Subsidy Project of Decommissioning and Contaminated Water Management

Upgrading of the Comprehensive Identification of Conditions inside Reactor

Accomplishment Report for FY2017

June 2018

International Research Institute for Nuclear Decommissioning (IRID)
The Institute of Applied Energy (IAE)

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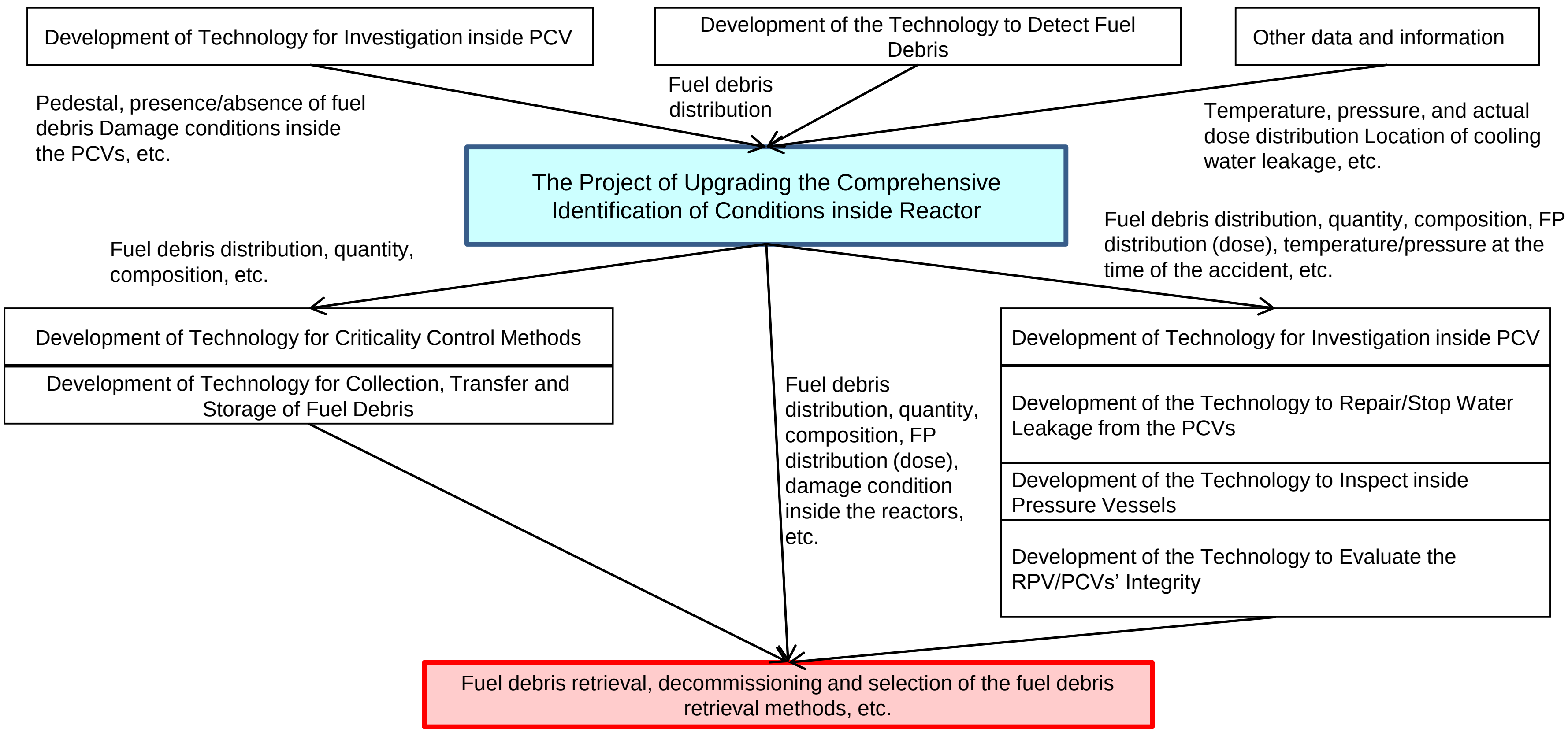
1-1. Project Needs

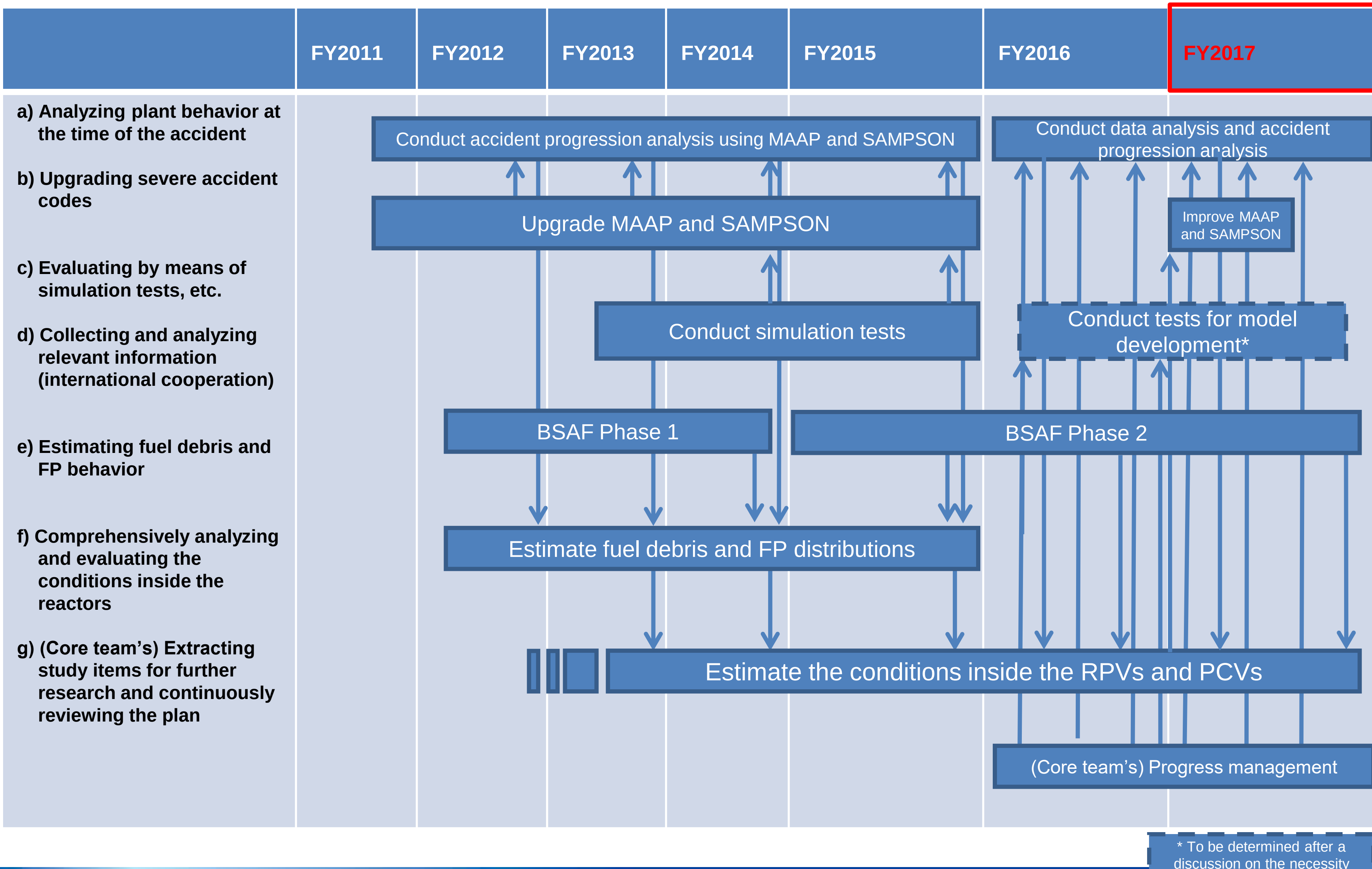
- For the decommissioning initiative for the Fukushima Daiichi Nuclear Power Station, it is crucial to estimate and understand the conditions of fuel debris and fission products (hereinafter FPs) inside the reactor pressure vessels (hereinafter RPVs). Direct observation inside the RPVs is still challenging due to the highly radioactive environment.
- A practical alternative measure is to comprehensively advance analysis and evaluation techniques to estimate the conditions inside the RPVs and the primary containment vessels (hereinafter PCVs) through accident progression analysis and evaluation of various measurement data and information, such as those collected on-site. This knowledge is expected to be used for the decommissioning process. To this end, initiatives for understanding the conditions inside the reactors have been pushed forward since FY2011, centering on improved severe accident analysis codes; the Modular Accident Analysis Program (MAAP) and the Severe Accident Analysis Code with Mechanistic (SAMPSON).
- Since FY2016, comprehensive activities have been carried out to advance identifying of the conditions inside the RPVs and PCVs. Not only the analytical results, using the accident progression analysis codes, but also plant parameters obtained at the time of the accident and other actual measurements will be used. Fact-based, on-site information collected through the internal PCV investigations, or dose investigations inside the R/Bs, in addition to the material science knowledge enhanced by recent studies will also be used.
- Amid such circumstances, the following technical development was conducted in FY2017.
  - In FY2016, the review was pushed forward with no improvement of the analysis codes based on the understanding of their capabilities and limits. In the end, it was necessary to upgrade the analysis codes to enhance the analysis results' reliability. Therefore, it was decided to make partial improvements to the analysis codes in FY2017.
  - Estimating the conditions inside the RPVs and PCVs by using the results obtained up to FY2016 were advanced. The study items selected for further research to reduce uncertainties of the estimated conditions inside the RPVs and PCVs were also analyzed and evaluated.

1-2. Application of Project Achievements and Project Target

- **The aim of the project is to estimate the conditions inside the RPVs and PCVs through comprehensive analysis and evaluation of the study on issues to be solved. Specifically, major outputs of the project are shown in the below diagram; fuel debris and FP distributions that were estimated based on comprehensive analysis and evaluation. The diagrams were occasionally updated to reflect the latest results and provide them for other related research projects.**
- **In other words, the purpose of the project is to provide information that will be useful for fuel debris retrieval and decommissioning work.**



1-3. Overall Plan: Implementation Schedule



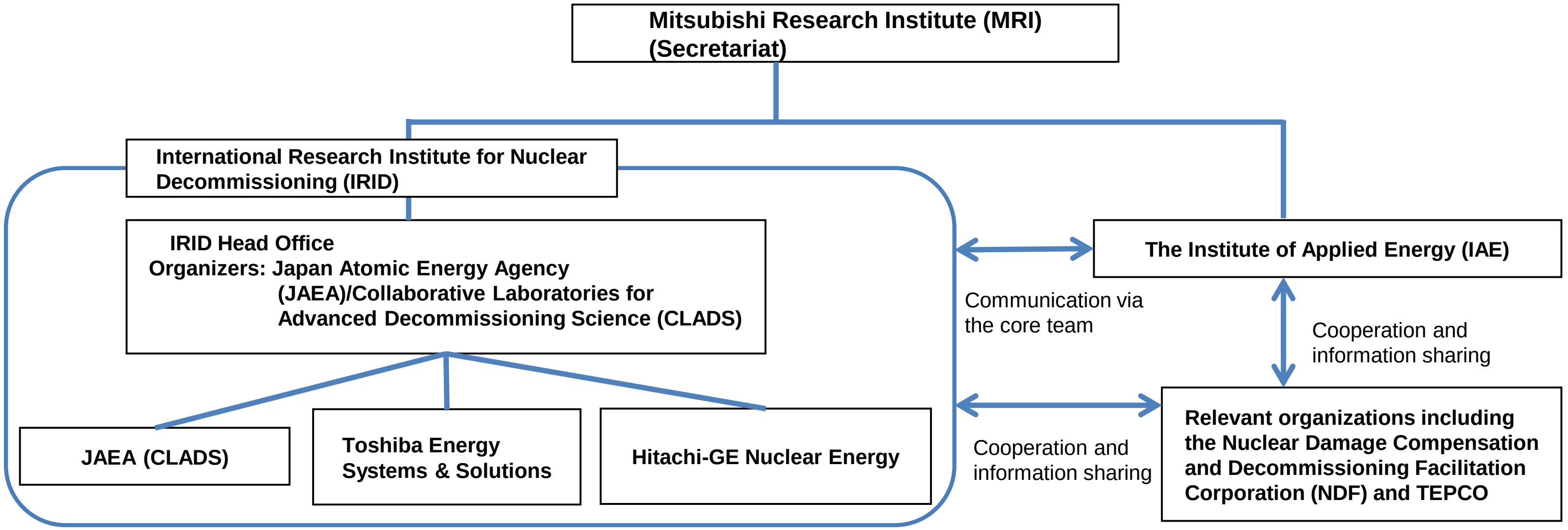
1-4. Project Overview and Structure for FY2017

○ Project Overview

Identifying of the conditions inside the reactors has been sought by gathering knowledge from Japan and abroad and cooperating with overseas institutions. Accident progression analysis technology, analyzed actual data and on-site investigation results were used, as well as material science knowledge, etc. By using the highest level of technology available, the fuel debris and FP distributions inside the RPVs and PCVs as an input for the decommissioning task was estimated.

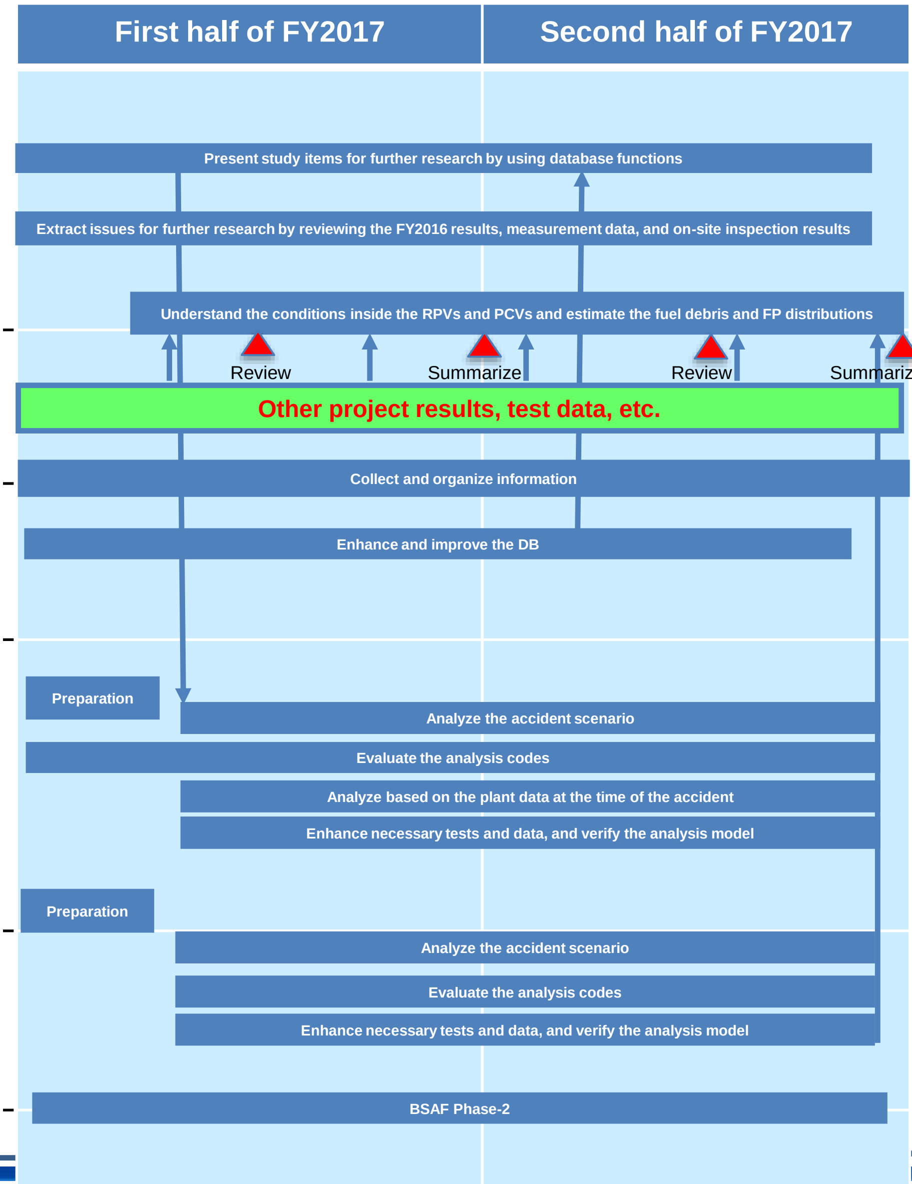
○ Structure of the Project

This project is conducted under the cooperative framework between IRID and IAE, in which IRID manages and executes the overall project in cooperation with IAE. IRID conducts the project in cooperation with the IRID member organizations, which are JAEA (CLADS), Toshiba Energy Systems & Solutions, and Hitachi-GE Nuclear Energy. In addition, the project was promoted by closely sharing information with relevant organizations, such as NDF and TEPCO, and thereby obtaining cooperation from them.



Core team: An organization consisting of project participating organizations. Responsible for project policymaking and estimating conditions inside the RPVs and PCVs.

1-5. Implementation Items and Schedule for FY2017



(1) Comprehensively analyze and evaluate the conditions inside the reactors

(1) Comprehensively analyze and evaluate based on the Units data and other R&D project results

[1]-1 Extract issues for further research to reduce uncertainties of estimated conditions inside the reactors

[1]-2 Comprehensively analyze and evaluate based on the Units data and other R&D project results

[2] Establish a database for comprehensive analysis and evaluation

(2) Estimate and evaluate fuel debris behavior and fission product behavior and their characteristics for comprehensive analysis and evaluation

[1] Reduce uncertainties by using analytical techniques, etc.

[2] Evaluate FPs' chemical characteristics

[3] International joint research to use knowledge inside and outside Japan

2. Project Achievements for FY2017

2-1. Overall Structure of the Project for Identifying the Conditions inside the Reactors

(1) Comprehensively analyze and evaluate the conditions inside the reactors

[1] Comprehensively analyze and evaluate based on the Units data and other R&D project results

1 Extraction of issues for further research to reduce uncertainties about the estimated conditions inside the reactors

2 Comprehensive analysis and evaluation based on the Unit data and other research development results

[2] Establishment of a database for comprehensive analysis and evaluation (IAE)

(2) Estimation and evaluation of fuel debris behavior, and fission product behavior/characteristics toward comprehensive analysis and evaluation

[1] Reduce uncertainties by using analytical techniques, etc.

1 Evaluation of the concrete erosion at the pedestal using sensitivity analysis (Hitachi GE)
2 Inverse problem evaluation using virtual reactors and a compiled database (JAEA)
3-(a)-1 Estimation of the amount of fuel debris that remains in the pedestal space obstacles (IAE)
3-(a)-2 Detailed evaluation of debris behavior inside the post-damage lower head (IAE)
3-(a)-3 Evaluation of the condensation behavior of water vapor containing hydrogen in the S/P (IAE)
3-(a)-4 Event sequence analysis at the time of core material slumping (JAEA)
3-(a)-5 Three-dimensional evaluation of MCCI behavior (IAE)
3-(a)-6 Determination of factors influencing debris distribution (IAE)
3-(a)-7 Analysis of each unit over three weeks after the accident (IAE)
3-(a)-8 Detailed accident progression analysis using accident progression analysis codes (Toshiba Energy Systems & Solutions)
6 Simulated fuel assembly degradation test (JAEA)

2-1. Overall Structure of the Project for Identifying the Conditions inside the Reactors

(2) Estimation and evaluation of fuel debris behavior, and fission product behavior/characteristics toward comprehensive analysis and evaluation

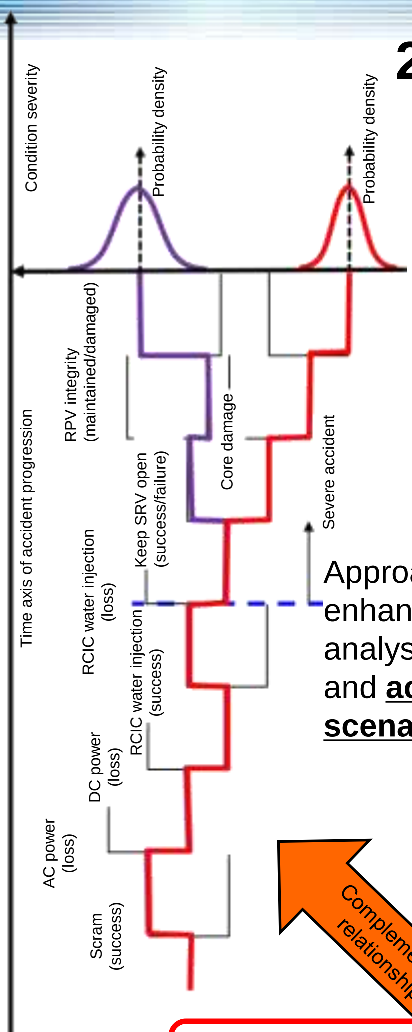
[2] Evaluation of FPs' chemical characteristics (JAEA, IAE)

1 Reaction between cesium and steel materials, and re-evaporation
2 Evaluation of particulate cesium compounds
3 Optimization of the cesium compounds evaluation model
4 Analysis and evaluation of the units using the upgraded model
5 Analysis of samples collected at Fukushima Daiichi Nuclear Power Station (NPS)

[3] International joint research to make use of knowledge inside and outside Japan (IAE)

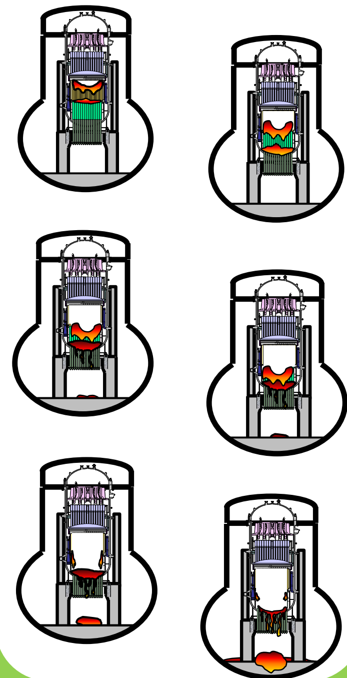
2-2 Three Approaches for Further Research

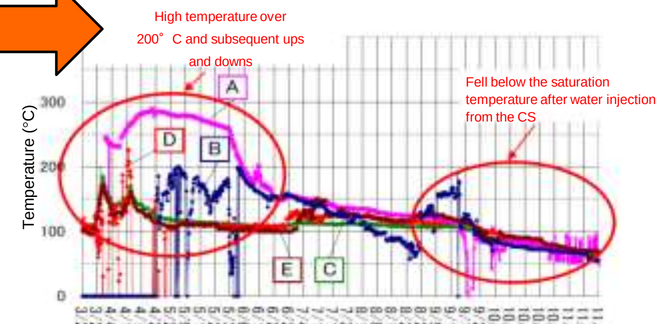
Complementary relationship



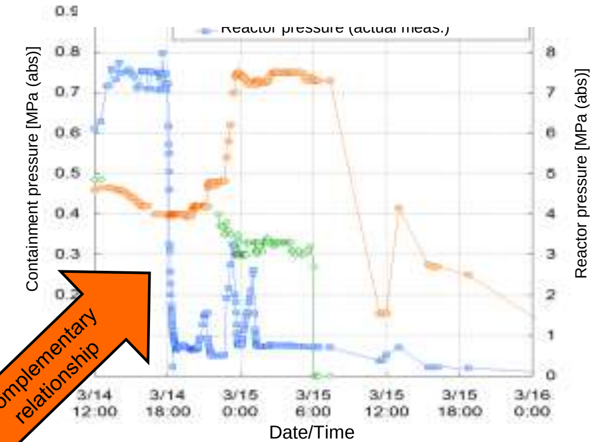
Approach by means of enhanced reliability of analysis code evaluation and **accident progression scenario analysis**

Identifying reactor conditions





Approach promoting the estimate by deepening the Identifying of the phenomenon using **data analysis and inverse problem analysis**

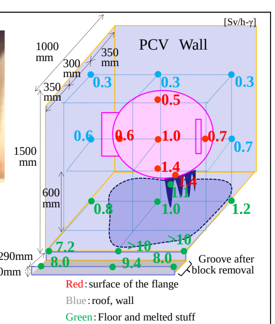


Complementary relationship

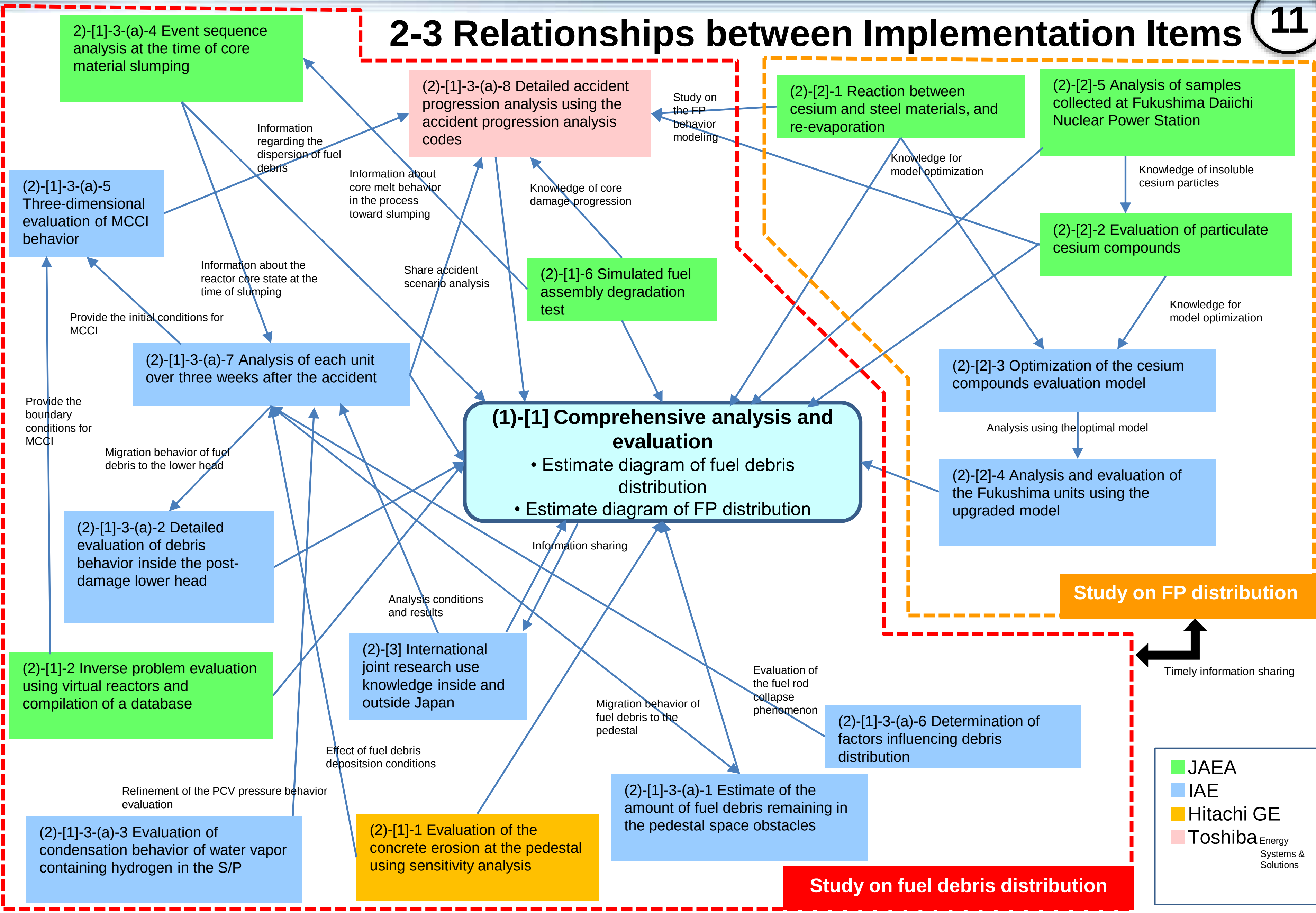
Complementary relationship

Information obtained by on-site inspection and estimate about the surrounding area

X-6 penetration flange (After steel plate removal)



2-3 Relationships between Implementation Items



2-4. Results of Comprehensive Analysis and Evaluation

Unit 1: Diagram of Summarized Information

- **Measurement results and observation data available**
- ***Estimated based on the measurement results and observation data***
- Accident analysis or qualitative estimation

Reactor building (1F-4F)

- **The ambient dose rate is several to tens of mSv/h.**
- **There is high radiation around the RCW piping**
- **There is significant contamination of 100 mSv/h or greater in the TIP room near the containment penetration of the PCV**
- *The inspection on the 1st floor of the reactor building revealed high radiation around the AC piping (presumably due the effect of the vent).*
- **The southwest section of the 4th floor is heavily damaged, while the damage is minor on the east side. The ceiling (5th floor) of the northwest section has collapsed.** *The major damage was assumed to be affected by blast and rubble caused by the hydrogen explosion on the 5th floor.*
- **Hydrogen of 10,000 ppm or greater was detected from the PCV spray piping (Sep 2011)**
- **In the MSIV room, the max dose was 1,096 mSv/h near the top plate and duct of HVH (Nov 2015).** *The source is estimated to be the upper duct of HVH.*

PCV top head

- *The sealing has degraded, and a leakage opening exists (from which vapor and FPs were discharged).*

Operation floor

- **60 mSv at 2-2.5 m above the shield plug (2011)**
- **The roof collapsed, almost keeping its flat shape.**
- **Significant contamination with max 121 mSv/h at approx. 1 m above the roof right on the top of the reactor (Aug 2015)**
- **The temperature right above the reactor was higher than in any other areas (Oct 2011),** *possibly due to the effect of heat from the FPs attached around the shield plug.*
- **The top and middle layers of the shield plug were lifted.**
- **In addition to the top and middle layers of the shield plug, the bottom one had shifted from its original position.**
- **The top north section of the shield plug had shifted 720 mm to the west. Also, the top layer sagged down by 84 mm at its north section and by 155 mm at its middle section.**
- **The vertical dose distribution around the shield plug indicated that the dose rises toward the gap below the top layer of the shield plug (max 565.8 mSv/h)(2017). The horizontal dose distribution inside the gap between the top and middle layers of the shield plug indicated that the dose is higher toward the center of the plug.**
- **The surface dose rate on the shield plug was max 200 mSv/h, and the rate was high around the center of the plug. (Jun 2017)**
- **Gamma-ray spectra measurement detected only two nuclides, which were Cs-134 and Cs-137, in the gap and on the plug surface.** ***(Possibly other nuclides were not detected because Cs was a too strong source)*****(Jun 2017)**

Upper part of the pressure vessel

- **The temperature reading was almost the same at the upper and lower parts of the RPV.**
- A large amount of FPs have been attached.
- The chemical forms of the attached FPs are unknown (ex. solubility/insolubility, etc.).
- The degree of re-evaporation of the attached FPs is unknown.
- Assume a leak opening was created near the upper part of the pressure vessel (MS piping, for example)

In-core structures

- Analysis results indicate the structures' downward shift because of creep and support deformation.
- Large amounts of FPs are assumed to have been attached on the separator and dryer.
- Cesium has possibly been taken into the steel materials' oxide layers.
- Cesium may have been combined with molybdenum, boron, and silicon.

Reactor core

- The remaining volume seems to be almost zero, as shown by the analysis results.
- **Muon measurement was performed (Feb-May 2015). No significant fuel mass was detected in the reactor core section.**
- **The water level in the reactor core area was zero**

CRD

- *A heat source is estimated to exist near the CRD piping based on the variation in the HVH temperature.*
- **There is a correlation between the feedwater system's water injection flow rate and a part of HVH temperature behavior.** *(A heat source is estimated to exist near the CRD piping in the northwest, northeast, and southwest.)* **(***Debris is estimated to exist around the control rod drive piping because the HVH temperature's increment rate was partially greater than the readings of other PCV thermometers when the volume of water injection into the reactor was reduced from Dec 2016 to Jan 2017).*
- **The contact condition of the control rod position indicator was checked. There is no clear tendency allowing us to estimate the lower RPV section's condition.**

Reactor well

- High radiation is expected, presumably because leakage exists in the PCV top head section.

RPV lower head

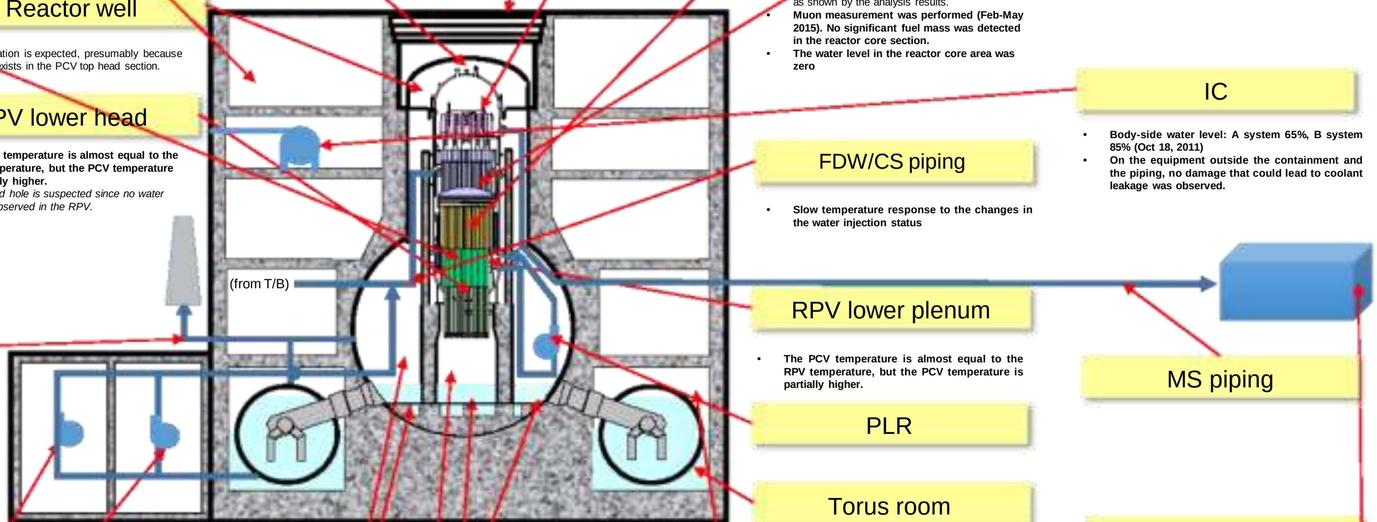
- **The PCV temperature is almost equal to the RPV temperature, but the PCV temperature is partially higher.**
- *A ruptured hole is suspected since no water level is observed in the RPV.*

IC

- **Body-side water level: A system 65%, B system 85% (Oct 18, 2011)**
- **On the equipment outside the containment and the piping, no damage that could lead to coolant leakage was observed.**

FDW/CS piping

- **Slow temperature response to the changes in the water injection status**



RPV lower plenum

- **The PCV temperature is almost equal to the RPV temperature, but the PCV temperature is partially higher.**

PCV vent/exhaust stack

- **Significant contamination of up to 5,000 mSv/h in front of the SGTS room on 2F (Aug 2011)**
- **High contamination exceeding 10 Sv/h was observed at the exhaust stack used for both Units 1 and 2 (near the SGTS piping joint)(Aug 2011). The dose dropped to 2 Sv/h (Oct 2015).**
- **Deformation and fracture were found on a part of the brace (support) used for the exhaust stack for both Units 1 and 2.** *The damage was probably caused by a hydrogen explosion.*
- *Contamination with several Sv/h at the SGTS in Unit 2 is presumably attributed to Unit 1 based on the estimation that there was no damage to the rupture disk of Unit 2.*
- **Cs134 (8.26 kBq/cm3) and Cs137 (51.9 kBq/cm3) were detected from the accumulated water inside the pit (Sep 12, 2016).**

MS piping

PLR

Torus room

- **The water level has been maintained below O.P.3,200 to prevent water from flowing into the turbine building.**
- **Water is leaking from the vacuum break line bellows and the sand cushion drain line.**
- **Dose on the catwalk: 200–2,400 mSv/h (May 2014)**
- *There is some kind of penetration exists between the R/B and turbine building.*
- **There is significant contamination in the northwest section, with max 920 mSv/h in air and max. 800 mSv/h underwater**

Turbine building

- **The ambient dose rate was several μSv/h to several-hundred mSv/h.**
- **High radiation was detected in the underground floor.**
- **Significant contamination of up to 5,000 mSv/h in front of the SGTS room on 2F (Aug 2011)**
- *Highly contaminated water was assumed to have flowed out of the vacuum break line bellows and sand cushion.*

Drywell

- **The inspection on the 1F grating outside the pedestal indicated no major structural damage, although there were some fallen objects. The average dose was several Sv/h. There is a mass, seemingly a piping shield, around the PLR piping.** *(Presumably a molten lead mat)*
- **Slow temperature response to the water injection changes. Regardless of water injection, some PCV thermometers have shown higher readings due to the effect of nitrogen injection, etc.**
- The measured temperature is lower than the estimated decay heat.
- The pressure suggests some leakage in the gas phase of the containment (incl. the PCV top head).
- *Water-filling conducted in the water level gauge piping (the water level gauge piping is presumably sound).*

RPV pedestal

- *Due to the dropped fuel debris, damages are estimated on the grating, TIP piping, CRD exchanger, etc.*
- *Water injected into the RPV is presumably falling into the pedestal.*

CS pump

HPCI

- **A white, powdery deposits exists around the HPCI's vapor line penetration X-53 (on the bellows cover, floor, and wall) (sample collected). A leak trace exists in the gap between the HPCI piping and bellows cover as well as at the root portion of the bellows cover and biological shielding wall. The root portion has the highest radiation (max 7 Sv/h).**

Drywell floor

- **A water level of approx. 3m (leakage was found from the vacuum break line bellows, which is at about the same water level)**
- **A camera inspection observed a deposits on the floor, and the second inspection indicated increased volume.**
- Analysis indicated that the debris had spread on the D/W floor, reaching the PCV shell.
- **At the observation point, the dose in the water in the PCV tends to rise toward the bottom, and it reaches the level of several Sv/h to more than 10 Sv/h at about 0.3-1.6 m above the floor surface. The dose near the pedestal opening (at approx. 1 m from the floor surface) was up to 9.4 Sv/h (2017).**
- **Simplified fluorescent X-ray analysis of the deposits detected U, as well as Fe, Ni, Zn and Pb, which probably originated from the structural materials and paint inside the containment. Although various gamma-ray analyses identified Cs-134, 137, Co-60, and Sb-125,** *small amounts in the deposits are estimated, given that none of them was detected in the simplified fluorescent X-ray analysis.*
- **Based on the captured image and the length of the wire sent, the height of the deposits was estimated to be approx. 0.3 m or lower on the opposite side of the opening and approx. 1.0 m or lower near the opening.**
- **No major damage was observed on the sump pump, valve, etc. on the opposite side of the opening.**

RPV pedestal floor

- *The high radiation observed near the RCW piping in the reactor building was it guessed to be caused by debris which damaged the equipment drain sump in the pedestal and allowed radioactive materials to enter the RCW system. Therefore, the pedestal is guessed to have debris.*
- The dropped debris are guessed to be eroding the pedestal floor and sump to some extent.

Suppression chamber

- **The S/C is almost filled up with water (a slight amount of gas generated in the early phase of the accident still exists; nitrogen shielding is underway).** *The S/C is estimated to be almost sound.*
- *From the results of the dose inspection at the torus room, it is estimated that the Cs concentration level is high at the building and S/C walls or in the stagnant water inside the S/C.*

Estimate Diagram of the Debris Distribution in Unit 1 and Its RPV and PCV Conditions

- A molten pool may have been formed inside the reactor during the accident. (general assumption)
- The shroud may be deformed, damaged or buckled. (general assumption and analysis)
- The shroud may be damaged, possibly causing molten fuel enter the down-comer area and damaging the jet pump. (general assumption)

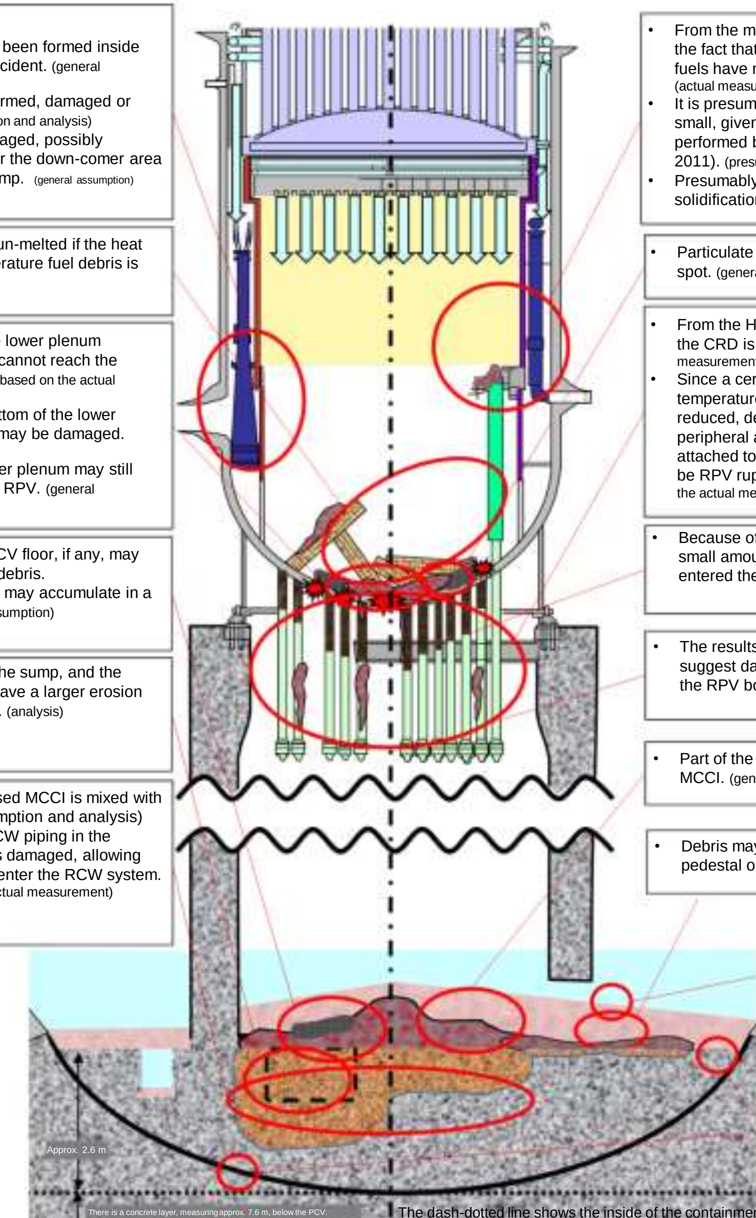
- The CRGT may remain un-melted if the heat transfer from high-temperature fuel debris is small. (general assumption)

- There are ruptures in the lower plenum because the water level cannot reach the reactor core. (presumption based on the actual measurement)
- The drain, etc., at the bottom of the lower plenum are fragile; they may be damaged. (general assumption)
- The fuel fallen to the lower plenum may still exist at the bottom of the RPV. (general assumption)

- A pool of water on the PCV floor, if any, may have formed particulate debris.
- Particulate debris, if any, may accumulate in a stagnant spot. (general assumption)

- The pedestal wall near the sump, and the concrete below it, may have a larger erosion area than other portions. (analysis)

- The fuel debris that caused MCCI is mixed with concrete. (general assumption and analysis)
- In our estimation, the RCW piping in the equipment drain sump is damaged, allowing radioactive materials to enter the RCW system. (presumption based on the actual measurement)



The dash-dotted line shows the inside of the containment is asymmetrical with the pedestal.

- From the muon measurement and analysis results, and the fact that the water level is zero, it is presumed most fuels have melted ,and there are no remaining fuel rods. (actual measurement and analysis)
- It is presumed that the remaining amount of debris is small, given the fact that the cooldown was successfully performed before starting the CS water injection (Dec 10, 2011). (presumption based on the actual measurement)
- Presumably general oxide debris, a result of molten fuel solidification. (general assumption)

- Particulate debris, if any, may accumulate in a stagnant spot. (general assumption)

- From the HVH temperature, the presence of debris near the CRD is presumed. (presumption based on the actual measurements and analysis)
- Since a certain HVH thermometer showed a significant temperature increase when the FDW flow rate was reduced, debris possibly exists near the CRD in the peripheral area (but there is no telling whether it is attached to the outer surface or it is inside), and there may be RPV ruptures immediately above it. (presumption based on the actual measurement)

- Because of damage to the CRGT and CRD housing, a small amount of fuel debris or molten metal may have entered the CRD housing. (general assumption and testing)

- The results of the internal PCV investigation of Unit 3 suggest damage to the CRD housing, the platform, and the RPV bottom. (presumption based on the actual measurement)

- Part of the fuel debris may have been solidified without MCCI. (general assumption)

- Debris may have spread on the D/W floor through the pedestal opening. (general assumption and analysis)

- There are mounds of deposits on the D/W floor, and the closer they are to the pedestal opening, the higher they tend to be. (actual measurement)

- In our estimation, the PCV is damaged because water is leaking from the sand cushion drain pipe. (actual measurement and analysis)

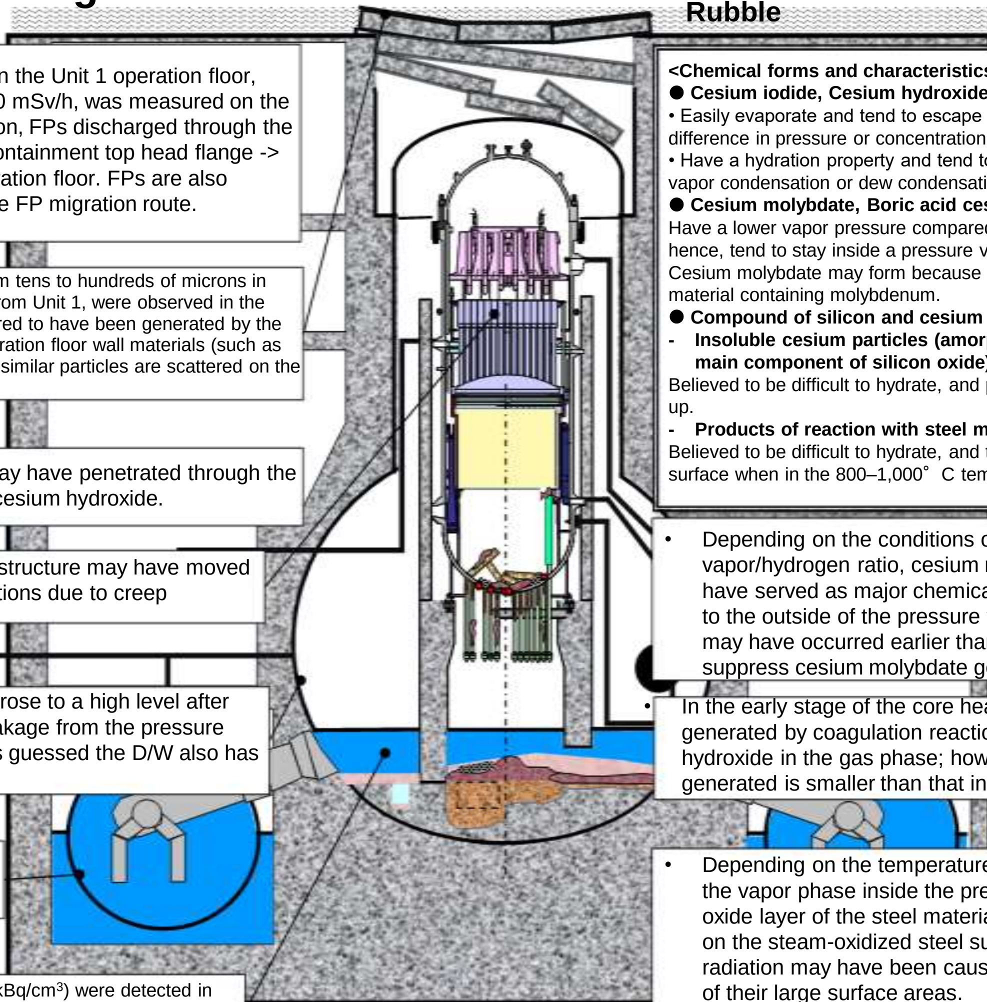
Legendss

	Remaining fuel rod and its debris*
	Oxide debris (porous)
	Particulate debris
	Fuel debris (rich in metals)*
	Concrete mixed debris
	CRGT
	Damaged CRGT
	CRD
	CRD (with debris inside)
	Shroud
	Damaged shroud
	Pellet*
	Ruptured hole of the RPV
	Upper tie plate*
	deposits (material unknown)
	Ballooning fuel*
	Oxide debris*
	Heavy metal debris*
	Particulate pellet*
	Cladding residue*
	Molten reactor internals*
	Solidified B4C*
	Control rod mixed melt*

* Not used in the estimate diagram of Unit 1

Unit 1: Estimate Diagram of FP Distribution

*The estimate was made focusing on cesium, a major source nuclide



- A hydrogen explosion occurred on the Unit 1 operation floor, and high radiation, exceeding 100 mSv/h, was measured on the rubble. Therefore, in our estimation, FPs discharged through the following route: containment -> containment top head flange -> reactor well -> shield plug -> operation floor. FPs are also figured to exist unevenly along the FP migration route.

- Cesium-bearing particles, ranging from tens to hundreds of microns in size, which seem to have originated from Unit 1, were observed in the environment. Those particles are figured to have been generated by the reaction between cesium and the operation floor wall materials (such as Glasron Wool), and it is guessed that similar particles are scattered on the operation floor.

- Hydrated cesium compounds may have penetrated through the concrete surface in the form of cesium hydroxide.

- In a high temperature, the upper-structure may have moved downwards from the original locations due to creep deformation, etc.

- It is likely that the D/W pressure rose to a high level after the accident due to the direct leakage from the pressure vessel to the D/W; therefore, it is guessed the D/W also has a large amount of FPs.

- High contamination is estimated on the building wall, the S/C wall, or the stagnant water in the S/C.

- Cs134 (19 kBq/cm^3) and Cs137 (35 kBq/cm^3) were detected in the samples collected from the stagnant water in the PCV. (Oct 9-13, 2012)

<Chemical forms and characteristics of major types of cesium estimated>

● Cesium iodide, Cesium hydroxide, Cesium chloride

• Easily evaporate and tend to escape from the pressure vessel as vapor due to a difference in pressure or concentration.

• Have a hydration property and tend to move with moisture by, for example, water vapor condensation or dew condensation (on the wall surface).

● Cesium molybdate, Boric acid cesium

Have a lower vapor pressure compared to the afore-mentioned substances and, hence, tend to stay inside a pressure vessel.

Cesium molybdate may form because of a reaction between cesium and steel material containing molybdenum.

● Compound of silicon and cesium

- Insoluble cesium particles (amorphous particles including cesium with its main component of silicon oxide)

Believed to be difficult to hydrate, and particles the size of a few microns tend to soar up.

- Products of reaction with steel material (crystalline substances)

Believed to be difficult to hydrate, and tend to stay in the oxide layer on the steel surface when in the 800–1,000° C temperature range.

- Depending on the conditions of the gas phase, such as the water vapor/hydrogen ratio, cesium molybdate or boric acid cesium may have served as major chemical species, suppressing their discharge to the outside of the pressure vessel. Also in Unit 1, water vapor loss may have occurred earlier than in other units, which may have helped suppress cesium molybdate generation.

- In the early stage of the core heating, insoluble cesium particles were generated by coagulation reaction between silicon oxide and cesium hydroxide in the gas phase; however, it is estimated that the amount generated is smaller than that in Unit 2.

- Depending on the temperature/atmosphere history, some cesium in the vapor phase inside the pressure vessel may have penetrated the oxide layer of the steel material due to the reaction with silicon oxide on the steam-oxidized steel surface. Considering this factor, the high radiation may have been caused at the separator and dryer because of their large surface areas.

- Cesium possibly remains inside the fuel debris although its amount is expected to be small.

- FPs may unevenly exist in fuel debris.

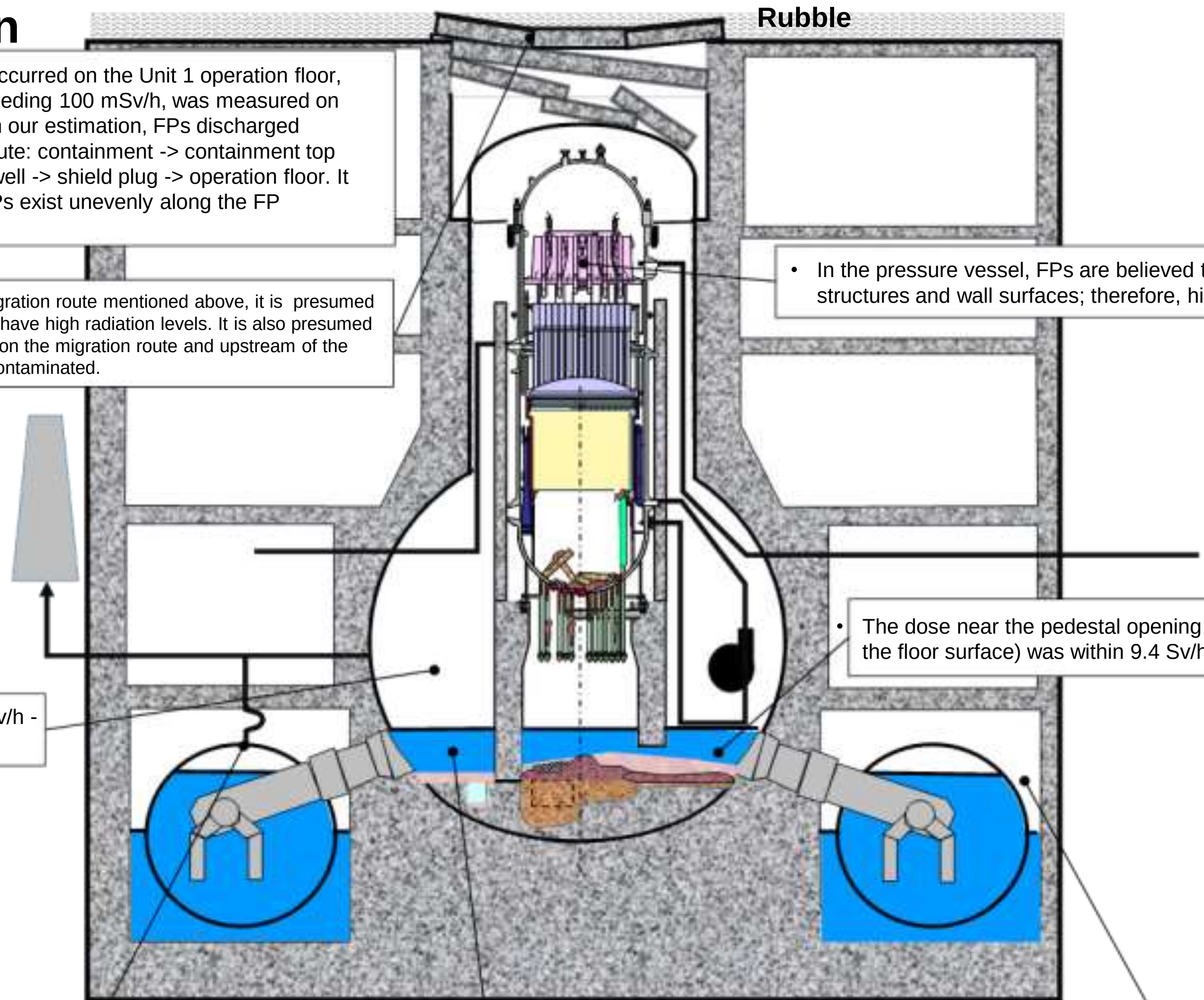
- In Unit 1, the leakage in the S/C is located at a high level; therefore, FPs released into the S/C may still be trapped in the stagnant water.
- A huge amount of FPs may still exist, given the large volume of stagnant water in the S/C.

Unit 1: Estimate Diagram of Dose Distribution

- A hydrogen explosion occurred on the Unit 1 operation floor, and high radiation, exceeding 100 mSv/h, was measured on the rubble. Therefore, in our estimation, FPs discharged through the following route: containment -> containment top head flange -> reactor well -> shield plug -> operation floor. It was also figured that FPs exist unevenly along the FP migration route.

- Based on the estimated FP migration route mentioned above, it is presumed the gaps between shield plugs have high radiation levels. It is also presumed the entire reactor well, located on the migration route and upstream of the operation floor, is also highly contaminated.

- In the pressure vessel, FPs are believed to have attached to the structures and wall surfaces; therefore, high radiation is expected.



- The dose near the pedestal opening (at approx. 1 m from the floor surface) was within 9.4 Sv/h (Mar 20-22, 2017)

- On the 1F grating: 4.7 Sv/h - 9.7 Sv/h (Apr 10, 2015)

- It is presumed that the S/C inner wall is highly contaminated since the area in the vicinity of the atmospheric control piping, which was used for PCV ventilation, is highly contaminated.

- The dose increased toward a deposits and reached about several Sv/h to more than 10 Sv/h at a location about 0.3–1.6 m above the floor (Mar 18, 19, 22, 2017)

- The max. dose was measured at the upper water level in the torus room, at 920 mSv/h. (Feb 20, 2013)
- High contamination is estimated on the building wall, the S/C wall, or the stagnant water in the S/C.

Unit 2: Diagram of Summarized Information

- **Measurement results and observation data available**
- *Estimated based on the measurement results and observation data*
- Accident analysis or qualitative estimation

Reactor well

- *Likewise in Unit 3, the gap between the top and middle layers of the shield plug may have highly-contaminated areas.*
- *In our estimation, the lower layer and the entire reactor well are also highly contaminated.*
- Particulate FPs may exist.

Operation floor

- **Max dose: Approx. 880 mSv/h (Jun 30, 2012)**
- *The dose is distributed along the direction from the well to the blowout panel.*

Upper part of the pressure vessel

- **At some location, the temperature is higher than in the lower section of the RPV.**
- In our estimation, a lot of FPs are attached.
- The chemical forms of the attached FPs are unknown (ex. solubility/insolubility, etc.).
- The degree of re-evaporation of the attached FPs is unknown.

In-core structures

- **The PLR system pressure rose when the FDW flow rate increased, presumably because the water level outside the shroud rose.**
- **It is presumed that the shroud still exists because the temperature dropped as a result of CS water injection and also because the water level rose outside the shroud when the amount of water injection was increased.**
- Analysis results indicate that severe deformation is likely to occur at 1,000° C or greater.
- FPs are figured to be attached to the separator and dryer.
- Cesium may have been absorbed into the oxide layer of the steel materials.
- Cesium may have combined with molybdenum, boron and silicon.

Reactor building (1F-4F)

- **The ambient dose rate was several to tens of mSv/h (Apr 2011~ Feb 2014).**
- **High radiation was locally observed at the penetration.**
- **No significant contamination was detected in the TIP room, at several mSv/h (Mar 2012).**

PCV top head

- *The sealing degraded, resulting in a leak opening (discharging vapor and FPs).*

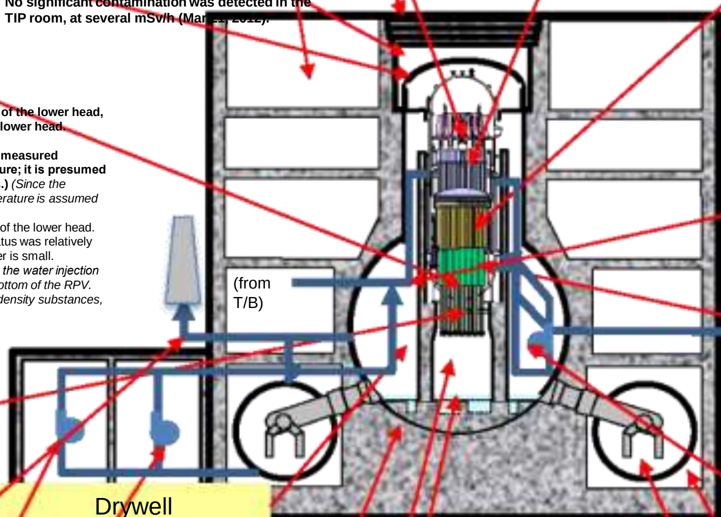
Reactor core

- **There is an obstacle inside the TIP piping.**
- **SEM analysis on the obstacle in the TIP piping detected Fe, Cr, Ni and Mn, which are the elements of iron and steel materials, apart from Zr, an element of the in-core structures and fuel cladding.**
- Zero water level in the reactor core area.
- Analysis often finds the remaining volume to be zero.
- Debris may get solidified at the upper portion of the speed limiter inside the CR guide tube and may still exist there (if the CRD piping is sound).
- *Muon measurement indicates there may be fuel in the peripheral area.*

RPV lower head

- **Because the CRD and cables remain in the peripheral area of the lower head, there is no massive damage such as collapse of the entire lower head.**
- **No water level is formed in the RPV, indicating ruptures.**
- **A thermometer was installed through the SLC piping. (The measured temperature was higher than the injected water's temperature; it is presumed that the temperature inside the RPV rose due to fuel debris.)** *(Since the measured temperature > injected water temperature, the temperature is assumed to have risen due to the fuel debris inside the RPV.)*
- The ruptures presumably exist at the center and the peripheral of the lower head.
- Temperature response to the changes in the water injection status was relatively fast, from which it is presumed that the amount of retained water is small.
- *The RPV and PCV's temperature responses to the reduction in the water injection volume in Mar-Apr. 2017 indicate fuel debris may exist at the bottom of the RPV.*
- *From the muon measurement results, it is presumed that high-density substances, seemingly fuel debris, exist.*

- **As a result of water injection using the FDW system alone, the temperature tended to increase by a few degrees C at some part of the pressure vessel bottom.**



FDW/CS piping

- **A temperature decrease was observed at the start of CS water injection.**
- *Existence of debris is estimated along the route from the CS through the reactor core to the lower plenum.*

RPV lower plenum

- **An obstacle exists in the SLC piping inside the RPV.**
- **A thermometer was installed through the SLC piping.** *(Since the measured temperature > injected water temperature, the temperature is assumed to have risen due to the fuel debris inside the RPV.)*
- The heat balance evaluation indicated that about 30% still exists.

PLR

MS piping

Turbine building

- **The ambient dose rate was several tens of μSv/h (Apr 2011 through Feb 2014).**
- **High radiation was detected in the underground floor (Apr 2011 through Feb 2014).**
- *Highly contaminated water is assumed to have flowed in through the rupture at the bottom of the S/C.*

CRD

- **The CRD and cables still exist in the lower head periphery.**
- **There was a deposits on the CRD replacement rail.**
- **There was no visible damage to the CRD housing support.**

Drywell

- **The ambient dose rate is higher compared to other units (70 Sv/h).**
- **Fallen powdery matter exists.**
- **The Cs concentration level in the stagnant water in the PCV is lower than that in the stagnant water in the building, while the Sr and H3 are at the same level.**
- From the pressure and oxygen concentration levels, it is presumed that leakage in the gas phase is relatively small.
- **The max reading of the D/W CAMS was 138 Sv/h at 16:10 on Mar 15, 2011.** *The trend of CAMS readings suggests that the maximum dose of 138 Sv/h occurred because the pressure vessel was damaged and fuel debris fell into the containment.*
- **Near the CRD rail, high radiation, ranging from approx. below 10 to 80 Gy/h, was observed (Jan 26 to Feb 9, 2017).**
- **Near the CRD replacement rail and the pedestal opening, a dose ranging from approx. 24 to 36 Sv/h was observed (Aug 12, 2013).**

RPV pedestal

- **The CRD and cables still exist in the lower head periphery.**
- **There was no visible major damage to the existing structure (the CRD exchanger)**
- **Water injected in the RPV is falling to the pedestal.**
- **There is defective grating on the platform in the pedestal (possibly due to the heat effect from fuel debris). There are attached depositss at various spots.**
- **Near the pedestal inner wall, the dose is below approx. 10 Gy/h (Jan 30, 2017).**
- **There is no visible major damage to the pedestal inner wall surface**
- **There is no visible change near the pedestal inner wall (on the CRD rail side) at the height from the platform to an area about 2 m below it, at the dose rate level of 7–8 Gy/h, and the temperature at about 21° C.**

Torus room

- **The S/C and torus room have almost the same water level.**
- **The difference in the water levels comes from the difference between the S/C pressure and atmospheric pressure.**
- *There is some kind of penetration exists between the R/B and turbine building.*
- **The inspection found air dose of 4.3–134.0 mSv/h and the water dose of 18.7–23.7 mSv/h (Apr 2013).**
- **Cs134 and Cs137 were detected from the collected stagnant water. On the order of 10^4Bq/cm³ (Apr 12, 2013).**
- **The inspection found no water leakage near the vent pipe.**

PCV vent/exhaust stack

- **High contamination, ranging from several to tens of Sv/h, was detected in the joint exhaust stack for Units 1s and 2, the vent line for Unit 2, and the SGTS for Unit 2 (Mar 26-27, 2012).**
- **High contamination, ranging from several tens of mSv/h to about 1 Sv/h, was detected in the lower part of the exhaust stack, and from 500 mSv/h to 2 Sv/h in the SGTS piping joint (Aug 6, 2014 to Oct 21, 2015).**
- *Assuming no damage to Unit 2's rupture disk, it is likely to have originated from Unit 1.*
- Contamination stays at the same low level both at the front and back of the rupture disk. *Hence, it is figured that the ventilation was unsuccessful.*

Suppression chamber

- **The S/C and torus room have almost the same water level.**
- *There is a rupture at the lower level of the S/C (estimated to be O.P.512 or less).*
- After the core damage occurred in Unit 2, forced depressurization was performed using the SRV, and a large amount of FPs are assumed to have migrated to the S/C, as observed by the S/C CAMS. Nonetheless, the FPs stuck in the pooled water are believed to have migrated to the building through a rupture, which is assumed to exist at the S/C's lower level.

RPV pedestal floor

- **Part of the fuel assembly was found to have fallen to the bottom of the pedestal,** *and the depositss found in its surrounding area are assumed to be fuel debris.*
- **depositss detected on the pedestal floor** *may contain fuel debris.*

Drywell floor

- **No debris was found during the internal PCV investigation (except for a small number of depositss).**
- **Water level: Approx. 30 cm (located at the lower edge of the PCV vent pipe)**
- *Because no water has been accumulated in the S/C, no water accumulation is expected in the D/W.*

CS pump

RCIC/HPCI

Estimate Diagram of Debris Distribution and the RPV and PCV Conditions in Unit 2

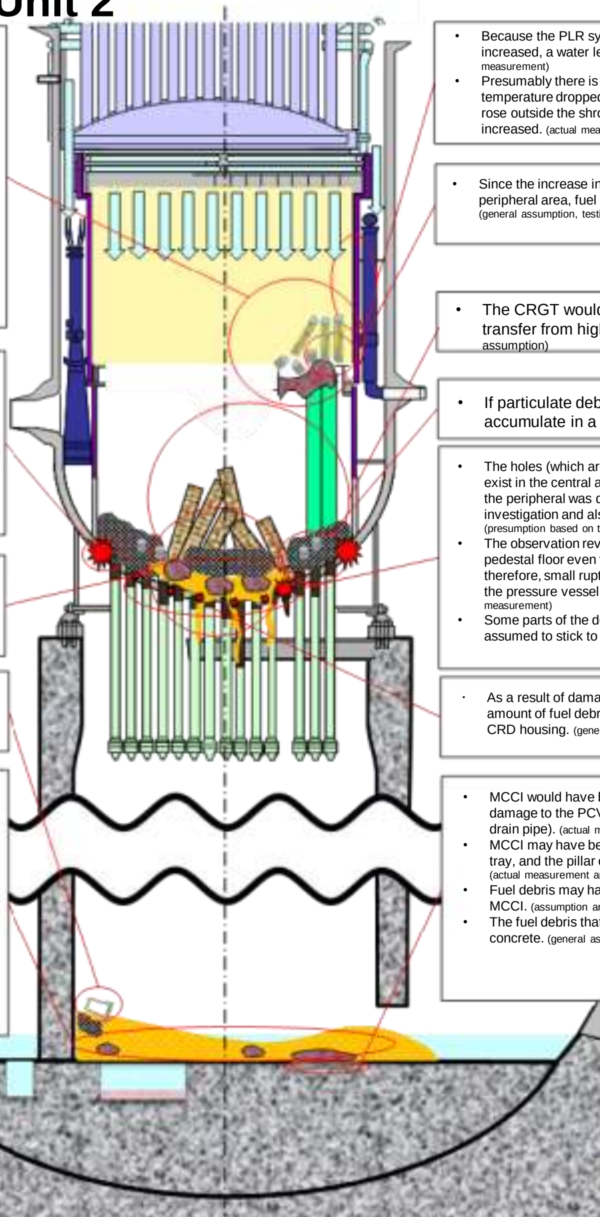
- By estimating the energy amount, based on the increased PCV pressure from hydrogen generation, most fuel is assumed to have been damaged or collapsed. (actual measurement and analysis)
- Since the temperature decreased during the CS water injection, fuel is assumed to exist in the reactor core periphery, where low-flow water injected from the CS could reach (yet the detailed location of debris cannot be estimated because a similar behavior is expected if molten fuel fell onto the fuel support metal fittings or CRGT and was solidified, which could also act as a heat source). (actual measurement)
- Muon measurement indicates fuel may exist in the reactor core periphery. (actual measurement)
- Even if some fuel rods are present, they should be just a small portion of the peripheral area. (general assumption)
- It is estimated to be general oxide debris from molten fuel solidification. (general assumption)

- The upper tie plate fell onto the pedestal periphery, indicating a pressure vessel rupture in the peripheral area. (presumption based on the actual measurement)
- It is at least presumed that there is a rupture large enough for the upper tie plate to fall through. (actual measurement)
- The CRGT and CRD in the peripheral area may have partially melted and collapsed due to fuel debris accumulated at the pressure vessel bottom. (presumption based on the actual measurement)

- Muon measurement showed shadows of high-density substances, which appear to be fuel debris, located at the pressure vessel bottom. The fuel dropped to the lower plenum may still exist at the bottom of the RPV. (actual measurement)

- The fallen upper tie plate was found on the pedestal floor, and the depositss in its surroundings are assumed to be fuel debris that fell through the RPV ruptures, just as the tie plate did. (actual measurement)

- There is no observable variation in the dose and temperature in the area from the pedestal floor to the platform, and no notable damage has been found in the pedestal sub-structure; therefore, the fuel debris on the pedestal floor seems to have relatively low dose and decay heat, and may contain large quantities of metals. (presumption based on the actual measurement)
- depositss containing fuel debris are assumed to have spread all over the pedestal bottom. (actual measurement)
- A pool of water on the PVC floor, if any, results in particulate debris formation. (general assumption)
- Particulate debris, if any, is likely to be found in a stagnant spot. (general assumption)



- Because the PLR system pressure rose when the FDW flow rate increased, a water level may be formed outside the shroud. (actual measurement)
- Presumably there is no major damage to the shroud because the temperature dropped by CS water injection, and the water level rose outside the shroud when the amount of water injection was increased. (actual measurement)

- Since the increase in fuel temperature might not be so great in the peripheral area, fuel rod debris and pellets may still exist there. (general assumption, testing, and analysis)

- The CRGT would remain un-melted with little heat transfer from high-temperature fuel debris. (general assumption)

- If particulate debris or pellets are present, they may accumulate in a stagnant spot. (general assumption)

- The holes (which are not so big) in the RPV are estimated to exist in the central and peripheral areas because the CRD in the peripheral was detected during the internal PCV investigation and also because of the damage to the grating. (presumption based on the actual measurement)
- The observation revealed that water drops fell all over the pedestal floor even though its severity varies by location; therefore, small ruptures may exist near the CRD housing at the pressure vessel bottom. (presumption based on the actual measurement)
- Some parts of the debris that fell through the ruptures are assumed to stick to the CRD housing, etc. (general assumption)

- As a result of damage to the CRGT and CRD housing, a small amount of fuel debris or molten metal may have entered the CRD housing. (general assumption and testing)

- MCCI would have been limited as there are no signs of damage to the PCV shell (no leakage from the sand cushion drain pipe). (actual measurement)
- MCCI may have been limited since the pedestal wall, the cable tray, and the pillar of the CRD exchanger stay un-melted. (actual measurement and general assumption)
- Fuel debris may have solidified, in most cases, without causing MCCI. (assumption and analysis based on the actual measurements)
- The fuel debris that caused MCCI may have been mixed with concrete. (general assumption)

The dash-dotted line shows the inside of the containment is asymmetrical with the pedestal.

Legendss

- Remaining fuel rod and its debris
- Oxide debris (porous)
- Particulate debris
- Fuel debris (rich in metals)
- Concrete mixed debris
- CRGT
- Damaged CRGT
- CRD
- CRD (with debris inside)
- Shroud
- Damaged shroud*
- Pellet
- Ruptured hole of the RPV
- Upper tie plate
- deposits (material unknown)
- Ballooning fuel*
- Oxide debris*
- Heavy metal debris*
- Particulate pellet*
- Cladding residue*
- Molten reactor internals*
- Solidified B4C*
- Control rod mixed melt*

* Not used in the estimate diagram of Unit 2

IRID IAE

Unit 2: Estimate Diagram of FP Distribution

*The estimate was made focusing on cesium, a major source nuclide

- Gas may have leaked from the pressure vessel to the D/W during the accident progression, thereby allowing FPs to easily migrate directly to the D/W side. This would explain the dose measurement data in the D/W and S/C in Unit 2.

- On the Unit 2 operation floor, high radiation was detected at the shield plug. In addition, the pictures at the time of the accident show a large amount of vapor was released from the blowout panel. Therefore, FPs are assumed to have been discharged through the following route: pressure vessel -> containment -> pressure vessel top head flange -> reactor well -> shield plug -> operation floor. It was also figured that FPs exist unevenly along the FP migration route.

- Insoluble cesium particles, which seem to have originated from Unit 2, were detected in the environment (the particle diameter is about a few μm). In the sample analysis of the protective covering just above the shield plug, particles appearing similar to the insoluble cesium particles were found; based on this, the reactor well may have been one of the routes through which the insoluble cesium particles were discharged. From the protective covering, multiple types of uranium-bearing particles were detected.

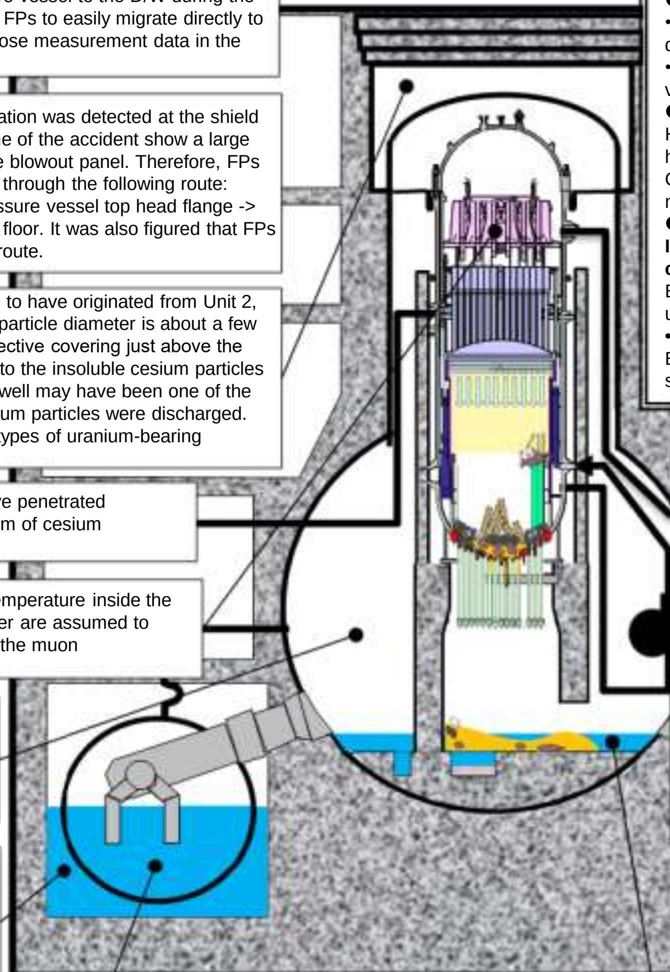
- Hydrated cesium compounds may have penetrated through the concrete surface in the form of cesium hydroxide.

- Although the fuel melt caused a high temperature inside the pressure vessel, the separator and dryer are assumed to have maintained their shape based on the muon measurement results of Unit 2.

- The internal PCV investigation detected localized high radiation, suggesting an FP concentration mechanism.

- Cs134 and Cs137 were detected from the collected stagnant water in the torus room. (Apr 12, 2013)
  Cs134: 1.3E + 04 Bq/cm^3
  Cs137: 2.4E + 04 Bq/cm^3

- Based on the accident progression in Unit 2, a large number of FPs are assumed to have migrated to the S/C via the SRV. Nonetheless, most of the FPs stuck in the pooled water are estimated to have moved out of the containment through the lower section of the S/C, and it is presumed that they scarcely exist in the S/C water.



- Cesium possibly remains inside the fuel debris although its amount is expected to be small.

- FPs may unevenly exist in fuel debris.

<Chemical forms and characteristics of major types of cesium estimated>

● Cesium iodide, Cesium hydroxide, Cesium chloride

• Easily evaporate and tend to escape from the pressure vessel as vapor due to a difference in pressure or concentration.

• Have a hydration property and tend to move with moisture by, for example, water vapor condensation or dew condensation (on the wall surface).

● Cesium molybdate, Boric acid cesium

Have a lower vapor pressure compared to the afore-mentioned substances and, hence, tend to stay inside a pressure vessel.

Cesium molybdate may form because of a reaction between cesium and steel material containing molybdenum.

● Compound of silicon and cesium

Insoluble cesium particles (amorphous particles including cesium with its main component of silicon oxide)

Believed to be difficult to hydrate, and particles the size of a few microns tend to soar up.

• Products of reaction with steel material (crystalline substances)

Believed to be difficult to hydrate, and tend to stay in the oxide layer on the steel surface when in the 800–1,000° C temperature range.

- As the fuel temperature increased, boric acid in the control rod material was possibly taken into the steel melt as a result of the eutectic reaction with steel materials, suppressing the generation of boric acid.
- Generally, as the fuel temperature goes higher, molybdenum evaporation becomes significant, and cesium molybdate would be generated; as a result, the ratio of cesium staying inside the RPV would increase. Nonetheless, in Unit 2, cesium possibly moved out of the pressure vessel more easily because stable cesium chloride was produced by an injection of seawater. This could be one reason for the high radiation currently observed in the containment.

- In the early stage of the core heating, insoluble cesium particles were generated by a coagulation reaction between silicon oxide and cesium hydroxide in the gas phase. The produced amount would be limited to the order of 10 kg.
- Some part of the insoluble cesium particles may still exist inside the containment or the reactor building.

- Depending on the temperature/atmosphere history, some cesium in the vapor phase inside the pressure vessel is thought to be taken into the oxide layer of the steel material due to the reaction with silicon oxide on the steam-oxidized steel surface. Considering this factor, the high radiation may have occurred at the separator and dryer, which have large surface areas.

- In Unit 2, the water inside the D/W has leaked out of the containment through the lower section of the S/C. Cs concentration in the water inside the D/W should be low because the water is circulating.
- Actual measurements (Aug 7, 2013)
  Cs134: 2.14E + 03 Bq/cm^3
  Cs137: 4.38E + 03 Bq/cm^3

Unit 2: Estimate Diagram of Dose Distribution

- 401 mSv/h–880 mSv/h on the shield plug on the operation floor (Jun 13, 2012)

- On the Unit 2 operation floor, high radiation was detected at the shield plug. In addition, the pictures at the time of the accident show a large amount of vapor was released from the blowout panel. Therefore, FPs are assumed to have been discharged through the following route: pressure vessel -> containment -> pressure vessel top head flange -> reactor well -> shield plug -> operation floor. It was also figured that FPs exist unevenly along the FP migration route.

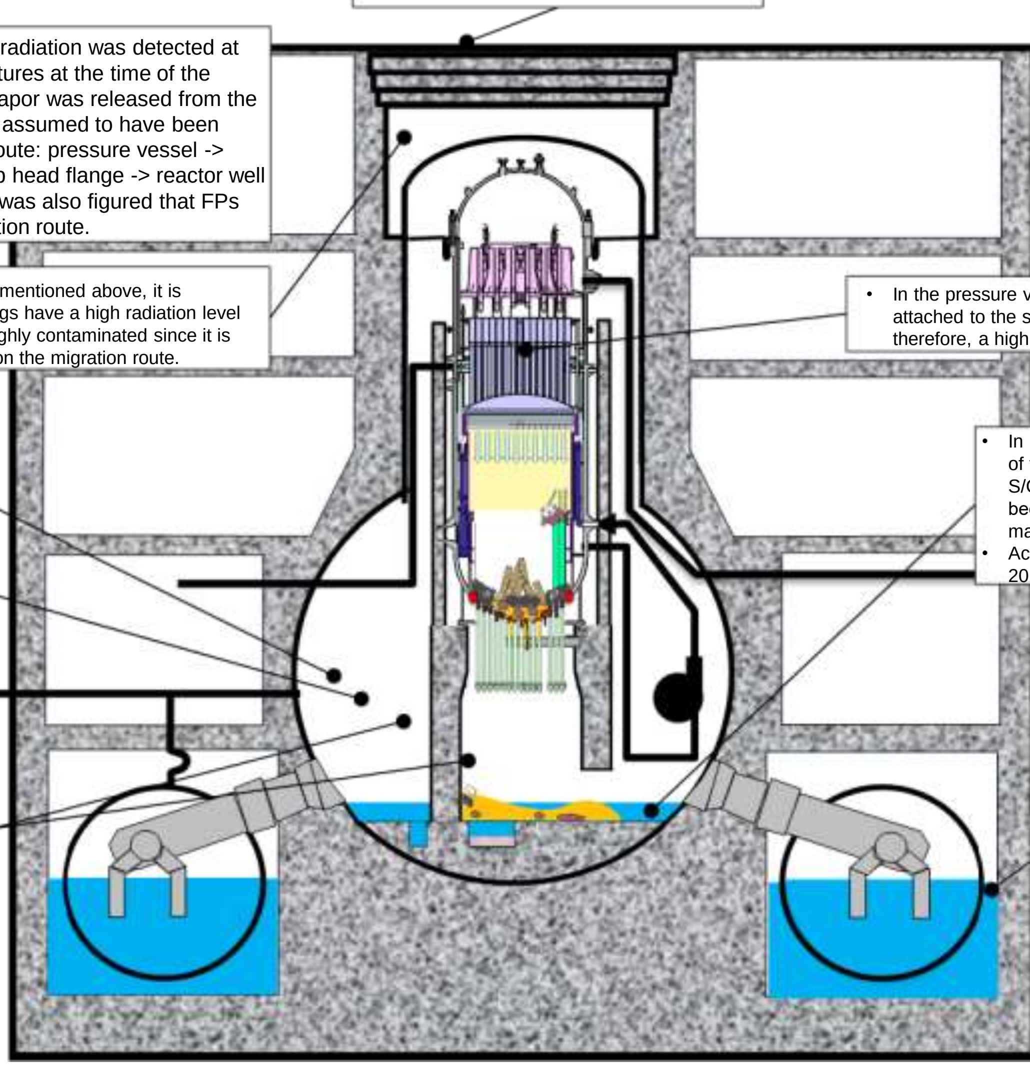
- Based on the estimated migration route mentioned above, it is presumed that the gaps in the shield plugs have a high radiation level and that the entire reactor well is also highly contaminated since it is located upstream of the operation floor on the migration route.

- In the pressure vessel, FPs are believed to be attached to the structures and wall surfaces; therefore, a high dose is expected.

- In Unit 2, the water inside the D/W has leaked out of the containment through the lower section of the S/C. The dose in the D/W water is presumably low because the water is flowing through the building's main circulation line.
- Actual measurement: 1.0 mSv/h or less (Aug 7, 2013)

- 31.1 Sv/h–72.9 Sv/h in the gas phase in the D/W (Mar 27, 2012)

- 24 Sv/h–36 Sv/h on the CRD replacement rail (Aug 12, 2013)



The following doses were detected during the internal PCV investigation.
- Approx. below 10 Gy/h ~ approx. 70 Gy/h at the space in the upper portion of the CRD rail (Jan 26, 2017)
- Approx. 10 Gy/h or below near the pedestal inner wall (Jan 30, 2017)
- Around 7–8 Gy/h near the pedestal inner wall (on the CRD rail-side), from the platform level to about 2 m below it (Feb 2018)
- Approx. 70 Gy/h to approx. 80 Gy/h on the CRD rail (Feb 9–16, 2017)

- 4.3–134 mSv/h in the gas phase in the torus room (Apr 11, 2013)
- 18.7–23.7 mSv/h in the liquid phase in the torus room (Apr 11, 2013)

Unit 3: Diagram of Summarized Information

- **Measurement results and observation data available**
- *Estimated based on the measurement results and observation data*
- Accident analysis or qualitative estimation

Reactor well

- *A highly contaminated area is assumed to exist in the gap between the top and middle layers of the shield plug.*
- *High contamination is also estimated at its bottom layer as well as the whole reactor well.*

Operation floor

- **The max dose is approx. 2 Sv/h on the shield plug (2013)**
- **A steam-like substance was observed at the shield plug gap. The temperature there was about 10° C higher than in its surroundings.**
- **The gate of the spent fuel pool is partially deformed. High radiation was observed in its vicinity.**
- Airborne radiation in the shield plug gap and joint on the operation floor: Approx. 200–300 mSv/h (2013–2014)
- **The CUW F/D hatch fell into the pool** *(possibly due to a hydrogen explosion).*
- The center section of the shield plug is deformed *(approx. 300 mm).*
- **Radionuclides contributing to the dose: Cs-137 and Cs-134.**
- **Cs-bearing particles were observed during the dust filter inspection. From the results of the elemental composition analysis, it us presumed that they originated from Unit 2.**

Upper part of the pressure vessel

- A lot of FPs are believed to have attached.
- Chemical forms of the attached FPs are unknown (ex. solubility/insolubility, etc.).
- The degree of re-evaporation of the attached FPs is unknown.

In-core structures

- Analysis results indicate the structures' downward shift because of creep and support deformation.
- FPs seem to be attached to the separator and dryer.
- Cesium has possibly been taken into the steel materials' oxide layers.
- Cesium may have been combined with molybdenum, boron, and silicon.

Reactor core

- **A temperature decrease was observed at the start of CS water injection (Sep 2011).**
- *Existence of debris is estimated along the route from the CS through the reactor core to the lower plenum.*
- **Meanwhile, no temperature increase occurred when the CS system was stopped for 15 days (Dec 2013); instead, the FDW system alone was used for water injection.**
- **The temperature increase occurred when the amount of water injection was increased in May 2011.**
- **No water level has been detected in the RPV.**
- Debris may have solidified and, hence, still may exist in the upper portion of the speed limiter inside the CR guide tube (if the CRD piping was sound).
- **The muon measurement results (interim report) indicate part of the fuel debris still may exist, but no presence of huge, high-density material has been detected in the reactor core (Jul 2017).**
- **The muon measurement results indicate there may be no large mass of fuel debris in the original reactor core area.**

Reactor building (1F-4F)

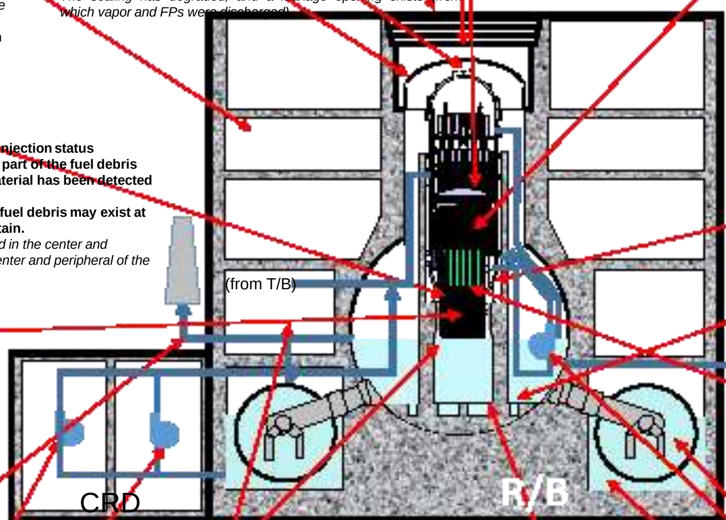
- **The ambient dose rate was several tens of mSv/h.**
- **Local high radiation was observed at the equipment hatch.**
- **The door of the TIP room was blown off.**
- *The shielding plug at the equipment hatch shifted, presumably due to propagated pressure from a hydrogen explosion in the building.*
- **The building's 4th floor was damaged due to a hydrogen explosion (the northwest section suffered significant damage).**

PCV top head

- *The sealing has degraded, and a leakage opening exists (from which vapor and FPs were discharged).*

RPV lower head

- **Slow temperature response to the changes in the water injection status**
- **The muon measurement results (interim report) indicate part of the fuel debris still may exist, but no presence of huge, high-density material has been detected at the pressure vessel bottom (Jul 2017).**
- **The muon measurement results suggest that part of the fuel debris may exist at the bottom of the RPV; however, the possibility is uncertain.**
- *Inside the pedestal, water surface fluctuations were observed in the center and peripheral of the RPV; therefore, ruptures may exist in the center and peripheral of the RPV.*



Drywell floor

- **The water level was approx. 6.5 m from the D/W floor (leakage has occurred from the MSIV room).**
- *From the water level behavior in the D/W, it is figured that there has been almost no leakage from the lower part.*

FDW/CS piping

- **CS water injection started (in Sep 2011)**
- **Water injection from the CS system was stopped for 15 days (Dec 2013).**
- **The volume of water injection was reduced (Feb-Mar 2017).**

CRD

- **The contact behavior of the control rod Position Indicator Probes (PIPs) was checked, and there were no PIPs that showed sound conduction reaction (Sep 2011).**
- *The CRD or cables are presumably severely damaged.*
- **A deposits of solidified molten material was observed on the metal fittings to support the CRD housing.**
- **A structure that appears to be the CRGT was observed.**

RPV lower plenum

- *The measured temperature was higher than the injected water's temperature; it is presumed that the temperature inside the RPV rose due to fuel debris.*
- The temperature at the RPV lower part was higher than the PCV temperature in the past, but currently their readings are almost at the same level.

MS piping

- **Leakage occurred around the extension joint of the main vapor pipe D.**
- *The leak location is presumably the main vapor pipe D only, according to the camera inspection and to the observation of the water flow on the floor.*

PCV vent/exhaust stack

- **The dose at the SGTS filter train was several mSv/h (Dec 2011).** *Contamination is presumably low, and there would be no backflow through the filter train.*
- *The estimate suggests that, due to the effect of the vent in Unit 3, hydrogen flowed into Unit 4 and caused the explosion in the Unit 4 building.*

Drywell

- **The internal investigation showed no damage to the structures around the X-53 penetration.**
- **The airborne radiation in the D/W was relatively low (approx. 1 Sv/h).**
- **Cs concentration in the stagnant water in the PCV is lower than that of the stagnant water in the building, while Sr and H3 are at the same level. An alpha nuclide was detected (Oct 2015).**
- *Based on the containment oxygen concentration (approx. 8%) and the D/W pressure, Unit 3 is assumed to have the highest leak level in the PCV's gas phase among other units (presence of air in-leak is assumed).*
- **The temperature measured during the internal PCV investigation was approx. 26–27° C in the gas phase and approx. 33–35° C in the water.• Cs134 concentration [Bq/cm³]: 4.0E + 2 (near the water surface), 2.3E + 2 (approx. 0.7 m below the water surface)**
- **Cs137 concentration [Bq/cm³]: 1.6E + 3 (near the water surface), 9.4E + 2 (approx. 0.7 m below the water surface)**

PLR

Torus room

- **The water level in the torus room was about O.P.3000.**
- **No corrosion was observed on the catwalk handrails.**
- *Water vapor leaking into the torus room is presumably minor.*
- *Given the stagnant water level, there may be some kind of penetration between R/B and turbine building.*

Turbine building

- **Ambient dose rate was several-hundred μSv/h to several mSv/h. • There was high radiation in the underground floor.**
- *Highly contaminated water presumably flowed into the turbine building from the MSIV room during the early phase of the accident.*

CS

RCIC/HPCI

- **Cs concentration in the stagnant water in the HPCI room (Feb 2017)**

Cs-134: 6.8E + 7 [Bq/l]
Cs-137: 4.3E + 8 [Bq/l]

RPV pedestal

- Due to the dropped fuel debris, damages are estimated on the grating, TIP piping, CRD exchanger, etc.
- Water injected into the RPV **is presumably dropping into the pedestal.**
- **Damaged multiple structures were observed inside the pedestal, with some substances that seem to be solidified molten material (Jul 2017).**
- **Structures that are expected to be, but not limited to, a speed limiter or upper tie plate were found.**
- **Coarse surfaces were observed on some walls inside the pedestal.**
- **A substance that is seemingly a result of molten material solidification was observed near a pedestal wall.**
- **Cables of some thermometers at the RPV bottom were found to be broken.** *Those thermometers are believed to be measuring PCV temperatures.*

RPV pedestal floor

- *Hydrogen caused by MCCI may have contributed to the hydrogen explosion in Unit 3, and fuel debris may have dropped to the pedestal floor.*
- **Unlike Unit 1, no contamination has been detected in the RCW system.**
- **Multiple fallen objects and pebble or sand depositss were detected in the lower part of the pedestal.**
- **A deposits was observed near the workers' access opening in the pedestal.**

Suppression chamber

- No water level has been detected in the S/(but it is presumed to be almost full).

Estimate Diagram of Debris Distribution and the RPV and PCV Conditions in Unit 3

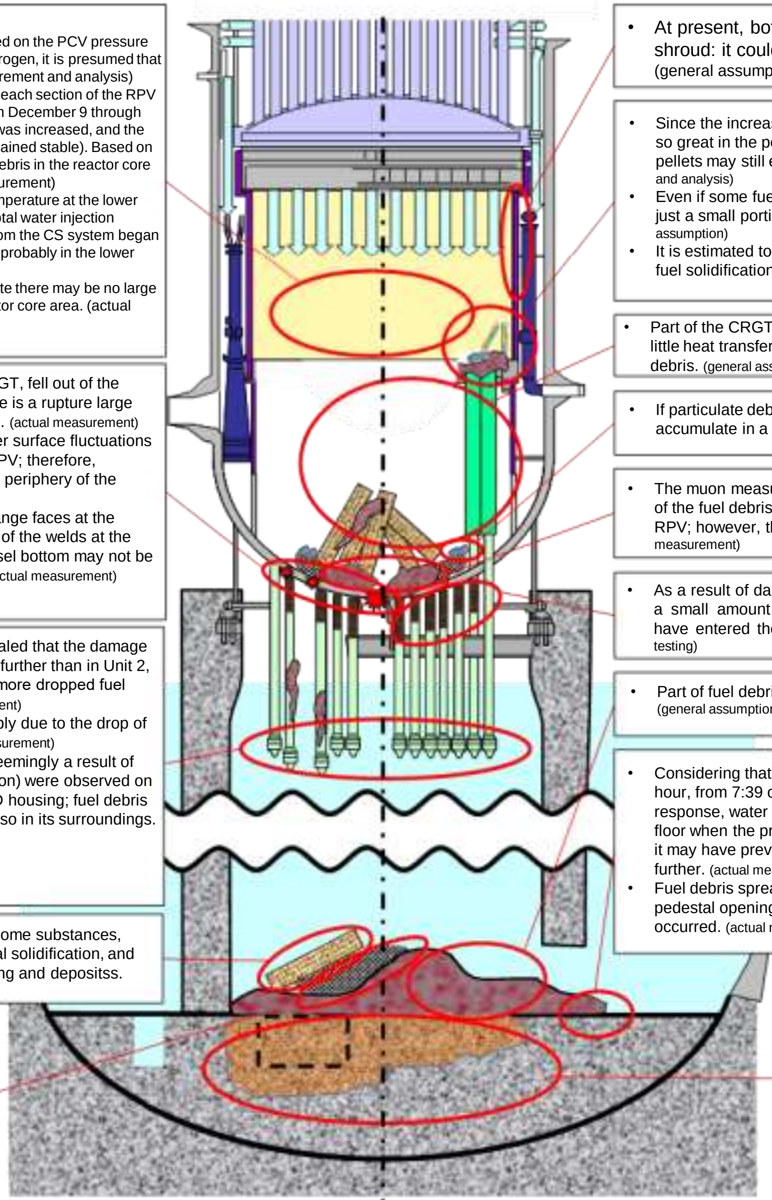
- From an estimate energy amount based on the PCV pressure increase caused by the generated hydrogen, it is presumed that most of the fuel melted. (actual measurement and analysis)
- There was no temperature increase in each section of the RPV when the CS system was stopped from December 9 through 24, 2013 (the flow rate from the FDW was increased, and the total volume of water injection had remained stable). Based on this finding, there should be little fuel debris in the reactor core (less than that in Unit 2). (actual measurement)
- In addition to the above finding, the temperature at the lower part of the RPV decreased when the total water injection volume increased as water injection from the CS system began (on Sep 1, 2011); hence, fuel debris is probably in the lower plenum. (actual measurement)
- The muon measurement results indicate there may be no large mass of fuel debris in the original reactor core area. (actual measurement)

- Because a structure, seemingly CRGT, fell out of the pressure vessel, it is presumed there is a rupture large enough for the CRGT to fall through. (actual measurement)
- Inside the pedestal, there were water surface fluctuations in the center and periphery of the RPV; therefore, ruptures may exist in the center and periphery of the RPV. (actual measurement)
- Since there is a gap between the flange faces at the lower part of the CRD housing, part of the welds at the CRD housing and the pressure vessel bottom may not be firmly fixed. (presumption based on the actual measurement)

- The internal PCV investigation revealed that the damage inside the pedestal had progressed further than in Unit 2, suggesting that the unit's PCV has more dropped fuel debris than Unit 2's. (actual measurement)
- Damage to the platform is presumably due to the drop of high-temperature debris. (actual measurement)
- Damage and depositss (the latter seemingly a result of solidified molten material solidification) were observed on the metal fittings to support the CRD housing; fuel debris may exist above and below it and also in its surroundings. (actual measurement)

- The lower part of the pedestal has some substances, seemingly a result of molten material solidification, and other dropped objects such as grating and depositss. (actual measurement)

- A pool of water on the PCV floor, if any, may have formed particulate debris. (general assumption)
- Particulate debris, if any, is likely to be found in a stagnant spot. (general assumption)



- At present, both possibilities exist for the shroud: it could be sound or damaged. (general assumption and analysis)

- Since the increase in fuel temperature might not be so great in the peripheral area, fuel rod debris and pellets may still exist there. (general assumption, testing, and analysis)
- Even if some fuel rods are present, they should be just a small portion of the peripheral area. (general assumption)
- It is estimated to be general oxide debris from molten fuel solidification. (general assumption)

- Part of the CRGT remains un-melted when there is a little heat transfer from high-temperature molten debris. (general assumption)

- If particulate debris or pellets are present, they may accumulate in a stagnant spot. (general assumption)

- The muon measurement results suggested that part of the fuel debris still may exist at the bottom of the RPV; however, the possibility is uncertain. (actual measurement)

- As a result of damage to the CRGT and CRD housing, a small amount of fuel debris or molten metal may have entered the CRD housing. (general assumption and testing)

- Part of fuel debris may have solidified without MCCI. (general assumption)

- Considering that D/W was sprayed for more than an hour, from 7:39 on March 13 as part of the accident response, water may have accumulated on the D/W floor when the pressure vessel suffered damage, and it may have prevented fuel debris from spreading further. (actual measurement and general assumption)
- Fuel debris spread out of the pedestal through the pedestal opening, but presumably no shell attack occurred. (actual measurement and analysis)

- An explosion occurred in Unit 4 and then in Unit 3, possibly due to hydrogen generated by MCCI. (actual measurement)

Legends

- Remaining fuel rod and its debris
- Oxide debris (porous)
- Particulate debris
- Fuel debris (rich in metals)*
- Concrete mixed debris
- CRGT
- Damaged CRGT
- CRD
- CRD (with debris inside)
- Shroud
- Damaged shroud*
- Pellet
- Ruptured hole of the RPV
- Upper tie plate*
- deposits (material unknown)*
- Ballooning fuel*
- Oxide debris*
- Heavy metal debris*
- Particulate pellet*
- Cladding residue*
- Molten reactor internals*
- Solidified B4C*
- Control rod mixed melt*

The dash-dotted line shows the inside of the containment is asymmetrical with the pedestal. * Not used in the estimate diagram of Unit 3

Unit 3: Estimate Diagram of FP Distribution

*The estimate was made focusing on cesium, a major source nuclide

- The gamma-ray spectra measurement on the operation floor in Unit 3 indicated high dose levels in the shield plugs' gaps and joints. The pictures at the time of the accident also show a large amount of vapor was discharged from the damaged reactor building.
- Therefore, it is presumed that FPs were discharged through the following route: pressure vessel -> containment -> containment top head flange -> reactor well -> shield plug -> operation floor. It was also figured that FPs exist unevenly along the FP migration route.

<Chemical forms and characteristics of major types of cesium estimated>

● Cesium iodide, Cesium hydroxide, Cesium chloride

•Easily evaporate and tend to escape from the pressure vessel as vapor due to a difference in pressure or concentration.

•Have a hydration property and tend to move with moisture by, for example, water vapor condensation or dew condensation (on the wall surface).

● Cesium molybdate, Boric acid cesium

Have a lower vapor pressure compared to the afore-mentioned substances and, hence, tend to stay inside a pressure vessel.

Cesium molybdate may form because of a reaction between cesium and steel material containing molybdenum.

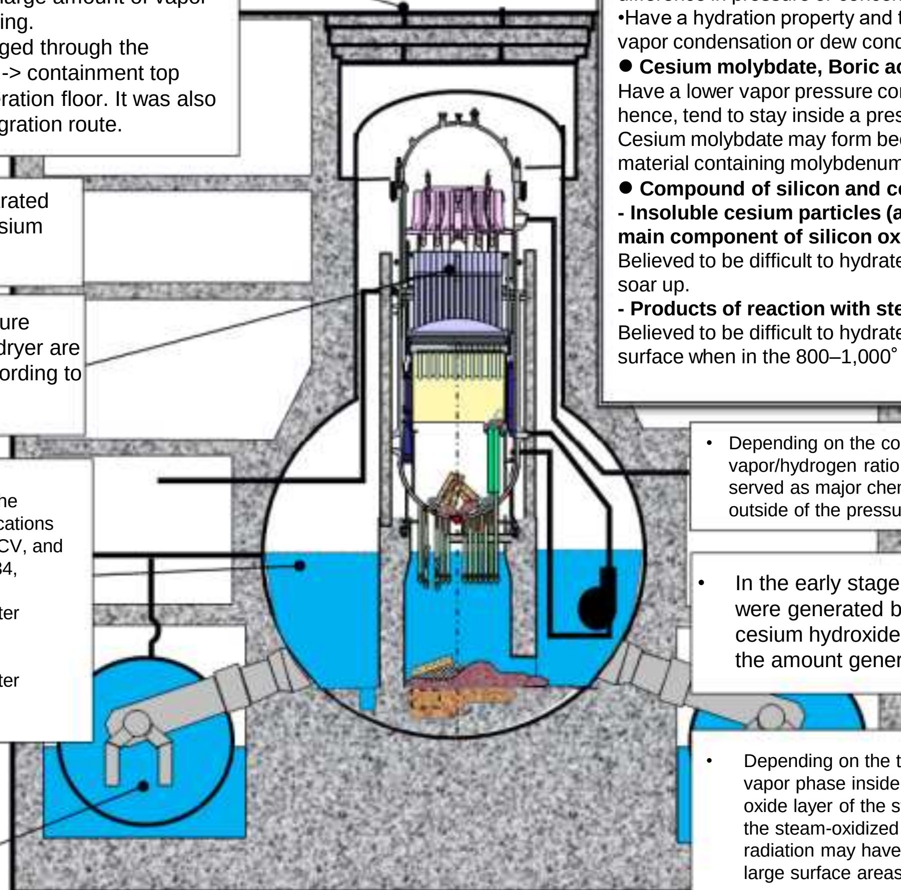
● Compound of silicon and cesium

- Insoluble cesium particles (amorphous particles including cesium with its main component of silicon oxide)

Believed to be difficult to hydrate, and particles the size of a few microns tend to soar up.

- Products of reaction with steel material (crystalline substances)

Believed to be difficult to hydrate, and tend to stay in the oxide layer on the steel surface when in the 800–1,000° C temperature range.



- Hydrated cesium compounds may have penetrated through the concrete surface in the form of cesium hydroxide.

- Although the fuel melt caused a high temperature inside the pressure vessel, the separator and dryer are highly likely to have retained their shapes, according to the muon measurement results of Unit 3.

- The level of activity at 0.7 m below the surface of the stagnant water in the PCV was lower than that near the water surface. The specimens collected at the two locations near the water surface of the stagnant water in the PCV, and about 0.7 m below the water surface, contained Cs134, Cs137, tritium, and Sr90.(Oct 30, 2015)
- Cs134 concentration [Bq/cm^3]: 4.0E + 2 (near the water surface), 2.3E + 2 [7] (approx. 0.7 m below the water surface)
- Cs137 concentration [Bq/cm^3]: 1.6E + 3 (near the water surface), 9.4E + 2 [7] (approx. 0.7 m below the water surface)

- Depending on the conditions of the gas phase, such as the water vapor/hydrogen ratio, cesium molybdate or boric acid cesium may have served as major chemical species, suppressing their discharge to the outside of the pressure vessel.

- In the early stage of the core heating, insoluble cesium particles were generated by coagulation reaction between silicon oxide and cesium hydroxide in the gas phase; however, it was estimated that the amount generated is smaller than that in Unit 2.

- Depending on the temperature/atmosphere history, some cesium in the vapor phase inside the pressure vessel is thought to be taken into the oxide layer of the steel material due to the reaction with silicon oxide on the steam-oxidized steel surface. Considering this factor, the high radiation may have occurred at the separator and dryer, which have large surface areas.

- The FPs that migrated to the S/C and got stuck in the pooled water may still exist in the stagnant water.
- A huge amount of FPs may still exist, given the large volume of stagnant water in the S/C.

- Cesium possibly remains inside the fuel debris although its amount is expected to be small.

- FPs may unevenly exist in fuel debris.

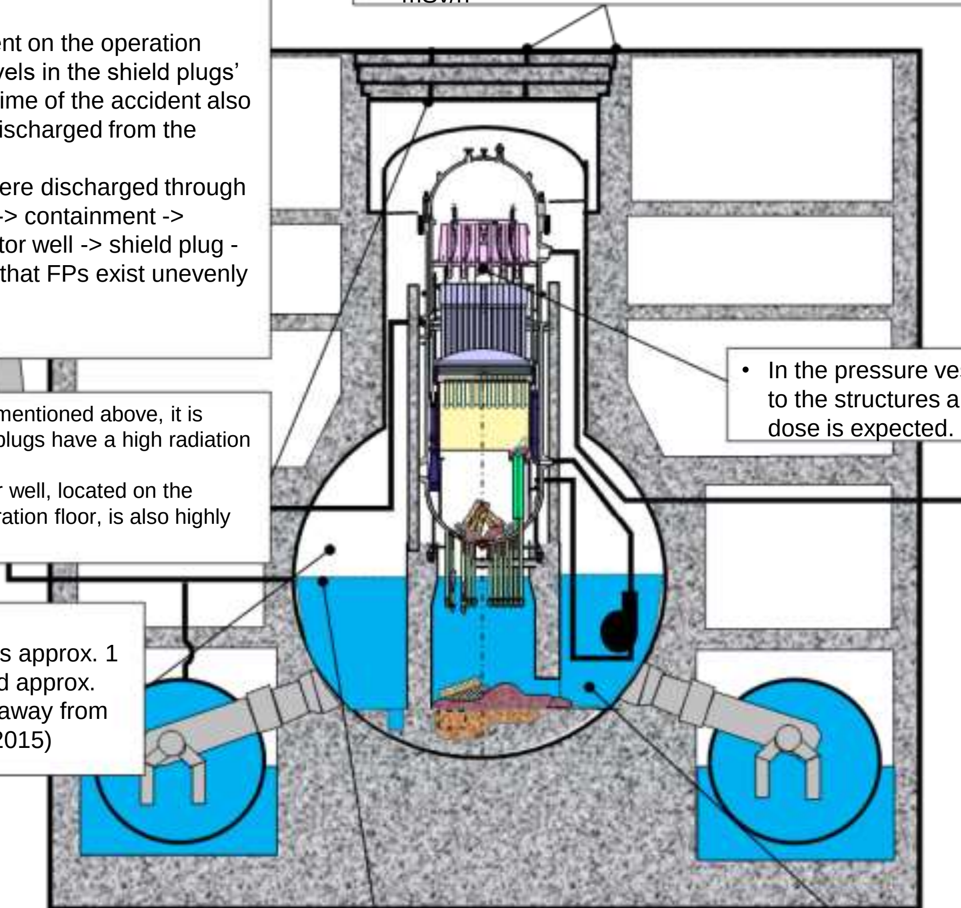
Unit 3: Estimate Diagram of Dose Distribution

- The gamma-ray spectra measurement on the operation floor in Unit 3 indicated high dose levels in the shield plugs' gaps and joints. The pictures at the time of the accident also show a large amount of vapor was discharged from the damaged reactor building.
- Therefore, it is presumed that FPs were discharged through the following route: pressure vessel -> containment -> containment top head flange -> reactor well -> shield plug -> operation floor. It was also figured that FPs exist unevenly along the FP migration route.

- Based on the estimated migration route mentioned above, it is presumed that the gaps between shield plugs have a high radiation level.
- It is also presumed that the entire reactor well, located on the migration route and upstream of the operation floor, is also highly contaminated.

- The dose in the PCV gas phase was approx. 1 Sv/h near the PCV wall surface, and approx. 0.75 Sv/h at a point about 550 mm away from the X-53 penetration exit. (Oct 30, 2015)

- Airborne radiation near the gaps and joints of the shield plugs on the operation floor: approx. 200–300 mSv/h



- In the pressure vessel, FPs are believed to be attached to the structures and wall surfaces; therefore, a high dose is expected.

- The FPs attached to the D/W wall surface may have fallen along the wall into the stagnant water in the PCV; therefore, a high dose could be expected near the water surface.

- Because the leakage of the PCV of Unit 3 is located around the extension joint of the main vapor pipe D, the dose in the lower part of the PCV's stagnant water could be higher than that on the water surface.

(1)-[2] Establish a Database Necessary for Comprehensive Analysis and Evaluation

[Overview and Objectives]
•Promote the enrichment of the database by maintaining and managing it and updating its contents in a continuous and timely manner so that the latest measurement data and research results can be viewed, analyzed, and downloaded from the database. Also, upgrade the display functions to present the analysis results in a more user-friendly manner.
•The website for the OECD/NEA BSAF project (English site) is run separately; however, since the Japanese and English sites have a lot of information in common, the two sites will be managed in an integrated manner. By doing so, information sharing in English will be enhanced.

[FY2017 Research Details]
- Improve the operability of the graph display functions (including the zoom function)
- Integrated management of the English and Japanese databases
- Timely update and management of the contents

[Final Report]
• Detailed display range settings for the graph display function were introduced. A display selection function to provide an easy-to-view display of the measurement data collected within the first three weeks after the accident was also introduced.
•The English version of the document search function to enable the search of English documents on the internal PCV investigations, etc., was developed.
•The timely update and management of the database contents was introduced.
•By implementing the afore-mentioned measures, making the comprehensive analysis and evaluation mentioned in (1)-[1] more efficient was contributed to.

[Database URL]
https://fdada.info/

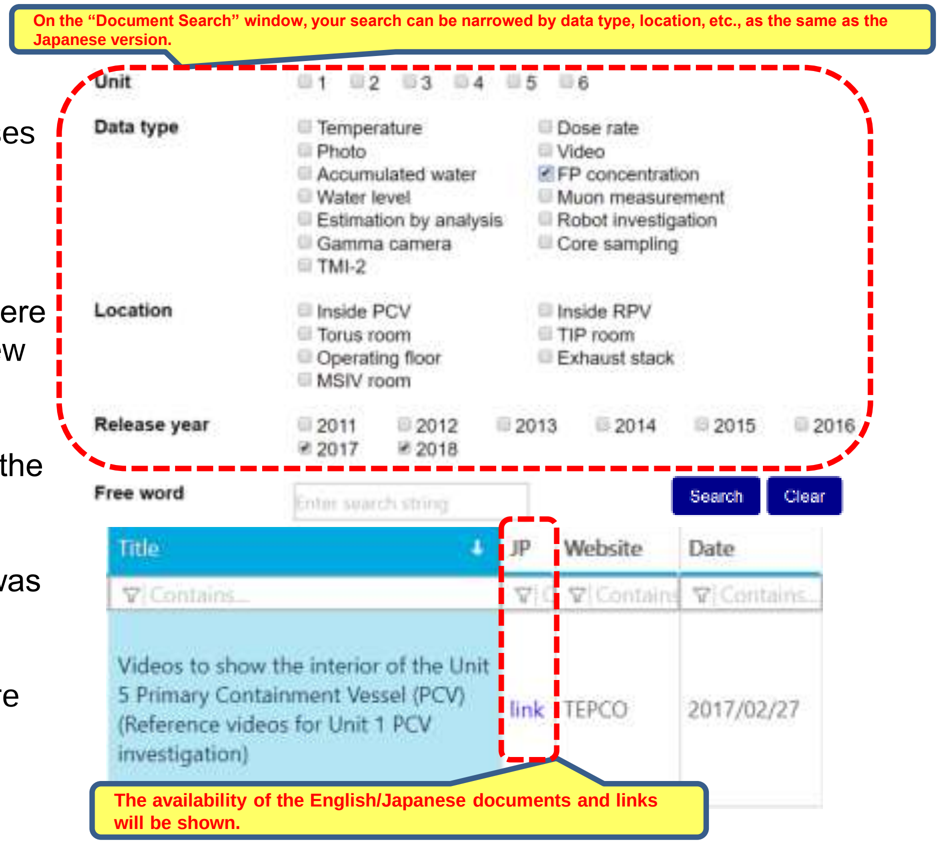


Figure: Developing the English version of Document Search

(2)-[1]-1 Sensitivity Analysis of the Concrete Erosion at the Pedestal

[Overview and Objectives]
Based on the results of the investigation inside Unit 1, a sensitivity analysis of MCCI was conducted to help estimate the fuel debris conditions in the pedestal.

[FY2017 Research Details]

- Since the internal PCV investigations indicate that the pedestal is standing on its own, the key parameters that affect concrete erosion using the MAAP codes were studied.
- The significant effect of a concrete melting when no water was injected from a fire engine was verified and reflected the finding in the analysis conditions for long-term erosion.
- By using three-dimensional codes, long-term erosion analysis was performed to consider the differences in the falls of fuel debris and their resultant deposits formulation.

[Final Results]

- The pedestal erosion tendency for each fuel debris fall location and deposits state was grasped. The tendencies indicated the possibility that, even in a situation where no water is injected from a fire engine, no major pedestal erosion occurs, although it may depend on the concrete's properties.
- The sensitivity analysis results were reviewed, including those obtained up until the last fiscal year, and reflected the pedestal erosion tendencies in the fuel debris distribution diagram.

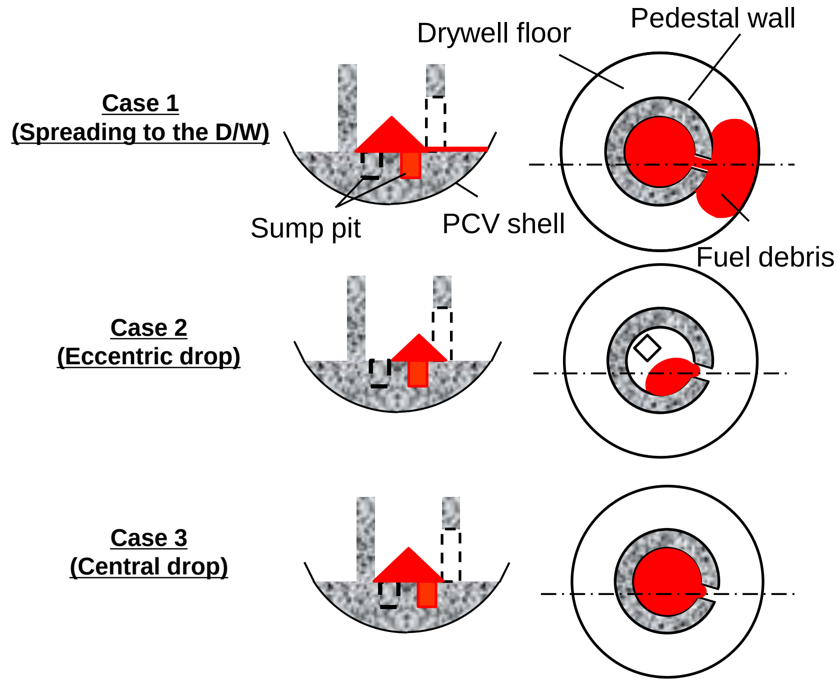


Figure 1. Estimated condition of fuel debris depositss

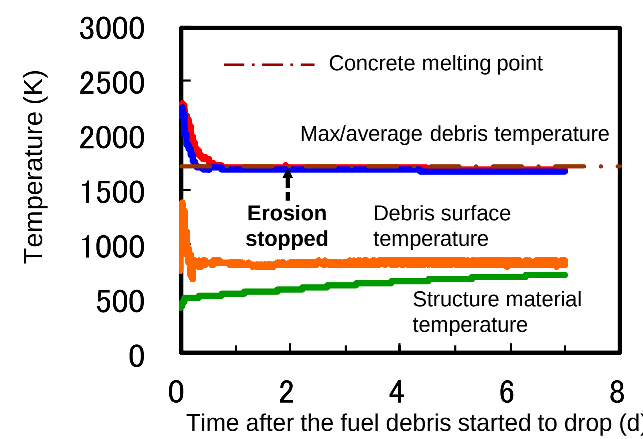
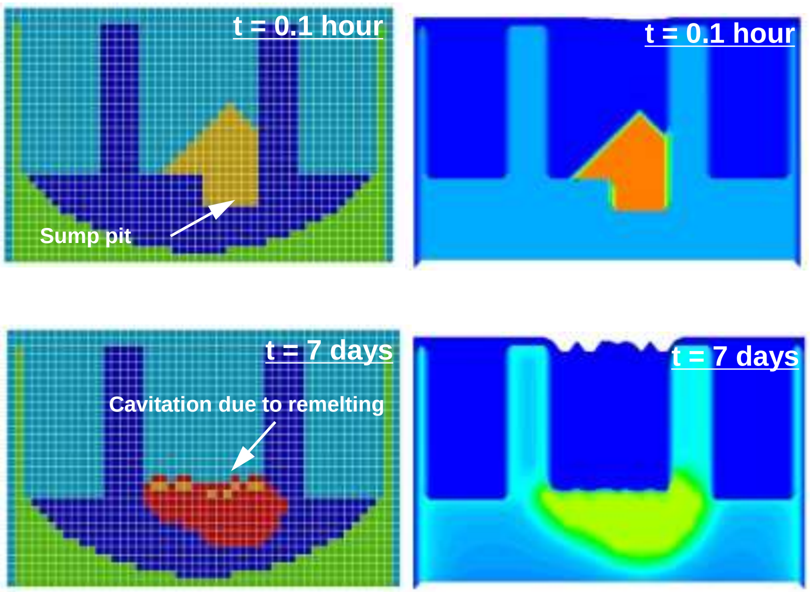


Figure 2. Temporal change in fuel debris temperature and structure material temperature (Case 2)

(2)-[1]-1 Sensitivity Analysis of the Concrete Erosion at the Pedestal

[Application to the estimate diagram of fuel debris distribution] Fuel debris shape and concrete erosion behavior

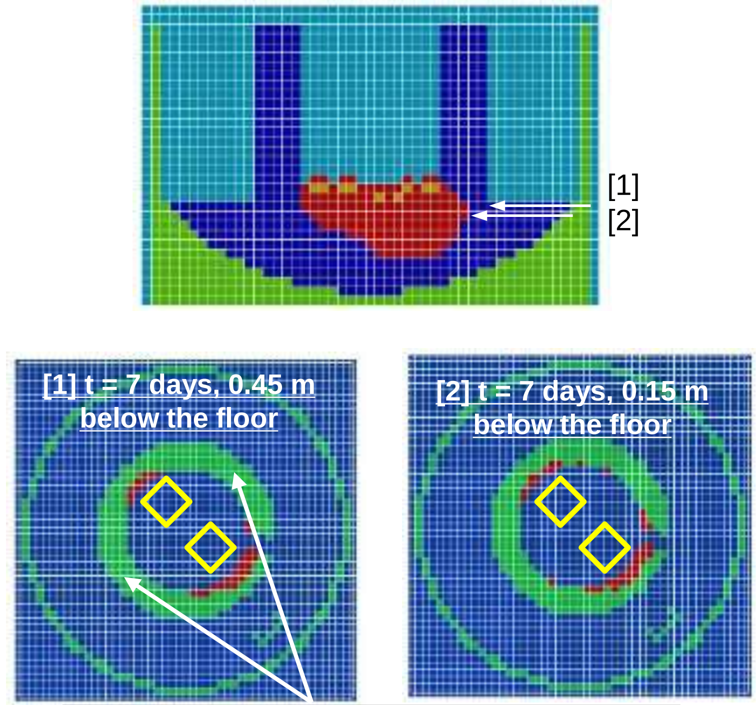
[Solid-liquid distribution] [Temperature distribution]



- Even if fuel debris depositss solidify, they may remelt because of decaying heat and flatten.
- It is possible that no major pedestal erosion occurs, even without water injection from a fire engine, although it depends on the concrete's melting point. (Case 1 and 3 showed the same tendency.)

Molten debris | Concrete floor | Crust | Concrete outside the shell

500 1,000 1,500 2,000 2,500
Temperature (K)



- Erosion of the pedestal wall is minor in the areas away from the sump pit.

Eroded area (concrete -> fuel debris)
Pedestal wall/base and PCV shell

Figure 3. Case of eccentric fall and depositsion (Case 2)

Figure 4. Review on the damage condition of the pedestal base and wall (Case 2)

(2)-[1]-2 Inverse Problem Evaluation Using Virtual Reactors and Compilation of a Database

[Overview and Objectives]

From right after the accident until today, huge and various kinds of data were released to the public, including the temperature and pressure data inside the PCV. By solving inverse problems from those data, fuel debris locations are estimated.

[FY2017 Research Details]

Virtual reactors were created by making use of three-dimensional CFD codes (StarCCM+). In addition to the creating more RPV models, the temperature distribution from one month to six months after the accident was evaluated, with the forward problem analysis approach, and the fuel debris locations inside the plant was estimated.

[Final Results]

[Unit 1]

Almost no heat source inside the RPV is expected. Meanwhile, given the leakage of superheating vapor from the RPV near the safety valve, the presence of a heat source in the RPV needs to be assumed.

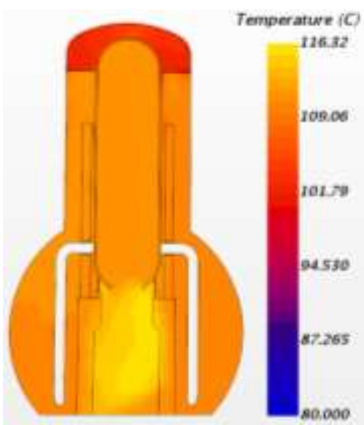
[Unit 2]

For an assumed situation where most of the heat source exists in the RPV and some 0.8MW (about 25% of the estimated total heat generated) is present in the pedestal, the temperature distributions in the steady and transient states were successfully simulated.

[Unit 3]

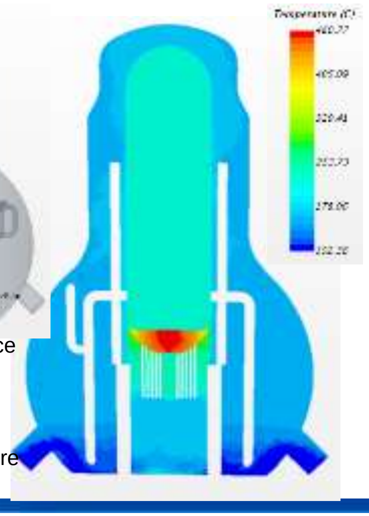
The temperature distribution suggested the possibility that the heat source of about 0.25–0.3 MW (about 25–30% of the estimated total heat generated) is present in the lower part of the RPV. According to a calculation, a significant amount of heat, which is estimated to be 0.8MW (81% of the estimated total heat generated) or greater, was removed by feedwater.

Estimated heat source distribution

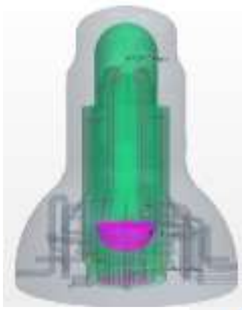


Apr 6, 2011 Estimated temperature distribution

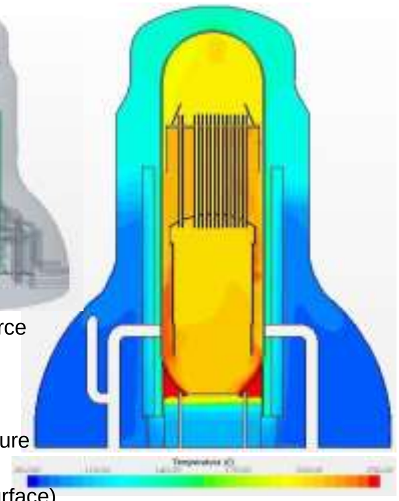
Estimated heat source distribution



Apr 1, 2011 Estimated temperature distribution



Estimated heat source distribution



Jun 30, 2011 Estimated temperature distribution (above the water surface)

(2)-[1]-3-(a)-1 Estimate of the Amount of Fuel Debris Remaining in the Pedestal Space Obstacles

[Overview and Objectives]
Fuel debris may have attached to structures, such as the grating and the penetration pipe, in the pedestal space.
From the Unit 2 internal PCV investigation results, thermal balance data, etc. the amount of fuel debris stuck in those obstacles are estimated.

[FY2017 Research Details]
Through comprehensive analysis of temperature distribution and temperature behavior upon changes in the water injection operation, the presence or absence of fuel debris in the pedestal space was evaluated. A heat transfer analysis was also conducted by referring to the results of the internal PCV investigation held in January and February 2017 and to the accident scenario, and how much fuel debris may have got stuck in the structures near the CRD housing was evaluated.

[Final Report]
Since the temperature in the lower part of the CRD housing is almost the same as that of the D/W (Figure 1), it is estimated that there are no large amount of fuel debris near the CRD housing. Possible locations on the structures where fuel debris may have got stuck (Figure 2) were extracted and a heat transfer analysis (Figure 3) was performed. The results indicate the possibility that the structures did not melt and that a small amount of fuel debris still exists.

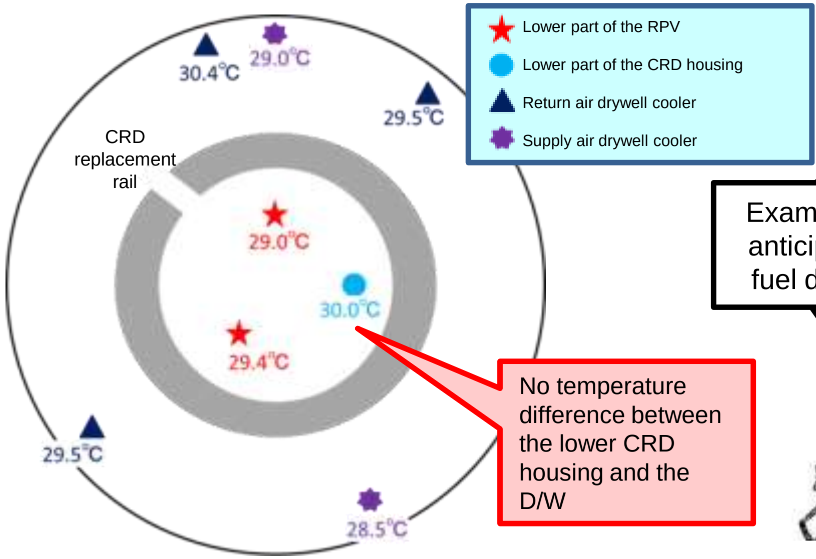


Figure 1. Temperature distribution near the pedestal (Jun 1, 2015)

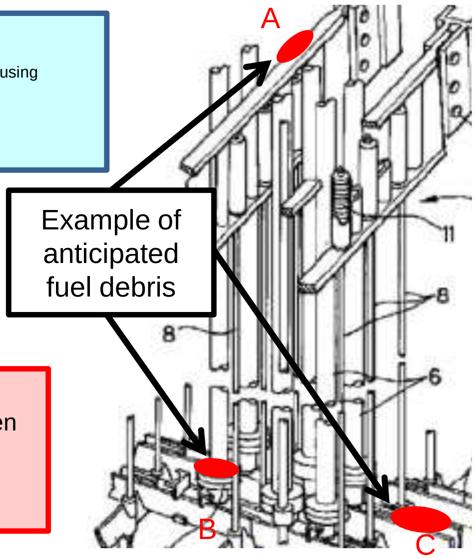


Figure 2. Structures near the CRD housing

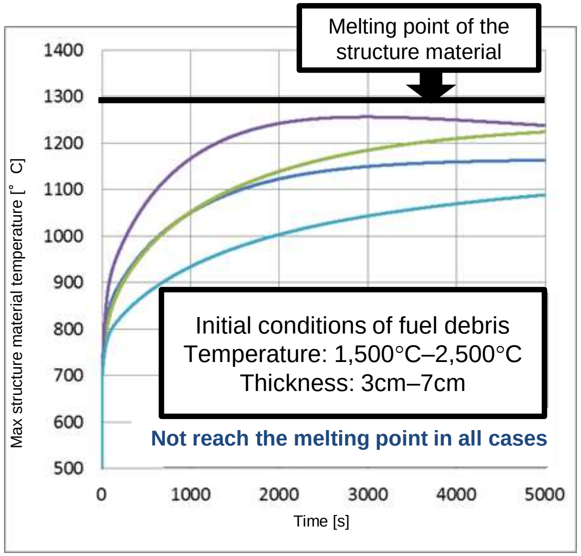


Figure 3. Example of heat transfer analysis (hypothesized fuel debris in Figure 2's A)

(2)-[1]-3-(a)-2 Detailed Evaluation of Debris Behavior inside the Lower Head after Damage of the Lower Head

[Overview and Objectives]

It is important to find out whether fuel debris still exists inside the CRD housing in the lower part of the RPV for debris removal purposes.

[FY2017 Research Details]

Based on the specimen cutting inspection in the CRD housing melting test, the model parameter was optimized and the fuel debris behavior in the lower head (Figure 1) was analyzed. To evaluate the presence of fuel debris near the CRD and based on the KAERI test results, a sensitivity analysis was performed with the detailed meshes showing fuel debris temperatures, the presence/absence of water in the lower head, decay heat, etc. (Figure 2 and 3).

[Final Report]

The amount of fuel debris penetrating the CRD housing largely depends on the fuel debris temperature when melting occurred in the upper part of the CRD housing and, hence, involves significant uncertainties. The detailed analysis results showed that a high temperature persisted near the weld, even when water was present in the lower head, indicating that fuel debris may have flowed out through the gap under the weld of the RPV and CRD housing.

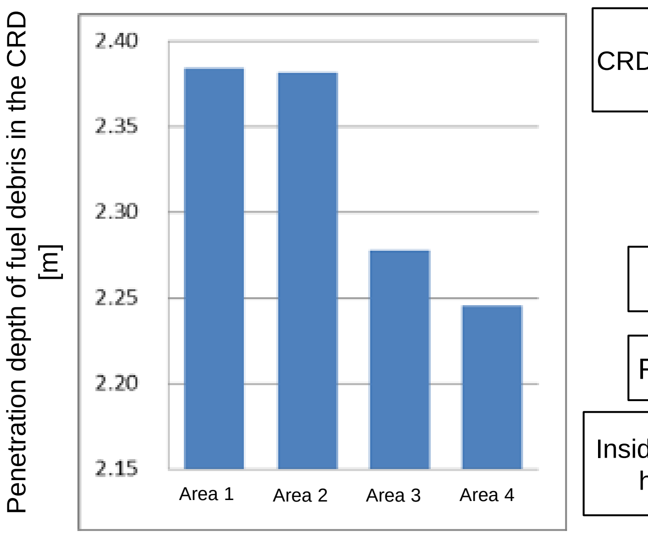


Figure 1. Analysis of fuel debris behavior in the Unit 3 lower head

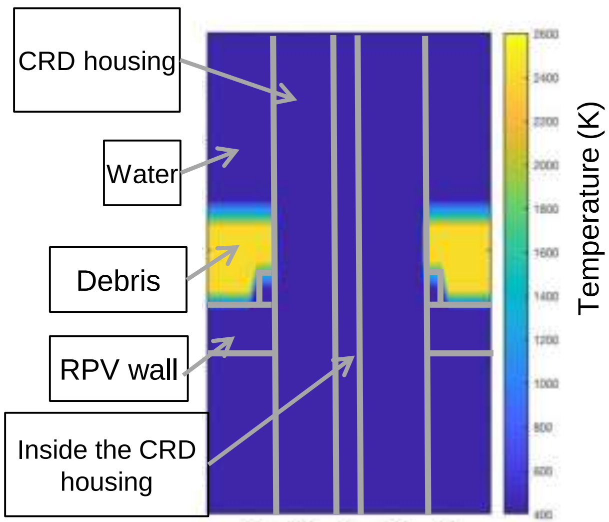


Figure 2. Initial fuel debris distribution according to the detailed analysis

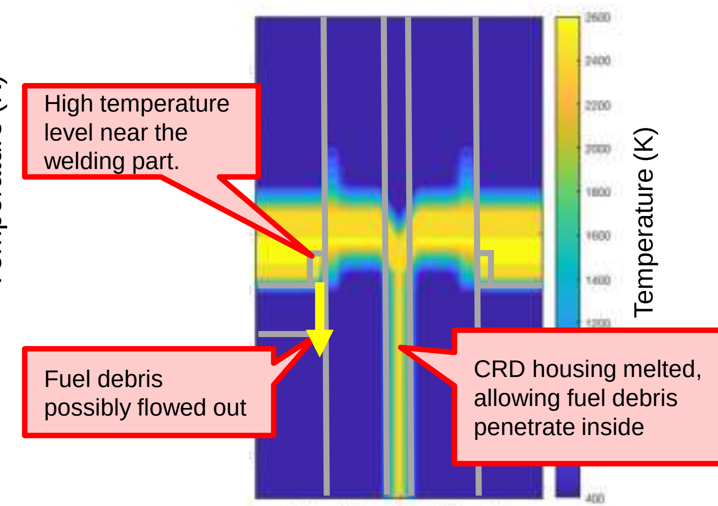


Figure 3. Fuel debris distribution after the melt of CRD housing

(2)-[1]-3-(a)-3 Evaluation of Condensation Behavior of Water Vapor Containing Hydrogen in the S/P

[Overview and Objectives]
Vapor condensation behavior (a phenomenon of incomplete condensation) in the S/C is important in identifying the accident progression. Especially, the behavior of mixed gas of hydrogen and vapor blowing into the S/C is important for evaluating the PCV/RPV pressures. Through evaluating this behavior, and projecting the amount of generated hydrogen, etc., a reduction in the the uncertainties of fuel debris migration behavior is sought.

[FY2017 Research Details]
By combining SAMPSON and the CFD tool POOL-3D with the temperature distribution and the condensation experiment on steam containing non-condensable gas (LINX), the condensation behavior of water vapor containing hydrogen in the S/C is studied. Based on the study results, the distribution of S/C water temperatures are provided, a key parameter regarding condensation behavior, in the three-week analysis “(2)-[1]-3-(a)-7.”

[Final Report]
The three-dimensional temperature distribution inside the S/C during the period in which vapor and hydrogen were discharged from the SRV, was evaluated. It was found that, in Unit 2, there was a temperature difference of approx. 30° C up until the RCIC stopped (70 hours after the scram). After forced depressurization (75 hours after the scram), the temperatures inside the S/C equalized; however, it was found that temperature stratification occurred again due to vapor and hydrogen erupting from the SRV. It was found that, in Unit 3, temperature stratification occurred while the RCIC was in operation and it persisted while the HPCI was working. It as also found that, after the HPCI stopped, temperature stratification was resolved by a huge amount of hydrogen generated during the progression of core melt.

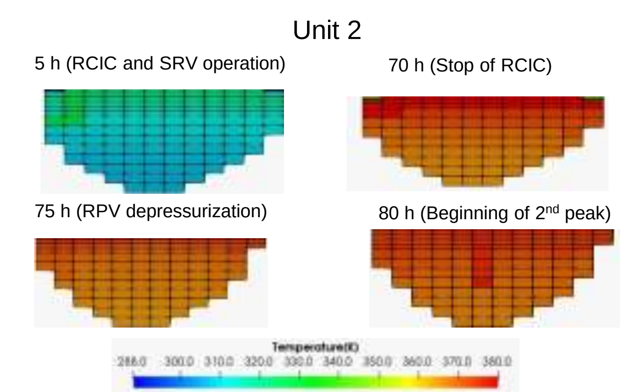


Figure 1. Unit 2 temperature stratification at indicative event

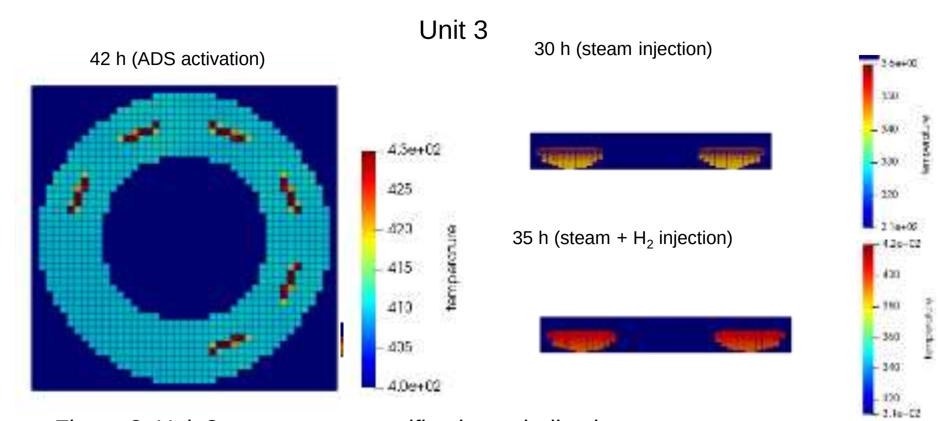


Figure 2. Unit 3 temperature stratification at indicative event

2)-[1]-3-(a)-4 Event Sequence Analysis at the Time of Core Material Slumping

[Overview and Objectives]

In all units of Fukushima Daiichi NPS, the core materials are believed to have relocated to the lower plenum (slumping) and to the pedestal. The core state during the slumping affects the RPV failure mode and has a significant impact on the eventual material distribution and their properties. The corresponding accident progression is therefore determined by conducting the SA code analysis and using the plant data (such as pressure) to improve the accuracy of the evaluation of the final materials distribution and their properties.

[FY2017 Research Details]

A detailed MELCOR code analysis (Waseda Univ.) of the accident progression in Units 2 and 3 was performed, with a focus on the consistency with the plant data. The most probable transitions and uncertainties was also evaluated by comparing the results with the MAAP (EPRI), SAMPSON (IAE) and SCDAPSIM (JAEA) results all the while shedding light on core energy.

[Final Results]

The following knowledge was obtained and provided in the estimate diagram of fuel debris distribution.

- Based on the evaluation of core energy, it was concluded that most of core fuel did not melt when slumping occurred in Unit 2 (the best estimate). It was concluded, meanwhile, that in Unit 3, where it took a long time from the initiation of core damage to slumping, a considerable amount of fuel melted due to accumulated energy (the best estimate) (Figure 1).
- Based on the energy state described above, the effect on the heat transfer behavior of fuel debris and structural materials in the lower plenum (Figure 2) was assessed and a result matching the internal investigation's results was obtained.

Reflection in the estimate diagram of fuel debris distribution

The combined knowledge with the results of the "simulated fuel assembly degradation test" indicates a considerable amount of un-melted fuel pellets may still exist in the reactor core or at the lower plenum area in Unit 2. The knowledge indicated a relatively large mass of solidified fuel may still exist in the pedestal or CRD area in Unit 3. The Unit 3 data analysis results also indicate the possibility that MCCI was mitigated by the water in the PCV.

*Core energy: decay heat + oxidation reaction heat - heat transfer to the coolant or structure material

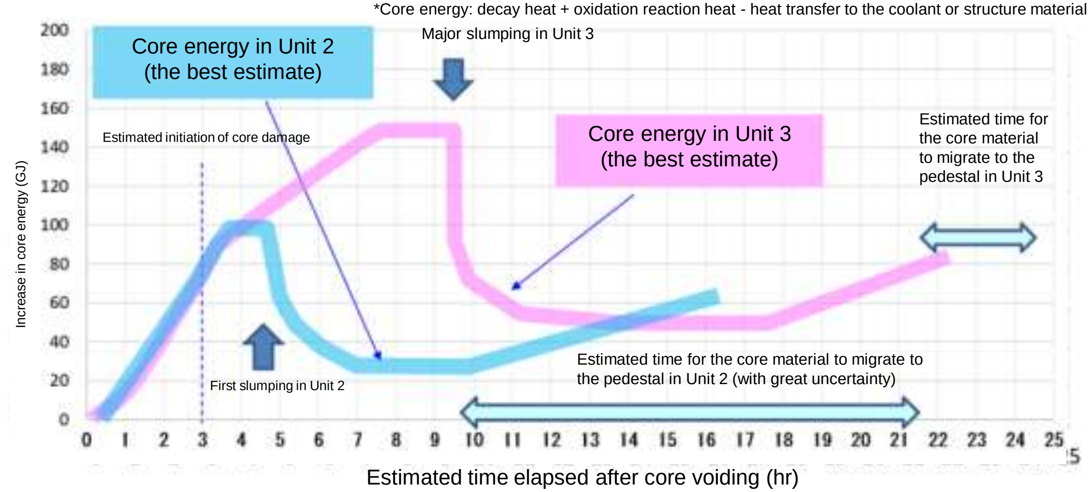


Figure 1. Comparison of the accumulated core energy (the best estimate) and timing of core material migration to the lower plenum and pedestal in Units 2 and 3

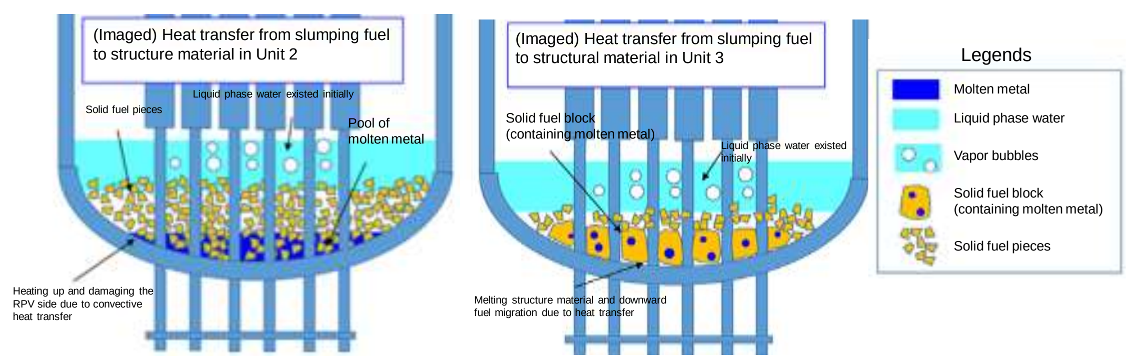


Figure 2. Effect on the heat transfer behavior of fuel debris and structure material estimated according to the energy during slumping

(2)-[1]-3-(a)-5 Three-Dimensional Evaluation of MCCI Behavior

[Overview and Objectives]

A quantitative evaluation of the state of concrete erosion caused by MCCI that is assumed to have occurred in the pedestal was performed.

[FY2017 Research Details]

In the last fiscal year, a sensitivity analysis on Unit 1 was conducted, finding that the volume and temperature of dropped fuel debris are the key parameters of fuel debris diffusion. In this fiscal year, with regard to Units 2 and 3, a three-dimensional evaluation on the MCCI behavior under a condition where water remains on the pedestal floor was performed. A sensitivity analysis using parameters such as the remaining water volume was also conducted.

[Final Report]

- It was verified that, in Unit 2, the whole fuel debris would remain in the pedestal if the dropped debris was 20 tons (12.8 tons of fuel debris) or 40 tons (25.6 tons of fuel debris) and concrete erosion stopped within 10 hours.
- With regard to Unit 3, an analysis for a condition where the dropped debris was 140 tons (89.4 tons of fuel debris) was performed, by setting the sensitivity parameters at the initial water level and the drop velocity of fuel debris. The results of the analysis with SAMPSON/DSA showed the initial water level had a limited impact on the diffusion of fuel debris, while the drop velocity of debris had a large impact on fuel debris diffusion.

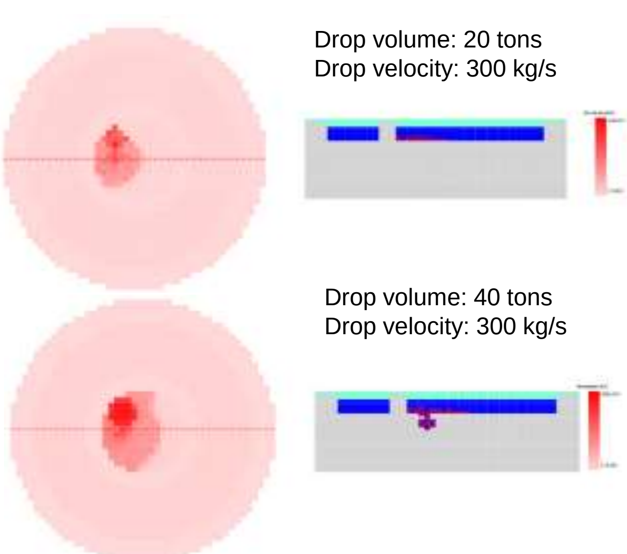


Figure 1. Sensitivity evaluation of dropped fuel debris volume (Unit 2)

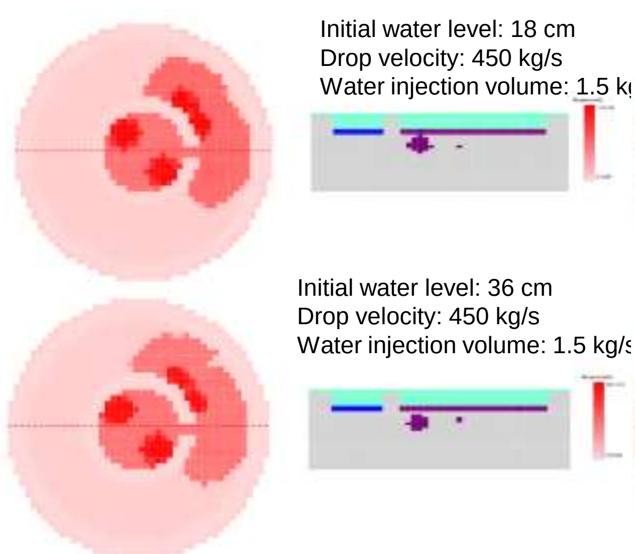


Figure 2. Sensitivity evaluation of initial water level (Unit 3)

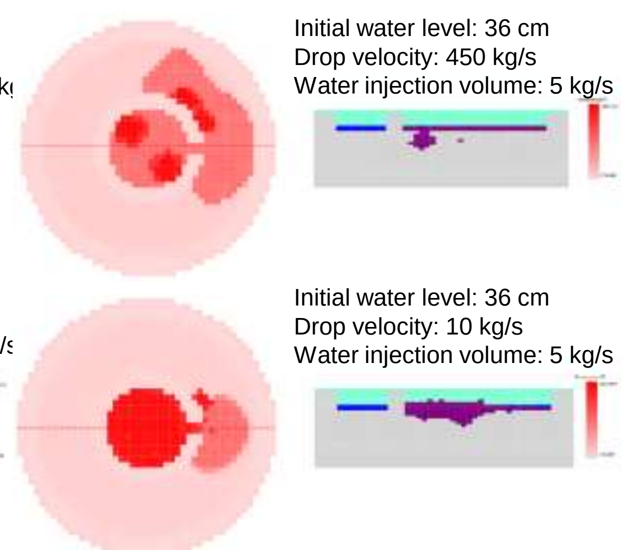


Figure 3. Sensitivity evaluation of drop velocity (Unit 3)

(2)-[1]-3-(a)-6 Confirmation of Factors Influencing Debris Distribution

[Overview and Objectives]

A quantitative evaluation of the effect on fuel debris distribution is performed, assuming the fuel rods collapsed before melting, to ensure reliability of the fuel debris distribution information that could be obtained through the accident progression analysis, thus reducing uncertainties.

[FY2017 Research Details]

A model representing the pre-melt collapse of fuel rods was introduced and a quantitative evaluation on the effect of this phenomenon on the fuel debris distribution was conducted. The fuel debris distribution was comprehensively evaluated while taking into account the results of the two-dimensional temperature distribution evaluation that studied the output per fuel assembly.

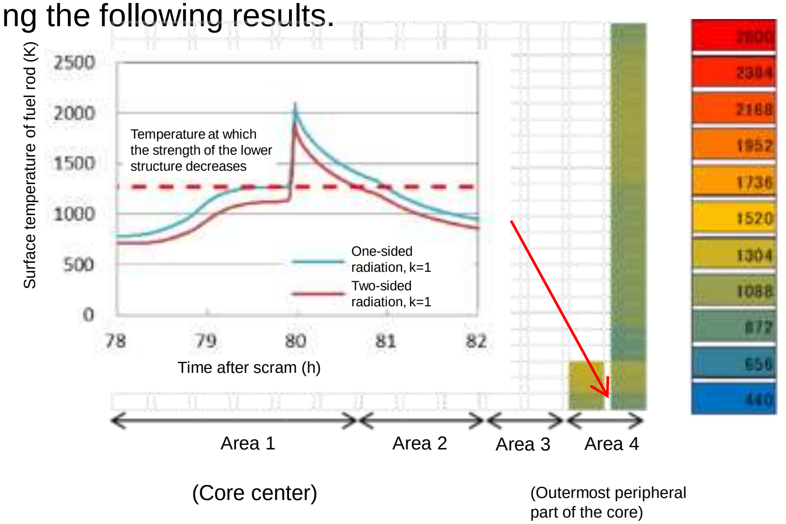
[Final Report]

- In the VERCORS test, a fuel rod collapsed at approx. 2,600 K, meaning that it would take about 100 hours for a Larson-Miller parameter model to collapse when the fuel rod temperature is around 2,300 K. Therefore such factors as the loss of metal cladding, due to melting and oxidation, and the vibration of fuel rods, due to the sudden generation of vapor, were taken into account and a model using the fuel rod temperature and time history for the SAMPSON code was introduced to simulate a fuel rod that collapses within a few hours even at 2,300 K.
- To evaluate the possibility of localized collapse of a fuel rod that cannot be evaluated through the SAMPSON analysis, the temperature distribution at the horizontal section of Unit 2’s core was evaluated. For the latter evaluation, factors such as radiation and the Zr-water vapor reaction were taken into account, getting the following results.

Evaluation of the two-dimensional temperature distribution based on the SAMPSON three-week analysis

1. **The fuel rod temperature increased as shown in the figure; however, it is more likely that the supporting strength in the lower part of the fuel rods was lost before the fuel failure occurred.**
2. When the emissivity and the amount of hydrogen generation to an extent consistent with the knowledge and actual measurements in the past was changed, a result in which part of the fuel at the outermost ring collapsed was gotten.

Presumably the support structure at the lower part of a fuel rod starts losing its strength before the fuel rod starts collapsing.
Also presumably, a fuel rod collapses downwards when the fuel falls toward the center; or collapses onto the shroud when the fuel falls toward the shroud.



Evaluation results of two-dimensional temperature distribution (at 82 hrs)

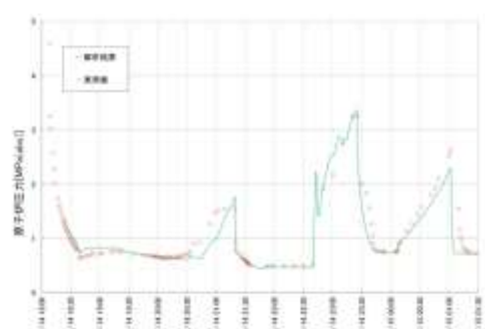
(2)-[1]-3-(a)-7 Analysis of Each Unit over Three Weeks after the Accident

[Overview and Objectives]
Analysis was performed for up to three weeks, by which time the MCCI was expected to have reached a completely steady state, in order to evaluate the final fuel debris distribution all the while taking uncertainties into account.
[FY2017 Research Details]
Sensitivity parameters affecting the distribution, composition, and properties of fuel debris were extracted and a sensitivity analysis was performed. A three-week analysis was performed with each sensitivity parameter group that matches the actual measurements obtained from the units and the results of the on-site inspections including the internal PCV investigations.
[Final Report]

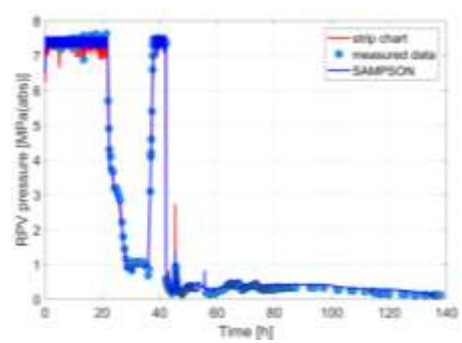
- To reduce uncertainties in the accident progression scenario, which is the prerequisite for evaluating fuel debris migration behavior and the final fuel debris distribution, accident progression scenarios to the most likely ones were narrowed down. This process was completed by comprehensively evaluating results of studies conducted inside and outside Japan, such as internal investigations using muon and robots, measurements of pressure and water levels in the reactors, the knowledge and scientific considerations gained or made until today, and BSAF.
- This initiative was conducted by IAE, the implementation organization for this particular study item, and in cooperation with the parties involved in the Project (JAEA, Hitachi GE, Toshiba, and TEPCO) through discussions at study meetings (which were held six times).
- The SAMPSON analysis results, which were obtained based on this accident progression scenario, matched well with the actual measurement results, significantly contributing to reducing the uncertainties for the final evaluation of the fuel debris distribution.

RPV pressure behavior (Unit 1)

RPV pressure behavior (Unit 2)



RPV pressure behavior (Unit 3)

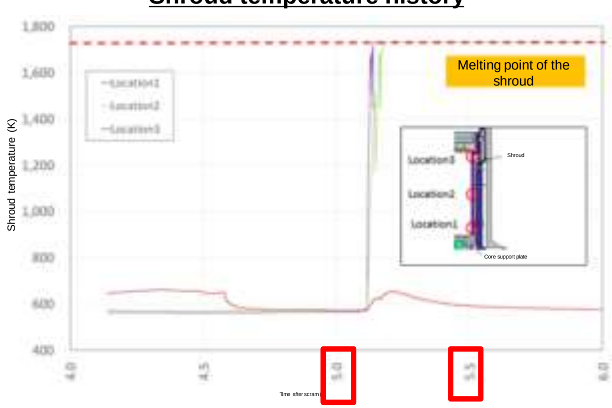


(2)-[1]-3-(a)-7 Analysis of Each Unit over Three Weeks after the Accident

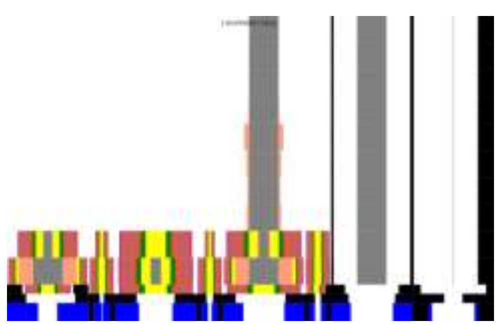
[Final report (evaluation results of Unit 1)]

- From the discussions made at the accident-scenario study meetings and the SAMPSON analysis results, it was presumed that the fuel rods started fracturing about 4 hours after the scram and started melting in about 5 hours. Based on these presumptions, fuel debris fell on the lower plenum in about 11 hours and further down to the pedestal in about 15 hours. It was concluded, therefore, that the fuel rods and fuel debris (containing fuel and structure materials) stayed in the reactor core area for about 5 hours in the high-temperature state due to decay heat and oxidation reaction.
- The VERCORS and CMMR tests indicate the possibility that the fuel rods maintain their column shape up to around 2,600 K. Considering the melting point of the channel box covering the fuel rods is 2,100 K (or 1,500K or even lower due to eutectic formation), the shroud would be directly exposed to radiation from fuel rods when their temperature far exceeds the melting point of 1,700 K. For Unit 1, no water injection was conducted while core damage progressed; therefore, it was concluded that the water in the down-comer may have fallen to the BAF or lower in 5 to 7 hours after the scram.
- The SAMPSON analysis reached the same conclusion, indicating that the shroud would fracture significantly and melt. Considering the phenomenon in which a shroud loses its structural strength at around 1,000 K before it melts, it was concluded that the shroud is deformed, damaged or buckled by the weight of the structures at the upper portion (such as the separator).
- It was also concluded that the structures in the lower plenum (such as the CRGT) are damaged or melted further than their counterparts in Units 2 and 3. It was also presumed that the concrete erosion by MCCI, even the largest one, will stop near the PCV shell.

Shroud temperature history

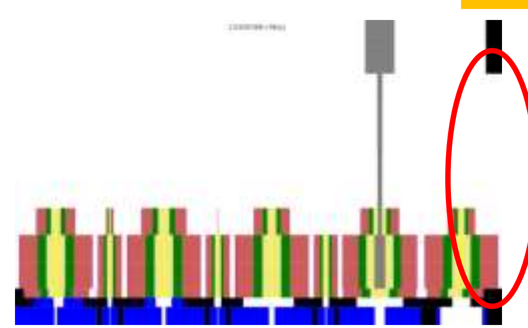


5 hours after scram
(before fracture of the channel box)



5.5 hours after scram
(after fracture of the channel box)

Shroud damage was detected

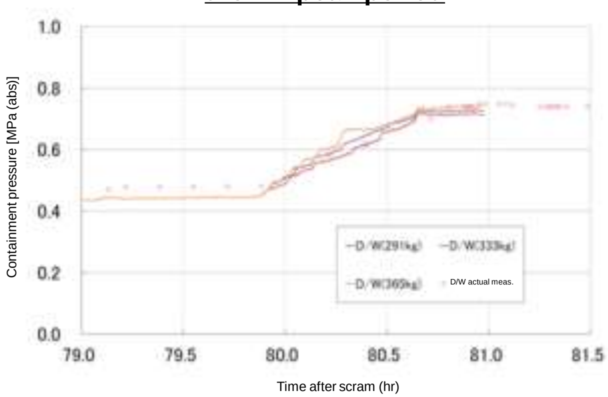


(2)-[1]-3-(a)-7 Analysis of Each Unit over Three Weeks after the Accident

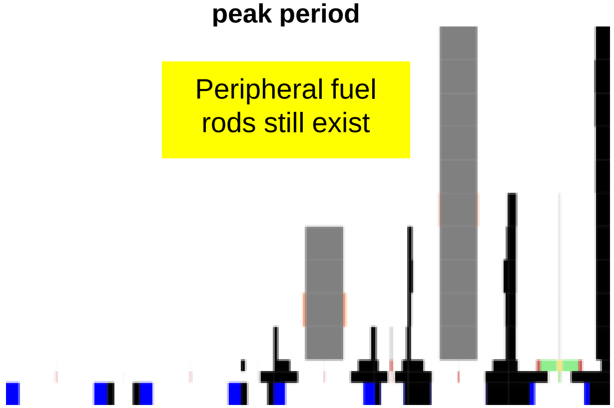
[Final report (evaluation results of Unit 2)]

- From the discussions made at the accident-scenario study meetings and the SAMPSON analysis results, it was concluded that the fuel rods started fracturing about 77 hours after the scram and started melting in about 78 hours. Then, it was presumed that about 100 tons of fuel debris (containing fuel and structure material) fell on the lower plenum in about 80 hours and some part of fuel debris fell further down to the pedestal in about 92 hours.
- The PCV pressure presumably rose between Hour 80 and 81, generating hydrogen and accelerating damage to, and melting of, the fuel rods. The LINX test results indicate the possibility that non-condensable gases, such as hydrogen, inhibit the condensation of water vapor. As is shown by actual values, the S/C water temperature rose to a high level during the hour, indicating our need to take incomplete condensation into account when evaluating the amount of hydrogen generated.
- A SAMPSON analysis was performed that took account of the actual measurement data and the phenomenon, getting a result showing that at least nearly 300 kg of hydrogen was generated. The analysis results showed that peripheral fuel rods may not melt completely under some conditions.
- A two-dimensional analysis ((2)-[1]-3-(a)-6) of the in-core temperature distribution was conducted, taking into account such factors as the oxidation reaction heat resulting from hydrogen generation, water injection and vapor temperature that was obtained from the SAMPSON analysis, to closely evaluate the fuel rod temperatures. A presumption was reached that the support structure of the peripheral fuel rods nearly rose to the melting point and that almost all of them eventually collapsed. However, some fuel rods may still exist, leaning on the shroud side.

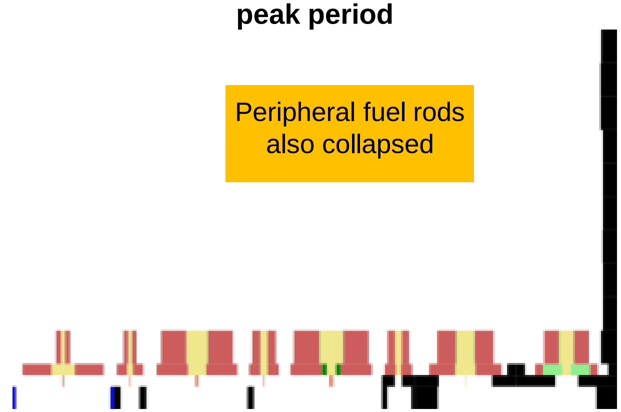
Sensitivity analysis of PCV pressure in the 2nd peak period



Case of 291 kg of hydrogen generated during the 2nd RPV peak period



Case of 365 kg of hydrogen generated during the 2nd RPV peak period

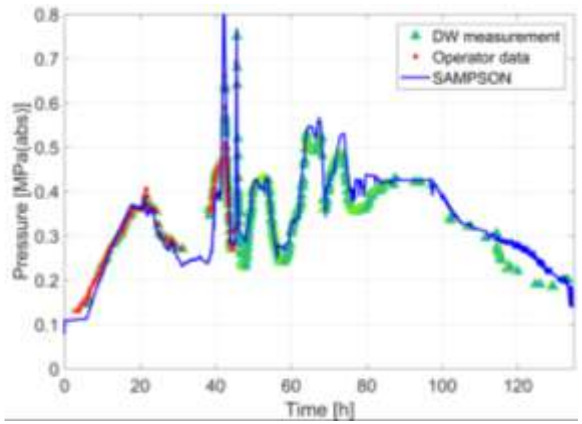


(2)-[1]-3-(a)-7 Analysis of Each Unit over Three Weeks after the Accident

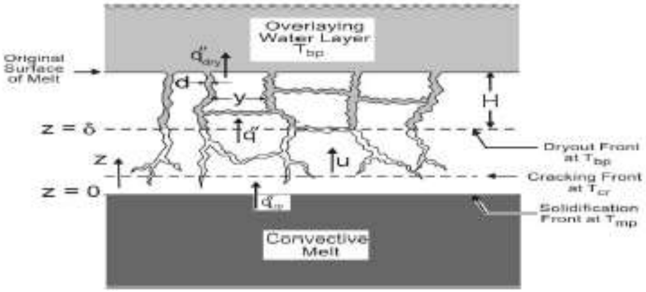
[Final report (evaluation results of Unit 3)]

- From the discussions made at the accident-scenario study meetings and the SAMPSON analysis results, it was concluded that fuel debris (fuel and structure material) slumping occurred at 43 hours and 45 hours after the scram. It was concluded that the PCV pressure continued to rise after the PCV vent closed, but that increase stopped as a result of balancing with the leakage from the D/W. Given the fact that the water injection from the fire engine was limited, it was concluded that the PCV pressure finally started decreasing when the inside of the RPV dried out. It was concluded that, during that period, water started migrating from the S/C to the D/W, creating, along with the water due to the D/W spray, some pools of water on the D/W floor.
- It was concluded that the lower head damaged 50 hours after the scram and that fuel debris dropped to the pedestal. From the analysis results showing almost zero residue of fuel debris in the reactor core, it was concluded there are about 30 tons of crusted fuel debris in the lower plenum and about 120 tons of fuel debris in the pedestal.
- A relatively large amount of water was present when the fuel debris migrated to the pedestal. Considering the possibility that water seeped in the cracks of the crust on the surface, it was concluded that the fuel debris cooled down quickly, resulting in almost zero concrete erosion.

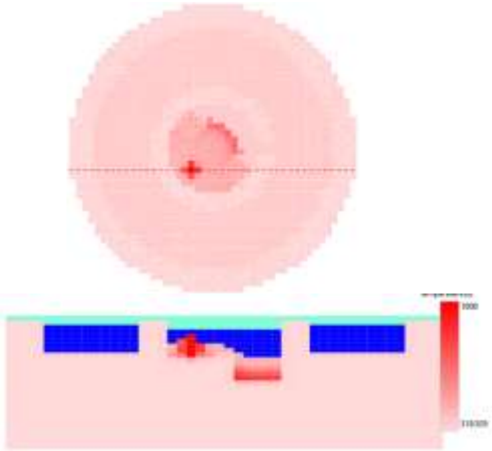
PCV pressure behavior



Schematic diagram of crustal fracture



Progression state of MCCI



(2)-[1]-3-(a)-8 Detailed Accident Progression Analysis Using the Accident Progression Analysis Codes

[Overview and Objectives]

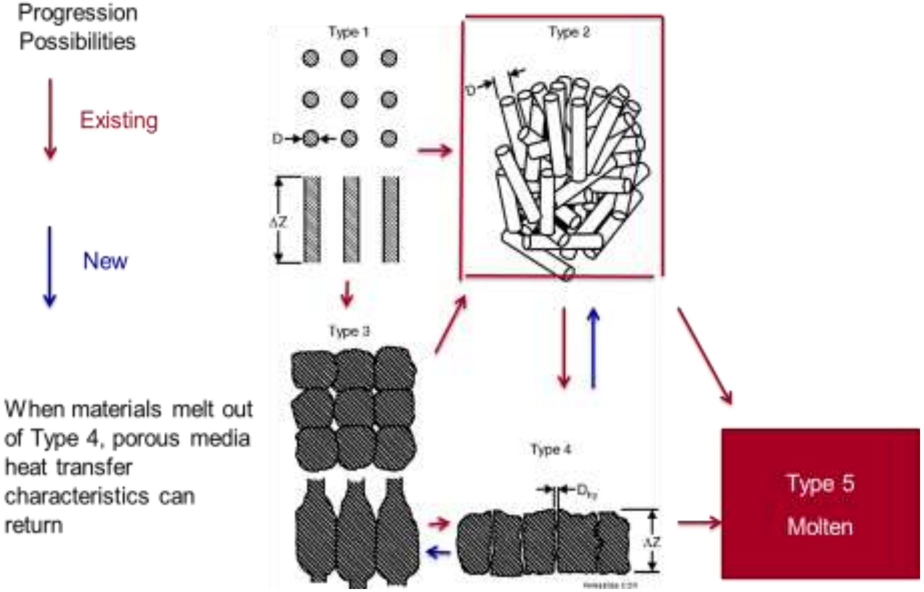
- In the detailed accident progression analysis conducted in FY2016 by using the MAAP code, the accident progression scenario and the fuel debris distribution was studied while taking into account the code's limit and for a physical phenomenon model. The issues of reduction and quantification of the fuel debris distribution were also extracted.
- In FY2017, some of the physical phenomena models of the code were changed. The renovated models in the accident progression analysis were used to reduce the uncertainties involved in estimating the accident progression scenario and the fuel debris distribution.

[FY2017 Research Details]

- Improvement of the MAAP physical phenomena models
  - ➢ Model of fuel debris outflow when the RPV was damaged
  - ➢ Model of B_4C oxidation of the residual heat control rods
  - ➢ Model of core blockage and corium migration
  - ➢ Model of fuel debris flowing into the CRD housing
- Detailed evaluation of fuel debris dispersion behavior for the study on MAAP boundary conditions
- Estimation of the accident scenario and quantification of the fuel debris distribution based on the accident progression analysis on Units 2 and 3 using the improved code

[Final Results]

- Improved the MAAP physical phenomena models and testing its operation and validity
- Performed the detailed accident progression analysis in Units 2 and 3 using the improved code
- Updated the accident progression scenario and the estimated fuel debris distribution



Example of model improvement (core blockage and corium migration model)

The model initially did not return to Type 2 or 3 once it reached the blockage node (Type 4); a core damage progression model was made allowing such return in order to simulate the accident that presumably occurred at Fukushima Daiichi. The impact on the amount of hydrogen generated in the core damage process was particularly significant, and the reproducibility of the PCV pressure behavior was improved.

(2)-[1]-3-(a)-8 Detailed Accident Progression Analysis Using the Accident Progression Analysis Codes

Unit 2

Area	Results of improved MAAP analysis	Estimated fuel debris distribution based on the analysis results
Reactor core area	In the reactor core, 14.5% of the initial UO_2 mass still exists.	Most parts of the reactor core fell to the lower plenum. The outermost peripheral area of the reactor core with low output may still exist on the core support plate because of the long-term water injection from the fire engine.
Lower plenum	In the lower plenum, 65% of the initial UO_2 mass still exists as particulate fuel debris.	Due to the localized damage to the weld of the instrumentation guide tube, and solidification on the RPV sub-structure surfaces, the molten core may have gradually fallen to the lower part of the PCV. A rupture may exist at a location closer to the side of the RPV boom than to the center, possibly limiting the fall of the molten core only from a position at or higher than the height of the rupture.
Lower part of the PCV	Of the initial mass of the reactor core, 20% falls to the pedestal.	At around 14:00 on March 15, the molten core, presumably part of the molten core in the lower plenum, possibly fell to the lower portion of the PCV. Concrete erosion may have occurred to a very limited extent.

Unit 3

Area	Results of improved MAAP analysis	Estimated fuel debris distribution based on the analysis results
Reactor core area	Fuel no longer exists in the reactor core.	Most parts of the molten core may have fallen, in a large scale, from the reactor core area to the lower plenum; as a result, the amount of the fuel that still exists in the reactor core area may be limited.
Lower plenum	Of the initial mass of the reactor core, 20% still exists in the lower plenum.	Due to the major damage at the RPV bottom, the amount of the molten core that still exists in the lower plenum may be small.
Lower part of the PCV	Of the initial rector core mass, 80% fell to the pedestal and then part of it flowed into the drywell.	Due to the massive molten core falling to the lower part of the PCV, the molten core may have flowed from the pedestal to the drywell, but in a limited manner. There may be more concrete erosion toward the floor.

(2)-[1]-6 Simulated Fuel Assembly Degradation Test

[Overview and Objectives]

Regarding the core melting and migration behavior at the time of the accident on the 1st floor, it might have had a residual, heat-type progression, different from the TMI-type. Factors such as (1) gas permeability in the high-temperature core, and (2) entry of decayed fuel into the lower support structure area substantially affect the presence or absence of molten fuel in the core. These have greatly contributed to the uncertainties in evaluating the accident progression and the fuel debris distribution and characteristics. To reduce such uncertainties, a specimen simulating the residual heat-type reactor core and lower support structure was heated up with a plasma torch and the decay, melt, and migrating behavior of the core substance was observed.

[FY2017 Research Details]

The specimen was heated up simulating the residual heat system up to the assumed temperature in the lower core area at the time the core substance in Unit 2 slumped (migrated to the lower plenum) and the results were evaluated.

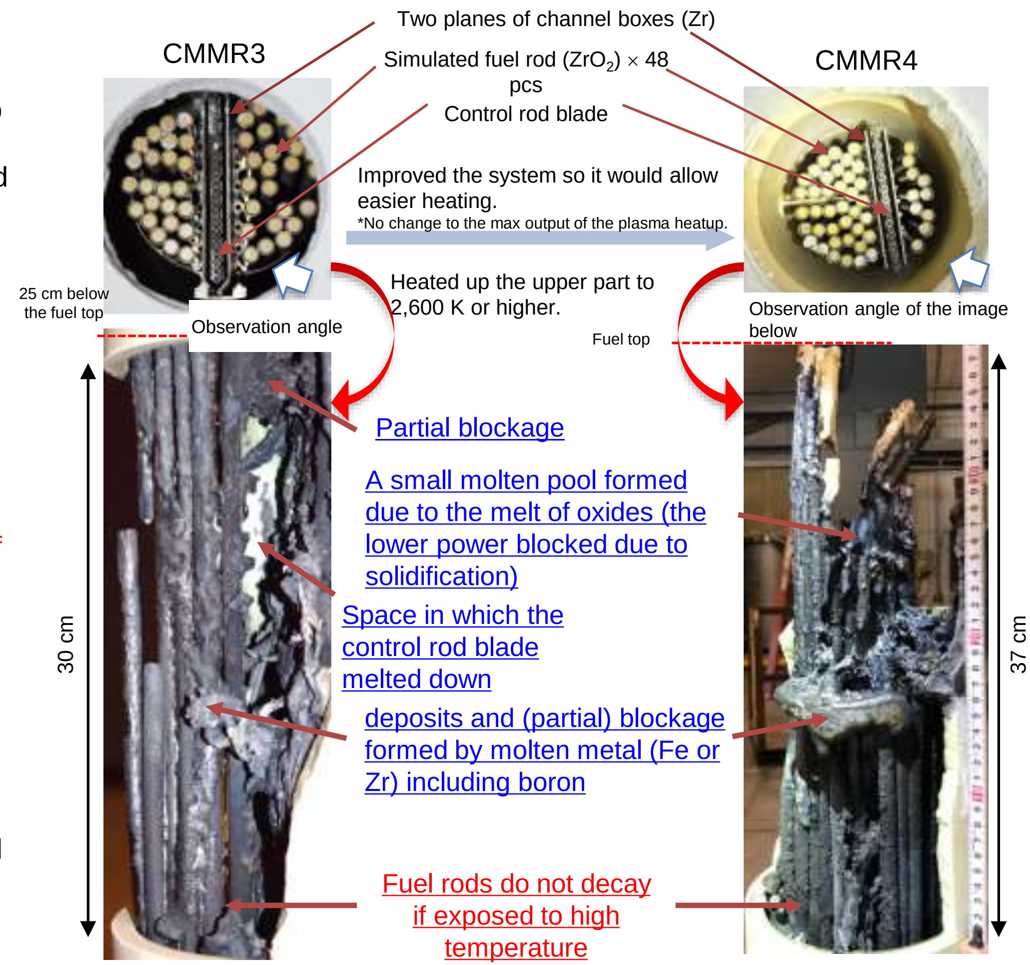
[Final Results]

The following knowledge was obtained and provided it in the estimate diagram of fuel debris distribution.

- [1] High-temperature core fuels form partial blockage; however, residue fuel columns tend not to fuse with each other and presumably have macroscopic permeability against the gas phase (and the liquid phase) including when they have decayed (similar to the loose core debris in TMI-2).
- [2] High-temperature fuels would maintain their column shape up to nearly the melting point, and they hardly move in the form of a pellet. If fuels in units decay due to a load, etc., the decayed high-temperature fuel would stay in the reactor core (with no change to the axial position), presumably allowing only part of the high-temperature fuels to enter the support structure in the process leading to the core melting.

Reflection in the estimate diagram of fuel debris distribution

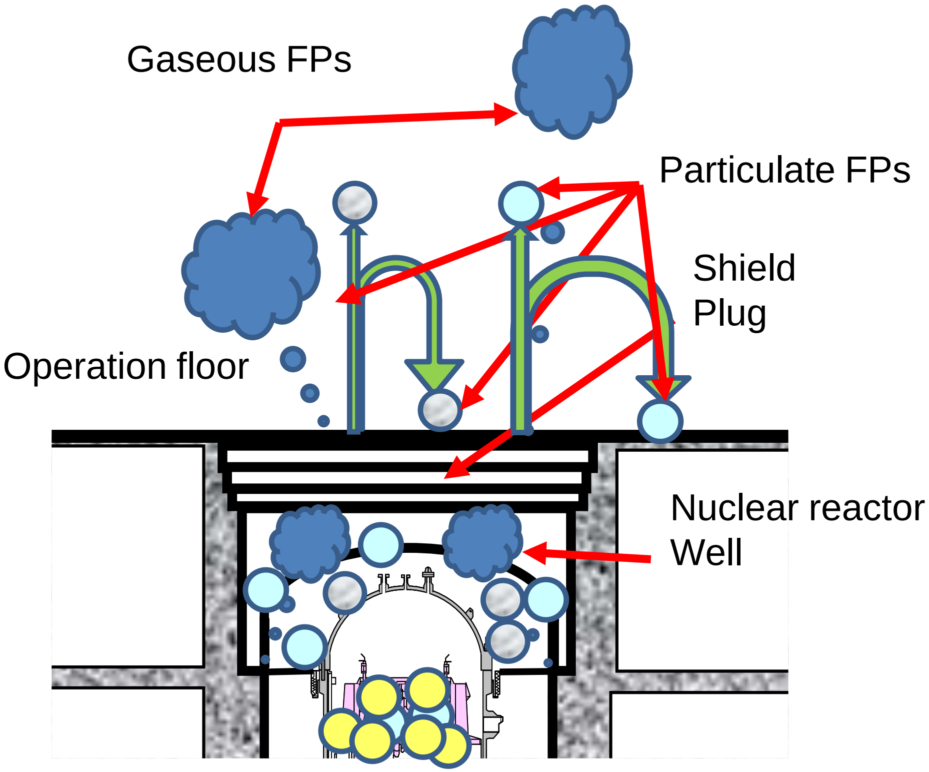
- ➢Combined with the knowledge obtained in the “event sequence analysis at the time of core material slumping,” the results were used to estimate the core sub-structure fracture mode in Unit 2 and the properties (mainly solid) of the fuels that still exist in the lower plenum.
- ➢The knowledge that would enable more accurate evaluation of the RPV fracture mechanism, the fracture mode, the fuel debris properties when the RPV was damaged, and the outflow property.



2.b) Evaluation of Chemical Characteristics of FPs

Purpose

To decommission the Fukushima Daiichi Units 1 to 3, it is essential to identify the distribution and chemical properties of cesium accounting for a major part of the gamma source. Under this project, apart from the typical chemical species in the evaporation, migration, and condensation processes (CsI: cesium iodide, and CsOH: cesium hydroxide), a study on the possible presence of compounds that are believed to be relevant to evaluating Fukushima Daiichi (molybdate compound, silicate compound, boronic acid compound, etc.) and their impact levels is conducted. If there is uneven distribution of cesium with poor solubility --- a factor that would negatively influence the top-entry method --- in the upper reactor core region due to the reaction with the upper-structure material (steel), was also tried to be found out. Furthermore, a study was conducted on the mechanism by which insoluble cesium particles/powder dust are generated and their migration routes.



Unit 2 Operation floor

(2)-[2]-1 Reaction with Cesium and Steel Materials, and Re-evaporation

[Overview and Objectives]

Overview: Conduct a test on Cs adsorption in steel materials and its re-evaporation, using parameters such as temperature and atmosphere to enhance knowledge of the Cs chemisorption behavior, thereby developing a Cs adsorption model that takes major compounds generated by adsorption into account.

Objectives: Verify the existing model through a replication experiment on Cs adsorption and evaluate the chemical and thermodynamic properties of relevant compounds to develop a model that takes various chemical conditions, such as re-evaporation and Cs concentration variation, into account.

[FY2017 Research Details]

(i) Evaluation test on adsorption and re-evaporation behavior
An experiment on Cs absorption was performed, using parameters such as temperature and atmosphere, and the effects of various chemical conditions was evaluated.

(ii) Development of an adsorption and re-evaporation model
The results of the Cs adsorption experiment based on the mass transfer theory and thermodynamic data on Cs adsorption products were analyzed and modeling to consider the effects of chemical conditions such as the CsOH concentration in the gas phase was performed.

[Final Results]

- ✓ The influence of chemical conditions were experimentally clarified, such as CsOH concentration in the gas phase, atmosphere, Si concentration in steel and temperature, on the Cs adsorption volume. A chemisorption model based on the test values was developed.
- ✓ By using the model formula, the sensitivity (fluctuation range) of Cs adsorption volume to the dryer and separator according to changes in chemical conditions, such as CsOH concentration in the gas phase, was exhibited.

Chemisorption model developed in this study

$$v_d \cong 7.027\frac{\sqrt{C_B}}{{C_g}^{0.5225}}\exp\left(-\frac{109000}{2RT}\right)$$

(*): The adsorption volume was calculated using: $\frac{1}{A}\frac{dM_d}{dt} = C_g v_d$

v_d: reaction rate constant [cm/s]
C_g: CsOH concentration in the gas phase [μg/cc]
C_B: Si concentration in steel [wt.%]
A: Adsorption surface area [cm^2]
M_d: Adsorption volume [μg]
t: Adsorption time [s]

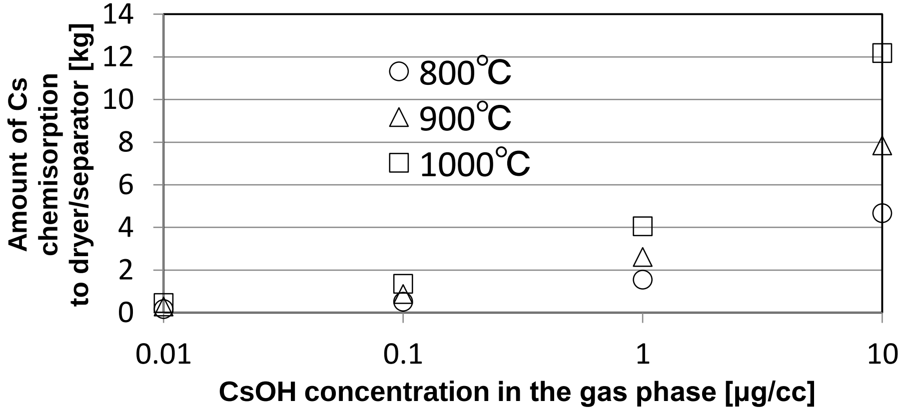


Figure 1. (Estimate) Cs adsorption volume to the dryer/separator in 1F2 where structural material temperature and CsOH concentration in the gas phase were constant for 3 hours

(2)-[2]-2 Evaluation of Particulate Cesium Compounds (1. Test on Reaction and Solidification in the Gas Phase)

[Overview and Objectives]

To clarify the generation mechanism of insoluble cesium particles, a test on reaction and solidification in the gas phase in FY2017 was conducted by using (1) Mo, (2) Zn coating, and (3) Ca as evaporation sources in addition to the standard simulation conditions of Cs-Si-O that were examined in FY2016.

[FY2017 Research Details]

An additional test was performed to examine the following possibilities:

[1] Generating insoluble Cs particles in the environment where molybdate coexists.

[2] Entry of the Zn coating component used for the S/C inner wall into insoluble Cs particles.

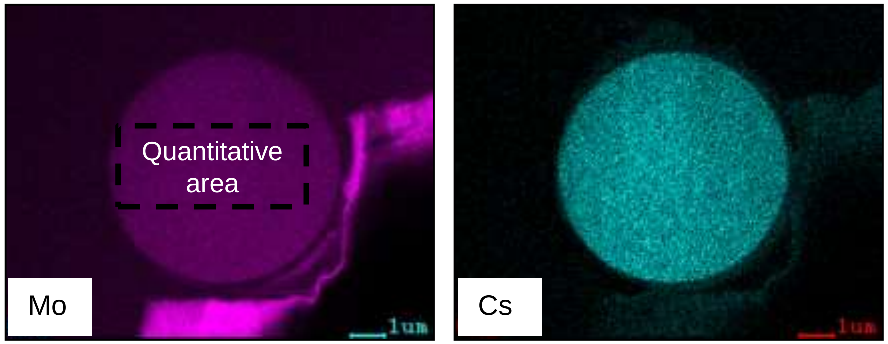
[3] An insulation material or concrete are the original component of insoluble Cs particles.

[Final Results]

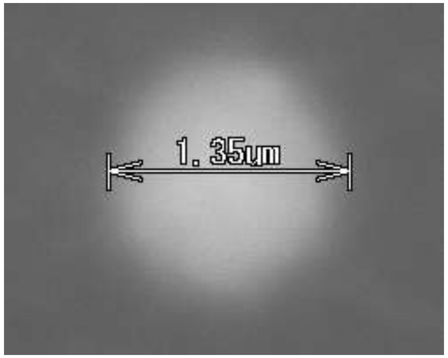
[1] When the evaporation source, Mo, was added to the simulation test conditions, soluble Mo-Cs-O particles were produced, but no insoluble cesium particles were produced (Right figure (a)). In the molybdate-bearing environment, it is believed that no insoluble Cs particles are produced; presumably they were produced in the early phase of core damage (where FP composition was Cs>>Mo).

[2] When the coating fragments of the S/C inner wall were added as an evaporation source, insoluble cesium particles came to contain Zn depending on the coating fragments' loading volumes (Right figure (b)).

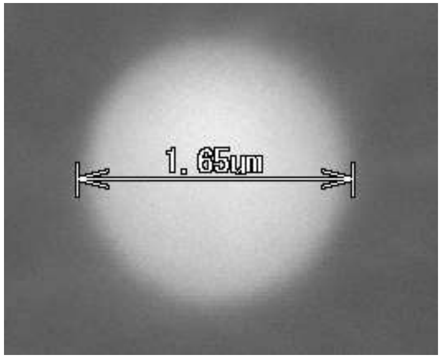
[3] When Ca-bearing material was added as an evaporation source, insoluble cesium particles came to contain Ca depending on the evaporation property of the Ca-bearing material (Right figure (c)). Presumably, neither concrete nor insulation was involved in generating insoluble Cs particles.



(a) Cross section of a Mo-Cs-O particle generated
(Mo: 46%, Cs: 53%, Si: 1%-metallic element, (at% in the metallic element))



(b) Zn-bearing cesium particles
(Si: 56%, Cs: 17%, Zn: 22%-metallic element)



(c) Ca-bearing cesium particles
(Si: 69%, Cs: 29%, Ca: 1%-metallic element)

(2)-[2]-2 Evaluation of Particulate Cesium Compounds (2. Study on the Microstructure)

[Overview and Objectives]
According to a report, a structure having nano-sized, network-like crystalline $ZnFe_2O_4$ was found in some insoluble Cs particles. The generation mechanism of those particles was studied by simulating and evaluating the microstructure of insoluble Cs particles.

[FY2017 Research Details]A simulated sample was prepared with the following two techniques to observe the microstructure.

[1] The sample was adjusted so it had the same composition as the insoluble Cs particles. It was loaded in a crucible and melted at 1,500°C in the air atmosphere, then quickly cooled it down with water.

[2] SiO_2-Fe_2O_3-ZnO was adjusted so it had a composition similar to that of the insoluble Cs particles and hence had the possibility of experiencing spinodal decomposition. Then, it was melted by gas levitation and laser heating and then solidified.

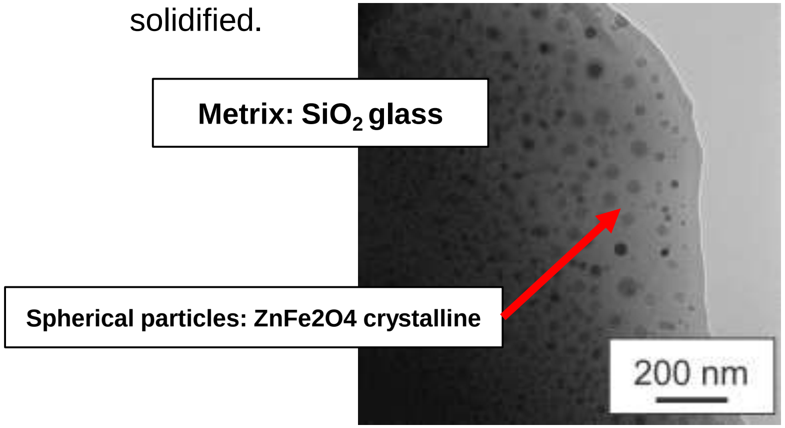


Figure 1: TEM image of sample 1 (Phase identification based on electron diffraction results)

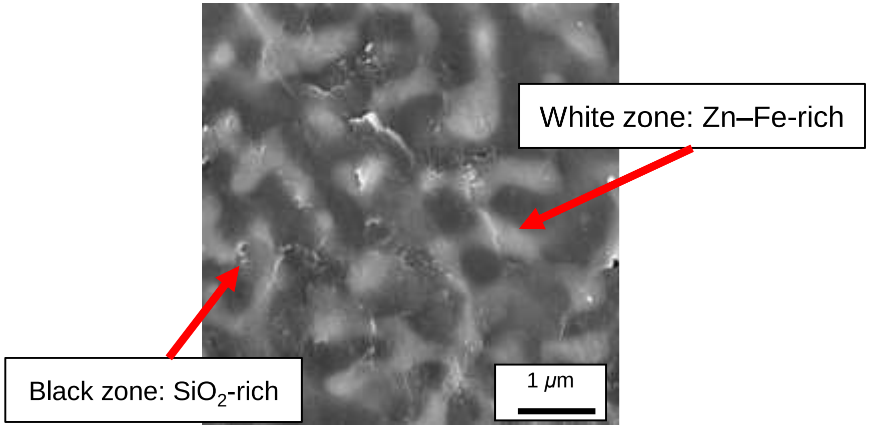


Figure 2: SEM image of sample 2 (Elementary analysis with EDS)

[Final Results]

- ✓ In the SiO_2 glass matrix of Sample [1], the generation of nano-sized crystalline $ZnFe_2O_4$ (Figure 1) was observed. The particle observed by Yamaguchi et al.*1 (a structure formed through the dynamic dissolution of $ZnFe_2O_4$) requires rapid cooling after it is generated at a high temperature. It suggests that this type of structure was generated at a high temperature in the RPV and then rapidly cooled in the S/C.
- ✓ Presumably, the network structure of the Zn-Fe-rich phase shown in Sample [2] (Figure 2) was generated by spinodal decomposition. The microstructure observed by Furuki et al.,*2 (a network crystalline structure of $ZnFe_2O_4$) is nano-sized and has a different dimension from the spinodal decomposition structure formed in the liquid phase mocked up under the conditions of Sample [2]. The origin of the nanostructure remains unknown.

*1: N. Yamaguchi et al., Sci. Rep. DOI: 10.1038/srep20548, 2016
*2: G. Furuki et al., Sci. Rep. DOI: 10.1038/srep42731, 2017

(2)-[2]-2 Evaluation of Particulate Cesium Compounds
(3. Evaluation of Inorganic Zinc Coating)

[Overview and Objectives]

A study on the possibility that inorganic Zn coating in the S/C inner wall was the material origin of the constituent elements (Zn and Si) of insoluble Cs particles, was conducted. A simulated coating sample was prepared by referring to construction specifications and the coating components' leaching behavior was examined under the test conditions simulating the accident (with purified water at 140°C for 30 hours).

[FY2017 Research Details]

- By using ICP and IC, the coating components before and after the leaching test (Table 1) was measured. Apart from Zn and Si, a small amount of K and Cl was detected.

Table 1. Chemical analysis results of the coating before and after the test

No.		ICP						IC	
		Zn	Si	K	Mg	Ti	Al	Cl	Balance
		mass%							
①	Before test	79	4.3	0.13	0.03	0.06	1.4	0.17	14.91
	After test	74	3.8	0.11	0.01	0.05	1.4	0.012	20.62
②	Before test	78	4.0	0.16	0.03	0.07	1.6	0.13	16.01
	After test	74	3.7	0.12	0.01	0.05	1.4	0.010	20.71

[Results]

✓ Inorganic Zn coating components can only be involved with Unit 2, where the RCIC water source was switched from the CST to the S/C. Hence, the RPV of Unit 2 is the site where the insoluble Cs particles occurred in the environment.

✓ From the amount of K_2O supplied from the coating (on the order of 100 g) and K_2O concentration in insoluble particles (approx. 1 wt%), the amount of insoluble Cs particles generated is estimated to be on the order of more than 10 kg (Table 2).

✓ As the ratio of SiO_2/K_2O supplied from the coating shows SiO_2 was insufficient, the evaporation of silica from the steel materials may have contributed to the amount.

Table 2. The estimated amount of insoluble Cs particles generated (based on the potassium amount)

Items	Evaluation values	Evaluation methods
1. Zinc coating weight	2,600 kg	Coating area (8,566 m^2) × Ratio of the smooth area (0.8) × Coating thickness (100 μm) × Coating density (3.8 g/cm^2)
2. K content	3.12 kgK (3.76 kgK_2O)	2,600 kg × 0.0012 (Table 1 average)
3. Amount of K_2O dissolved	0.752 kgK_2O	3.76 kgK_2O × 0.2 (dissolution rate; Table 1 average)
4. Brought into the RPV	0.125 kgK_2O	Dissolution amount × Amount of water evaporated/Total water amount (500 m^3/3,000 m^3)
5. Amount of Cs particles generated	**12.5 kg**	0.125 kgK_2O/0.01 (K_2O content by Kogure et al.)

(2)-[2]-2 Evaluation of Particulate Cesium Compounds
(4. Oxidation and Evaporation Behavior of Silicon in Steel Material)

[Overview and Objectives]
A study on the oxidation behavior of stainless steel and the existing state of silicon in such oxides under the temperature range and atmosphere containing hydrogen and water vapor that presumably simulate the temperature range and atmosphere at the time of the Fukushima Daiichi accident, was conducted. The thermodynamic properties of Cs-Si-(Fe)-O as a basic structure of cesium compounds are clarified.

[FY2017 Research Details]
- As shown in Figures 1 and 2, the Si distribution state in metallic phase near the metallic phase obtained by heating up SUS304 grade stainless steel was analyzed, in which Si content changed under water vapor atmosphere.
- As shown in Figure 3, using a thermodynamic database, the possibility that hydroxide and suboxide are the silicon sources of the afore-mentioned particles was estimated and studied.
- A Cs_2O-SiO_2 melt and copper, as basic structures, in a graphite crucible under coexistence of CO was equilibrated and the SiO_2 activity determined.

[Final Results]
- The existing state of Si oxides is divided mainly into Fe_2SiO_4 and SiO_2, largely depending on the state of the oxide layer of steel material.
- When the oxidation rate is high, the oxide layer is separated into two, and Si exists in the inner oxide layer as Fe_2SiO_4; then, it is transformed to SiO_2 as the oxide layer peeled off due to long-time heating.
- Thus light was shed on the atmospheric dependence of silicon source in insoluble cesium particles and the dependence of SiO_2 activity in cesium silicate melts on Raoult's law.

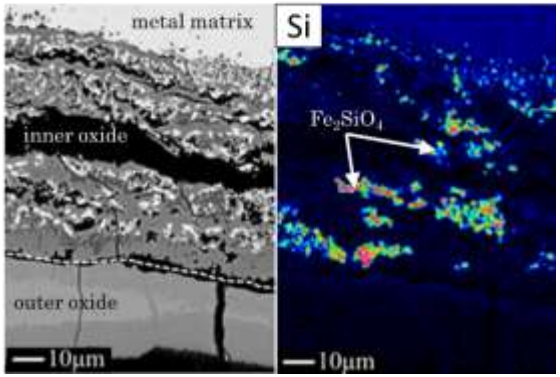


Figure 1. Si distribution in stainless steel containing 1% Si, which was oxidized for 60 minutes at 1,200°C under atmosphere of H_2O/Ar

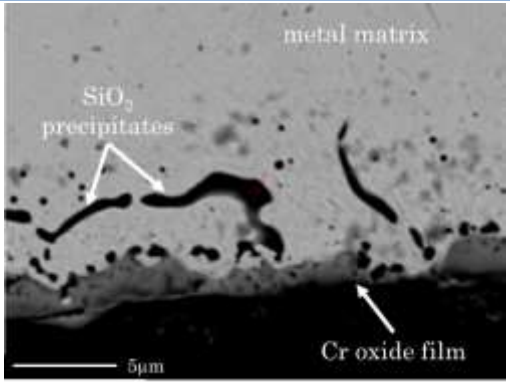


Figure 2. Si distribution in stainless steel containing 1% Si, which was oxidized for 60 minutes at 1,200°C under atmosphere of H_2/H_2O/Ar

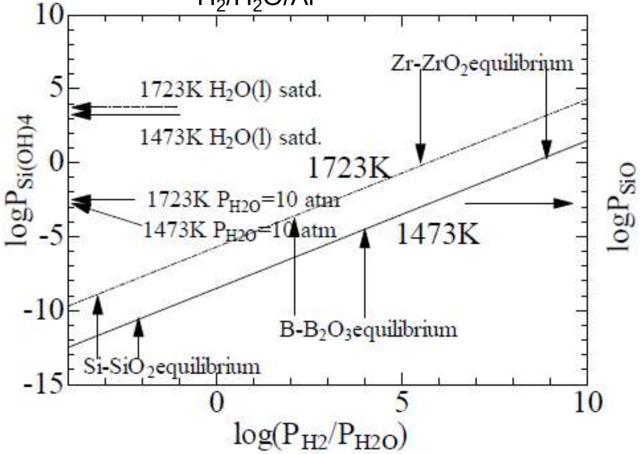


Figure 3. H_2/H_2O ratio dependence of vapor pressure of SiO and $Si(OH)_4$

(2)-[2]-2 Evaluation of Particulate Cesium Compounds
(5. Relation with the Accident Progression Sequence)

Time sequence	Generation process
During normal operation	• The major portion of Cs stays in the pellets, but some portion precipitated at the grain boundary. • Mo, as intermetallic compound, precipitated mainly in crystal grains.
Earthquake (14:46, Mar 11), Tsunami, Loss of AC power (15:40–15:41, Mar 11), Switching of RCIC water source (CST to S/C)	
While RCIC is operating	• S/C water temperature rose → Exceeded the heat-resistant temperature of the coating → Coating components leached → Coating components migrated due to the RCIC water flow → Evaporated to dryness (Si, Zn, K) on the boiling surface in the RPV
Stop of RCIC (around 9 am, Mar 14), Forced depressurization of RPV (18:00, Mar 14), Decrease in water level, Start of core heating	
Core heating	• Fuel cladding tube failure • Evaporation of Fe crud and coating components (Zn, Si, and K) attached to the surface of fuel cladding • Evaporation of Zircaloy component (Sn) • Meltdown of control rods (Meltdown forming Fe-B eutectic, without B's involvement) • Fuel heating during the meltdown of the upper tie plate, oxidation and evaporation of Si • Discharge of Cs due to grain boundary connectivity of pellets (>1,500^{o}C) • Liquid phase of Cs_2SiO_4-SiO_2 system → Densification and spheroidization of the condensation phase
Discharge from RPV	• As a result of opening the safety valve relief (21:20, Mar 14), spherical cesium-bearing particles were rapidly discharged from the RPV, along with a large amount of water vapor and non-condensable gas → Rapid cool-down→ Suppressed nucleation and spherical precipitation of $ZnFe_2O_4$ → Generation of insoluble Cs particles • Most of the particles were captured in the S/C water. Some of them were discharged to the environment.
Observed at environmental monitoring facilities in the Tokyo metropolitan area, including Meteorological Research Institute (7:00–16:00, Mar 15)	

(2)-[2]-2 Evaluation of Particulate Cesium Compounds
(6. Phenomenology of Material Mechanism)

Meltdown of control rods (CRs)
- Meltdown due to the eutectic reaction of Fe-B (at about 1,200 °C) (the evaporation of silicon was suppressed)
- HBO_2 evaporated, suppressed generation of $CsBO_2$

Meltdown of the upper tie plate
- Cast steel contains Si of approx. 1.5 wt%
- Presumably, the plate was heated up with fuel during the fall and became the source of Si

Involvement of S/C coating components (Unit 2)
- During the RCIC operation, the inorganic zinc coating component leached and migrated to the RPV
- Evaporated to dryness on the boiling surface (of fuel)
- Evaporated due to heat generated from the core

Insoluble Cs particles
SiO_2-$Cs_2Si_4O_9$
(ZnO-Fe_3O_4)

Si

Si, Zn, K, Cl

Fe, Sn

Cs

Mo

CsOH

Cs2MoO4

CsI

Cs

Cesium silicate on the steel material surface
$Cs_2Si_4O_9$, $CsFeSiO_4$

Phase stability of insoluble Cs particles
- Two immiscible liquid phases and dissolution gap
- Rapid cooling that suppressed the nucleation and spherical precipitation of $ZnFe_2O_4$
- Susceptibility for spinodal decomposition

Cs chemical form
- Under the environment where the generation of Cs_2MoO_4 and $CsBO_2$ was suppressed (CsOH environment), amorphous insoluble Cs particles and crystalline $Cs_2Si_4O_9$, $CsFeSiO_4$ were produced

Heat generation of core fuel
- Evaporation of iron crud
- Evaporation of cladding component (Sn)
- Discharge of Cs and I > 1,500 °C
- Discharge of Mo was limited < 2,300 °C

Vapor oxidation of the material of the structures at the upper portion
- Fe_2SiO_4 and SiO_2 were formed on the SUS surface
- Cesium silicate was generated by the reaction with Cs
- Possibility of cesium silicate evaporating at high temperature (>1,000 °C)

(2)-[2]-3 Optimization of the Evaluation Model for Cesium Compounds

[Overview and Objectives]

Overview: Optimize the adsorption model regarding the products of reaction between CsOH and SUS304 steel under an oxidized/reducing atmosphere based on the JAEA experiment results, and develop a generation model of particulate cesium.

Objectives: Introduce the adsorption model and the silicon generation model into SAMPSON, and develop a chemical characterization tool, based on the analysis results of the amount of silicon generated.

[FY2017 Research Details]

(i) Optimization of the adsorption model

A correlation equation for the temperature-dependent adsorption rate was used in a reducing atmosphere based on the existing data and it was determined that the effect of CsOH concentration can be evaluated using the corrected formula based on this project's results (2)-[2]-1.

(ii) Development of a generation model of particulate cesium

It was determined that the Si discharge sources were the evaporated Si component in SUS core structure materials, such as the control rod blade, core support plate, and the leached Si component in the S/C inner wall coating. Then, an Si discharge model and a particulate cesium generation model was developed.

[Final Results]

✓ The optimized adsorption model and the particulate Cs generation model in SAMPSON was incorporated.

✓ A tool to evaluate the validity of the branching ratio "f" was developed based on the study results of this project (2)-[2]-2.

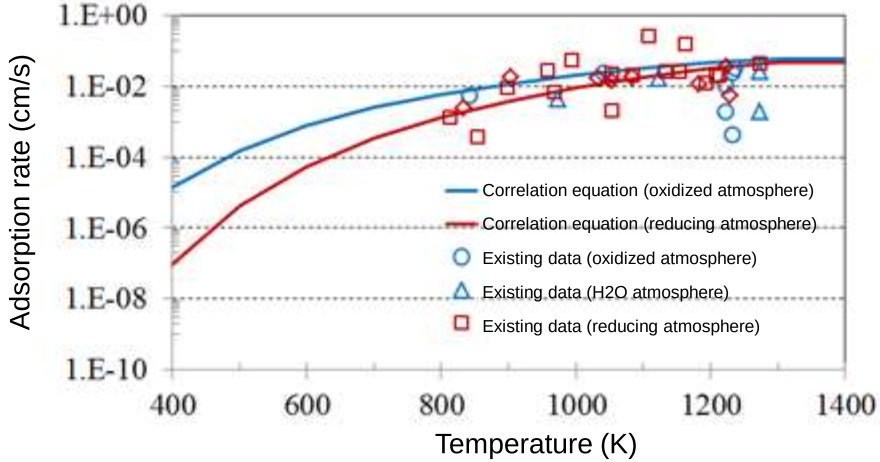


Figure 1. Temperature dependence of adsorption rate

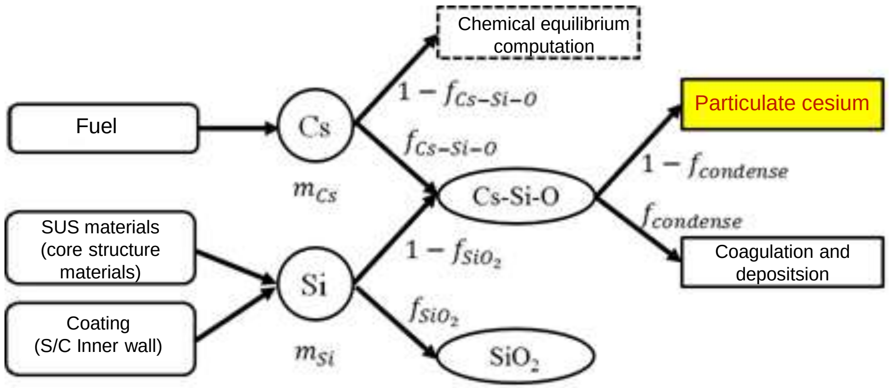


Figure 2. Generation model of particulate cesium

(2)-[2]-4 Analysis and Evaluation of the Actual Units Using the Upgraded Model

[Overview and Objectives]

Overview: Conduct analysis of Units 1 to 3 using SAMPSON, study the consistency with items such as actual measurement data that can affect the atmosphere, and evaluate the reliability of analysis results

Objectives: Evaluate the cesium distribution in the RPV and PCV based on the SAMPSON analysis results

[FY2017 Research Details]

(i) Analysis under the conditions of Fukushima Daiichi plant
The Units 1 to 3 using the SAMPSON were analyzed, for which the improved Cs behavior model was incorporated.

(ii) Study on the factors that affect Cs behavior

- Cs adsorption mainly occurs in an oxidized atmosphere, and adsorption in a reducing atmosphere ($P(H_2) > P(H_2O)$) lasted for a short time.
- Generating particulate Cs needs a greater amount of Si than that coming from the core structure materials, requiring Si from the coatings inside the S/C. Therefore, the study was conducted only on Fukushima Daiichi Unit 2, where RCIC was run by using the S/C as the water source.

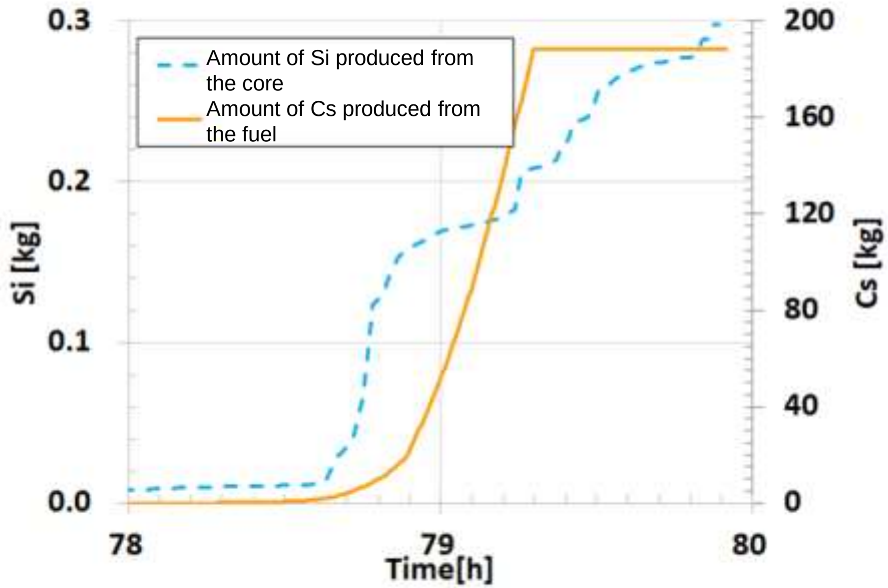


Figure 1. Time dependency of the amounts of Cs and Si generated in Fukushima Unit 2

[Final Results]

✓ Analysis was conducted and the Cs distribution in the RPV and PCV was calculated.

✓ It is believed that the D/W inside of Fukushima Daiichi Unit 2 has the highest contamination level when comparing Units 1–3. Meanwhile, Unit 2 analysis results showed almost no cesium on the D/W inner wall, inside the pedestal, or on each Fukushima Unit. This result indicates that the stage of simulating the current cesium distribution in each Fukushima Daiichi unit has not been reached.

✓ Current understanding and handling of the analysis codes are insufficient for analyzing the phenomenon.

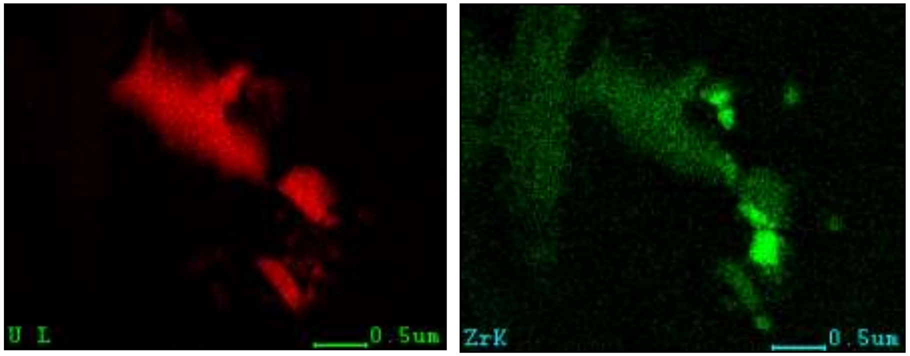
(2)-[2]-5 Analysis of Samples Collected at Fukushima Daiichi NPS

[Overview] The samples collected at Fukushima Daiichi NPS such as deposits in the contaminated water in the PCV of Unit 1, contaminant attached onto the blockage inside the TIP piping in Unit 2, and a surface of a protective covering piece on the Unit 2 operation floor were analyzed. A detailed observation by performing FE-TEM, with a focus on U-bearing particles was also conducted, and the samples' microscopic characteristics were sorted out.

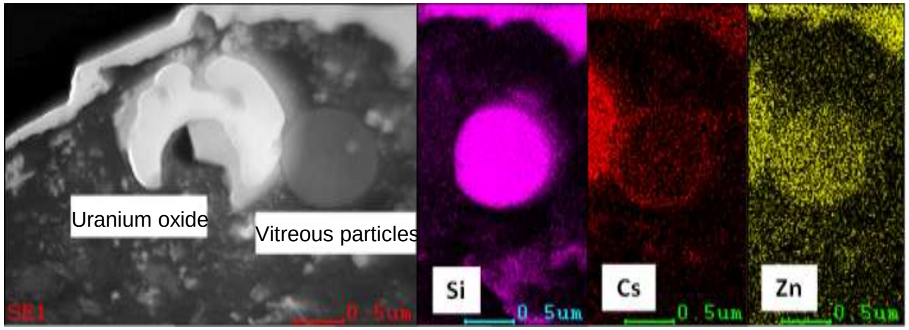
[FY2017 Research Details and Results]

[2] Detailed analysis of contaminated samples from Fukushima Daiichi NPS

- **The following was found in the deposits in the contaminated water in the PCV of Unit 1: rust containing Fe and Zn; C-bearing material; structure material component; glass material; Cu; Pb; Zr; and U-bearing particles containing Cu (Figure "a" in the right).**
- **In obstacles collected from the TIP piping at Unit 2, an area containing Zr apart from the structure material components and Mo was found.**
- **The following was observed on the protective sheet on Unit 2's operation floor: uranium oxide particles that do not contain Zr; and vitreous particles containing Zn, Cs, etc. (Figure "b" in the right).**



(a) U-bearing area inside the deposits in the contaminated water of PCV in Unit 1
(TEM-EDS mapping of the element U (right) and element Zr (left))



(b) Vitreous particles and uranium oxide on the protective sheet on Unit 2's operation floor

(2)-[3] Knowledge Utilization through International Joint Research

[Overview and Objectives]

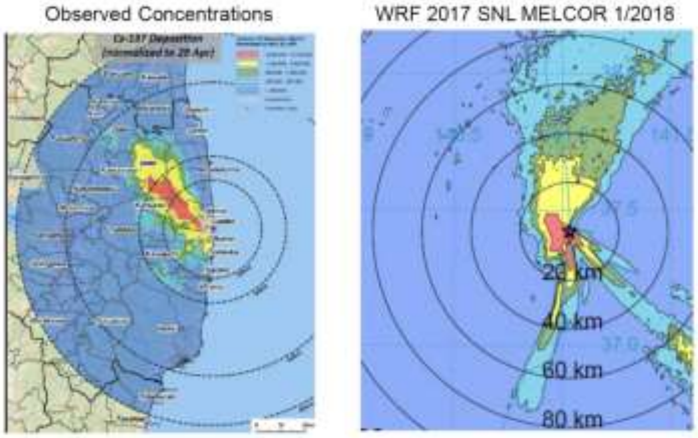
The Benchmark Study of the Accident at the Fukushima Daiichi NPS (BSAF) Project Phase-2 was established by the Organization for Economic Co-operation and Development (OECD)/the Nuclear Energy Agency (NEA). The BSAF project has been hosted by the several operating agents including the Institute of Applied Energy (IAE) which utilizes the data obtained through the project and understand the conditions inside the reactors.

[FY2017 Research Details]

In FY2017, the final year of the BSAF Project Phase-2, using major physical models, the analytical results provided by participating organizations on the accident progression behavior is compared and studied, within three weeks after the scram. Their consistency is also studied with the actual measurements and on-site investigation results. The results were shared among the BSAF participating organizations as the Phase-2 output.

[Final Report]

- In July 2017, the following meetings were held at the Kokukaikan hall in Tokyo: the PRG (Program Review Group) meeting; MB (Management Board) meeting; and the joint session with other projects for the Fukushima decommissioning. The past and latest results of the internal investigations and Japan's views regarding the accident scenario based on the knowledge obtained so far, were reported on . Opinions with experts from participating organizations were exchanged. The following were discussed: the results of the comparisons among the preliminary analyses, actual measurement data and on-site investigations concerning the accident progression, fuel debris distribution and FP distribution within the three weeks after the scram; and the structure of the final report.
- In January 2018, the PRG, MB and joint sessions at the OECD/NEA headquarters in Paris were had. The evaluation results that the participating organizations provided on the accident progression within three weeks after the accident were compared against the actual measurement data and the on-site investigation results. It was confirmed that our understanding of the accident progression significantly deepened from Phase-1, as is demonstrated, for example, by the basic consistency between the evaluation results concerning the FP discharge amount and the amount discharged to the environment. The knowledge obtained through such discussions with overseas institutions was used, for our comprehensive analysis and evaluation, which helped improve the estimation accuracy on the fuel debris distribution.



Comparative example of observation and analysis results of Cs-137 depositsion

3. Research Details and Results Summary for FY2017

3. Research Details and Results Summary for FY2017

(1) Comprehensive analysis and evaluation of the conditions inside the reactors

[1] Comprehensively analyze and evaluate based on the Units data and other R&D project results

The conditions inside the RPVs and PCVs were comprehensively analyzed and evaluated based on the results of the following initiatives: “analysis and evaluation based on on-site information;” “analysis and evaluation based on the data at the time of or after the accident and inverse problem analysis;” and “analysis and evaluation utilizing analysis codes.” Then, the estimate diagrams of fuel debris distribution and FP distribution was developed.

[2] Establish a database for comprehensive analysis and evaluation

To share knowledge with overseas institutions more efficiently, the English search function was added to the database and the graph display function for the measurement data, etc. collected mainly within three weeks after the accident was improved. The adding of documents and data with search tags to the database that was developed under this project, contributing to a more efficient, comprehensive analysis and evaluation was also continued.

3. Research Details and Results Summary for FY2017

(2) Estimation and evaluation of fuel debris behavior and fission product behavior and its characteristics that would be useful for the comprehensive analysis and evaluation

[1] Reduction of uncertainties through analytical techniques

Sensitivity and other analyses were performed using the accident progression analysis codes, taking the boundary conditions and analysis models into account, with regard to the events assumed to have occurred inside the reactors. Thus knowledge was obtained that will be useful for comprehensive analysis and evaluation.

[2] Evaluate FPs' chemical characteristics

When evaluating the FPs' chemical characteristics, light was shed on Cs in particular, as it was believed the information would be important for decommissioning. Cs distribution and its chemical characteristics were studied, such as: identifying chemical species requiring special consideration, apart from the typical chemical species of CsI and CsOH; the production amount of insoluble Cs particles observed in the environment; and the possibility of uneven distribution of Cs due to a reaction with the structures at the upper portion of the RPV. The samples collected on-site were also analyzed and the composition and spatial distribution of uranium and FPs to understand the conditions inside the reactors was studied.

[3] International joint research to use knowledge inside and outside Japan

By using a database that was developed for the project and through managing an international joint research project (OECD/NEA BSAF Project Phase-2), the accident progression scenario and plant-related information was shared with overseas institutions. The results provided by participating organizations concerning the accident progression, fuel debris distribution, and FP distribution in the first three weeks after the accident were also compared with the measurement data and the on-site investigation results. As a result, our shared understanding about the accident progression and plant status deepened compared to Phase 1, leading to a remarkable reduction in the variety of analysis results from participating organizations and to the basic consistency of the evaluated FP discharge amount with the measured discharge to the environment. This deeper understanding of the accident progression contributed to improved accuracy of fuel debris distribution estimates. The knowledge obtained through the discussions with overseas institutions was also used for our comprehensive analysis and evaluation.

Attachment

- Attachment 1. Project Organization Chart (detailed)

Attachment 1. Project Organization Chart (detailed)

International Research Institute for Nuclear Decommissioning (IRID)

IRID Head Office
- ○ Research management of implementation plans, their progress status, etc.
- ○ Project results summary and information dissemination
- (1)-[1]-1 Extraction of issues for further research to reduce uncertainties of the estimated conditions inside the reactors
- (1)-[1]-2 Comprehensive analysis and evaluation based on the data from the actual units and other research development results

IRID/JAEA
- (1)-[1]-1 Extraction of issues for further research to reduce uncertainties of the estimated conditions inside the reactors
- (1)-[1]-2 Comprehensive analysis and evaluation based on the data from the actual units and other research development results
- (2)-[1]-3 (a) (4) Event sequence analysis at the time of core material slumping
- (2)-[1]-6-1 Simulated fuel assembly degradation test
- (2)-[2] Analysis and evaluation of chemical forms of FPs

IRID/Hitachi GE
- (1)-[1]-1 Extraction of issues for further research to reduce uncertainties of the estimated conditions inside the reactors
- (1)-[1]-2 Comprehensive analysis and evaluation based on the data from actual units and other research developments results
- (2)-[1]-1 Evaluation of the concrete erosion state at the pedestal using sensitivity analysis
- (2)-[1]-3 Accident progression analysis using the analysis code

IRID/Toshiba Energy Systems & Solutions
- (1)-[1]-1 Extraction of issues for further research to reduce uncertainties about the estimated conditions inside the reactors
- (1)- [1]-2 Comprehensive analysis and evaluation based on the data from the actual units and other research development results
- (2)-[1]-3 Accident progression analysis using the analysis code

(outsourcing) Electric Power Research Institute (EPRI), USA — (2)-[1]-3 Accident progression analysis using the analysis code

Organizations under IRID/JAEA:

- (Consignment research) The University of Tokyo — (2)-[1]-2 Inverse problem evaluation using virtual reactors
- (outsourcing) Hitachi Power Solutions — (2)-[1]- 2 Inverse problem evaluation using virtual reactors (Analysis and evaluation of the cooling characteristics of debris considering water injection)
- (Consignment research) Waseda University — (2)-[1]-3-(a)-4 Event sequence analysis at the time of core material slumping [MELCOR analysis]
- (Consignment research) The University of Tokyo — (2)-[1]-3-(a)-4 Event sequence analysis at the time of core material slumping (Analysis of pipe deformation)
- (outsourcing) Sojitz Machinery Corporation — (2)-[1]-6-1 Simulated fuel assembly degradation test (plasma heatup experiment)
- (outsourcing) Inspection Development Company Ltd. — (2)-[1]-6-1 Simulated fuel assembly degradation test (Service contract for CT imaging in the plasma melting test)
- (Consignment research) Osaka University — (2)-[2]-2 Evaluation of particulate cesium compounds (Property evaluation test for cesium-bearing glass)
- (Consignment research) Tokyo Institute of Technology — (2)-[2]-2 Evaluation of particulate cesium compounds (Study for evaluation of thermodynamic quantities of Cs-Si-(Fe)-O system compound)
- (Consignment research) Tohoku University — (2)-[2]-2 Evaluation of particulate cesium compounds (Study of silicon behavior in steel materials)
- (outsourcing) Art Kagaku Co.,Ltd. — (2)-[2]-1 Response between cesium and steel materials, and re-evaporation (Evaluation test for adsorption and re-evaporation behavior)
- (outsourcing) Nippon Nuclear Fuel Development Co., Ltd. — (2)-[2]-2 Evaluation of particulate cesium compounds (Study on the production conditions of particulate cesium)
- (outsourcing) Nippon Nuclear Fuel Development Co., Ltd. — (2)-[2]-5 Analysis of samples collected at Fukushima Daiichi Nuclear Power Station (Analysis of 1F contamination samples)

The Institute of Applied Energy (IAE)

IAE
- (1)-[1] Comprehensive analysis and evaluation based on the data from the actual units and other research development results
- (1)-[2] Establishment of a database for comprehensive analysis and evaluation
- (2)-[1]-3 Accident progression analysis using the analysis code
- (2)-[2] Analysis and evaluation of chemical forms of FPs
- (2)-[3] International joint research to make use of knowledge inside and outside Japan

(outsourcing) Mizuho Information & Research Institute, Inc.
- (1)-[2] Management and maintenance of DB site and upgrade of display functions
- (2)-[2] Coding work associated with the FP model introduction
- (2)-[3] Management and maintenance of web portal for NEA BSAF project

(outsourcing) Advance Soft Corporation
- (2)-[2] Coding work associated with the FP model introduction

(Affiliated organizations)
- TEPCO
- NDF