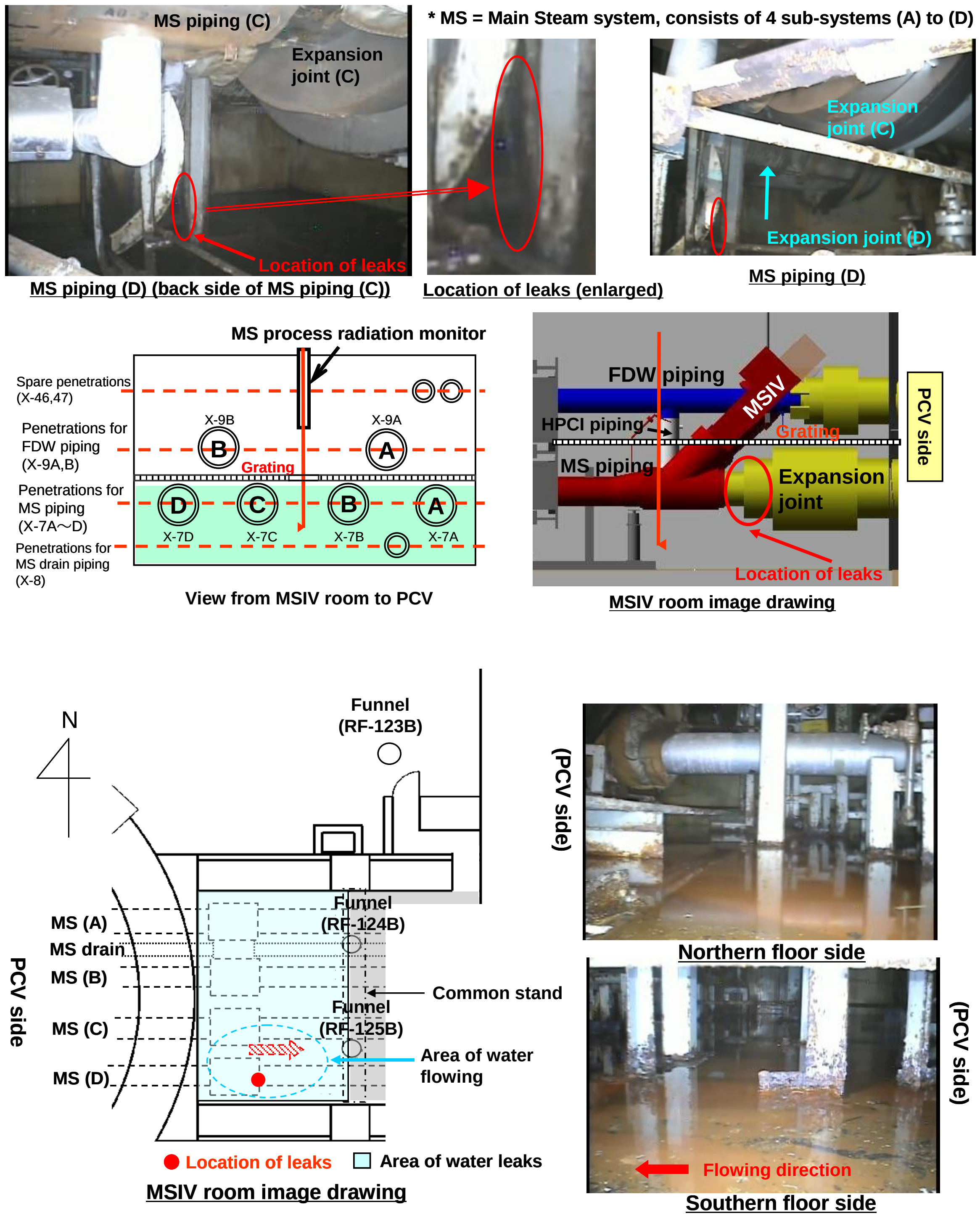


Figure 4-3 Water leaks from main steam line D in MSIV room

The MSIV room was confirmed to be difficult to enter due to rubble, and the 2014 investigation confirmed that the dose rate at the entrance was over 100mSv/h. The leakage of water contained in the PCV had spread outside the MSIV room as of February 2018, but by June 2019, the leakage was contained inside the MSIV room. The leakage was stopped and there were traces of water outside the MSIV room, and the dose rate at the entrance of the MSIV room was confirmed to be about 12 to 20mSv/h (Figure 4-4).

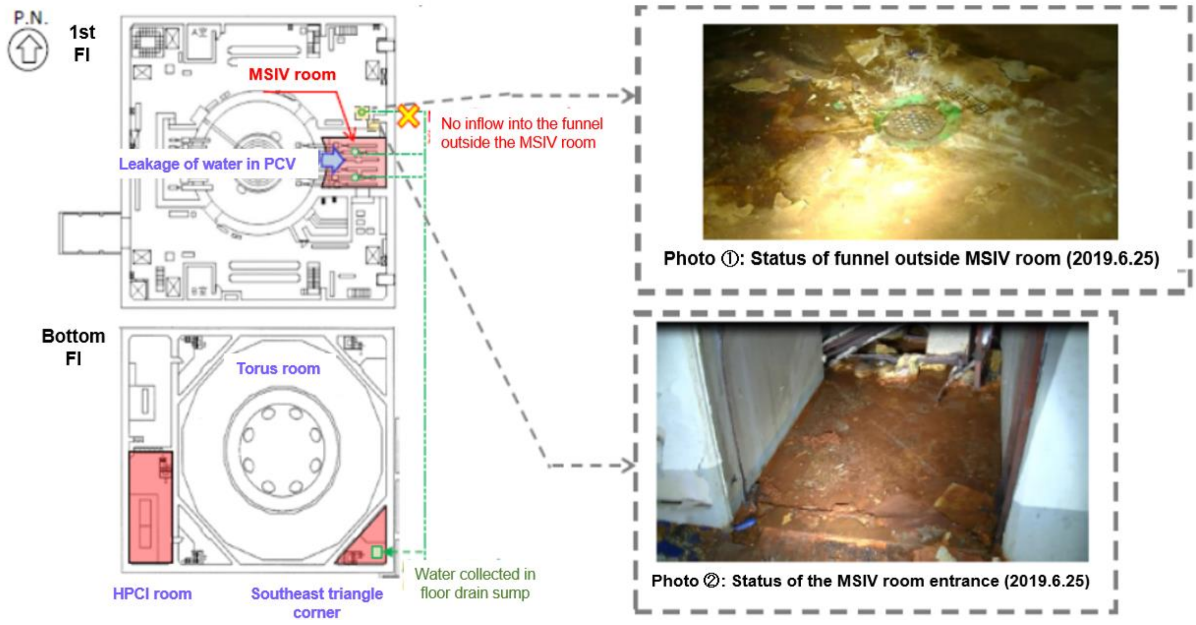


Figure 4-4 Water traces outside the MSIV room in Unit-3 [4-5]

In addition, investigation of the intermediate basement floor was conducted to confirm the feasibility of installing a pump and other equipment for the PCV at the southeast corner of the Unit-3 reactor building. It was confirmed that there was space for installing a pump and temporary hoses at the stairs in the southeast triangular corner and by the ascending catwalk and the stairs in the torus room (Figure 4-5). The dose rate at the site ranged from about 11 to 30mSv/h.

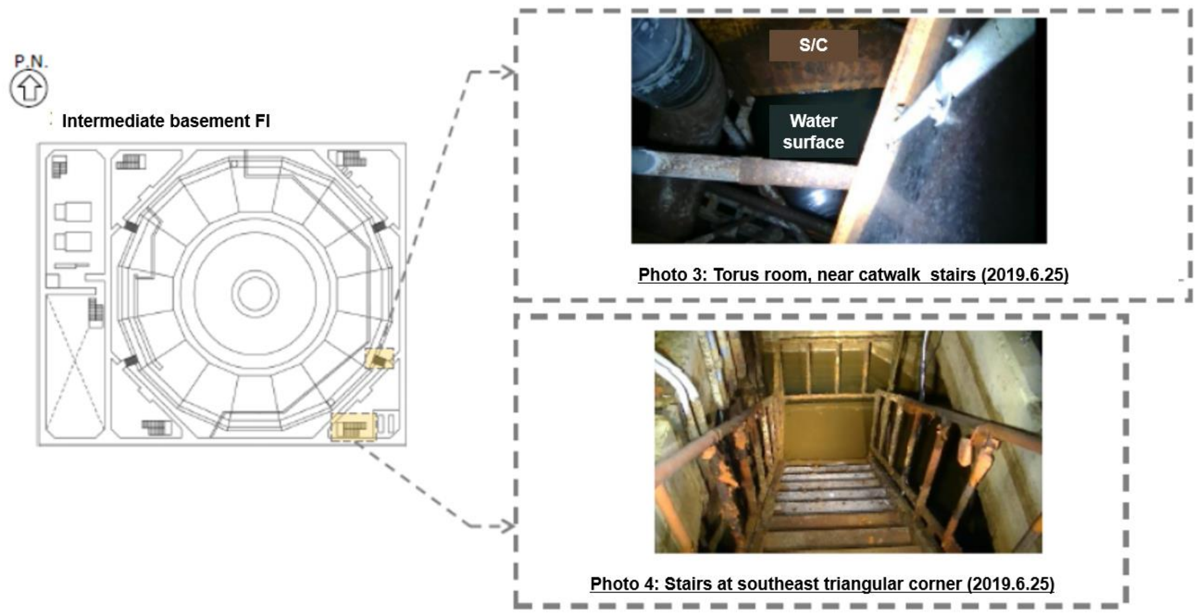


Figure 4-5 Site investigation of the intermediate basement floor of Unit-3 [4-5]

In April 2021, the MSIV room was investigated again, and it was confirmed that the leakage from

[4-5] Handout document: Progress status of the stagnant water treatment in the building, 73rd Specific Nuclear Facility Supervising and Evaluation Committee Meeting, July 22, 2019.

the joint of the main steam piping D, which was confirmed during the investigation in May 2014, was no longer present (Figure 4-6). On the other hand, although the joint of the main steam piping A was in the blind spot of the camera and could not be observed, it was confirmed that there was a possibility of leakage due to fluctuation of the water surface below the piping (Figure 4-7). The main steam piping A was located far from the camera and was not clearly imaged in the previous investigation, but it was imaged in this investigation because of the better camera performance. Main steam pipings A and D were installed at the same location (height), but the location (height) of the leak might be different. Since the current reactor containment vessel (PCV) water level is lower than in the previous investigation, it was thought that the leak on the piping D side has stopped while the leak on the piping A side is lower than the PCV water level, and therefore continuing.[4-6]

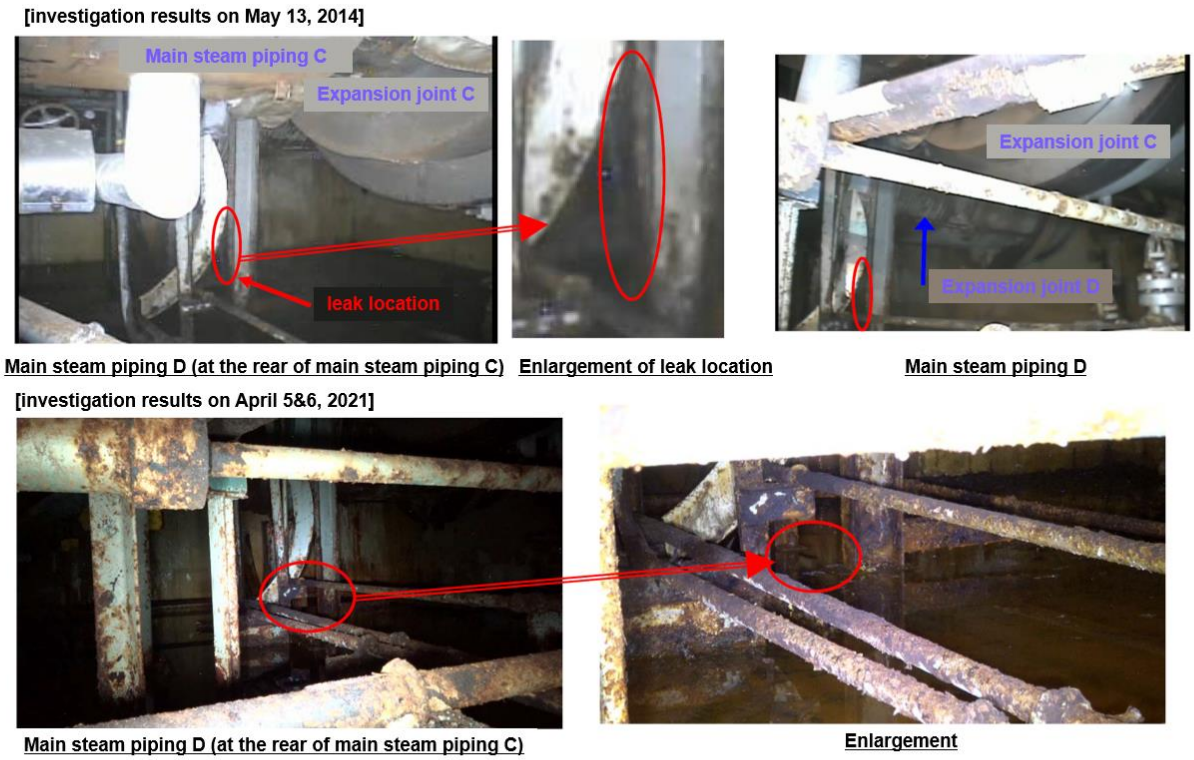


Figure 4-6 Investigation result of main steam piping D [4-6]

[4-6] Unit-3 reactor water injection stop test results (preliminary report) and status inside MSIV room during reactor water injection stop test, TEPCO Holdings, April 7, 2021

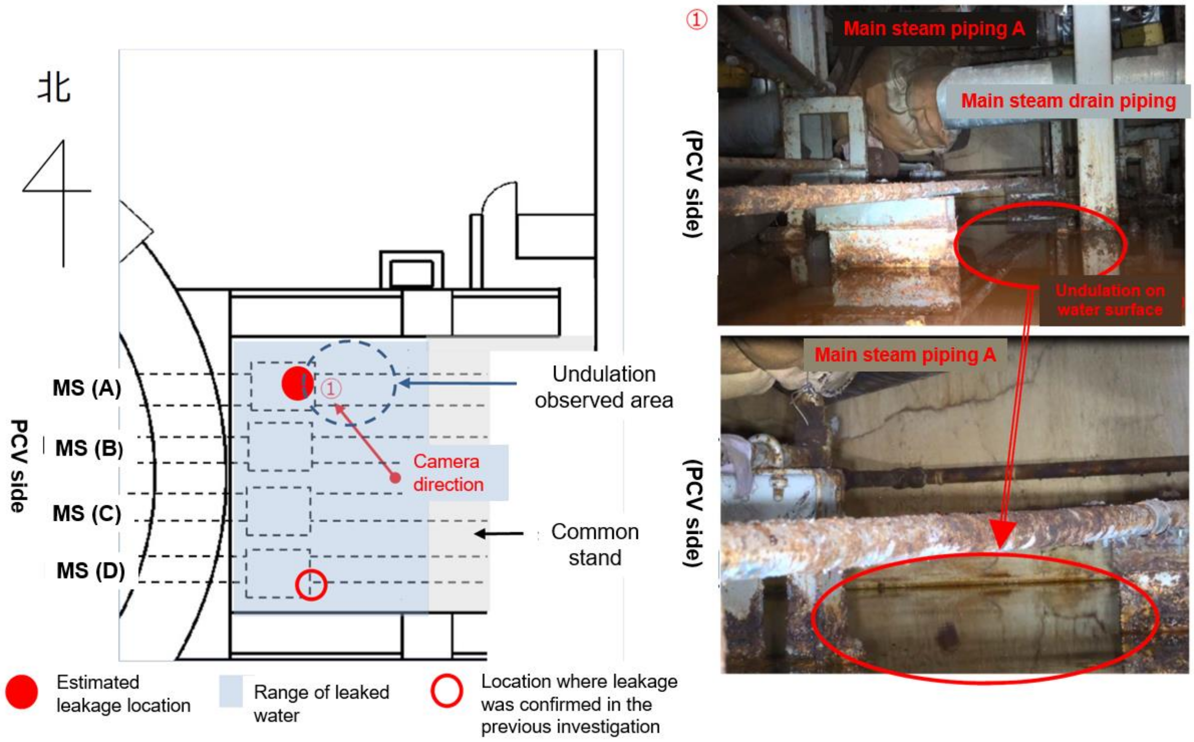


Figure 4-7 Investigation result of main steam piping A (obtained in April 2021) [4-6]

In June 2022, an investigation around the expansion joint of the main steam piping A was conducted again for the purpose of identifying the leakage point around this joint. It was confirmed that no water was dripping from the joint or from the main steam drain piping in the vicinity of the joint, and that there was no water puddle on the floor below the joint (Figure 4-8). This investigation did not lead to identification of the leakage point around the joint.

Since the calculated PCV water level has continued to decline slowly as of June 2022, it is assumed that some leakage points exist at a lower location than the PCV water level, in addition to the leakage points near the joints in question.

Furthermore, based on the results of the water injection stop test conducted in June 2022 and the PCV water level declining trend, it was estimated that the leakage point is about 4.2m (T.P. 8264) from the bottom of the PCV (Figure 4-9).

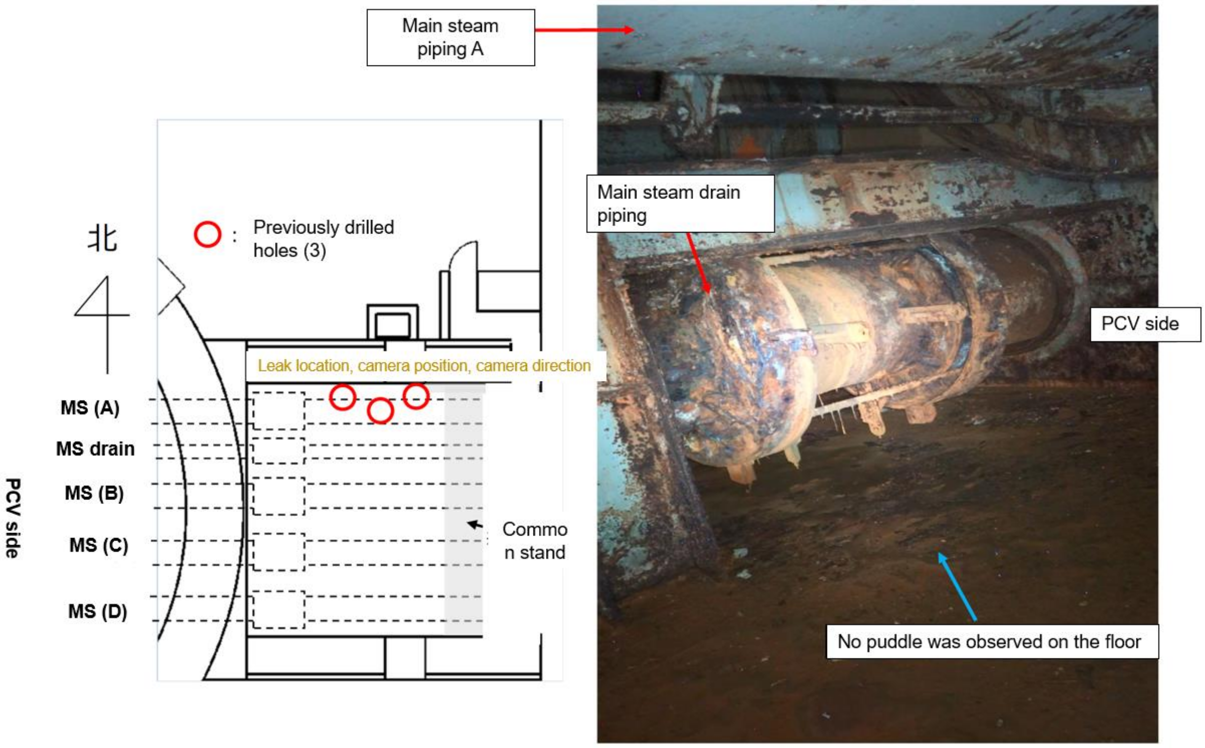


Figure 4-8 Investigation results of main steam piping A (obtained in June 2022) [4-7]

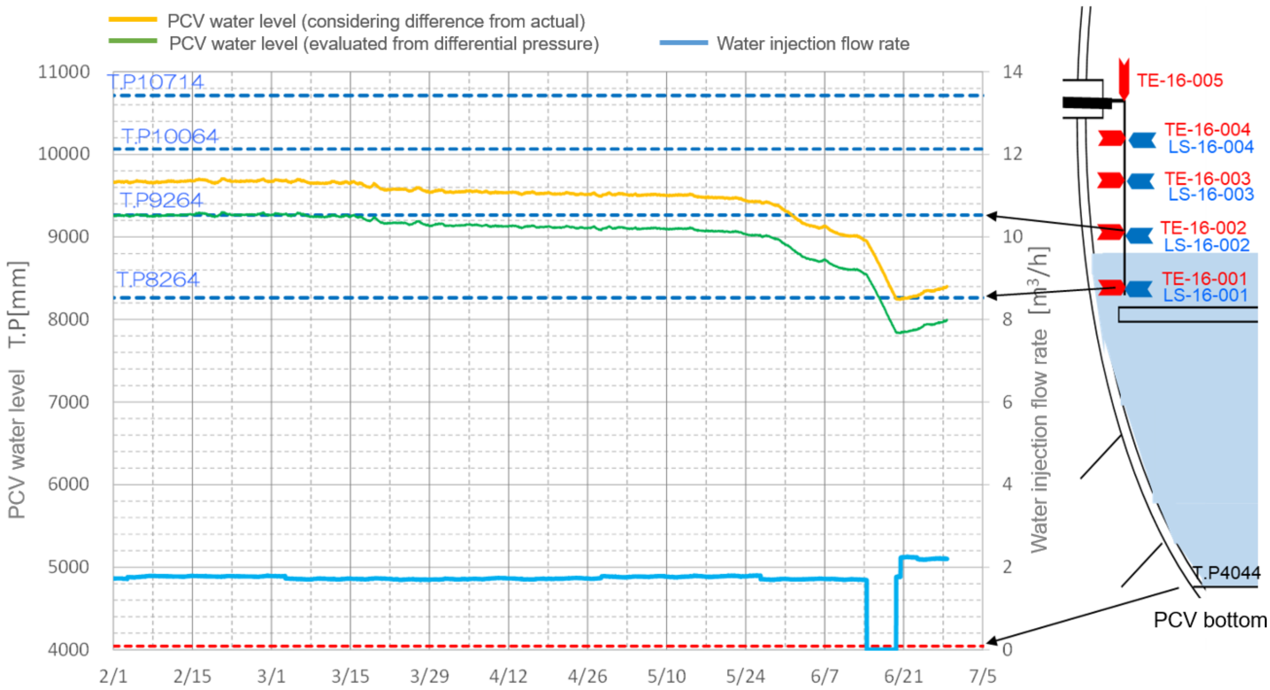


Figure 4-9 Change in PCV water level due water injection halting test (long-term trend) [4-8]

[4-7] Investigation results of the MSIV room of Fukushima Daiichi Nuclear Power Station Unit-3, TEPCO Holdings, June 10, 2022
[4-8] Handout document: Unit 3 reactor water injection stop test, handout for the 103rd secretariat meeting of the Decommissioning, Contaminated Water, and Treated Water Response Team Joint Meeting, June 30, 2022.

(4) Investigation of PCV equipment hatch of Unit-3

In order to locate the leak path from the PCV, the PCV equipment hatch on the R/B ground floor was investigated on September 9, 2015. In this area of the equipment hatch, it has been known from the earlier investigation (in 2011) that the shield plug, which was to be located in front of the equipment hatch during normal operation and be placed forward for maintenance during outage, was unexpectedly moved onto the rails and high dose rate water was confirmed to have been left in the ditches for the shield plug rails and in their vicinities. A possibility was pointed out therefore that the water retained in the PCV had leaked out through the equipment hatch seals.

In the current investigation, a small camera was inserted through the space between the shield plug and the wall originally surrounding the shield plug and the equipment hatch conditions were surveyed. Figure 4-10 presents photos taken of the hatch and around it. No water leaks from the hatch were recognized and no deformation of the hatch itself was recognized, either. Furthermore, no damage was found on the materials for periodic inspections that had been stored in front of the equipment hatch. Rather, it was noticed that the surface coating on the equipment hatch had come off and fragments of coating and other things had been accumulated on the floor in front of the equipment hatch.

At the original location for the shield plug, water (raindrops or condensed dew) was dripping from above and the floor seemed wet. Water puddles were recognized in the ditches for the shield plug travelling rails.

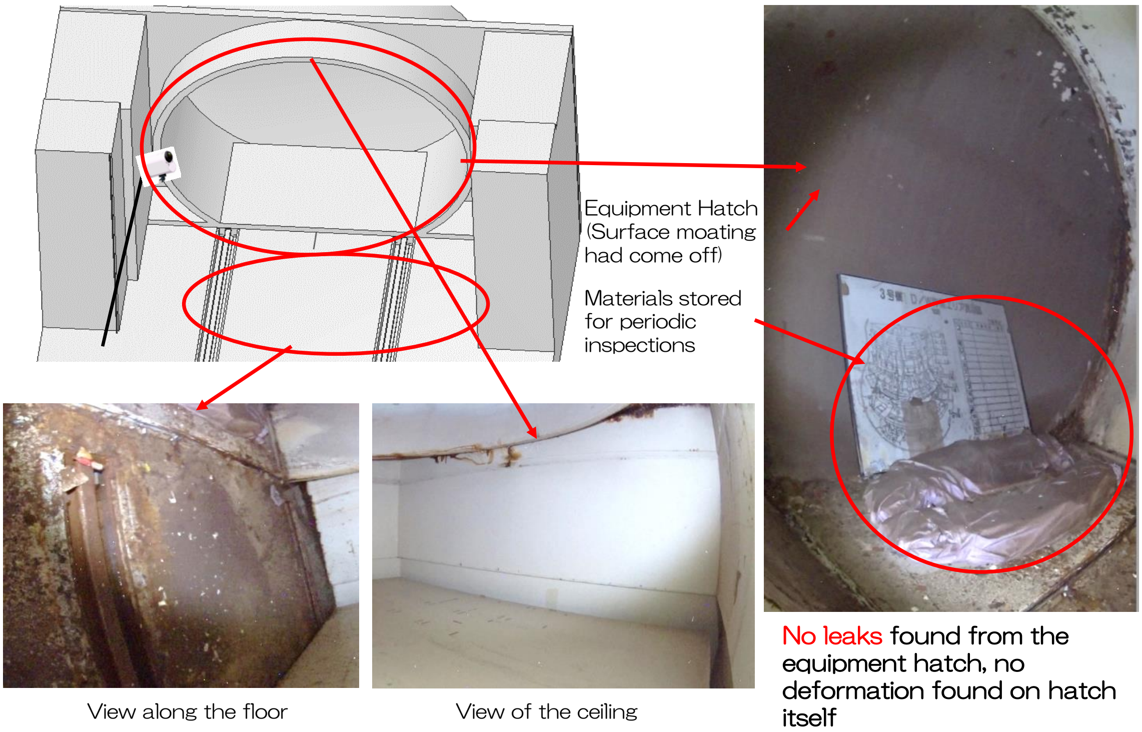


Figure 4-10 Photos of the equipment hatch[4-9]

[4-9] Handout document: Unit 3 PCV equipment hatch investigation results (using a small camera), 22nd secretariat meeting of the Decommissioning and Contaminated Water Response Team Joint Meeting, October 1, 2015.

(5) Investigation inside the PCV of Unit-3

The interior of the PCV of Unit-3 was investigated on October 20 and 22, 2015, by inserting an investigation probe through penetration X-53 to take photos, check water levels and measure temperatures and dose rates. The retained water was sampled for water quality analysis.

Figure 4-11 shows photos of the penetration X-53 taken from the front. Structures like piping, a ladder and other things were visible, but no damage was recognized. Including the underwater photos, no damage was recognized in the PCV in the current investigation.

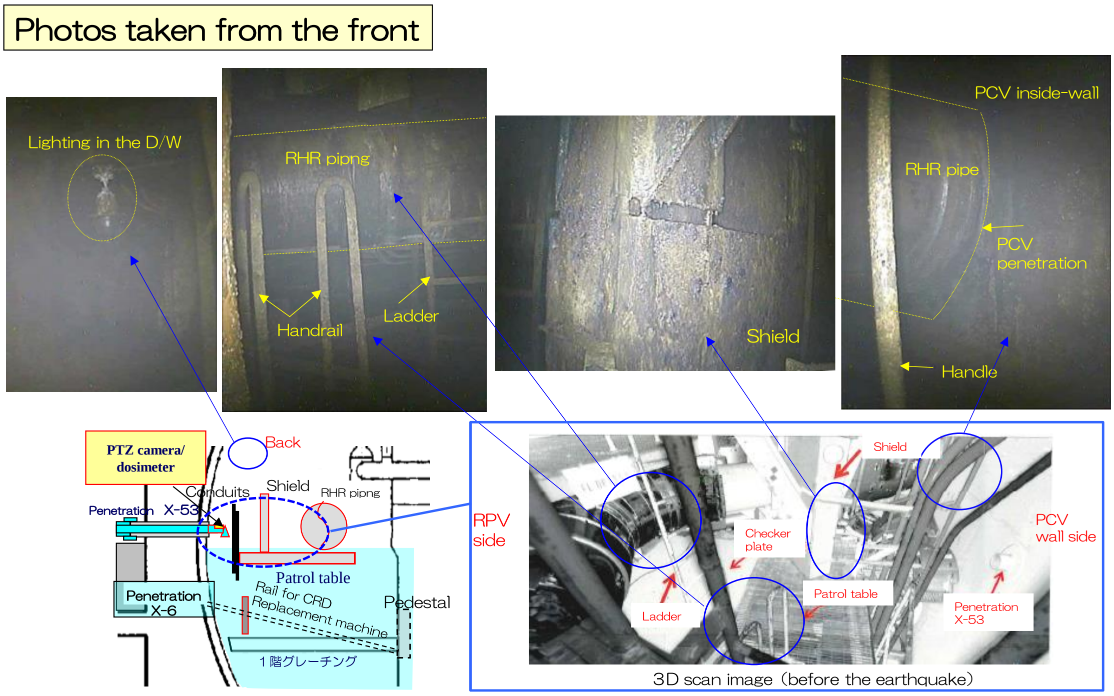


Figure 4-11 Photos of penetration X-53 taken from the front [4-10]

Figure 4-12 shows photos taken underwater by a PTZ camera. The camera was inserted through penetration X-53. Sediments were recognized on the grating and the rails for the CRD replacement machine.

[4-10] Handout document: Result of the investigation inside the reactor containment vessel (PCV) of Unit-3 at Fukushima Daiichi Nuclear Power Plant, 23rd Decommissioning and Contaminated Water Response Team Joint Meeting, October 29, 2015.

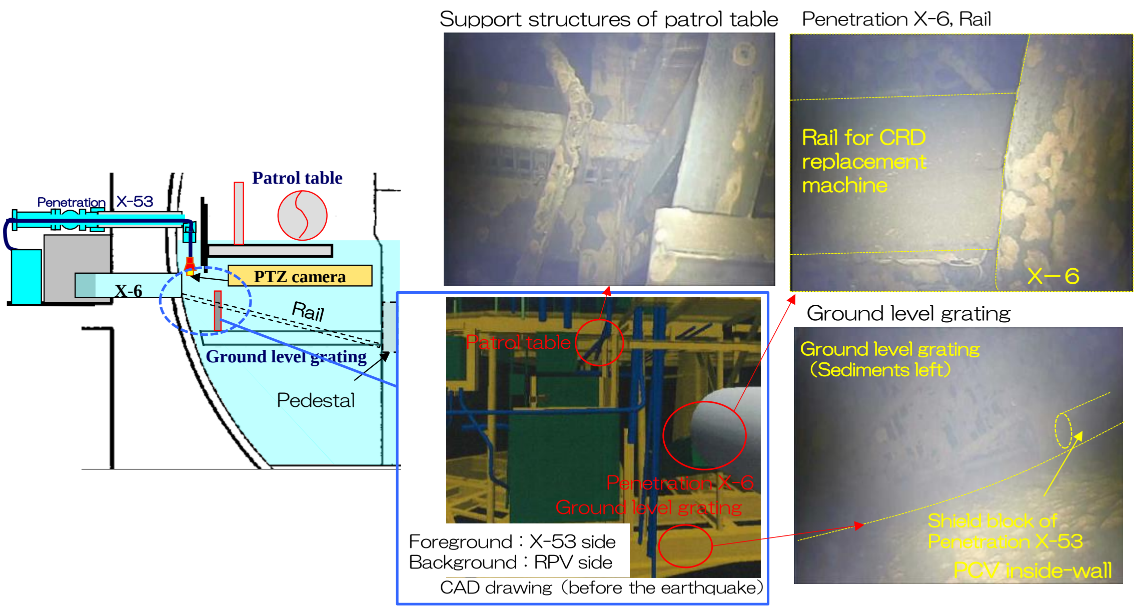


Figure 4-12 Photos taken in the water retained in the PCV [4-10]

The level of the water retained in the PCV was about OP11,800 or about 70cm below the penetration X-53. It was roughly in agreement with the value estimated from the PCV pressures. The temperatures in the PCV were about 26 to 27 ºC in the air and about 33 to 35ºC in the water. The air dose rate measured in the PCV was about 0.75Sv/h at a point about 55cm from the exit of penetration X-53, and 1Sv/h near the PCV inner wall.

The results of quality analysis of retained water that was sampled are given in Table 4-1. Samples were taken at two positions, one about 10cm below the surface and the other about 70 cm below it. The results showed that the water was not strongly corrosive. In the water, α-emitting nuclides were detected in addition to cesium and tritium.

Table 4-1 Results of water quality analysis of the water retained in the PCV [4-10]

Objective	Items of analysis (planned)		Near surface	About 70cm below surface	Evaluation
Corrosiveness of the environment	pH		6.8	6.3	Not strongly corrosive
	Conductivity (μS/cm)		14.0	10.2	
	Chlorine concentration (ppm)		Below detection limit (<1)	Below detection limit (<1)	
Radioactive materials released, nuclide migration behavior	γ-concentration (Bq/cm^3)	^{134}Cs	4.0E02	2.3E+02	
		^{137}Cs	1.6E+03	9.4E+02	
		^{131}I	Below detection limit (<8.1E+00)	Below detection limit (<5.3E+00)	
	Tritium concentration (Bq/cm^3)		2.7E+02	1.6E+02	
	^{89}Sr/^{90}Sr concentration (Bq/cm^3)		^{89}Sr: below detection limit (<8.4E+01) ^{90}Sr: 7.4E+03	^{89}Sr: below detection limit (<8.1E+01) ^{90}Sr: 3.9E+03	
	Total α concentration (Bq/cm^3)		2.1E+00*	9.7E+01*	

(6) Operation floor survey of Unit-3 reactor building

Gamma-ray spectra were measured, radiation source nuclides were identified, and source distribution was investigated in October 2015 on the operation floor of the Unit-3 reactor building.[4-11] The investigation was conducted to plan decontamination and shielding work, because high dose rates had remained despite decontamination by removing rubble, shaving or vacuum sucking. Collimators were used to measure gamma ray spectra only from below.

Figure 4-13 compares, as a typical example, the gamma ray spectra measured (blue points) at Location ⑭ in Figure 4-14 and the gamma ray spectra of a reference Cs-137 source (red points) obtained at the calibration facility. Relatively higher counting rates are observed in the Compton region in addition to the photo peaks due to Cs-134 and Cs-137 in the measured spectra for the operating floor. When the heights of photo peaks and Compton peaks are compared, the contribution of Compton effects is relatively higher in the measured spectra for the operating floor than in the spectra measured at the calibration facility using reference sources. Similar trends have been confirmed at the measurement locations of the operating floor other than ⑭. Higher contributions of Compton effect mean higher contributions of scattered gamma rays. Therefore, it is possible that higher contributions to the dose rates for the operating floor came from scattered

[4-11] Handout document: Results of γ-ray spectrum measurements at the Unit-3 reactor building operating floor 24th Decommissioning and Contaminated Water Response Team Joint Meeting, October 29, 2015.

gamma rays from the source deep under the floor surface.

Figure 4-14 shows the distribution of dose rates (relative) evaluated from the measured gamma spectra. Higher dose rates were observed at the periphery of the reactor well-cover (⑥, ⑨, ⑪, ⑱, ⑳) and at the slab interfaces (⑫, ⑯).

It should be noted that the gamma ray spectra measurement and scattered gamma simulation results [4-12] obtained by the Nuclear Regulation Authority had indicated the possibility of a high cesium concentration having been deposited on the back face of the shield plug.

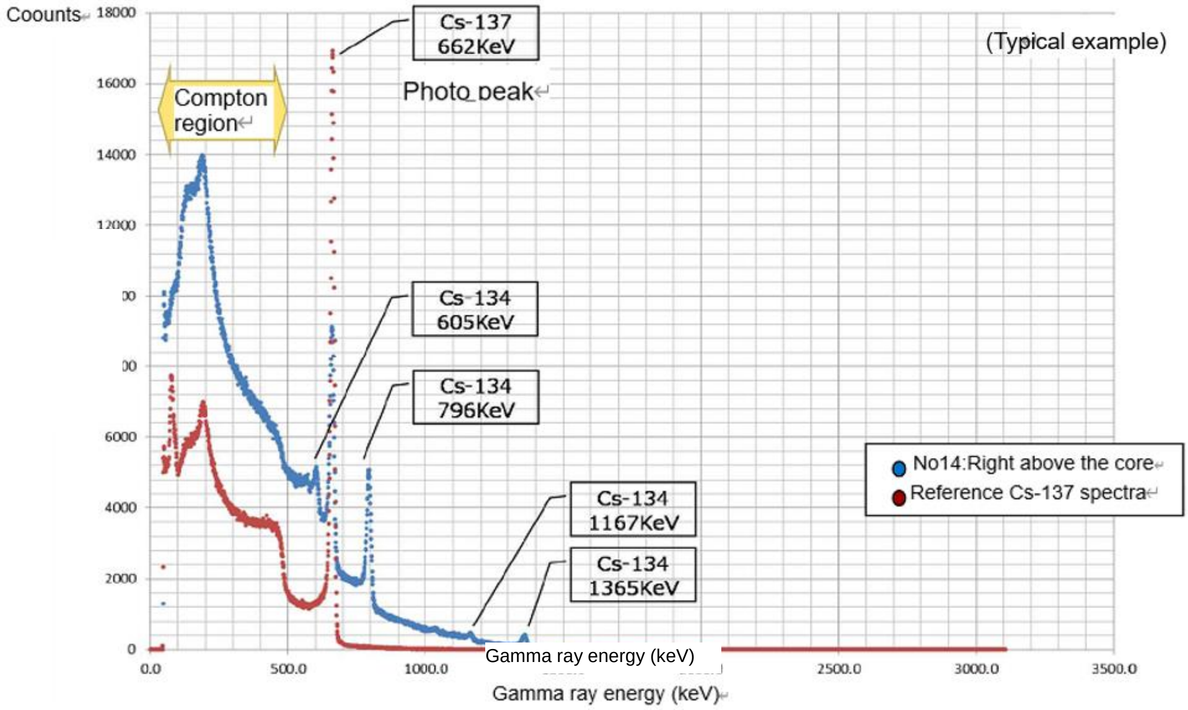


Figure 4-13 Gamma ray spectra measured (right above the core) [4-11]

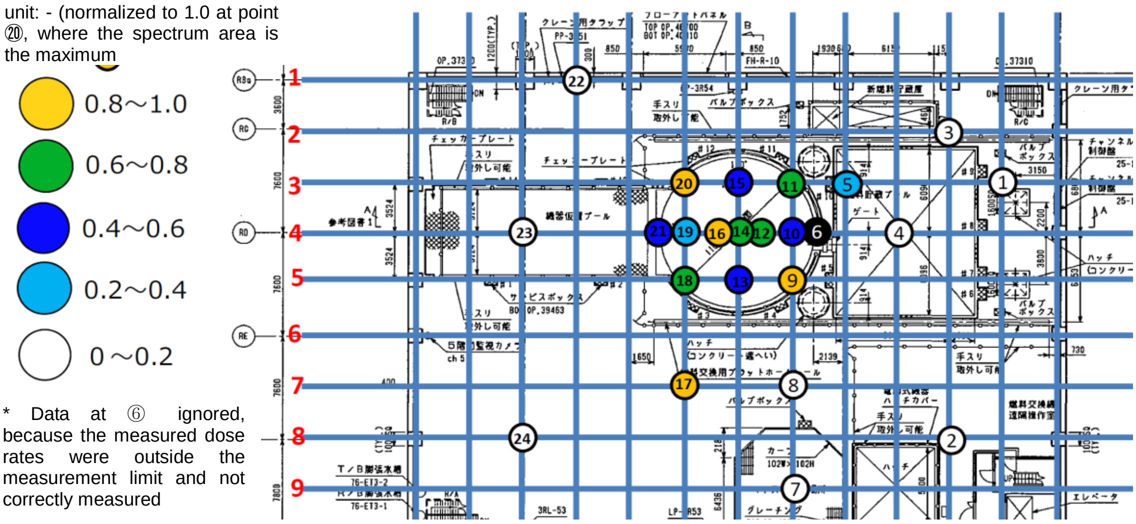


Figure 4-14　Dose rate distribution ratios evaluated from the gamma-ray total spectrum (Normalized to 1.0 at location ⑳) [4-11]

[4-12] Handout document: Investigation of radiation sources on operating floor of the Unit 3 reactor building (Preliminary report), 38th Specific Nuclear Facility Supervising and Evaluation Committee Meeting, December 18, 2015.

(7) Investigation inside the PCV of Unit-3

(Investigation of conditions inside the pedestal using an underwater ROV) [UPDATE]

The inside ot the Unit-3 pedestal was photo-surveyed in July 2017 by an underwater remotely operated vehicle (underwater ROV). Figure 4-15 shows a schematic diagram of the investigation. The underwater ROV was sent through a PCV penetration (X-53) to survey inside the PCV where it moved through the water taking photos and video films.

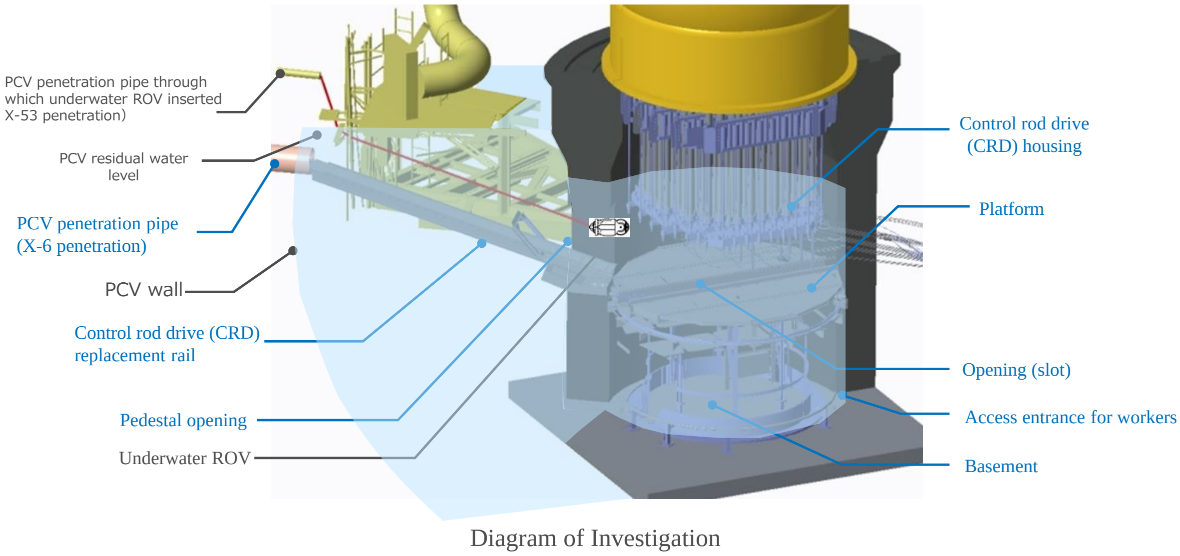


Figure 4-15 Schematic diagram for investigation inside the Unit-3 PCV [4-13]

The video camera filmed the CRD housing from below in Areas A1 and A2 inside the pedestal shown in Figure 4-16. Figure 4-17 and Figure 4-18 show photos taken. Figure 4-17 confirmed the CRD flanges were disarrayed; they should have been in place at uniform intervals and with the same elevation. Figure 4-18 confirmed an object looking like a control rod (CR) guide tube, which is part of the core internals and should be in the RPV, was now outside the RPV. Near the CRD housing damaged or fallen-off CRD housing support brackets were confirmed, and solidified molten objects were stuck to CRD flanges or other pieces. In Figure 4-17, ripples on the water surface, which could be photgraphed when looking from below toward the water surface, could be confirmed. This indicated a possibility that water was dripping from above. Similar water surface ripples could be confirmed near the pedestal inner wall, too. This indicated a possibility that, besides an opening at the center of RPV bottom head, another opening might exist in the RPV peripheral region, too.

[4-13] Handout document: Investigation inside the reactor containment vessel of Unit 3, 48th Decommissioning and Contaminated Water Response Team Joint Meeting, November 30, 2017.

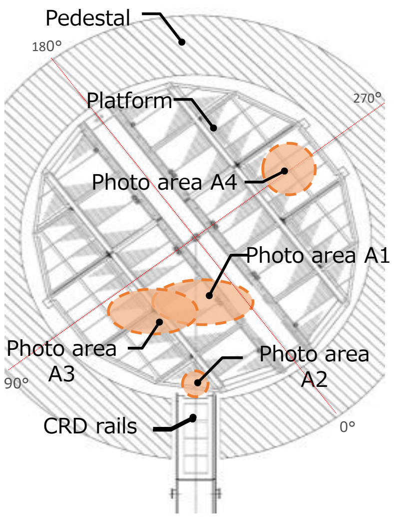


Figure 4-16 Photo area near CRD housing [4-13]

Photo area A1 (camera direction, upward)

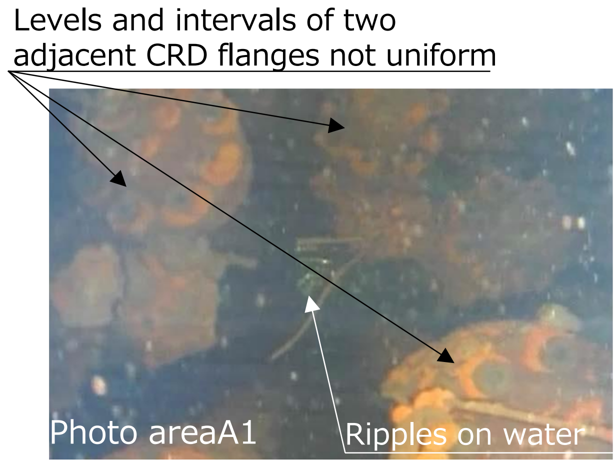


Figure 4-17 Photo [4-13]

(CRD flange photo taken near CRD housing, photo area A1)

Photo area A3 (camera direction, upward)

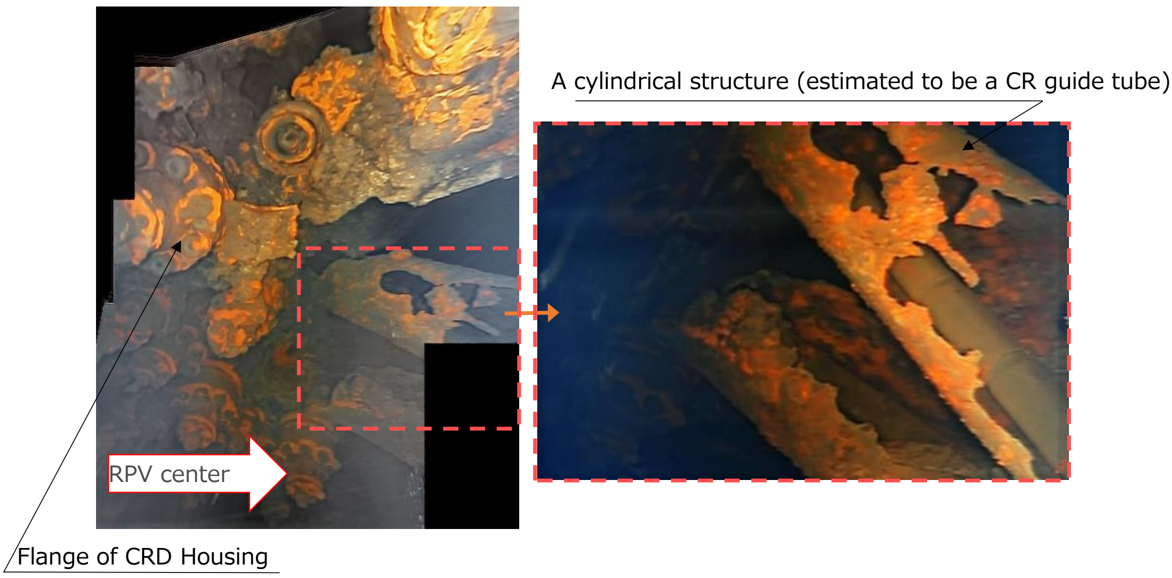


Figure 4-18 Photos

(A structure, apparently a CR guide tube; photo taken near CRD housing in photo area A3) [4-13]

Figure 4-19 and Figure 4-20 are photos of the pedestal inner wall. On part of the inner wall, the epoxy-based coating was stripped off and the wall surface was rough, but no big pedestal damage or deformation was confirmed as a whole (Figure 4-19). Part of the cables for the RPV bottom thermometers laid on the pedestal inner wall was found to be missing (Figure 4-20). It is estimated that high-temperature molten debris deposited on the cables when falling down.

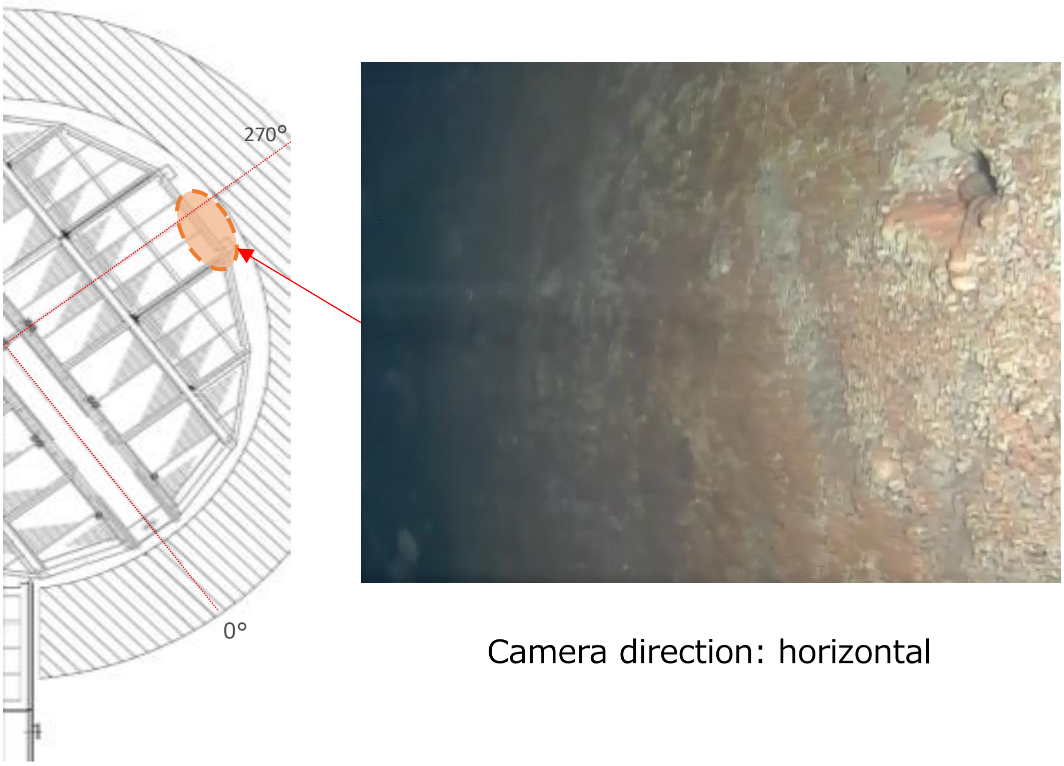


Figure 4-19　Photo (pedestal inner wall) [4-13]

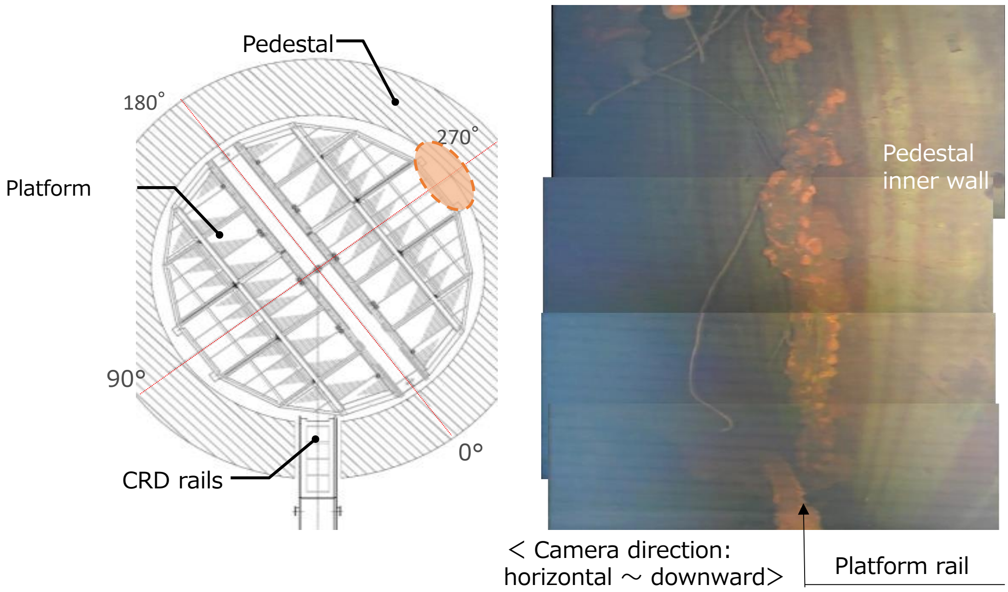


Figure 4-20　Photo (cables on the pedestal inner wall) [4-13]

Figures 4-21 and 4-22 are photos taken at a lower part inside the pedestal. Although not clearly identifiable, fallen objects looking like upper tie-plates of fuel assemblies or fuel support fitting plugs (Figure 4-21), or a control rod falling speed limiter (Figure 4-22) were found. In the pedestal lower part, pebble-like or lump sediments were confirmed besides sand-like sediments (Figure 4-22). The opening in the pedestal basement for human access was not checked visually, but sediments were confirmed to be present in the vicinity.

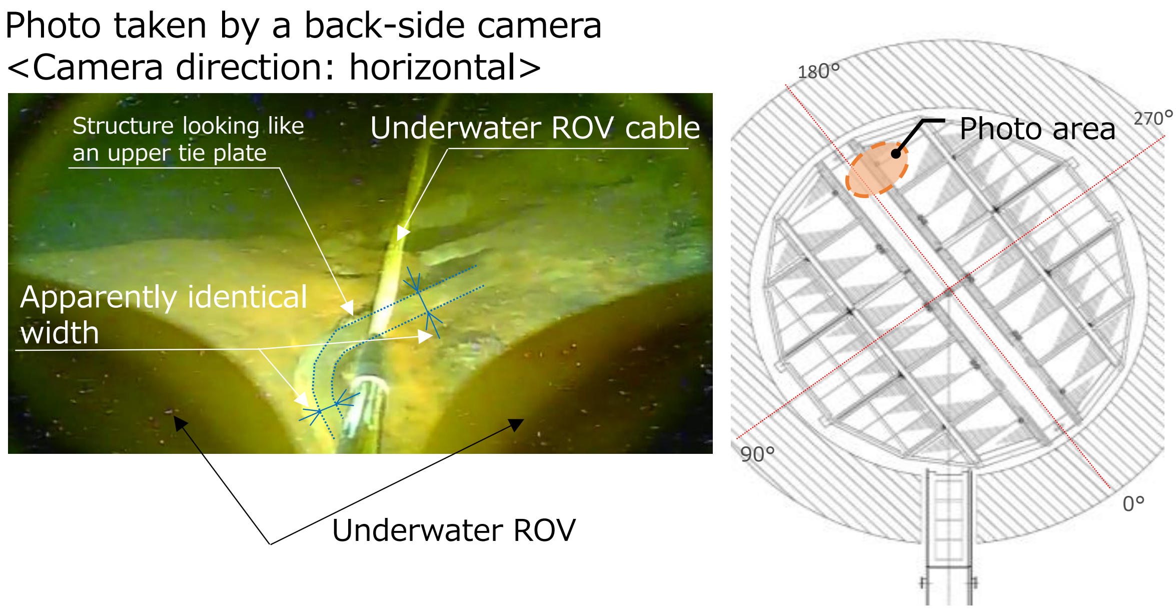


Figure 4-21 Photo

(A structure, possibly an upper tie-plate observed inside the lower pedestal) [4-13]

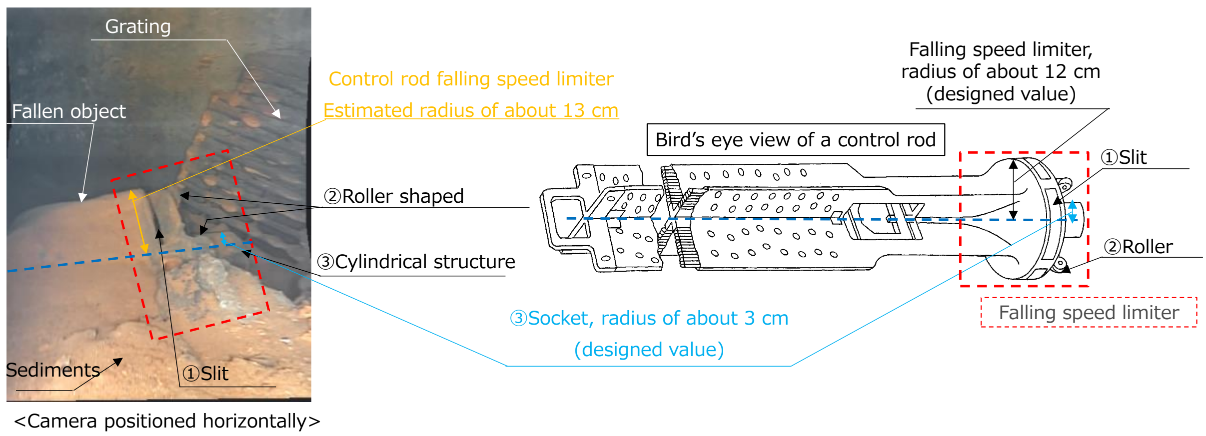
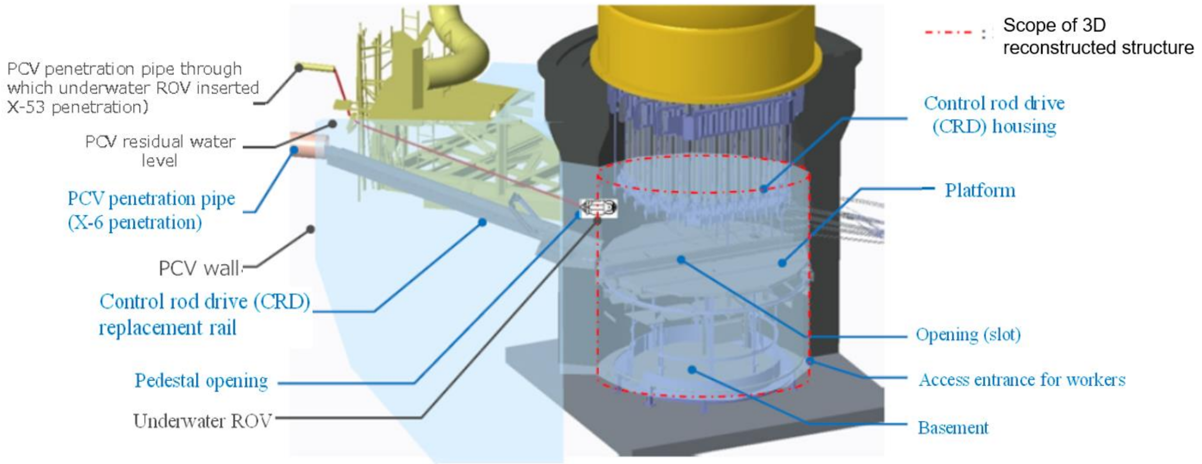


Figure 4-22 Photo

(A fallen object, possibly a control rod falling speed limiter observed inside the lower pedestal, Area C2) and schematic drawing of a control rod [4-13]

In addition, the entire pedestal was reconstructed in 3D using video images obtained from an underwater ROV investigation of the inside of the pedestal conducted in July 2017. For the areas not obtained by the internal investigation, 3D reconstruction was done from the design information of the structure. However, objects that could not be identified due to a short viewing time span, obscurity, or being only partial images, as well as objects whose positions could not be determined, were not reconstructed.The 3D restoration area is shown in Figure 4-23.



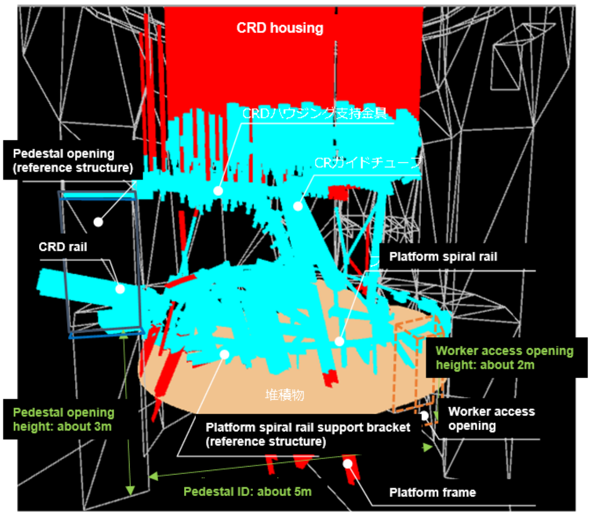
Scope of 3D reconstructed structure

Figure 4-23 3D reconstructed area from underwater ROV investigation [4-14]

In Figure 4-24, light blue represents structures that were confirmed in the video and red represents structures that were not confirmed. In Figure 4-25, color is assigned for each estimated structure. Since the pedestal openings and some of the swivel rail support brackets did not show significant damage, these structures were considered to be in the same position as they were before the accident and they were used as the basis for the 3D reconstructed positions. The structures were reconstructed to obtain an overall view of the situation in the pedestal, and their locations are approximate. Figure 4-26 was reconstructed by estimating the sediment height from the surrounding structures, and most of the sediments that could not be confirmed in the video images were reconstructed by interpolating from the estimated heights of the sediments that were confirmed.

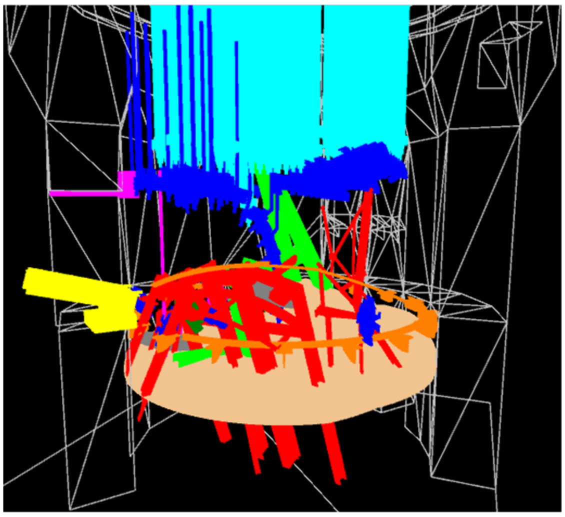
A summary of the information from the reconstructed data is shown in Figure 4-27. The reconstructed sediment height is highest at the center, and considering that the platform has fallen down and the CRD exchanger was not confirmed, it may be higher due to molten material falling on top of the CRD exchanger, which may contain fuel debris.

[4-14] Handout document: 3D reconstruction results from the investigation video inside the reactor containment vessel of Fukushima Daiichi Nuclear Power Plant Unit 3, 53rd Decommissioning and Contaminated Water Response Team Joint Meeting, April 26, 2018.



: Structures whose images were confirmed

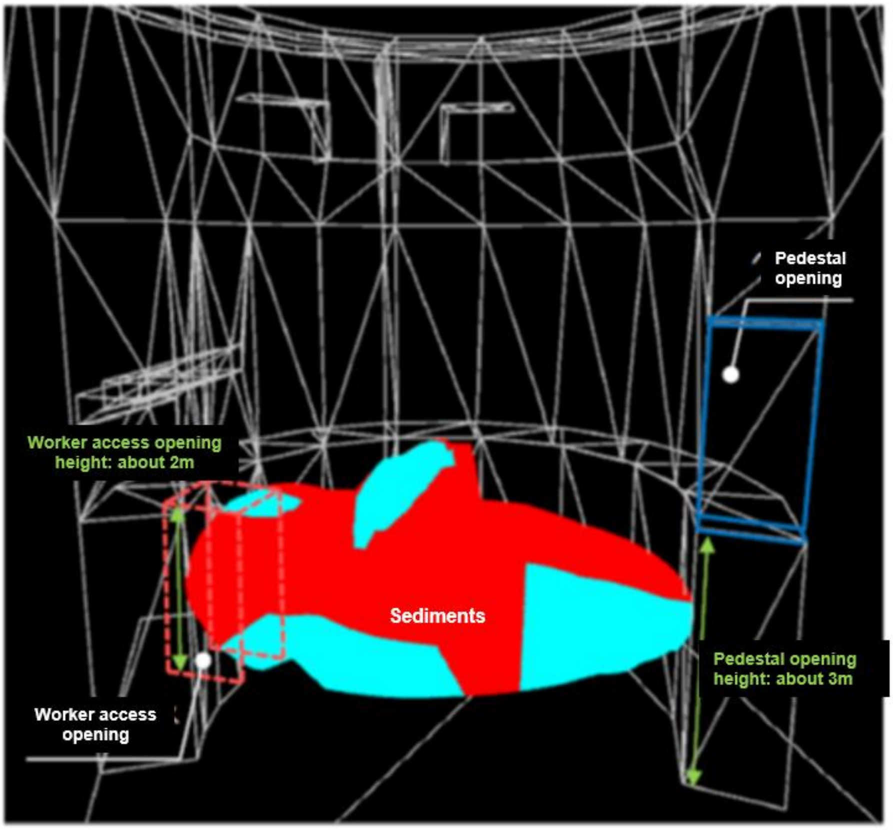
: Structures whose images could not be confirmed (estimated from design information)



Structures such as platforms and racks
Platform spiral motor
Control rod housing support bracket
Control rod housing
Control rod guide tube
Control rod drive mechanism changing rail
Platform spiral rail, support bracket
Terminal box, conduit, piping
Grating

Figure 4-24 Reconstucted figure from structure [4-14]

Figure 4-25 Structure distribution map [4-14]



: Range whose images were confirmed

: Range whose images could not be confirmed (estimated from design information)

Figure 4-26 Estimation figure of sediment height [4-14]

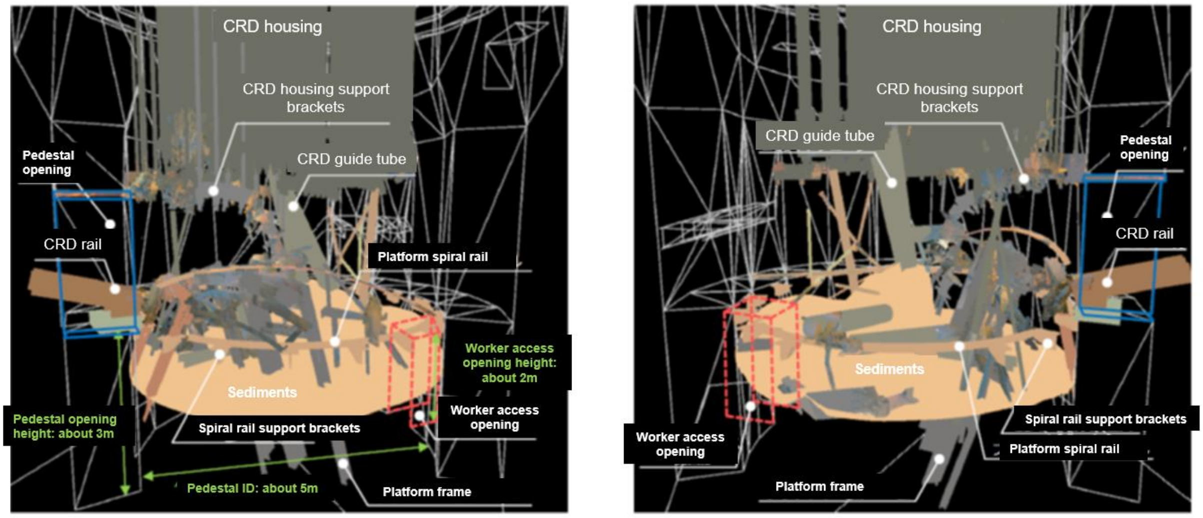


Figure 4-27 3D reconstruction figure of the inside of the pedestal [4-14]

(8) Investigation of Unit-3 with the muon device

An investigation with the muon device was conducted for Unit-3 in May to September 2017 following the investigations for Unit-1 and Unit-2. Figure 4-28 shows the location where the muon device was set up. Figure 4-29 compares the distribution of material quantities (density length) obtained from a simulation based on design and those obtained from muon measurements. Concrete in the PCV peripherals, spent fuel pool and reactor building walls could be identified in the distribution of material quantities from the muon measurements.

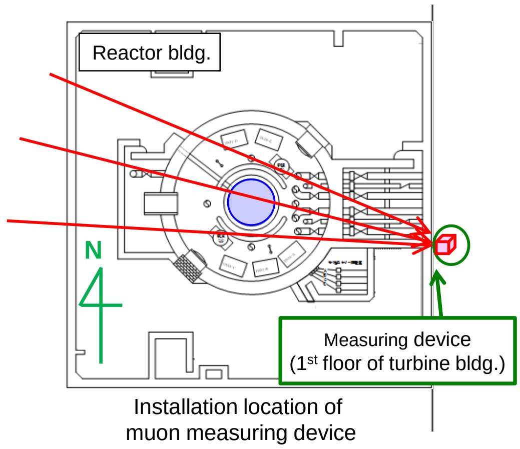
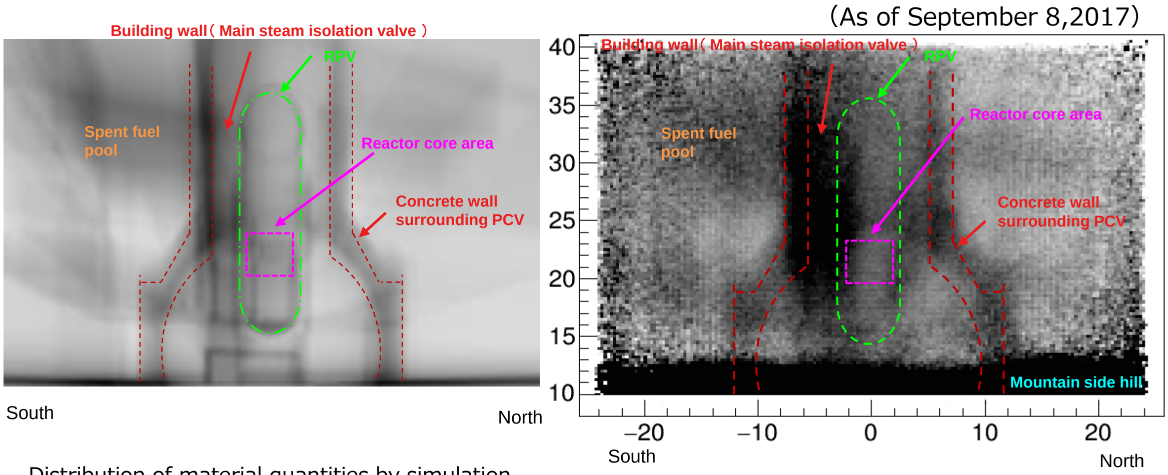


Figure 4-28 Location of muon device [4-15]

[4-15] Handout document: Determination of the location of fuel debris inside reactor by muon measurement in Fukushima Daiichi Nuclear Power Station Unit-3, 46th Decommissioning and Contaminated Water Response Team Joint Meeting, September 28, 2017.



Distribution of material quantities by simulation (fuel debris assumed in core and RPV bottom)

Materials quantities (density length) by muon measurements

Figure 4-29 Distribution of material quantities from simulation (left) and muon measurement (right) [4-15]

The distribution of material quantities in the RPV was derived from the measured results by eliminating the influences from structures like R/B walls and the PCV based on the simulation model. The results are given in Figure 4-30 (left). These were compared with the simulation results, in which high density materials (3g/cm^3, 1g/cm^3, and in addition 5g/cm^3 at the PCV bottom) were assumed to be present in the RPV or not present. The comparison led to the estimation of the presence of fuel materials at different RPV elevations (Figure 4-30 (right)). Materials of about 1g/cm^3 or below were present at the core region, far lower than the normal average densities in the core (about 3g/cm^3). On the other hand, more material quantities were identified than normal in some part of the bottom of the RPV. Quantitative evaluation results of material quantities are given in Figure 4-31 for respective positions in the RPV. By considering the big reduction of material quantities in the core region as compared with those before the accident, most of the fuel can be estimated to have been relocated downward together with structural materials. No big fuel debris is considered to be left in the core region. To the contrary, the material quantities at the RPV bottom increased from the quantities before the accident. This is an indication, with some uncertainties being admitted, that part of the fuel debris remains at the RPV bottom.

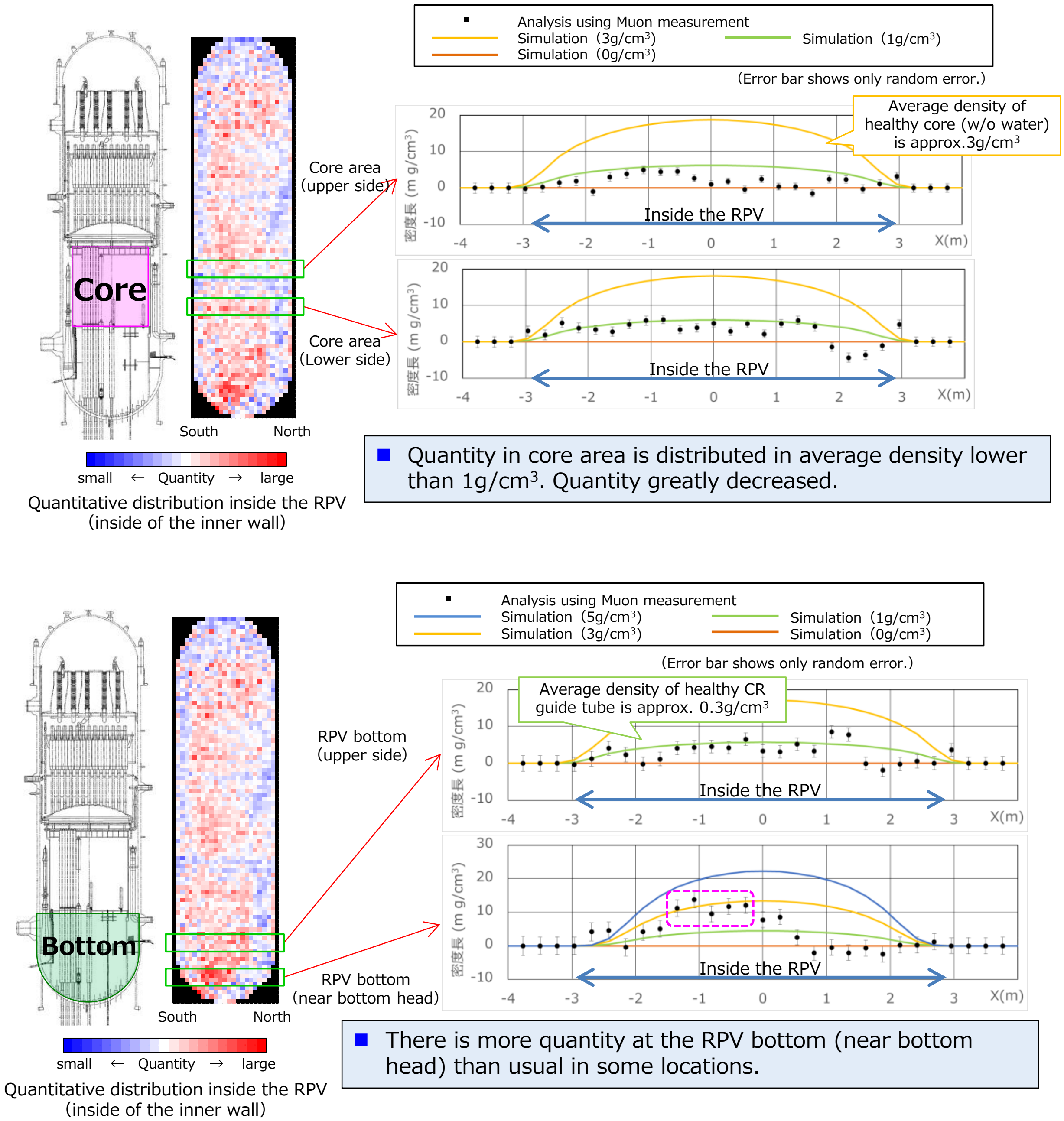
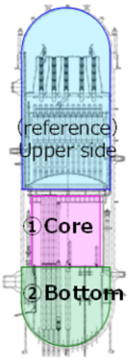


Figure 4-30 Distribution of material quantities in the RPV

(Top, core region; bottom, RPV bottom) [4-15]



<Result>

(As of September 8, 2017)

	quantity [ton]	Error [ton] Random error	Error [ton] Systematic error	(Reference) quantity before the accident[ton]
(Reference)Upper side	Approx.120	±Approx.6	Tens of tons	Approx.80 (reactor internals)
① Core area	Approx.30	±Approx.3		Approx.160 (fuel assembly) Approx.15 (control rod) Approx.35 (reactor internals)
② Bottom of the RPV	Approx.90	±Approx.5		Approx.35 (reactor internals) Without effect of water

Figure 4-31 Quantitative evaluation of material quantities in the RPV [4-15]

(9) Investigation results of the Unit-3 SGTS room [UPDATE]

In September 2020, dose rate distribution measurements using γ imager in the SGTS room, dose rate measurements around rupture disk, and investigation inside filter train were conducted.

As a result of the γ imaging measurements, contamination was confirmed along the SGTS piping leading from the vent line to the filter train and also downstream from the filter train (Figure 4-32). Furthermore, contamination was confirmed along the vent line (Figure 4-33). These results clearly indicated that there was a backflow of vent gas into Unit-3 itself.

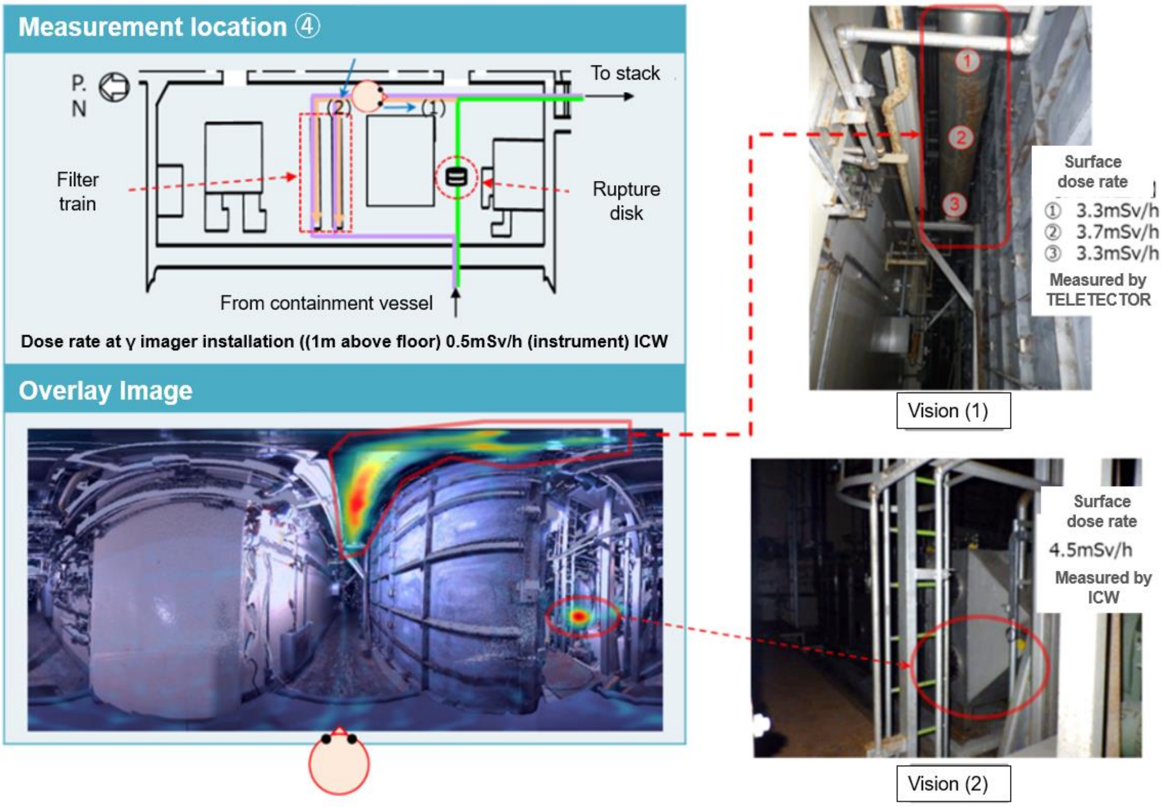


Figure 4- 32 Results of γ imager measurements in the SGTS room (Room No.1) [4-16]

[4-16] Handout document: Results of the SGTS room investigation for Units 1-4, 88th Decommissioning and Contaminated Water Response Team Joint Meeting, March 25, 2021.

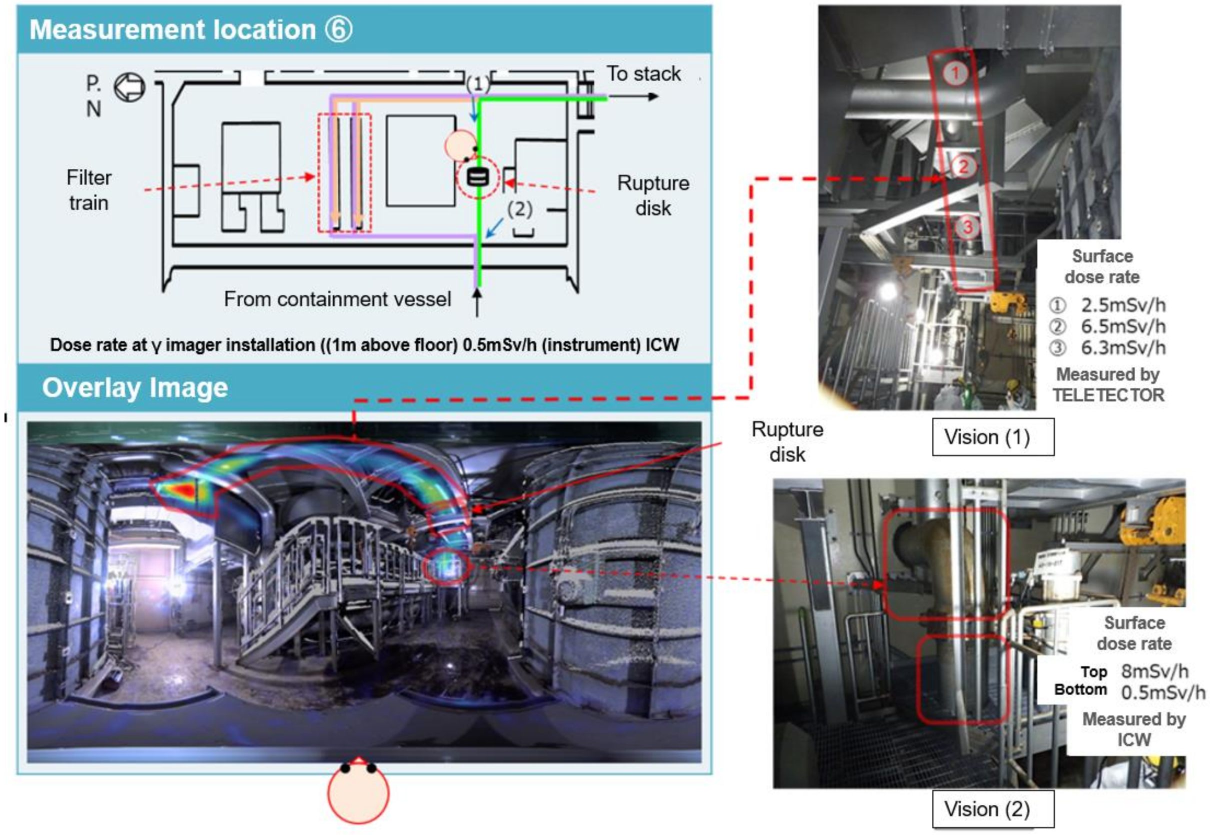
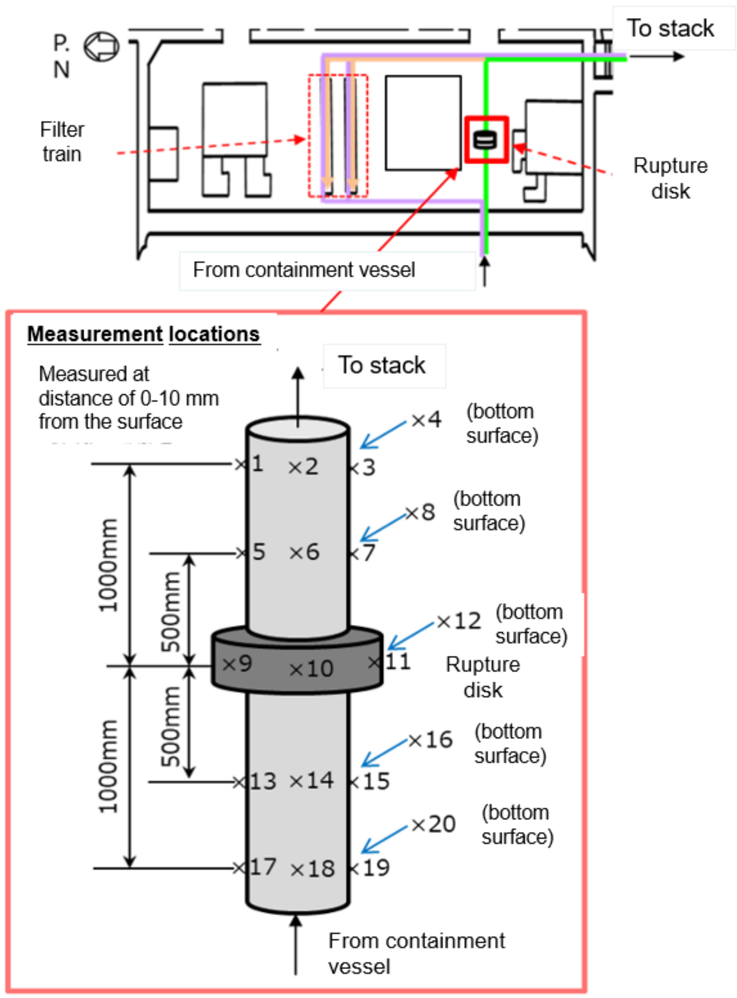


Figure 4-33 Results of γ imager measurements in the SGTS room (Room No.2) [4-16]

The dose rate measurements around the rupture disk showed that the measured dose rate increased in the order of "rupture disk **<** upstream from rupture disk **<** downstream from rupture disk" (Figure 4-34). This is significantly different from the fact that almost no contamination was observed around the rupture disk (closed due to non-operation) in Unit-2, where no venting was performed.

It was confirmed in an investigation conducted by the Nuclear Regulatory Agency in July 2019 that the dose rates were 2.5mSv/h downstream from the rupture disk, 8.0mSv/h on the surface of the rupture disk, and 5.5mSv/h upstream from the rupture disk.[4-17]

[4-17] Handout document: Outline of site investigation etc. - Contamination status at the reinforced pressure-resistant vent line -, 9th Meeting of the Study Group on Analysis of Accident at TEPCO's Fukushima Daiichi Nuclear Power Station, December 26, 2019.

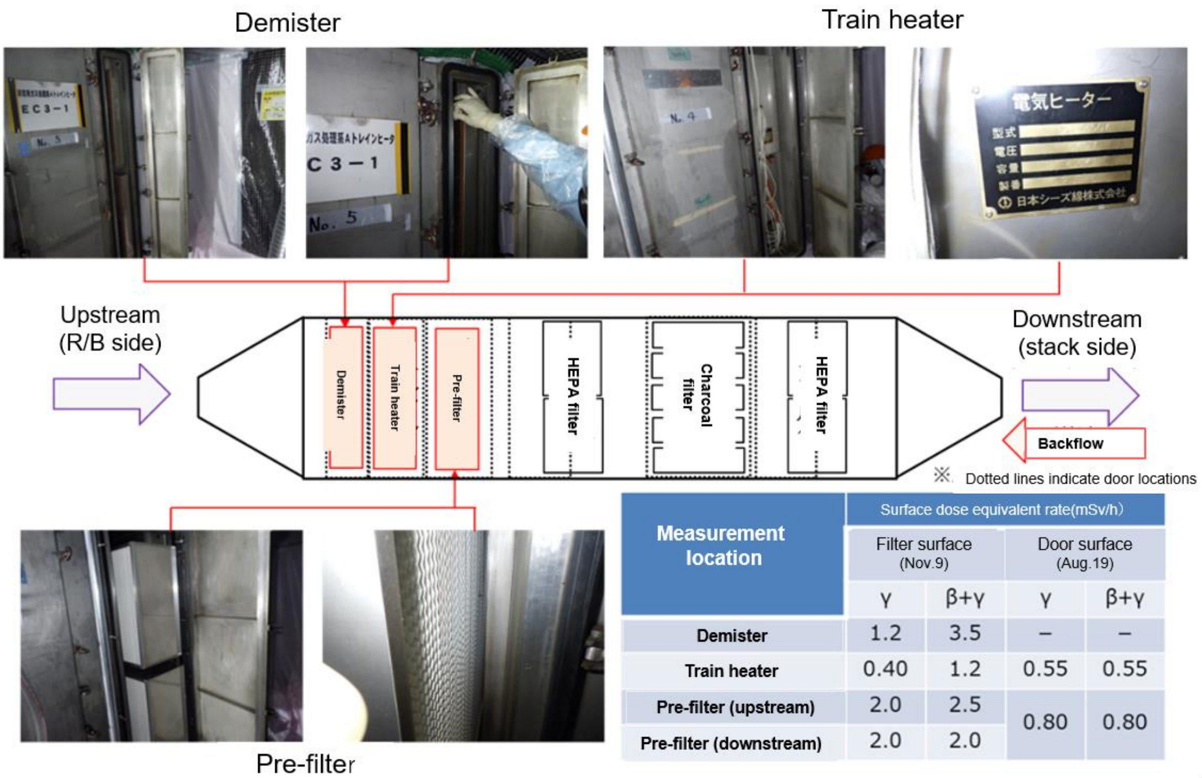


No	Surface dose equivalent rate(mSv/h)	
	γ	γ + β
×1	26	–
×2	32	–
×3	9.0	–
×4	18	–
×5	10	–
×6	30	–
×7	15	–
×8	30	–
×9	2.5	–
×10	5.0	–
×11	2.5	–
×12	2.0	–
×13	6.0	–
×14	5.0	–
×15	6.0	–
×16	6.0	–
×17	7.0	–
×18	9.0	–
×19	8.0	–
×20	7.0	–

Equipment
x1 - x4 : TELETECTOR
X5 – x20 : ICW

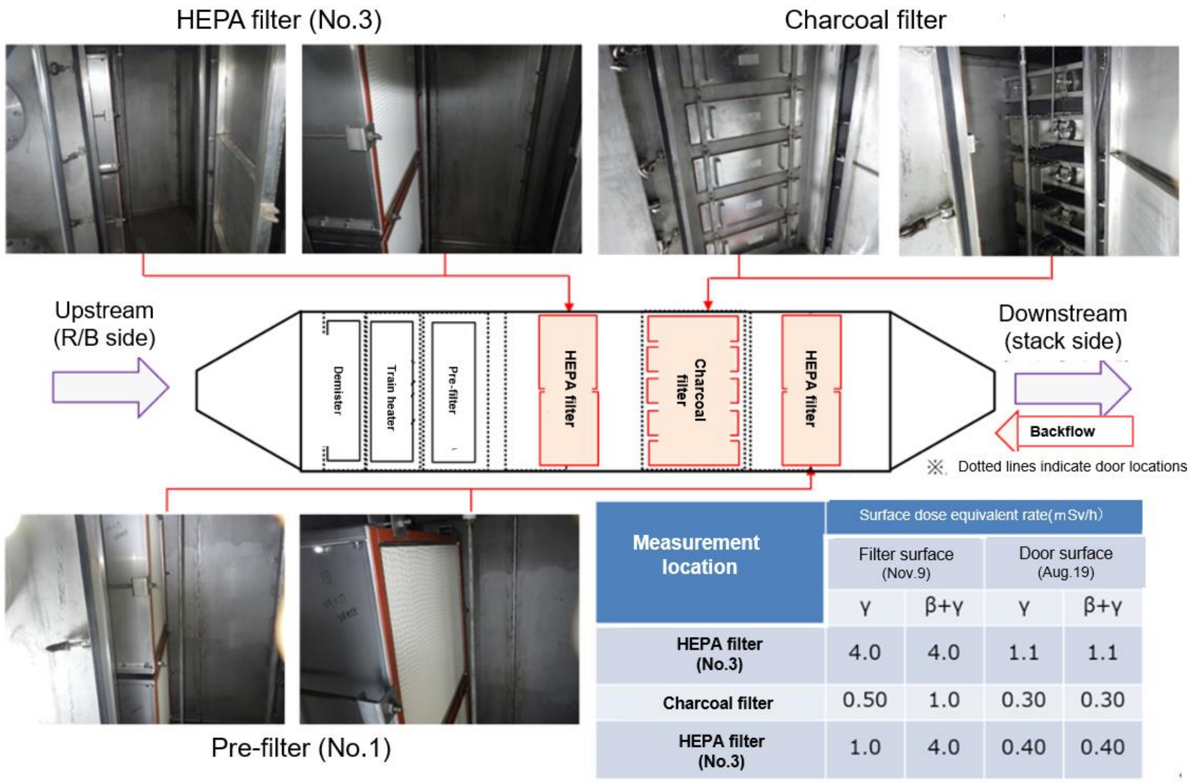
Figure 4-34 Dose rate measurement results around the rupture disk [4-16]

As a result of the surface dose rate measurement and smear sampling of the filters inside the filter train done by opening the train, it was confirmed that each filter was contaminated, but no damage was observed (Figures 4-35 to 4-38). In addition, there was a pool of water inside the B train when it was opened. It was estimated that this was condensed when the vent gas passed through.



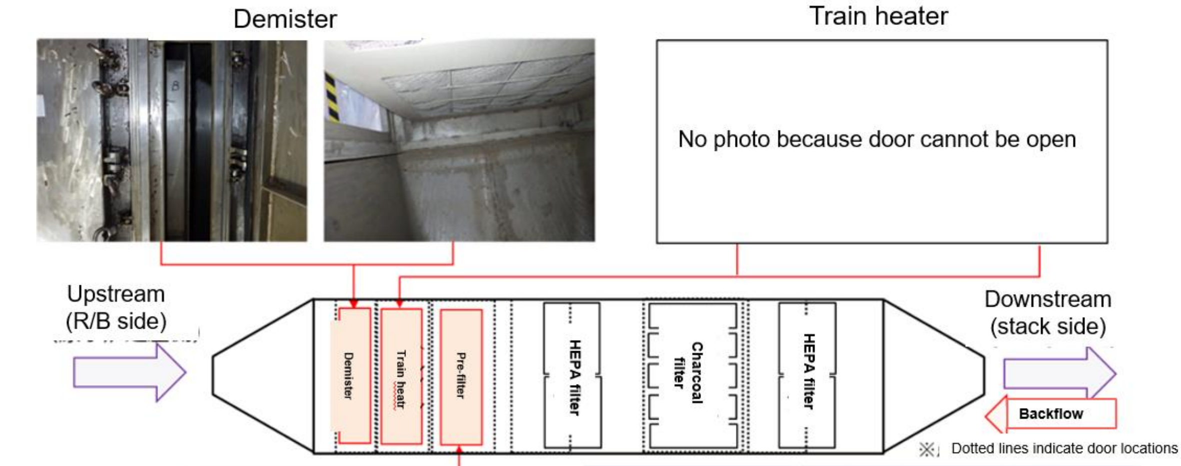
Measurement location	Surface dose equivalent rate(mSv/h) Filter surface (Nov.9) γ	Filter surface (Nov.9) β+γ	Door surface (Aug.19) γ	Door surface (Aug.19) β+γ
Demister	1.2	3.5	–	–
Train heater	0.40	1.2	0.55	0.55
Pre-filter (upstream)	2.0	2.5	0.80	0.80
Pre-filter (downstream)	2.0	2.0		

Figure 4-35 Investigation inside filter train A result ① [4-16]



Measurement location	Surface dose equivalent rate(mSv/h) Filter surface (Nov.9) γ	Filter surface (Nov.9) β+γ	Door surface (Aug.19) γ	Door surface (Aug.19) β+γ
HEPA filter (No.3)	4.0	4.0	1.1	1.1
Charcoal filter	0.50	1.0	0.30	0.30
HEPA filter (No.3)	1.0	4.0	0.40	0.40

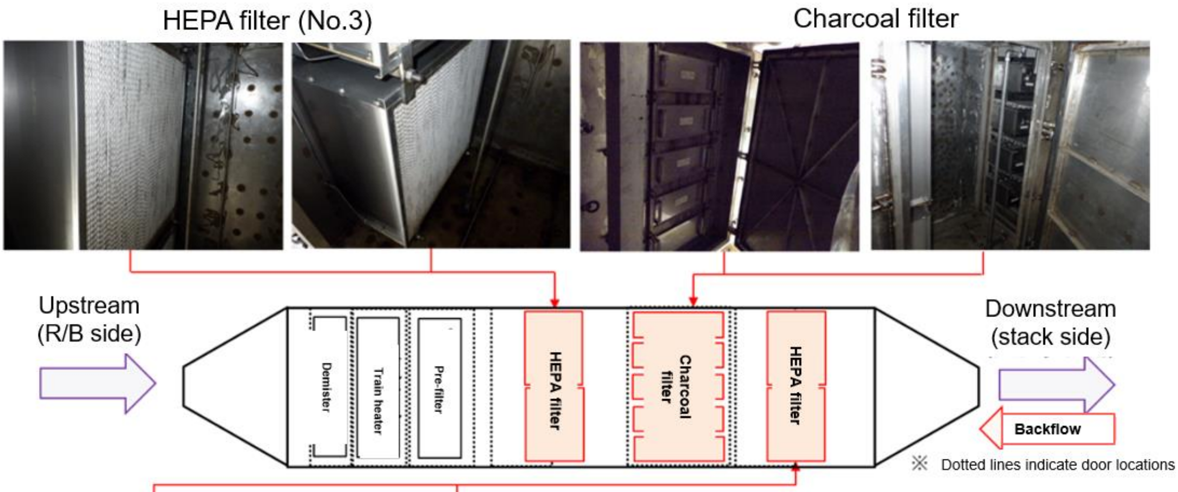
Figure 4-36 Investigation inside filter train A result ② [4-16]

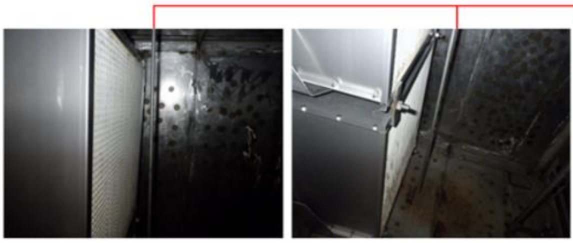


Pre-filter(Photo from downstream because door cannot be open)

Measurement location	Surface dose equivalent rate(mSv/h) Filter surface (Jan.7) γ	β+γ	Door surface (Aug.19) γ	β+γ
Demister	0.024	0.030	–	–
Train heater	–	–	0.030	0.030
Pre-filter (upstream)	–	–	0.045	0.045
Pre-filter (downstream)	0.090	0.25		

Figure 4-37 Investigation inside filter train B result ① [4-16]



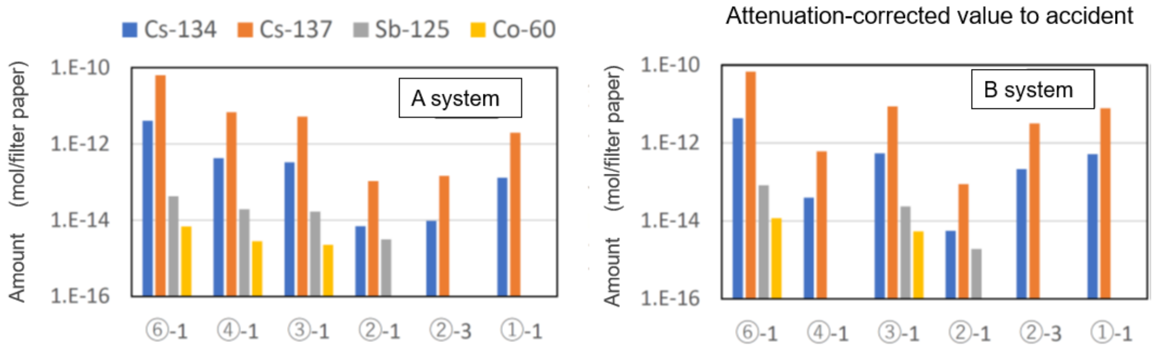


Pre-filter (No.1)

Measurement location	Surface dose equivalent rate(mSv/h) Filter surface (Jan.7) γ	β+γ	Door surface (Aug.19) γ	β+γ
HEPA filter (No.3)	0.12	0.30	0.055	0.055
Charcoal filter	0.50	2.0	0.25	0.25
HEPA filter (No.3)	4.5	20	1.1	1.1

Figure 4-38 Investigation inside filter train B result ② [4-16]

Analysis of smear samples from the filter surface confirmed that the demisters located furthest upstream and the HEPA filter located furthest downstream tended to have a higher presence ratio of nuclides (Figure 4- 39). It was estimated that this result suggests that in addition to the forward flow direction, vent gas containing nuclides flowed back into the filter train during venting operation.



- The most upstream filter (⑥)) and the most downstream filter (①) tend to have higher values.

⇒ In addition to the normal flow direction (from the reactor building side to the stack side), this suggests that the vent gas containing nuclides flowed back into the filter train when the containment vessel vent was activated.

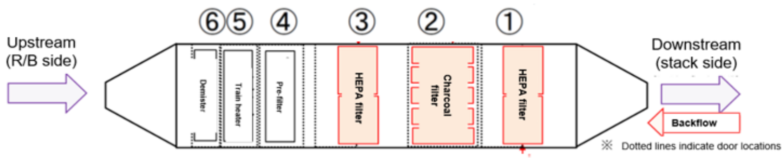


Figure 4-39 Analysis results of smear samples collected from the filter surface [4-18]

[4-18] Handout document: Summary of smear sample analysis by JAEA, 29th Meeting of the Study Group on Analysis of Accident at TEPCO's Fukushima Daiichi Nuclear Power Station, April 26, 2022.

(10) Analysis results of S/C water sampling and detection of stagnant gas in the RHR system [UPDATE]

In order to refine the design and operation of the PCV water intake system for the gradual decline of the PCV (S/C) water level as a measure to improve the seismic resistance of the reactor building, sampling was conducted in advance from late July to mid-September 2020 to determine the water quality of the S/C inner water (bottom part) (Figure 4-40). The analysis of the sampled water retained in S/C showed that the concentration of radioactive materials (Cs-137, total β activity) was higher than that currently found in the building stagnant water, and the concentration of total α activity was below the detection limit (Table 4-2).

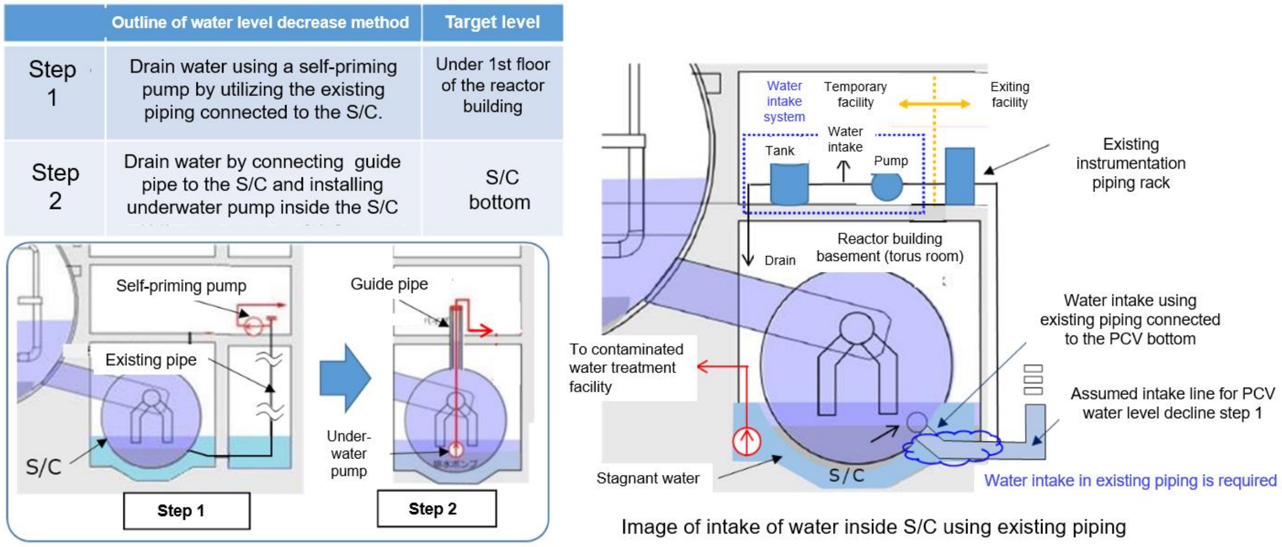


Figure 4-40 Image of S/C water intake using existing piping [4-19]

Table 4-2 Analytical results and effect of water containing S/C inclusions [4-19]

Properties of water in S/C and stagnant water in the building				Influence on stagnant water transfer and treatment	Reflected in equipment design of PCV intake system
Item		Water in S/C	Stagnant water ※1		
Total α*2	Bq/L	<5.73E+00	2.50E+01	None	None
Total β	Bq/L	7.88E+08	3.49E+07	Because of high radioactivity concentrations of Cs-137, etc., there is a possibility of affecting operation (frequency of adsorption column replacement) and adsorption performance of the contaminated water treatment facilities.	**Reflected in equipment design (radiation resistance)**
Cs-134	Bq/L	3.15E+07	1.16E+06		
Cs-137	Bq/L	6.07E+08	2.15E+07		
Cl	ppm	1800	600	Although concentration is higher than stagnant water, it was judged to have small impact based on past results.	**Reflected in equipment design (corrosion resistance)**
Ca	ppm	20	25	Since it is the same level as stagnant water in the building, it was judged to have no impact.	None
Mg	ppm	56	—		None
H-3	Bq/L	1.08E+07	—	None	None

[4-19] Handout document: Examination status of the Unit-3 PCV water level decrease, 88th Decommissioning and Contaminated Water Response Team Joint Meeting, October 29, 2020.

After the analysis, when proceeding with the installation of the PCV intake system, the vent valve around the residual heat removal system (RHR) heat exchanger (A) was opened in December 2021 as a preliminary preparation for draining the existing piping, and it was confirmed that a flammable gas was present in the exhaust. From collecting and analyzing the gas, Kr-85, a long half-life nuclide, was detected (Figure 4-41). The source of this stagnant gas is unknown, but it was thought that it might have been caused by the inflow of gas into the system due to some operation during the accident, radiolysis of water held in the system, or other sources.

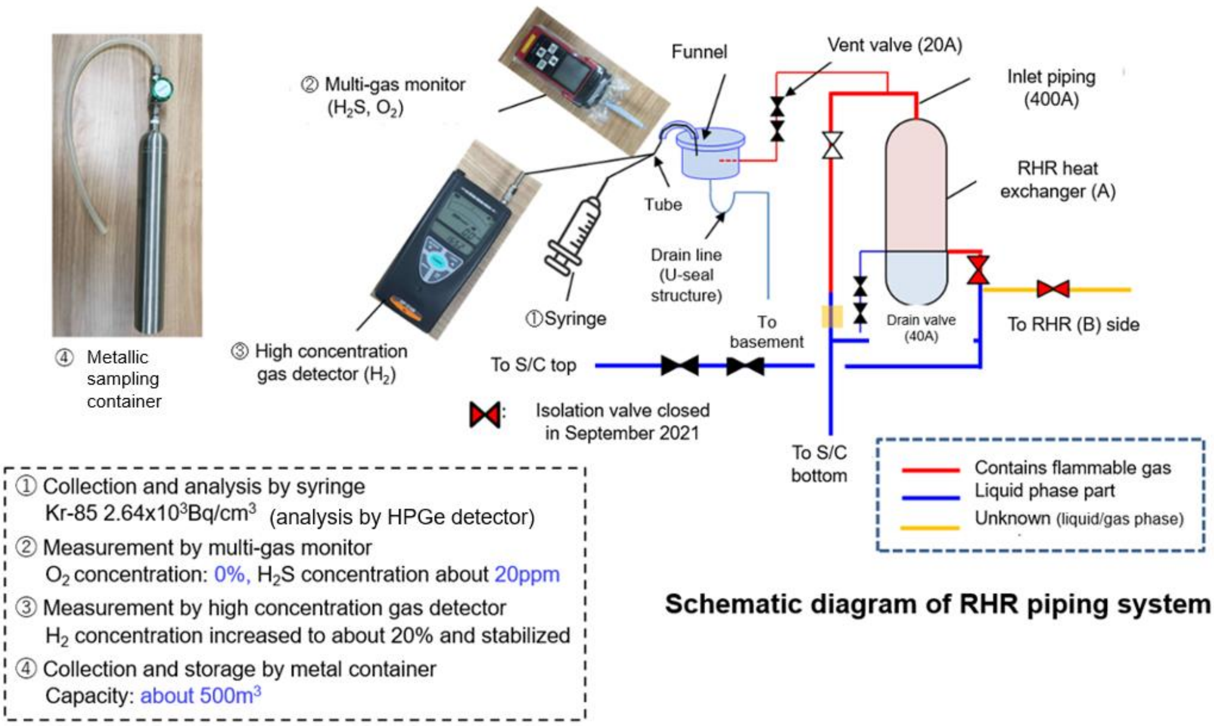


Figure 4- 41 Results of stagnant gas concentration measurement (H_2, etc.) and sampling [4-20]

[4-20] Handout document: Response to stagnant gas that was confirmed in the RHR piping of Unit-3, 97th Specific Nuclear Facility Supervising and Evaluation Committee Meeting, February 14, 2022.