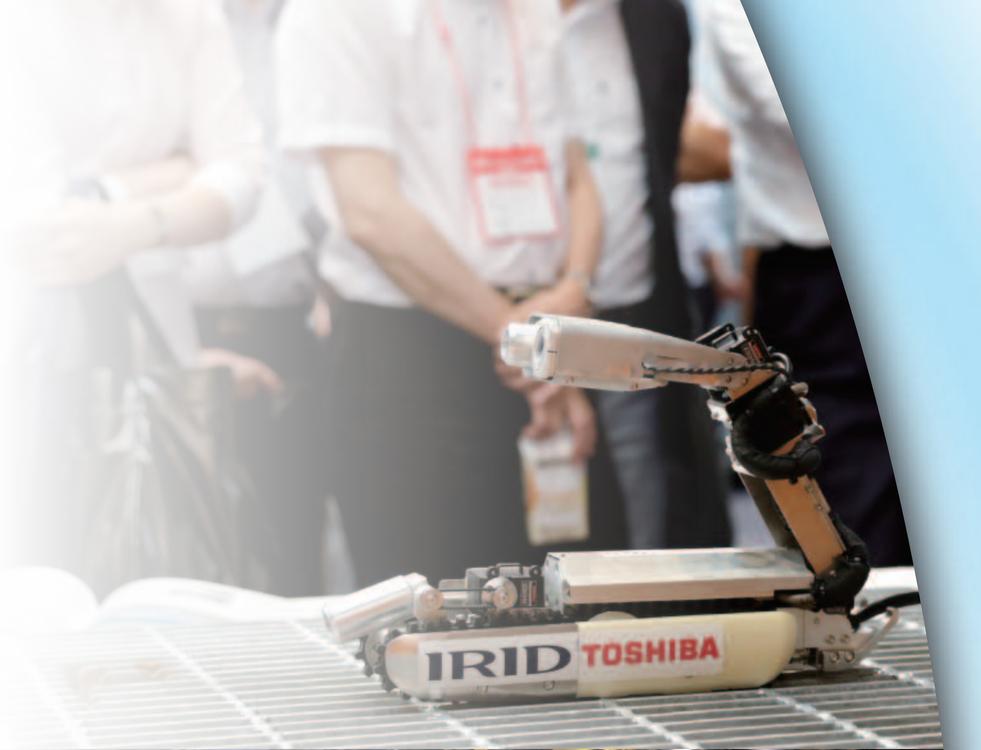
Annual Research Report 2016

IRID
International Research Institute for Nuclear Decommissioning



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Message from the President

More than six years have passed since the accident at the Fukushima Daiichi Nuclear Power Station (NPS) was caused by the Great East Japan Earthquake. The situation at the NPS has much improved after the accident; however, we are facing a crucial stage for decommissioning.

In the "Mid-and-Long-Term Roadmap towards the Decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Station Units 1-4," (hereinafter, "Mid-and-Long Term Roadmap") revised in June 2015, the latest milestones are clarified. Also, "Technical Strategic Plan for Decommissioning of the Fukushima Daiichi Nuclear Power Station of Tokyo Electric Power Company Holdings, Inc." which provides technical evidence for the Mid-and-Long Term Roadmap was updated by NDF in July 2016. As R&D themes and major technical issues to be solved are getting apparent, further achievements are being required.

The International Research Institute for Nuclear Decommissioning (IRID) has been addressing various themes in the following three R&D categories as the most urgent issues toward the decommissioning of the Fukushima Daiichi NPS since its start in August 2013.
(1) R&D for fuel removal from spent fuel pool
(2) R&D for preparation of fuel debris retrieval
(3) R&D for treatment and disposal of radioactive waste

In order to promote the R&D activities, we committed to gathering expertise from around the world under an integrated management which encouraged mutual coordination among projects, overlooking the whole R&D projects, and conducted fourteen subsidized projects and three in-house R&D projects in FY 2016.

As a result, robotic technologies have been developed; remotely operated robots for the decontamination of the reactor building and robots for the investigation inside the Primary Containment Vessel (PCV) have been developed and verified the feasibility of the technologies on-site, which brought us effective information inside the PCV. Furthermore, with the Muon tomography using permeation method, we estimated the conditions of debris inside the Reactor Pressure Vessel (RPV), and have achieved success in the development of technologies, which is essential for fuel debris retrieval.

This annual report is intended to introduce achievements of R&D which IRID had taken upon as a challenge since FY 2016. We hope this report will help you understand the results of our R&D.

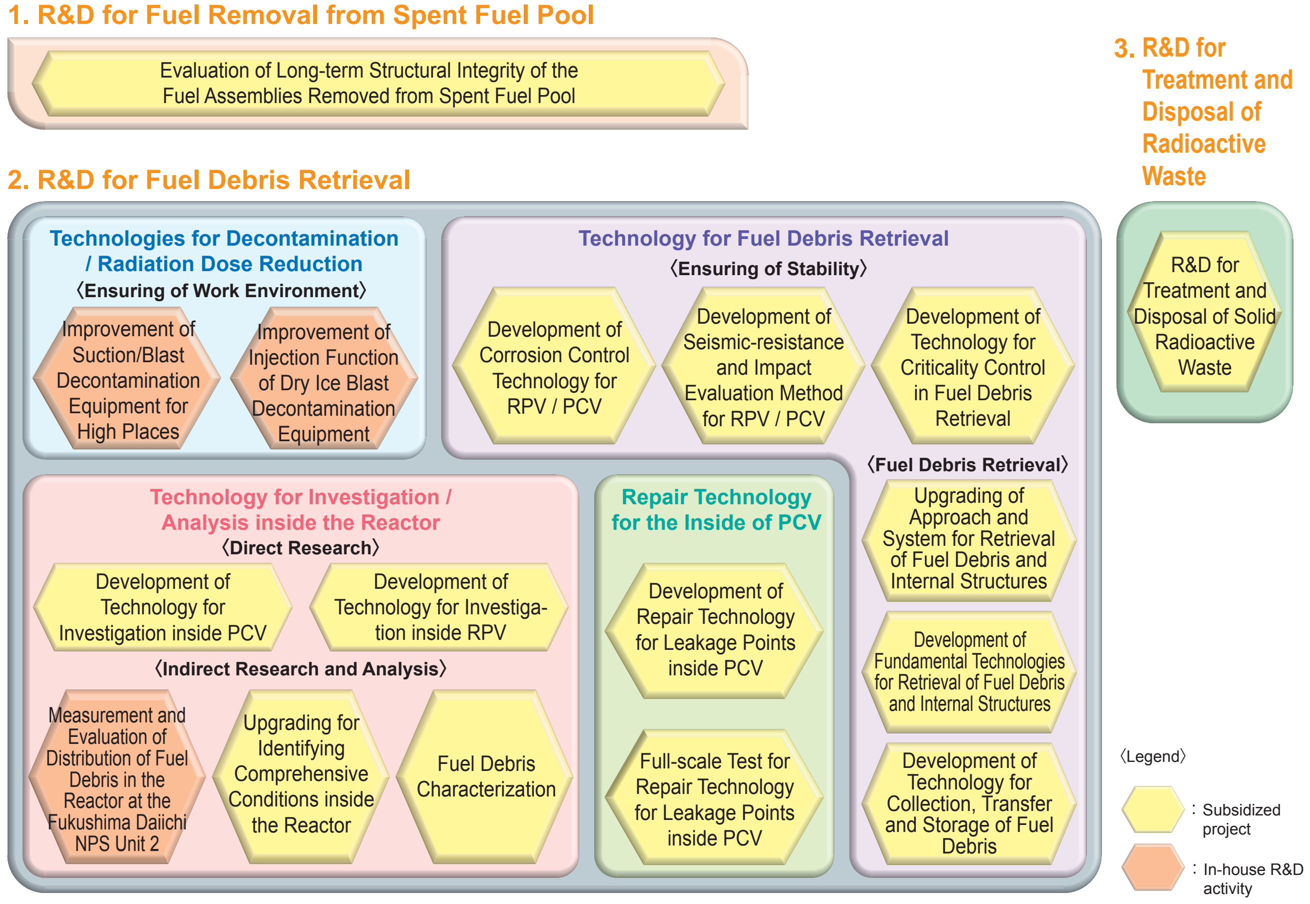
IRID will continue to take responsibility for the R&D on the steady and efficient decommissioning toward fuel debris retrieval which is facing a crucial stage. We are deeply grateful for the continuous support and guidance, and would like to express sincere gratitude and appreciation to all.

March 2017

Hirofumi Kenda

President, International Research Institute for Nuclear Decommissioning

IRID Research and Development Project, FY 2016 (Overview)



Contents

Key Challenge 1 R&D for Fuel Removal from Spent Fuel Pool

Evaluation of Long-term Structural Integrity of the Fuel Assemblies Removed from Spent Fuel Pool

Background

Fuel assemblies in the spent fuel pools at Units 1-4 at the Fukushima Daiichi Nuclear Power Station (NPS) have been stored in a water quality environment different from normal conditions due to the injection of seawater and rubble falling into the pool. To achieve long-term storage of fuel assemblies in a common pool and dry storage system in future, it is necessary to accurately evaluate the effects of these conditions on fuel assembly component materials and optimal storage environmental conditions.

Aims

In order to evaluate whether fuel assemblies retrieved from the spent fuel pools at Units 1-4 at the Fukushima Daiichi NPS can be stored safely in the common pool for a long time or not, we perform corrosion tests and investigate actual fuel in an environment that simulates real storage conditions. We also conduct a simulated test to evaluate the impact on the fuel integrity during dry storage.

Main Achievements and Approaches

1 Technical development for the evaluation of the long-term integrity of the fuel assemblies

① Evaluation on the deposits on the surface of the fuel assemblies

Materials of fuel assemblies (lock nuts) of Units 4 at the Fukushima Daiichi NPS stored in the common pool were transported to the post irradiation test facility, and component analysis for white deposits and measurement of corroded crevice re-passivation potential were carried out. Among the constituents of the white deposit, the amount of Mg was the largest, and the amounts of Al and Si were about half of it. The amount of Cl was below the detection limit. Since $Mg(OH)_2$ was separated and Cl was not captured, it is considered that there is no possibility of corrosion (Figure 1). In the electrochemical test, there was no crevice corrosion sensitivity in the area where the chloride ion concentration was lower than 100 ppm, and it was confirmed that there was almost no possibility of corrosion in the common pool (Figure 2).

② Evaluation of integrity of fuel in dry storage conditions

Assuming dry storage of the fuel assemblies from the spent fuel pool of Units 1-4 at the Fukushima Daiichi NPS, in order to confirm integrity of the fuel assemblies affected by rubble falling and seawater components, which were to be stored in the dry system, a hydride precipitation behavior verification test and a creep test were carried out, and the impact of the factors unique to Units 1-4 at the Fukushima Daiichi NPS on material properties was evaluated. It was confirmed that the impact on hydride precipitation behavior and creep behavior was small even in the condition affected by both rubble damages and seawater injection (Figure 3 and 4).

2 Basic tests for long-term structural integrity

A test to evaluate seawater components transfer behavior in the crevice structure of fuel components was carried out, and it was found that the seawater components did not concentrate in the crevice structure and it changed according to the salt concentration outside the crevice (completed in FY2015).

Future Developments

From the viewpoint of long-term integrity evaluation of fuel assemblies, the fuel assemblies of the spent fuel pool of Units 1-4 at the Fukushima Daiichi NPS were stored in the irregular water quality environment affected by seawater injection and rubble contamination, the impact of such special background on wet storage and dry storage was evaluated, and it was confirmed that the impact was small.

When a new issue is extracted at the time of spent fuel removal in the future, we will have to consider the necessity of new integrity evaluation.

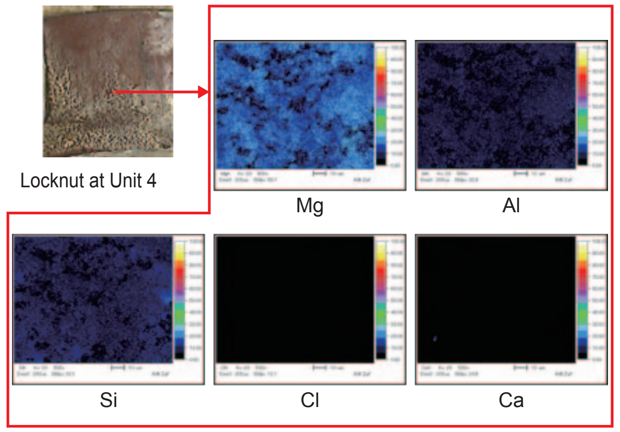


Figure 1: Component analysis result of white deposition area of locknut at Unit 4

Some metal components in seawater are detected in the white deposits, but Cl is not detected. This is considered to be a metal oxide rather than a chloride, and there is no possibility of corrosion.

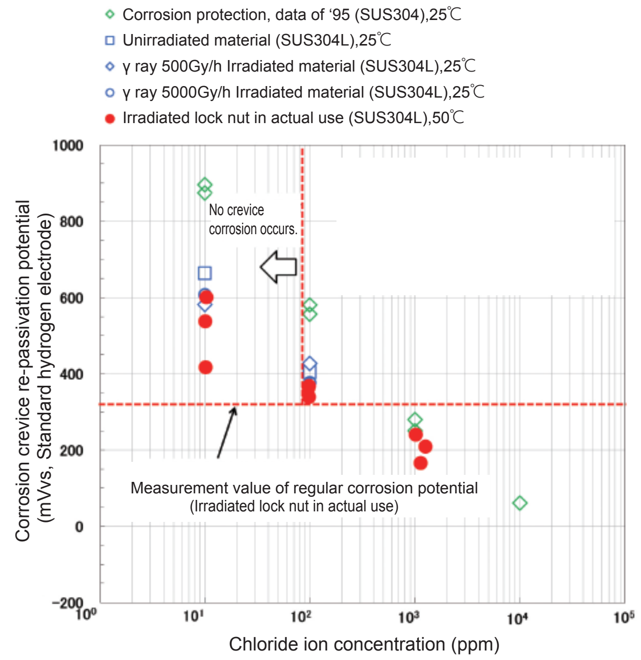


Figure 2: Re-passivation potential measurement result on corroded crevice of lock nut

When the chloride ion concentration is 100 ppm or lower, there is no possibility of corrosion because the potential where crevice corrosion occurs is in the region that is not corroded due to formation of oxide film (equal or lower than the corrosion crevice re-passivation potential).

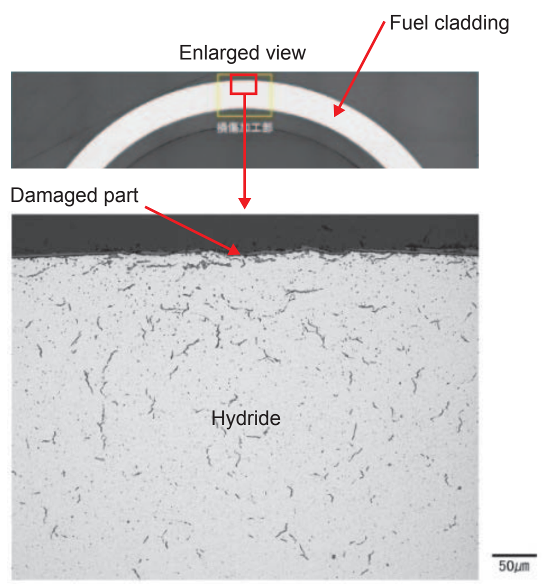


Figure 3: Hydride precipitation behavior verification test result

(Irradiated test piece, 300℃, cooling velocity 0.04℃/h, stress in a circumferential direction 70MPa, attached scratches and seawater, fixed rubble)

There was no clear difference in the hydride precipitation state of the damaged part even under the combined three conditions such as fixed rubble, seawater injection, and damage.

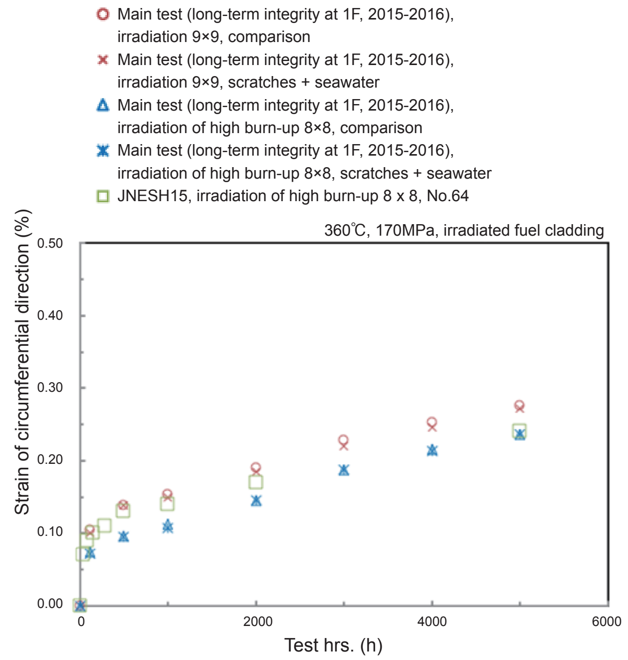


Figure 4: Creep speed test result

(Irradiated test piece, 360℃, stress in a circumferential direction 170MPa, attached scratches and seawater injection, 5000 h)

Long time creep tests of 5000 hours were conducted for the cases with/without seawater injection, and with/without damage. There was no large difference in their results.

Key Challenge 2 R&D for Fuel Debris Retrieval

Development of Technology for Investigation inside the Primary Containment Vessel

Background

At Units 1-3 at the Fukushima Daiichi NPS, after the reactor core has melted, nuclear fuel supposedly exists with some parts of reactor internals as fuel debris inside Reactor Pressure Vessel (RPV) and Primary Containment Vessel (PCV). It is considered that the fuel debris that dropped from the bottom of the RPV into the pedestal supporting the RPV were then spread from the opening at the bottom of the pedestal to the outside of the pedestal and distributed at the bottom of the PCV, but the condition of fuel debris is not still identified.

Aims

By accessing the inside of the PCV from X-100B penetration at Unit 1 and from X-53 penetration at Units 2 and 3, visual images, radiation dose, temperature, and other data of the inside of the PCV were collected. In addition to the severe environment with high dose and high humidity, it has been confirmed that visibility is restricted due to vapor and stagnant water in the dark. It is also possible that interference objects have been generated from the accident.

Therefore, it is necessary to develop technologies that can overcome these challenges and enable investigation inside the PCV.

Main Achievements and Approaches

1 Development of technologies to access inside / outside pedestal

① Technology to access outside of the pedestal at Unit 1 (B2 investigation technology)

We developed a B2 investigation device to access the inside of the PCV from Unit 1 X-100B penetration and investigate the distribution of the fuel debris on the basement floor outside the pedestal, and carried out preparations for on-site verification at Unit 1 (Appearance of device: Figure 1).

② Technology to access inside of the pedestal at Unit 2 (A2 investigation technology)

We developed a device that could remotely drill Unit 2 X-6 penetration and an A2 investigation device to access the inside of the PCV from the drilled X-6 penetration to investigate the condition inside the pedestal. We completed the on-site verification of the drilling device in December 2016, and the verification test related to the A2 investigation in February 2017 (Appearance of device: Figure 2).

③ Technology to access inside of the pedestal at Unit 3 (Unit 3 investigation technology)

We have completed test manufacturing of an underwater swimming device which accesses inside the PCV from Unit 3 X-53 penetration and investigates the condition inside the pedestal (Image of device: Figure 3).

2 Development of technology for investigation inside PCV in next phase

In addition to the technologies to access inside / outside the pedestal mentioned in the previous section 1, from necessity of investigation aiming at obtaining more detailed information, we have prepared a plan of technology for investigation inside the PCV in the next phase.

In planning, we organized and analyzed the latest needs found in each project to establish targets of investigations inside the PCV for the development of the fuel debris retrieval technology, extracted technical development tasks, and prepared for investigations and development inside the PCV.

Based on those plans, we examined the technology for establishing an access route to the inside of the PCV, access investigation technology, and measurement technology, and conducted element tests to construct the concept.

Future Developments

We completed the B2 investigation at Unit 1 in FY2016 and continue the development for the investigation inside the pedestal scheduled for Unit 3 in FY2017, aiming to complete the verification test.

Also, we will continue to develop technologies for investigations in the next phase to acquire further information about the inside of the PCV.

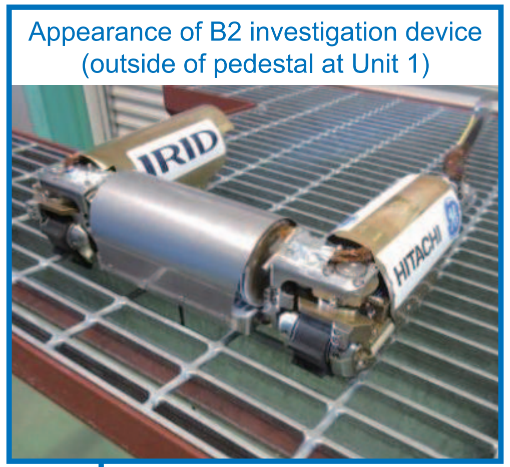


Figure 1

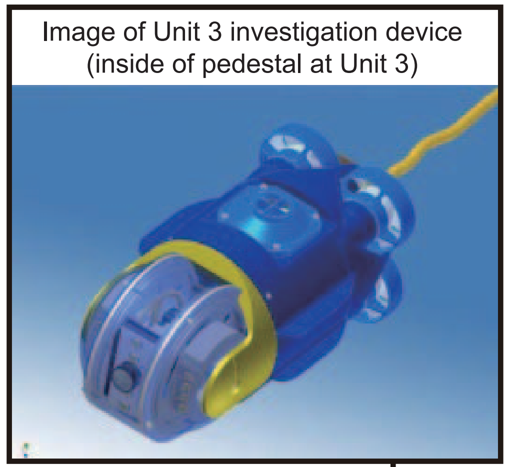


Figure 3

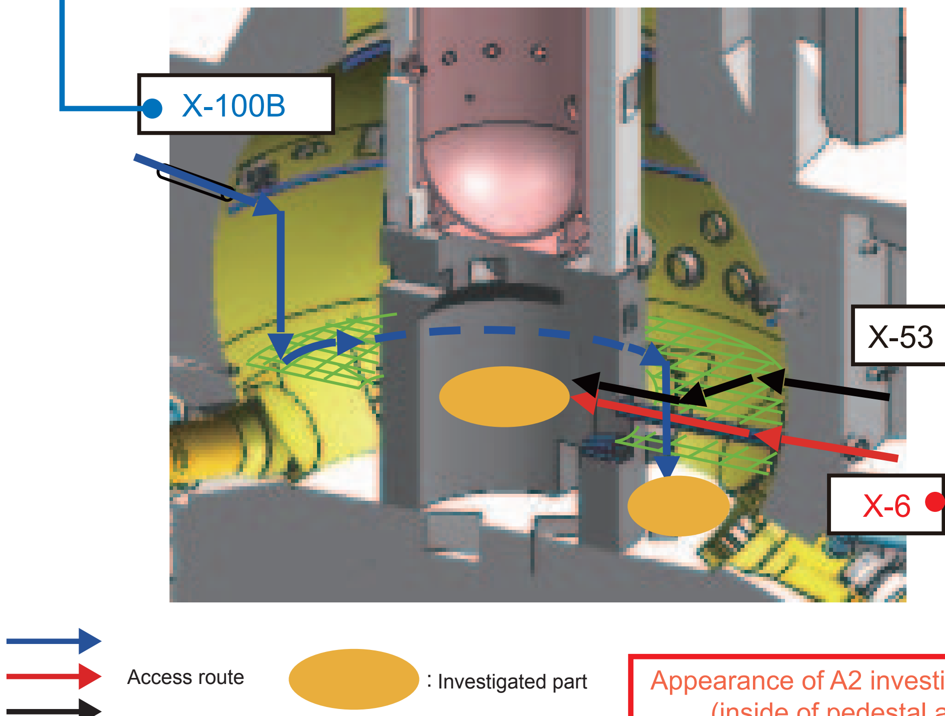


Figure 2

Key Challenge 2 R&D for Fuel Debris Retrieval

Development of Technology for Investigation inside the Reactor Pressure Vessel

Background

In order to retrieve the fuel debris and reactor internals, information about their locations, shapes and conditions must be understood in advance. However, since the inside of RPV has a complicated structure with extremely high radiation level, it is difficult to acquire directly information about the inside of RPV.

Aims

In order to promote the future decommissioning including retrieval of the fuel debris safely and smoothly, it is required to collect information on conditions inside the reactor in which are unclear now. Therefore, we clarify investigation targets, investigation methods and obtained results, and establish a comprehensive investigation method, to develop feasible technologies.

Main Achievements and Approaches

1 Formulation and updating of plan for investigation/development

We investigated and organized relevant development projects and on-site needs, and extracted useful information for developing a debris retrieval method such as appearance and radiation dose at the reactor core and reactor bottom for investigation. Also, regarding the access route for investigation of the reactor core, we selected not only the method allowing access from the operating floor above the RPV but also the method allowing access by drilling from the side of the reactor building as an applicable option, and evaluated feasibility of assessment of the building strength (Figure 1).

2 Development of equipment to access the reactor core from the top

① Development of drilling device to access through RPV head

We compared and evaluated several methods considered to be applicable at each work, concerning positioning of holes on the Primary Containment Vessel (PCV) head and the RPV head, processing of workpieces (leaving or collecting), etc. and selected applicable technologies and methods (Figure 2).

② Development of boundary function maintaining device / access device for work

We evaluated the exposure caused by dust scattering to air during processing of the reactor internals with a simple model. Also, after cleaning of the PCV head surface, we conducted an element test to verify the method to seal the connecting part between the PCV head and the guide pipe, with respect to maintenance of the boundary function to prevent contamination from spreading. In addition, we compared and evaluated several methods considered to be applicable at each work, concerning procedures for exchanging resin packings used for the connecting parts, and selected applicable technologies and methods (Figure 2).

③ Development of drilling equipment through upper grid plate

We manufactured a prototype of small tool head of Abrasive Water Jet (AWJ), which was remotely operated for opening work in the narrow part of reactor internals, and conducted an elemental test to confirm its drilling workability. Also, we evaluated fragments generated by the air AWJ for a mechanism that allowed to descend while opening on a complex shaped structure, and conducted a conceptual design (Figure 3).

3 Development and selection of investigation method up to the reactor core

We have confirmed applicability to the environment (high dose and fog) with the elemental test for the technology to investigate the internal situation and dose rate by passing through small diameter access routes. In addition, we conducted a conceptual design of the mechanism that brought the investigation devices closer to the deeper part in the reactor core.

4 Design of total investigation device system and method planning

Since the inside of the reactor well, individual structures, and condition of the access route to the reactor core part, the dose rate and others are unknown, we classified the work steps in confirming and establishing the access route and examined plans to be implemented together with the investigation, in order to implement the safe and reliable method.

Future Developments

Concerning the top and side hole drilling investigation method, we will conduct a detailed design of the equipment related to the applicable technologies and methods while ensuring safety, and verify the possibility of remote operation on site with a partial mockup test. In addition, we will formulate a series of work steps from installing the devices on site, conducting investigations, to processing after the investigations.

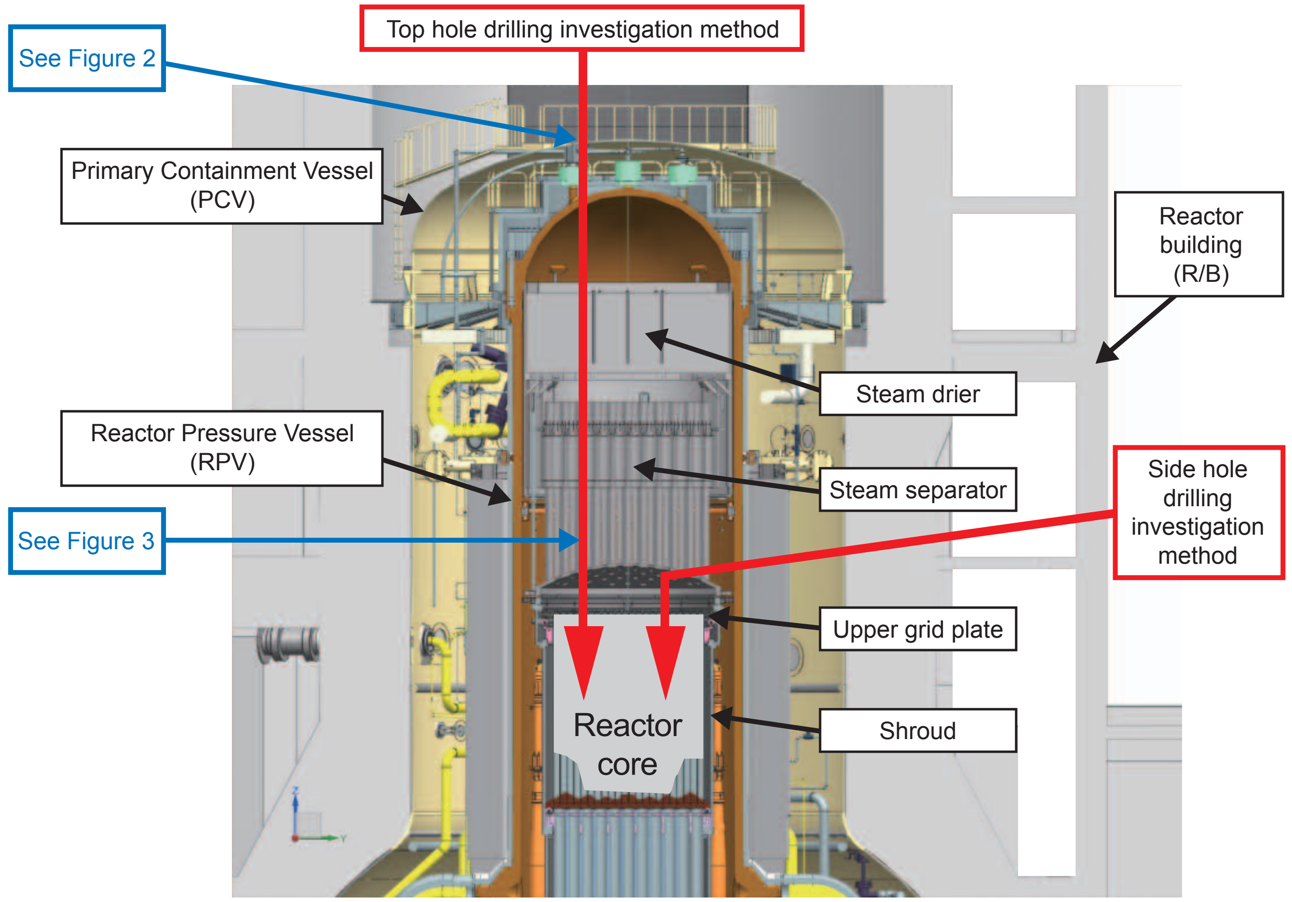


Figure 1: Concept of investigation method

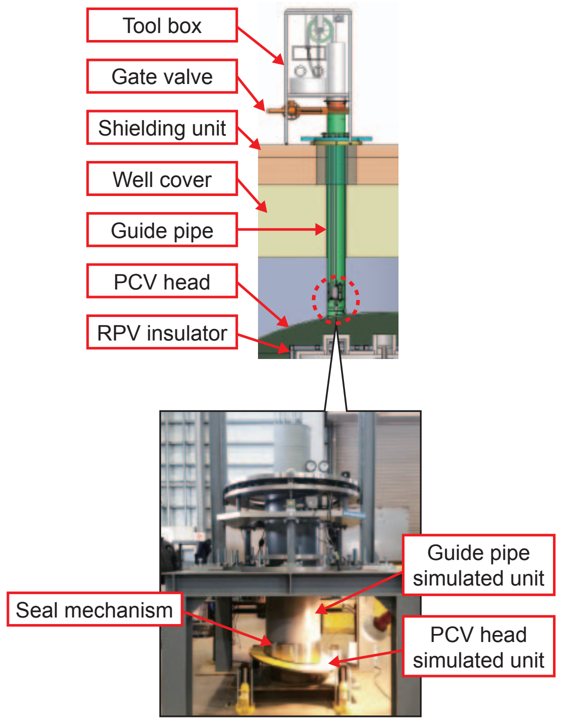


Figure 2: Drilling through RPV head / boundary function to prevent contamination from spreading

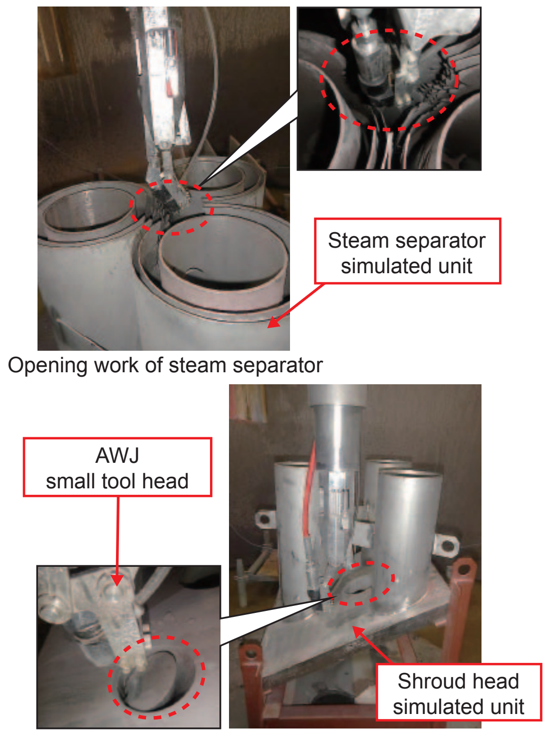


Figure 3: Verification test of hole drilling on reactor internals with air AWJ

Key Challenge 2 R&D for Fuel Debris Retrieval

Upgrading for Identifying Comprehensive Conditions inside the Reactor

Background

Identifying the conditions inside the RPV and the PCV is essential for planning the method of fuel debris retrieval and developing safety measures. However, Units 1-3 at the Fukushima Daiichi NPS have extremely high radiation dose, and it is difficult to investigate or observe them directly.

Aims

This project is intended to steadily proceed with decommissioning of the Fukushima Daiichi NPS to utilize results of the accident progression analysis and other R&D projects, analysis results of the measured data such as pressure and temperature at the time of accident, and information obtained on-site. We promote to estimate the conditions inside the RPV and the PCV based on comprehensive analysis and estimation on these information with the Institute of Applied Energy.

Main Achievements and Approaches

1 Comprehensive analysis and evaluation of condition inside the reactor

① Comprehensive analysis and evaluation based on actual data and results of other projects

In estimating the conditions inside the RPV and the PCV of each unit, we created an information summary map that comprehensively summarized various kinds of information in each section of the RPV, the PCV, and the reactor building. By comprehensively analyzing and evaluating the information, we created estimate diagrams of the fuel debris distribution (Figure 1), the FP (Fission Products) distribution, and the dose distribution.

② Establishment of database required for comprehensive analysis and evaluation

In order to effectively promote the activities shown in 1①, we established a database of totally collected and organized information such as actual measurement data and results of the on-site investigation. In addition, as a computational evaluation function contributing to comprehensive analysis and evaluation, we prepared the function to convert the FP mass calculated by the analysis code into the dose rate at the actual measurement location.

2 Comprehensive analysis and evaluation on behavior and characteristics of fuel debris and FP

① Uncertainty reduction by utilizing analysis method

Using the accident progression analysis code, we conducted sensitivity analysis and inverse analysis, etc., for the events presumed to have occurred inside the reactor based on the boundary conditions and analytical models, and acquired knowledge contributing to the comprehensive analysis and evaluation described in 1①. We also conducted a simulated fuel assemblies plasma heating test (Figure 2) and obtained knowledge that could reduce the uncertainty of phenomena such as core damage, core melt, and core move in the BWR system.

② Evaluation of FP chemical properties

In evaluating chemical properties of the FP, we focused on Cs with a large contribution of dose during decommissioning, and investigated the distribution of Cs and its chemical properties such as identifying the chemical species to be considered in addition to the standard chemical species such as CsI and CsOH, and the possibility of uneven distribution of poorly soluble Cs along with reaction with the structural materials.

We also started to analyze the samples taken at the site for identifying conditions inside the reactor (e.g. A curing sheet collected on the operating floor at Unit 2 (Figure 3)).

③ Utilization of domestic and international knowledge through international joint research

We operated the international joint research project (OECD/NEA BSAF Phase 2) as a host and compared achievements conducted by many organizations to identify the accident progress scenarios and the scope of uncertainty of analysis and evaluation, which were utilized for comprehensive analysis and evaluation shown in 1①.

We also conducted SAMPSON-MELCOR Crosswalk, and identified the common points and different points of both codes related to fuel temperature rise, melting progress, etc. As a result, we acquired knowledge to understand the accident progression analysis.

Future Developments

Following the activities in FY2016, we will promote comprehensive assessing of conditions inside the reactor, by conducting "improvement of the analysis code" as one of the approaches when needed, in addition to conducting "analysis and evaluation based on the information acquired on site," "analysis and evaluation of the data collected at the time of and after the accident and inverse problem analysis," and "analysis and evaluation based on the analysis code."

We will effectively utilize information to be acquired in scheduled investigations, such as investigation inside the PCV at Units 1 and 2, and continue to identify comprehensive conditions inside the reactor.

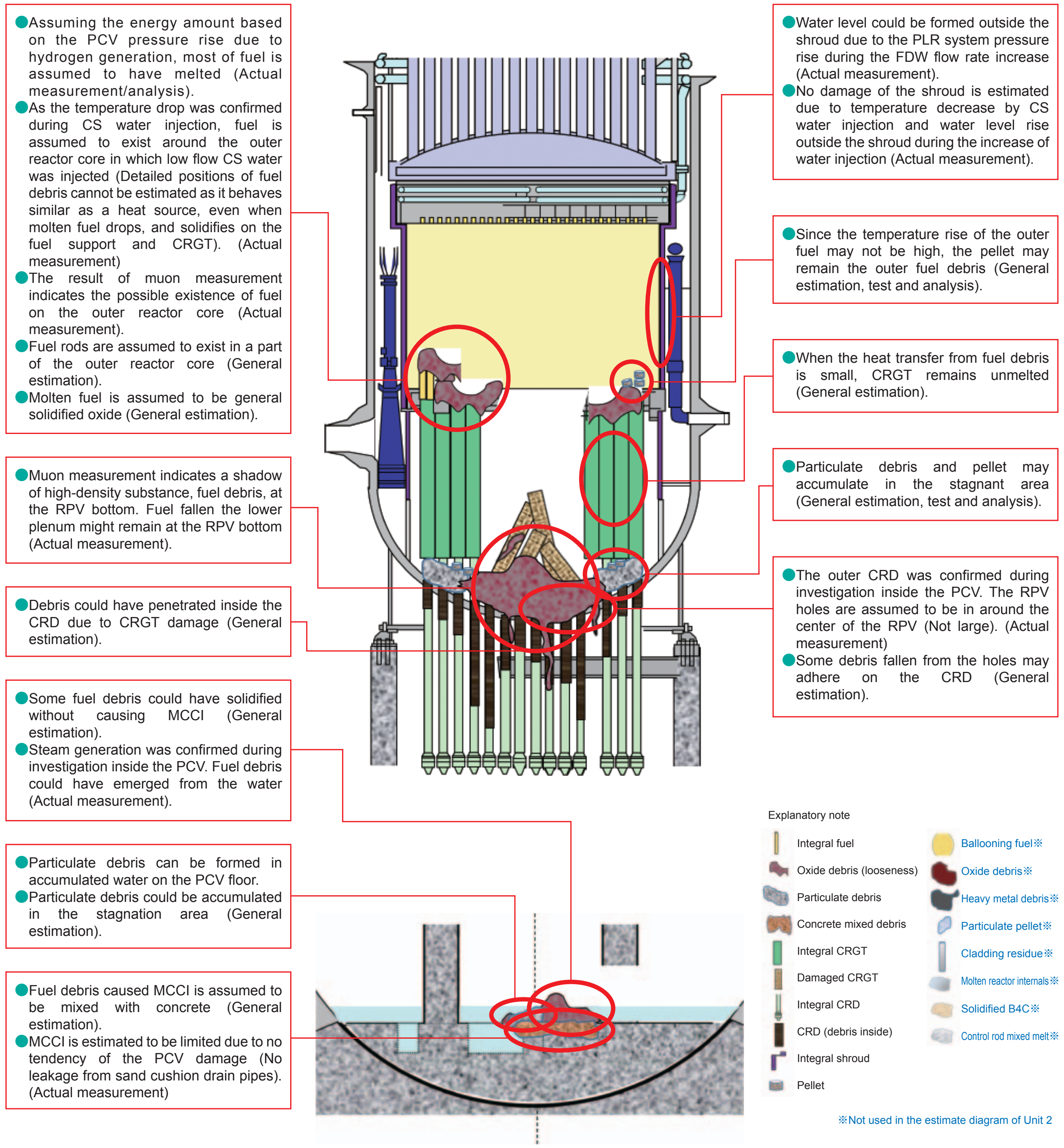


Figure 1: Estimate diagram of fuel debris distribution (example of Unit 2)

As of March 31, 2017

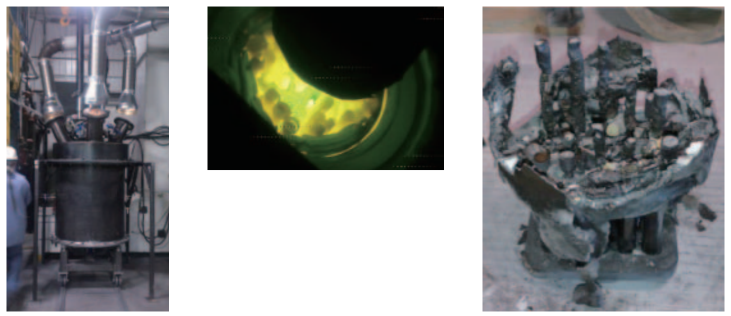


Figure 2: Simulated fuel assemblies plasma heating test (preliminary test device, heating condition, and tested unit)

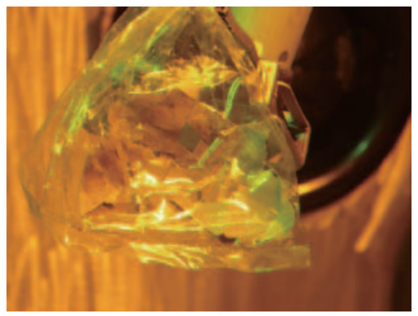


Figure 3: Curing sheet to be analyzed

Key Challenge 2 R&D for Fuel Debris Retrieval

Fuel Debris Characterization

Background

To ensure safe decommissioning process including retrieving, collecting, and storing fuel debris, it is necessary to investigate and estimate the fuel debris properties such as hardness and drying characteristics that can be used for development of retrieval equipment, and also to analyze the actual debris after obtaining the fuel debris sample to identify.

Aims

In order to provide information on the fuel debris required for the decommissioning works, we estimate properties of the fuel debris generated at the Fukushima Daiichi NPS with tests using simulated debris, and also develop elemental technologies for fuel debris analysis, so that we can analyze the actual fuel debris effectively for evaluating the fuel debris properties.

Main Achievements and Approaches

1 Estimate of fuel debris properties

Based on the results (molten core concrete interaction (MCCI) product information) obtained in FY2016, we updated the "fuel debris characteristics list" (summary of FY2015) that had estimated the fuel debris properties.

2 Characterization with the use of simulated debris

① Identifying fuel debris characteristics that affect collection and storage

We are developing technologies for drying characteristics, which are important for safety management, such as hydrogen generation management during collection and storage of fuel debris. In order to investigate the drying characteristics of concrete with MCCI products, we continued the drying behavior test using cement (main component of concrete), and evaluated impact that rehydratability of heat-deteriorated cement gave to the dry behavior in FY2016. Also, assuming powdered fuel debris that was expected to be difficult to dehydrate, we evaluated the drying behavior of the powdered fuel debris using particle size and others as parameters and acquired data on the powdering behavior of fuel debris to be used as the basic data for setting temperature during dehydration and heat treatment (Figure 1). In addition, we organized the existing knowledge on the dissolution behavior of actinide elements, compiled and provided them to the collection, transportation and storage projects as fuel debris properties data related to the storage canisters.

② Evaluation on unevenness of properties

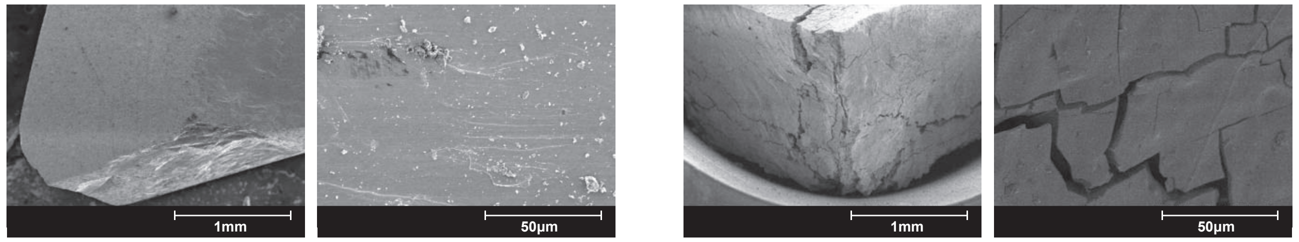
In cooperation with the Commissariat a l'Energie Atomique et aux Energies Alternatives (CEA), we used the MCCI test products prepared under past rapid cooling conditions stored in CEA to acquire the generated compounds, hardness, and other characteristic data, and expanded the property data in the rapid cooling conditions. We also conducted a large-scale MCCI test, considering the molten components and concrete components at Fukushima Daiichi NPS, and produced MCCI test products (Figure 2). As obtaining samples of different parts by disassembling the created test products, we obtained the knowledge that would be useful for retrieval, by confirming the erosion depth of the concrete and the state of the product, etc.

3 Development of elemental technologies for fuel debris analysis

In order to analyze fuel debris, we selected the elemental technologies for fuel debris analysis to be developed considering introduction of them into the Okuma Analytical Research Center, and are promoting technology development. For dissolution of poorly soluble fuel debris, we continued technology development using the alkali melting method, and evaluated dissolution conditions using oxides and various metal compounds which composed fuel debris (Figure 3). In addition, in developing the multi-element simultaneous analytical technology with inductively coupled plasma atomic emission spectrophotometry (ICP-AES), as conducting evaluation of spectral interference of uranium contained in large amount (Figure 4) and investigation of its solution method, we promoted development of technologies such as a quantitative evaluation method for porosity by X-ray CT, and a rationalization analysis method of multinuclear by ICP-MS, and identified the technical issues while verifying the feasibility of each technology. In addition, to study transport the fuel debris samples, we conducted safety analysis for transportation of fuel debris and obtained technical prospect of sample transport.

Future Developments

We will analyze and measure the hardness and others for each sample of the MCCI test product produced in FY2016, and incorporate the result in the "fuel debris characteristics list." We will also start to evaluate FP release behavior, which is necessary for examining drying equipment. Furthermore, as preparation for analyzing the fuel debris samples in the Ibaraki area, we continue development of analytical technology and investigation of transportation, and will prepare a guideline on the analysis method.

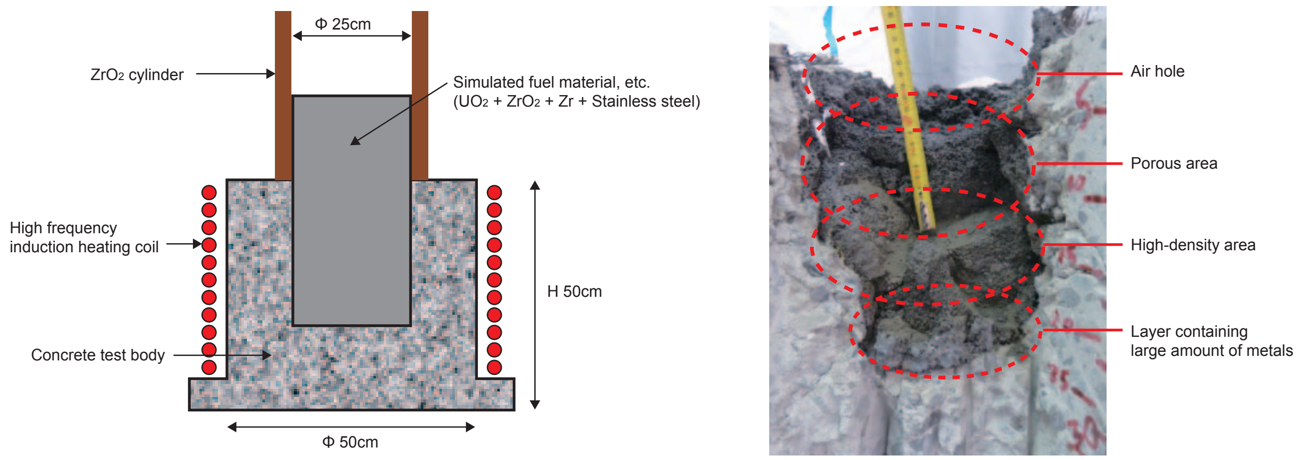


①Status after heat treatment at 300°C for 15 hours in a low oxygen atmosphere (Without crack generation)

②Status after heat treatment at 600℃ for 11 hours in a low oxygen atmosphere (Crack generation)

Figure 1: Appearance of simulated debris after heat treatment

With parameters such as temperature, time, and oxygen amount, we conducted a test using simulated debris. As a result, no crack was generated even in the heat treatment for 15 hours at 300°C with the oxygen amount under the environment where the decompression with a vacuum pump was assumed. Therefore, 300°C is one of rough targets in the drying temperature conditions.



①Cross sectional view of test equipment

②Status of MCCI test product after the test

Figure 2: Large MCCI test equipment and status of test products generated in the test in cooperation with CEA

A large MCCI test product was prepared by melting and reacting about 50 kg of simulated fuel material with the concrete using the test equipment ①. As shown in ②, the porous area, metal layer, and areas of the prepared test products were observed in the heterogeneous state. The test product was disassembled, which provided samples of individual parts. Detail analysis of each item including hardness is scheduled in FY2017.

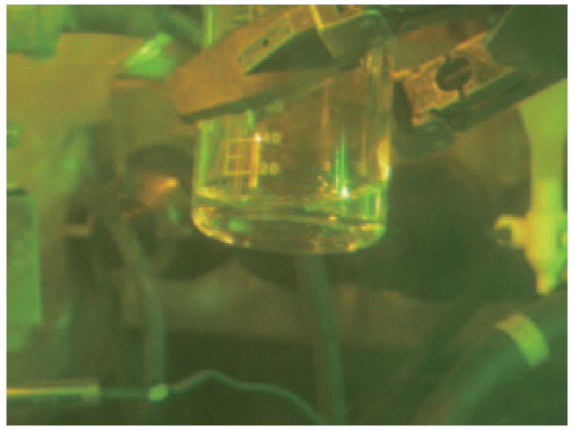


Figure 3: Alkali melting test in hot cell using simulated debris (status of solution)

Dissolution test of uranium simulated debris (sample prepared by melting UO_2 and Zircaloy) was conducted with the dissolution procedure using the alkali melting method. As a result of the test, it was confirmed that no residue was generated in the dissolution liquid and the total amount could be completely dissolved.

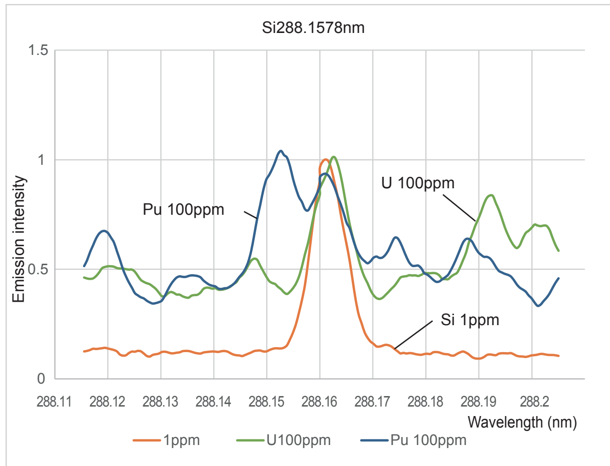


Figure 4: Interference of U and Pu in Si element analysis

Since fuel debris contains a large amount of uranium, spectral interference may be caused by uranium when other elements are analyzed. The figure shows the interference during analysis of silicon (Si) as an example of test results. The interference by uranium was evaluated for each element, and the solution method was examined.

Key Challenge 2 R&D for Fuel Debris Retrieval

Development of Repair Technology for Leakage Points inside the Primary Containment Vessel

Background

It is assumed that fuel debris at the Fukushima Daiichi NPS has not only fallen down within the Reactor Pressure Vessel (RPV) but has also made its way into the Primary Containment Vessel (PCV). In order to retrieve fuel debris, we plan to submerge them in the PCV. For that purpose, it is necessary to prevent water leakage from the PCV.

Aims

In order to substantiate fuel debris retrieval method by submersion or running water, this project intends to establish technology for repairing water leakage points in the PCV.

Main Achievements and Approaches

1 Development of PCV bottom repair technology (Figure 1)

① Water stoppage technology inside of the vent pipe facility

- Concerning inflatable sealing bags, sub-inflatable sealing bags, and waterproofing materials continuously developed until FY2015, we determined an action policy for the issues, planned a test schedule, and carried out each test.
- In addition to the conventional cement-based water stoppage materials, we tested rubber-based water stoppage materials in FY2016.

② Water stoppage technology injecting filling inside of the suppression chamber (S/C) (Figure 2)

- We completed the stiffening ring surmounting test, long distance pumping test, and elemental test under the assumed actual environment. In addition, we conducted a 1/1 scale concrete water stoppage test simulating the inside of S/C, and succeeded in stopping running water simulating the on-site environment.

③ Water stoppage technology injecting filling into vacuum break line (Particular application to Unit 1)

- We confirmed that the water stoppage plug developed in FY2015 had no leakage in the water-stoppage test at 0.45MPa, but some tasks such as improvement of installability of the water stoppage plug in the piping were extracted. In FY2016, we improved the developed water stoppage plug, and have been discussing how to improve on-site applicability including downsizing and simplification.

2 Boundary construction technology for connector pipes

- For water stoppage of the PCV connector pipes, we are developing water stoppage materials to be filled in the piping and remote devices. In FY2016, we have started designing each device according to the water stoppage method, and been conducting specific studies.

3 Development of water stoppage technology for torus room wall pipe penetrating part

- We selected a water stoppage material that could be used for spraying construction and tested it to confirm its water leak stoppage capability. Rubber materials were added and tested in FY2016.

4 Development of PCV top repair technology

Water stoppage technology for leakage from seal section (equipment hatch) (Figure 3)

- In order to prevent leakage from the flange area of the equipment hatch and secure airtightness, it is necessary to seal the flange joint, and we are examining a remote welding method. In FY2016, we removed rust of the welded part, manufactured a prototype of welding equipment and carried out elemental tests.

5 Study of environmental improvement concept for actual application of repair method

- We extracted high priority sections for repairing works, in which in the PCV bottom and top an environment improvement model was created as a representative. We also started to evaluate dose rates and to consider dose reduction.

Future Developments

It is necessary to examine applicability in actual use to reflect in the required performance of the devices, and also to study maintenance of the long-term water stoppage function.

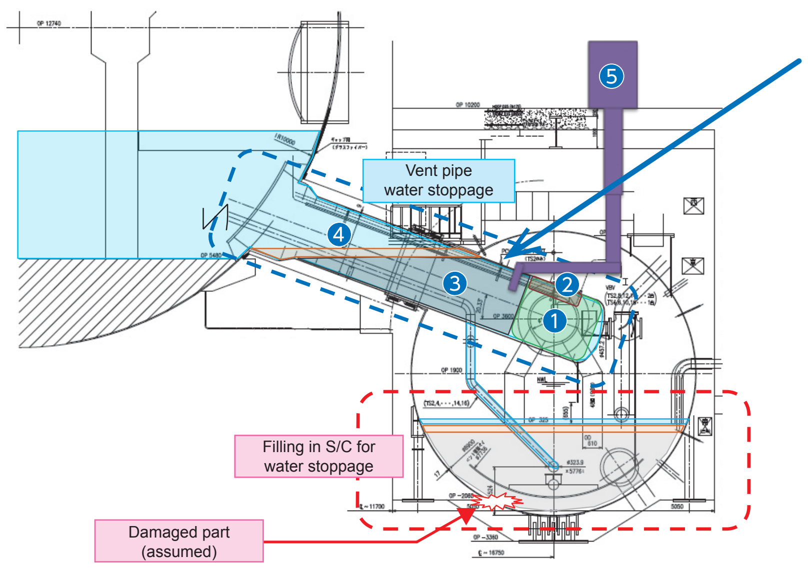


Figure 1: Overview of PCV bottom repair (example of Unit 2)

Equipment for water stoppage

Vent pipe — Water flow into S/C

Outer diameter: φ2057mm, plate thickness: 9mm
Number of pipes: 8, Material: Carbon sttel

Component	Application
① Inflatable sealing bag	Building a temporary dike for injecting water stoppage materials
② Sub-inflatable sealing bag	Sealing gaps after installation of the inflatable bag
③ Water stoppage material	Constructing a submersion range
④ Repairing material	Repairing cracks caused by long-term deterioration and external factors
⑤ Remote repairing device	Remotely operating for drilling in the S/C and vent pipes, removal of intervention, and installation of materials

Procedure of vent pipe stoppage: Starting construction using remote repairing device(⑤) from the 1st floor of the reactor building in the order, ①,②,③ and ④.

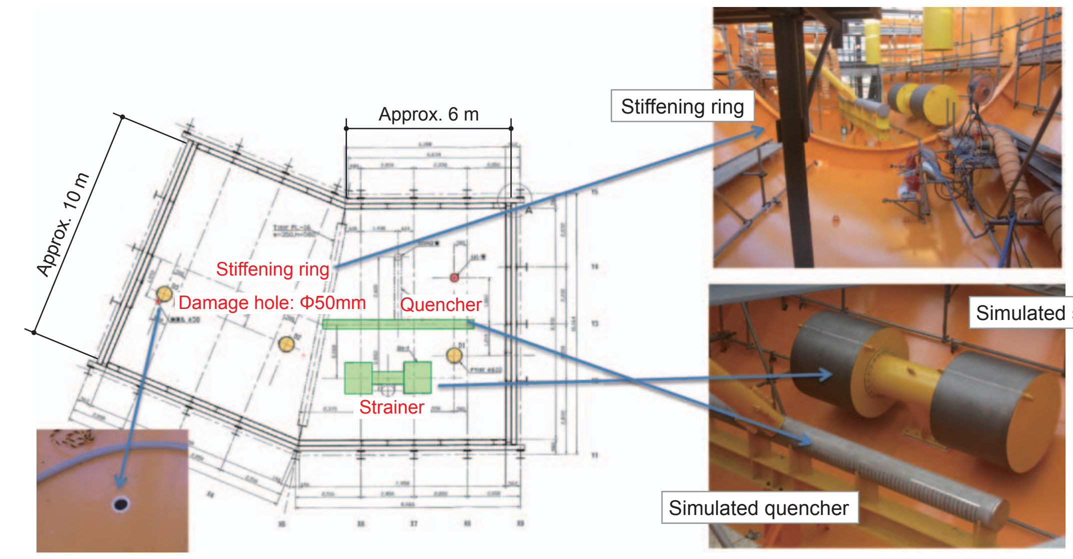
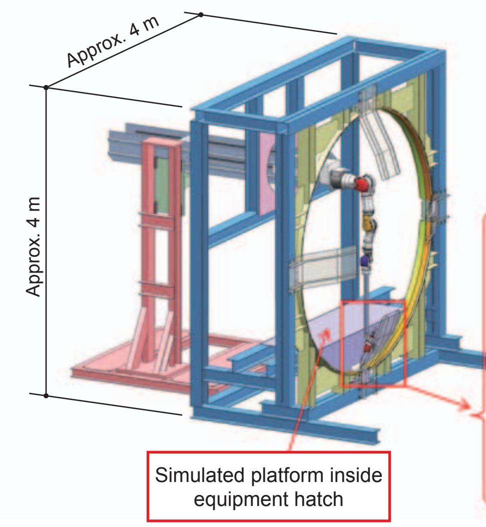


Figure 2: Water stoppage test by filling in S/C (1/1 scale)

Filling in S/C for water stoppage: Drilling of the upper part of the S/C and injecting concrete from that hole to stop water.
Injecting filling into the damage hole (φ 50), quencher, and strainer of the S/C to prevent water from flowing out of the S/C.



Test condition	Posture
Silicon G + Coating + Heating + Blast + Auto welding	Upward, side, downward
Silicon G + Coating + Heating + Laser + Auto welding	Upward, side, downward
Silicon G + Coating + Heating + Blast + Laser + Auto welding	Upward, side, downward
Silicon G + Coating + Heating + Blast or Laser + Water stoppage material	Upward, side, downward

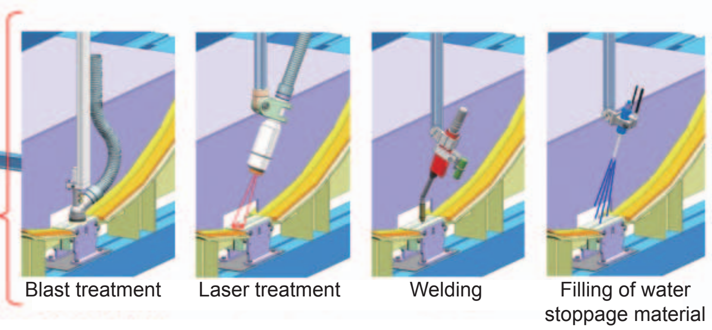


Figure 3: Equipment hatch test device (1/1 scale)

The PCV equipment hatch on the first floor of the reactor building is usually tightened by bolts, however they may be loosened after the accident. We are developing a construction method to weld the gap for water stoppage.

Main R&D Achievements Subsidized Projects
Main R&D Achievements In-house R&D Activities
Others Data Sheet

Key Challenge 2 R&D for Fuel Debris Retrieval

Full-scale Test for Repair Technology for Leakage Points inside the Primary Containment Vessel

Background

The Fukushima Daiichi NPS is in a harsh environment with high dose, narrow spaces, etc. There are many places where it is extremely difficult for people to approach and work for decommissioning. Therefore, in order to retrieve fuel debris, it is necessary to develop equipment/device for remote operations, which are implemented for decontamination and investigation or repair of each part, and to steadily promote decommissioning of the Fukushima Daiichi NPS.

Aims

To determine the policy for fuel debris retrieval, we have to conduct full-scale tests of the technology developed in the R&D of "Repair Technology for Leakage Points in the Primary Containment Vessel (PCV)" and to confirm its feasibility including its workability by remote operations. We also prepare a manual considering actual construction, and confirm its feasibility of the procedures and repair for water leak blockage capability. In this project, the test will be conducted in the Naraha Remote Technology Development Center of Japan Atomic Energy Agency.

Main Achievements and Approaches

1 Full-scale test of repair technology for lower part of PCV

As one of the criteria for verifying the applicability, we examined an evaluation method based on risk assessment. Also, we are conducting a full-scale test for the following items:

① Suppression chamber (S/C) support columns (Figure 1)

- We conducted a workability verification test as a full-scale test of S/C support columns reinforcement, and confirmed that there was no problem in works under high dose and remotely controlled operation, and applicability of equipment prepared for the PCV repair development. We will also discuss how to improve the workability based on the expertise obtained through the tests (Figure 2, Photo 1 and Photo 2).
- In order to determine necessity of reinforcement of S/C support columns according to the S/C support columns elasticity analysis result implemented in the seismic resistance / impact assessment project as a hold point for implementation of the placing test, we revised the implementation schedule of FY2017.

② Vent pipe water stoppage

- We began a workability verification test for vent pipe water stoppage such as setting of the remote operation, and drilling in the S/C and the vent pipes by the remote operation (scheduled to be continued until FY2017).

③ Filling in S/C for water stoppage (downcomer water stoppage)

- As sharing the information of the various technical development projects, we are revising a test plan.

④ Preparation for test (Photo 3 and Photo 4)

- As preparation for the tests, we implemented interference objectives to be used for test device and verified integrity by submerging the test device. Also we maintain operation and provide operation training for the equipment that supplies hot water and drains water after testing to simulate the test environment, in order to establish a system with which we can operate the tests thoroughly.

2 Preparation of VR data for preliminary simulation test

We are establishing an environment where a remote manipulator used for vent pipe water stoppage is simulated on a VR (virtual reality) system, allowing operation trainings. In FY2016, based on the interview of the designers and the operators of the remote devices, we decided required items for accuracy improvement simulated actual operations on the console, and implemented additional operation functions to the remote operation devices.

Future Developments

In order to judge the applicability of the actual equipment, we think that it is necessary to extract risks and tasks on the actual equipment construction not only from the R&D personnel but also the workers in site, and utilize them in improvement of the workability and construction process.

Also, we would like to establish an environment where information can be fed back to the remote operation devices in combination with the remote operation devices and VR technology effectively.

In FY2017, we will continuously conduct a full-scale test of each method and verify the test results.

Full-scale Test for Repair Technology for Leakage Points inside the Primary Containment Vessel

Outline of technology for reinforcement of suppression chamber support columns

Since weight increase due to filling a water stoppage material inside S/C is expected, technology aiming at seismic reinforcement for the support columns of the S/C is developed.

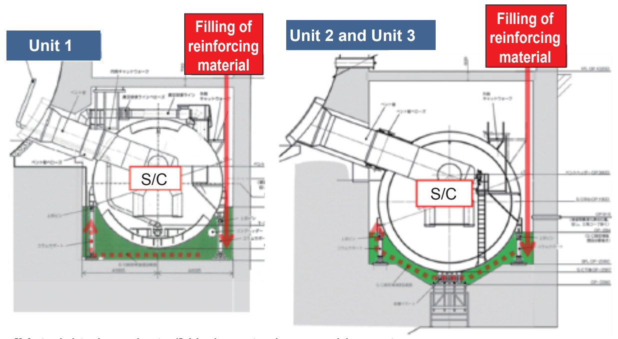


Figure 1: Overview of technology for reinforcement of S/C support columns

Workability verification test for reinforcement of S/C support columns

Major implementation of workability verification test

- Verification of workability of installation and collection of the placing hose with the device on the work floor
- Verification of construction procedures through water flow and remote monitoring performance

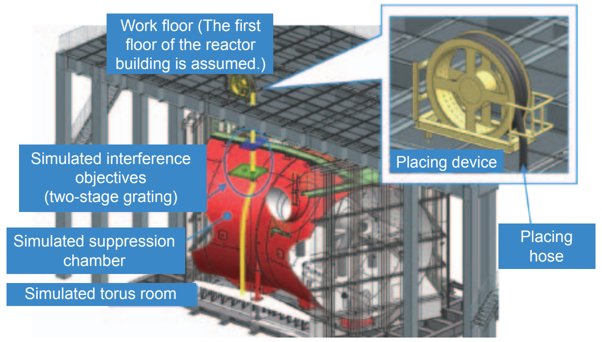


Figure 2: Overview of workability verification test

Photo 1: Implementation of reinforcing workability verification test for S/C support columns

Photo 2: Supply equipment of reinforcing material for S/C support columns (outdoor)

Photo 3: Appearance of test device

Photo 4: Inside of test device (inside of S/C)

Key Challenge 2 R&D for Fuel Debris Retrieval

Development of Corrosion Control Technology for RPV / PCV

Background

Due to severe events caused by the Great East Japan Earthquake, Reactor Pressure Vessel (RPV) and Primary Containment Vessel (PCV) at the Fukushima Daiichi NPS were exposed to high temperature and sea water, and fall of fuel debris and other issues are expected. Corrosion control technology is needed to prevent progress of corrosion of the structural materials and to maintain integrity for a long period until fuel debris retrieval from the reactor core.

Aims

Corrosion of the RPV / PCV is supposed to be controlled comparatively more than the normal environment due to deaeration treatment of water injection and nitrogen injection into the PCV, which have been carried out right after the accident. Since the PCV may be exposed to the atmosphere at the time of fuel debris retrieval, we are promoting development of corrosion control technology (rust inhibitor) instead of nitrogen injection, and the evaluation of its applicability.

Main Achievements and Approaches

1 Evaluation of corrosion control effect and impact (Figure 1, 2, 3)

① Evaluation of local corrosion resistance of rust inhibitor by electrochemical measurement

- We conducted electrochemical measurements for evaluating local corrosion resistance of carbon steel, the main structural material, (corrosion crevice re-passivation potential measurement, self-potential measurement, constant potential crevice corrosion test) under gamma irradiation and non-irradiation environment, and selected rust inhibitors that would not cause localized corrosion.

② Evaluation of fixation effect of phosphate rust inhibitor at high-temperature part

- We conducted a batch test and a water flow test to evaluate the fixation effect of a phosphate rust inhibitor at the high-temperature part, and checked the temperature and other conditions of fixation. We also conducted a combined impact assessment test with the use of the phosphate corrosion inhibitor and a sterilizing agent together to evaluate the impact of corrosion on carbon steel, and confirmed that there was no adverse effect.

③ Impact assessment on water treatment facility

- We conducted a test to verify impact of the rust inhibitor on the water treatment facility, and confirmed that it would be necessary to dilute collected water or remove the rust inhibitor from water in advance to reduce the impact on the water treatment facility, and to make the concentration of the rust inhibitor in the water to be treated lower than the concentration at the time of injection into the PCV.

2 Conceptual design of corrosion control system (Figure 4)

- We implemented conceptual design of a corrosion control system to apply the rust inhibitor (corrosion control measure) to actual equipment. In addition, we formulated management guidelines for applying the rust inhibitor.

Future Developments

- Development of corrosion control measures in the atmosphere based on the fuel debris retrieval method

In case that exposed parts to the air are required corrosion control by the fuel debris retrieval method, we will resume the study.

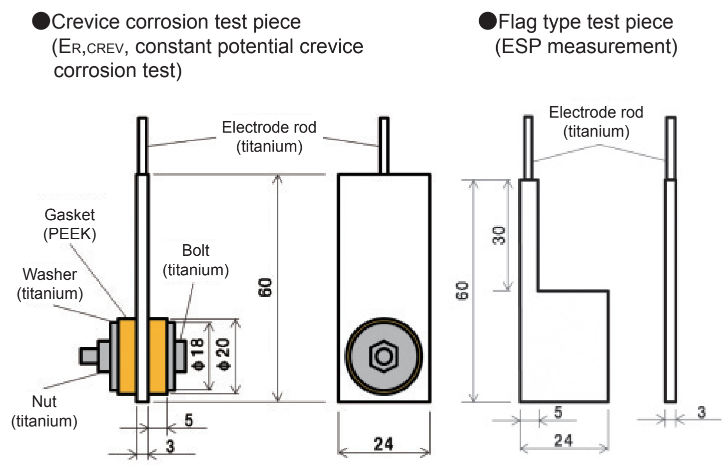


Figure 1: Electrode composition of corrosion test piece

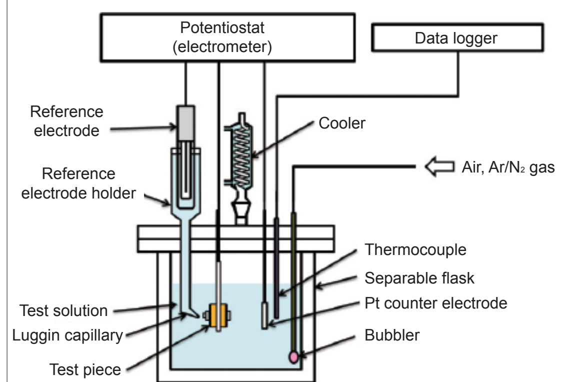


Figure 2: Configuration example of electrochemical measurement apparatus (irradiation)

●By using an electrochemical measuring device of three electrodes (specimen pole (test piece), counter electrode, reference electrode), a measurement was conducted.
●During self-potential (ESP) measurement, we used two electrodes such as a specimen electrode and a reference electrode without setting a Pt counter electrode.

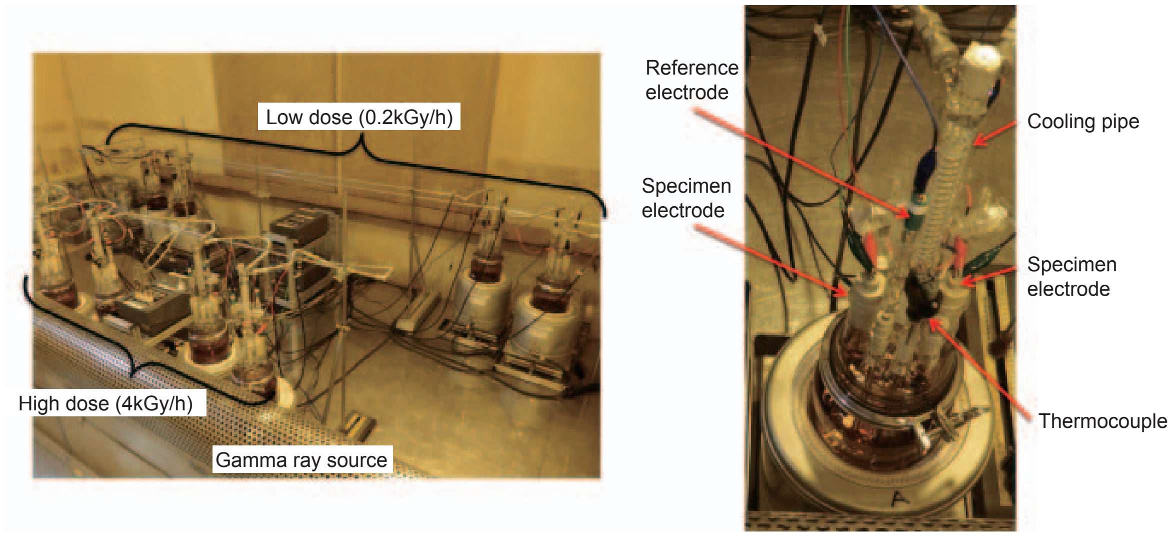


Figure 3: Test condition under irradiation

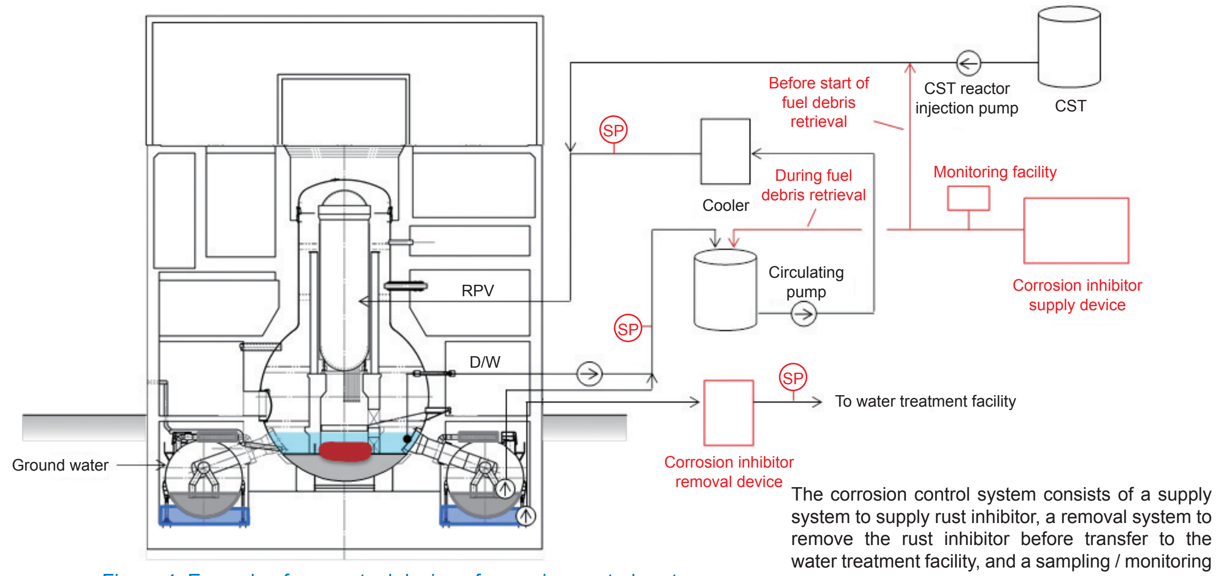


Figure 4: Example of conceptual design of corrosion control system

The corrosion control system consists of a supply system to supply rust inhibitor, a removal system to remove the rust inhibitor before transfer to the water treatment facility, and a sampling / monitoring facility to observe the water quality.

Key Challenge 2 R&D for Fuel Debris Retrieval

Development of Seismic-resistance and Impact Assessment Method for RPV / PCV

Background

Due to severe events caused by the Great East Japan Earthquake, Reactor Pressure Vessel (RPV) and Primary Containment Vessel (PCV) at the Fukushima Daiichi NPS were exposed to high temperature and sea water. It is estimated that fuel debris fall and other issues occurred. Some measures are necessary to maintain integrity of the PCV / RPV structures for a long period until fuel debris retrieval from the reactor core.

Aims

We clarify damage of major equipment in the RPV / PCV and the spreading impact at the time of occurrence of a large-scale earthquake, which affects the water level in the PCV assumed during fuel debris retrieval and the installation status of the critical equipment in the building, etc. We devise a countermeasure that can prevent or control such impact and confirm its effectiveness by assessment of seismic resistance.

Main Achievements and Approaches

1 Formulation of safety scenario for large earthquake (Figure 1)

We selected the suppression chamber (S/C) support columns and the RPV pedestal as damaged parts which may have spreading impact, based on the integrity evaluation and the analysis results of the PCV inside temperature during the accident and fuel debris erosion situation, and evaluated the damage and its spreading impact on the maximum assumption.

As a primary idea of the safety scenario against the risk of rise in the water level of the torus room due to water leakage of the PCV, we indicated measures to reduce the load by the PCV inner water level restriction without reinforcing the S/C support columns and the optimum water stoppage conditions of the S/C and the vent pipes.

Regarding seismic resistance of the RPV pedestal, we will conduct a detailed assessment on the seismic resistance under damage prediction conditions based on the latest knowledge.

2 Development of seismic resistance / impact assessment method for formulating safety scenario

We created a seismic resistance assessment condition based on the vent pipes and the S/C water stoppage conditions which are measures in the safety scenario (Figure 2), formulated a time history seismic response analysis model (elastic analysis) and a limit analytical model (elastic-plastic analysis) for the S/C support columns based on a total model where the both are coupled. We are analyzing with these models now (Figure 3).

We studied the pedestal temperature history and distribution, and the influence of the range eroded by fuel debris, based on the latest PCV accident progression analysis data. Considering these situations, FEM analysis model and analysis case draft were formulated to start analysis.

We examined the high temperature corrosion test condition for estimating the strength deterioration of the reinforcing bar in the concrete by the temperature history of the latest analysis, and fabricated a test body. In addition, we constructed a large-scale equipment coupled analysis model, which RPV, and a stabilizer are integrated.

3 Safety scenario upgrading

In order to upgrade the evaluation method mentioned in the section 2 above, we verify the confirmation measures such as analysis and tests based on seismic load and constraint conditions that can be supposed actually. As a method of evaluating and verifying resistance of the S/C support columns, we began studying a support column material test.

Future Developments

In order to formulate and upgrade the safety scenario, we will study the expanded material test data and the analysis method (model, elastoplastic analysis, etc.), and establish the seismic resistance / impact assessment method.

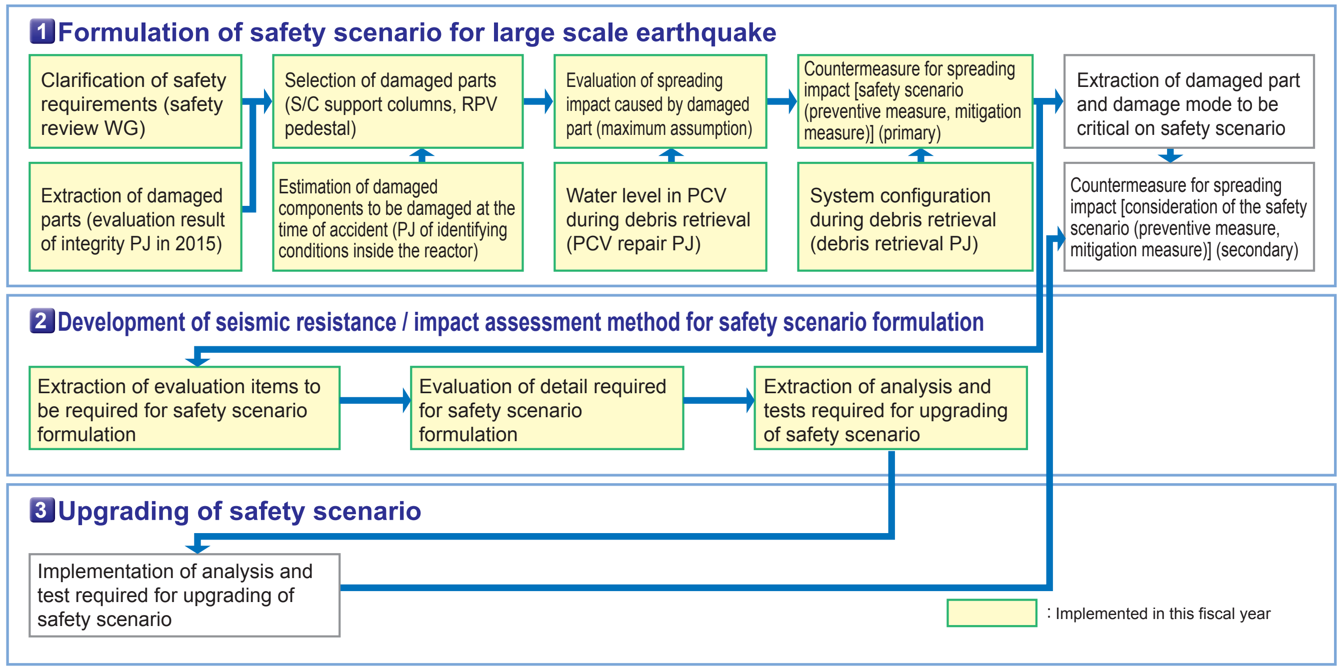


Figure 1: Total flow of study on seismic impact assessment PJ

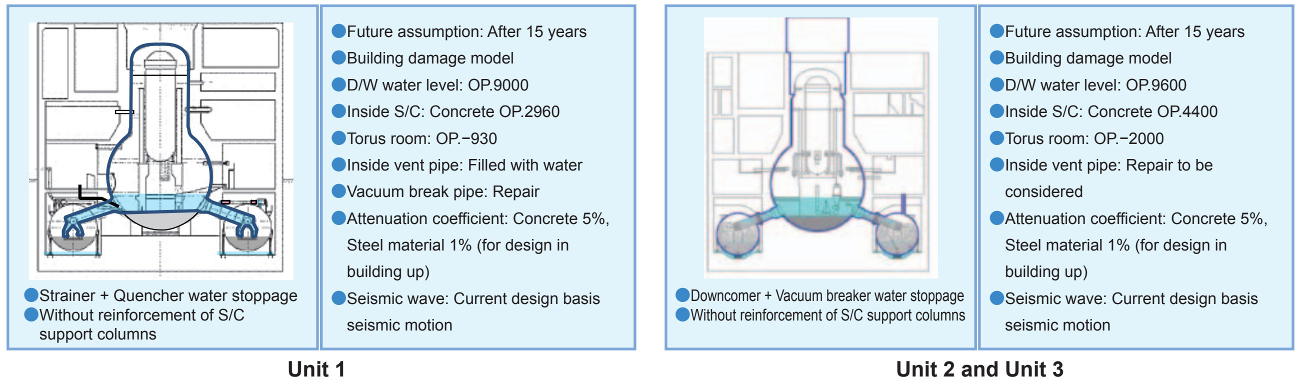


Figure 2: Example of seismic resistance assessment condition for S/C support columns

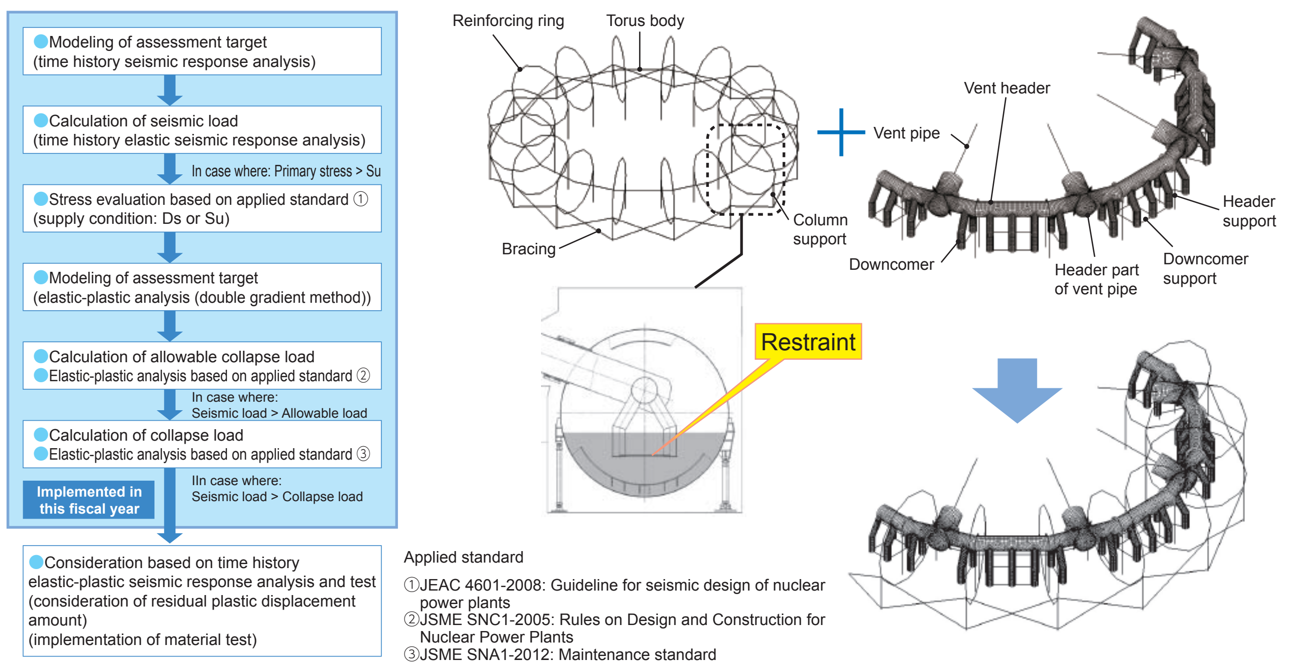


Figure 3: Development flow of seismic resistance impact assessment method, and total analysis model of S/C support columns

Key Challenge 2 R&D for Fuel Debris Retrieval

Development of Technology for Criticality Control in Fuel Debris Retrieval

Background

It is estimated that fuel debris is not currently in a critical state. Since the shape of fuel debris and water level may change during retrieval of fuel debris in the future, we secure the prevention of criticality and also promote development of criticality monitoring technologies and criticality prevention technologies so that even if criticality should occur, it would be safely terminated.

Aims

In the FY2016 projects, we examine and evaluate the criticality scenario based on the latest knowledge of condition inside the reactor for multiple retrieval methods to review criticality risks for each unit. In order to confirm feasibility of the technologies, the criticality monitoring technologies such as critical approach detection technology, and the criticality prevention technologies using soluble and non-soluble neutron absorbing materials was verified.

Main Achievements and Approaches

1 Establishment of criticality evaluation methods

We reviewed the critical scenario of Units 1 to 3 based on the latest information on remaining fuel and fuel debris distribution acquired in the project identifying conditions inside the reactor and the muon measurement of Unit 2 conducted in 2015, and re-evaluated critical risks (Table 1).

We developed a criticality behavior evaluation model (Figure 1) with an additional exposure evaluation model, evaluated sensitivity of equipment parameters related to criticality detection and critical shutdown (Figure 2), and examined equipment requirement specifications such as an FP gas leakage rate, which would be effective for exposure impact mitigation.

Based on the idea of defense in depth, we summarized an idea of the criticality control method where each element technology was applied to the protection system (PS) and the mitigation system (MS) for each fuel debris retrieval method (Table 2). Also, we summarized an idea of the method to consider uncertainty of the critical calculation.

2 Development of critical approach monitoring method

We are developing a method to estimate sub-criticality and to monitor critical approach (Figure 4) with the reactor noise method, the neutron source multiplication method, the period method, and the virtual neutron source multiplication method based on neutron measurements. We prepared and tested to confirm feasibility of these methods on site.

To verify operability of the B-10 neutron detector under high radiation environment, we installed the detector near the spent fuel assemblies (within the facilities of Nippon Nuclear Fuel Development Co., Ltd.) and prepared for verification tests for radiation dose and neutron detectability (Figure 3). We confirmed that neutrons could be detected under proper shielding. In addition, we prepared a verification test the principle of each method using a critical assembly (Kyoto University Critical Assembly: KUCA) that could actually simulate reactor core systems with different sub-criticality levels (Figure 5).

3 Development of re-criticality detection technology

We are promoting development of advanced gas sampling system technology which can measure Kr-87 and Kr-88 in addition to conventional Xe-135 in the small amount of FP gas existing in Primary Containment Vessel as a method to detect re-criticality at an early stage. This technology is based on gamma ray measurement with a germanium (Ge) detector.

For this reason, we installed an additional analyzer in the existing gas management system at Unit 1 equipped with a Ge detector, expanded the measurement range of gamma ray energy up to 3 MeV, and started collecting gamma ray data (Figure 6). We have confirmed the peaks of Kr-87 and Kr-88 in the gamma-ray energy spectrum. In the future, we plan to analyze the data and use a result for studying delay in critical detection time and for incorporating into the re-criticality detection model.

4 Development of criticality prevention technology

We are promoting development of a non-soluble neutron absorbing material as applicable technology to prevent criticality during fuel retrieval. Targeting the applicable materials extracted so far in the fundamental physical properties and radiation resistance performance test (B4C metal sintered material, B-Gd glass material, cement/Gd granulated powder material, etc. (Table 3)), we prepared for nuclear characteristics verification test with KUCA. Also, we are studying applied construction methods and equipment when implementing basic workability test to select applicable materials.

Furthermore, regarding the soluble neutron absorbing material sodium pentaborate (containing boron B) to be directly injected into cooling water, the previous research results indicate that when boron alone is used to ensure sub-criticality, the required B concentration is about 6,000 ppm. Since this value exceeds the actual concentration in the power generation reactors, we decided to prepare the nuclear characteristics verification test with KUCA. We also confirmed feasibility of boron concentration maintenance equipment of sodium pentaborate water.

Future Developments

Based on the latest knowledge about the condition inside the reactor, we will continuously refine the critical scenario and the critical risks, and examine a criticality control method for each work process of the multiple construction methods to decide a fuel debris retrieval policy.

We will also carry out a feasibility verification test and a practical applicability test in order to indicate how to operate the actual equipment, concerning element technologies such as the criticality monitoring technologies and the criticality prevention technologies.

Sections	Criticality risk	Unit 1	Unit 2	Unit 3
Reactor core	• Submersion of remaining fuel	Extremely low (almost no remaining fuel)	Medium (Fuel may remain in the periphery)	Low (Fuel may remain in the periphery)
Lower part of RPV	• Submersion of debris • Changes during retrieval	During submersion: Low During retrieval: Extremely low (RV: Small)	During submersion: Medium During retrieval: Low (RV: Large and exposed to the air)	During submersion: Medium During retrieval: Low (RV: Large and exposed to the air)
CRD housing	• Submersion of accumulated debris	Low - Extremely low (Risk: low)	Low - Extremely low (Risk: low)	Low-Extremely low (Risk: low)
PCV bottom	• Submersion of exposed debris • Change of state during retrieval(incl. curing)	Submersion: Low Retrieval: Low (RV: Large, exposure: small)	Submersion: Medium Retrieval: Small (RV: Large, exposure: large)	Submersion: Low Retrieval: Low (RV: Large, exposure: small)

Residual volume=RV

Table 1: Critical scenario and critical risk of each unit

We evaluated re-critical risks based on the latest prediction about the amount of fuel and fuel debris. The critical approach detection and criticality prevention technologies are applied to the debris retrieval works.

	Level 1 Prevention of abnormal operation (PS system)		Level 2 Control of abnormal operation and termination of failures (PS system)	
Target	Monitoring of criticality approach to prevent criticality		Detection and control of criticality	
Criticality control	Parameter monitoring	Prevention of abnormal operation	Detection of failures	Mitigation of impact
Specific measures (Primary issues)	• Criticality approach monitoring with the system • Monitoring water level / boric acid solution concentration.	• Restriction of debris retrieval amount • Application of Boric acid solution/ non-soluble absorbent	• Criticality detection by neutron assemblies or FP gas concentration	• Termination of criticality by injection of boric acid solution or non-soluble absorbent

Table 2: Idea of criticality control method based on defense in depth

To prevent criticality, and to detect to terminate it promptly even if it occurs, we summarize ideas of safety measures that have multiple criticality prevention systems and impact mitigation systems in the table above.

Category	Material	Evaluation: In air	Evaluation: In water	Future policy
Solid	B_4C / Metal sintered material	◎	○	Applicable to nuclear characterization
	Glass material incl. B-Gd	○	○	Applicable to nuclear characterization (advanced)
	Hollow boron	—	—	Withdrawal
	Gd_2O_3 particle	○	○	Applicable to nuclear characterization (advanced)
Liquid to Solid (solidified material)	Cement / Gd_2O_3 granulated powder material	△	△	Withdrawal
	Water glass / Gd_2O_3 granulated powder material	○	△	Applicable to nuclear characterization (inorganic)
	Underwater curing resin / Gd_2O_3 powder	○	○	Applicable to nuclear characterization (organic)
	Underwater curing resin / B_4C powder material	○	○	Applicable water stoppage material
Liquid (viscous material)	Slurry / Gd_2O_3 granulated powder material	△	○	Second option of nuclear characterization organic system of (Liquid to Solid) since it is solidified at 720 kGy or higher.
	B_4C gel material	△	—	Leaching characteristics of irradiated material must be improved.

Table 3: Evaluated applicable non-soluble neutron absorbing material

We decided to conduct a nuclear characterization test on materials with stable characteristics in water and air, among applicable non-soluble materials (solid, solidified material, viscous material).

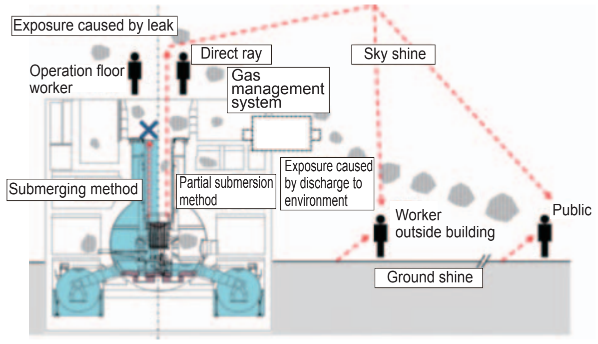


Figure 1: Image of exposure evaluation model

To examine the specifications of the safety equipment, we develop a model that can detect, terminate criticality and allows evaluation of the exposure amount of workers and the public. It will be incorporated in the design so that exposure will not occur even in case of criticality.

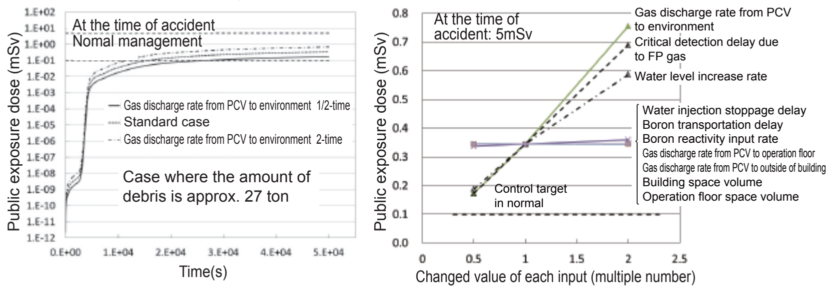


Figure 2: Evaluation example of exposure amount at the time of assumed criticality occurrence at PCV submerging

We examined the equipment specifications with sensitivity analysis by changing major parameters (critical detection delay time, gas leak rate, boric acid injection delay time, etc.) that would affect exposure evaluation.

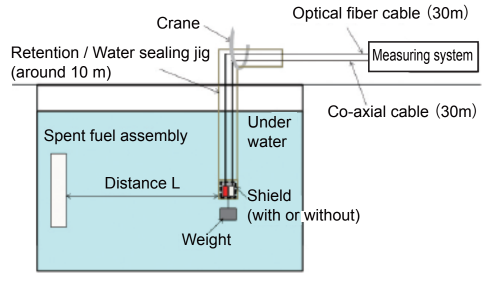


Figure 3: Detector operation verification test system under high radiation

The radiation dose is high in the vicinity of fuel debris. Therefore, we conducted a test in a pool of facilities with spent fuel to confirm whether the B-10 neutron detector could detect neutrons even under such circumstances.

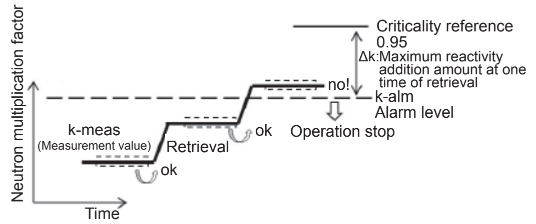


Figure 4: Image of monitoring critical approach

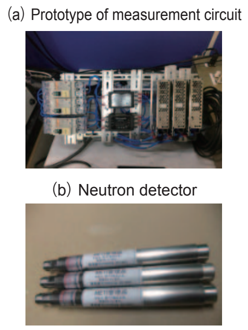


Figure 5: Preparation for feasibility verification test

We conducted several reactor cores with different sub-criticality levels with KUCA. We prepared for a test with which we would measure neutrons with the B-10 neutron detector, and examine how accurately the sub-criticality level could be predicted based on the obtained data.

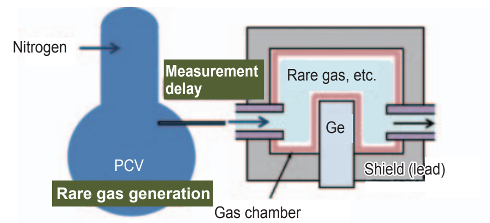


Figure 6: Image of gamma ray measurement test for FP gas inside PCV

Gas in the containment at Unit 1 into the gas chamber was introduced and measure gamma ray with a Ge detector to check whether Kr-87 and Kr-88 can be measured. Measurement is currently underway.

Key Challenge 2 R&D for Fuel Debris Retrieval

Upgrading Approach and System for Retrieval of Fuel Debris and Internal Structures

Background

It is assumed that fuel debris in the Reactor Pressure Vessel (RPV) and the Primary Containment Vessel (PCV) at the Fukushima Daiichi NPS has not currently reached criticality. The reactor buildings, the RPV and the PCV were damaged at the accident, and the reactors have been in unstable conditions. We aim to stabilize the reactors without diffusing radioactive materials, as retrieving fuel debris to maintain sub-criticality.

Aims

In order to retrieve fuel debris, we choose three retrieval methods; Submersion-Top entry method, Partial submersion-Top entry method, and Partial submersion-Side entry method. Toward finalizing the policy for retrieval of fuel debris and reactor internals, we clarify plant information, study methods, systems, and equipment for retrieval of fuel debris and reactor internals. To confirm feasibility of the retrieval methods and the entire system, we also formulate development plans of the retrieval system and equipment.

Main Achievements and Approaches

1 Clarifying plant information to determine policy for retrieval of fuel debris and internal structures

We organized plant data and development results of related projects, and clarified necessary information to determine the policy for retrieval of fuel debris.

2 Study of methods, systems, and equipment for retrieval of fuel debris and internal structures

① Study of feasibility methods

Concerning the three representative methods; Submersion-Top entry method, Partial submersion-Top entry method, and Partial submersion-Side entry method, we prepared a process flow showing a series of processes, and details of each work unit as a step diagram for fuel debris retrieval, and extracted technical issues (Figure 1).

In addition, we shared the information on latest situation of the other related projects, extracted common issues, and reviewed prerequisites.

② Conceptual study of system

In order to safely ensure fuel debris retrieval, we examined safety requirements during retrieving fuel debris, adopted the concept of defense in depth to organize functional requirements for the system, and considered the system configuration (Figure 2).

Furthermore, we assumed conditions such as a dispersion ratio at the time of fuel debris processing and a leakage rate from the boundary, carried out the exposure assessment, and examined required specifications of the main system.

③ Design of retrieval system

We studied a design of equipment for retrieval of fuel debris and reactor internals.

3 Formulation of development plans of systems and devices for retrieval of fuel debris and internal structures

We implemented technical investigations concerning remote operation, cutting, shielding, prevention of dust scattering, etc. We also organized the development plans formulated in the study of the methods, systems, and equipment for retrieval of fuel debris and reactor internals.

4 Formulation of development plans of system and device for sampling of fuel debris

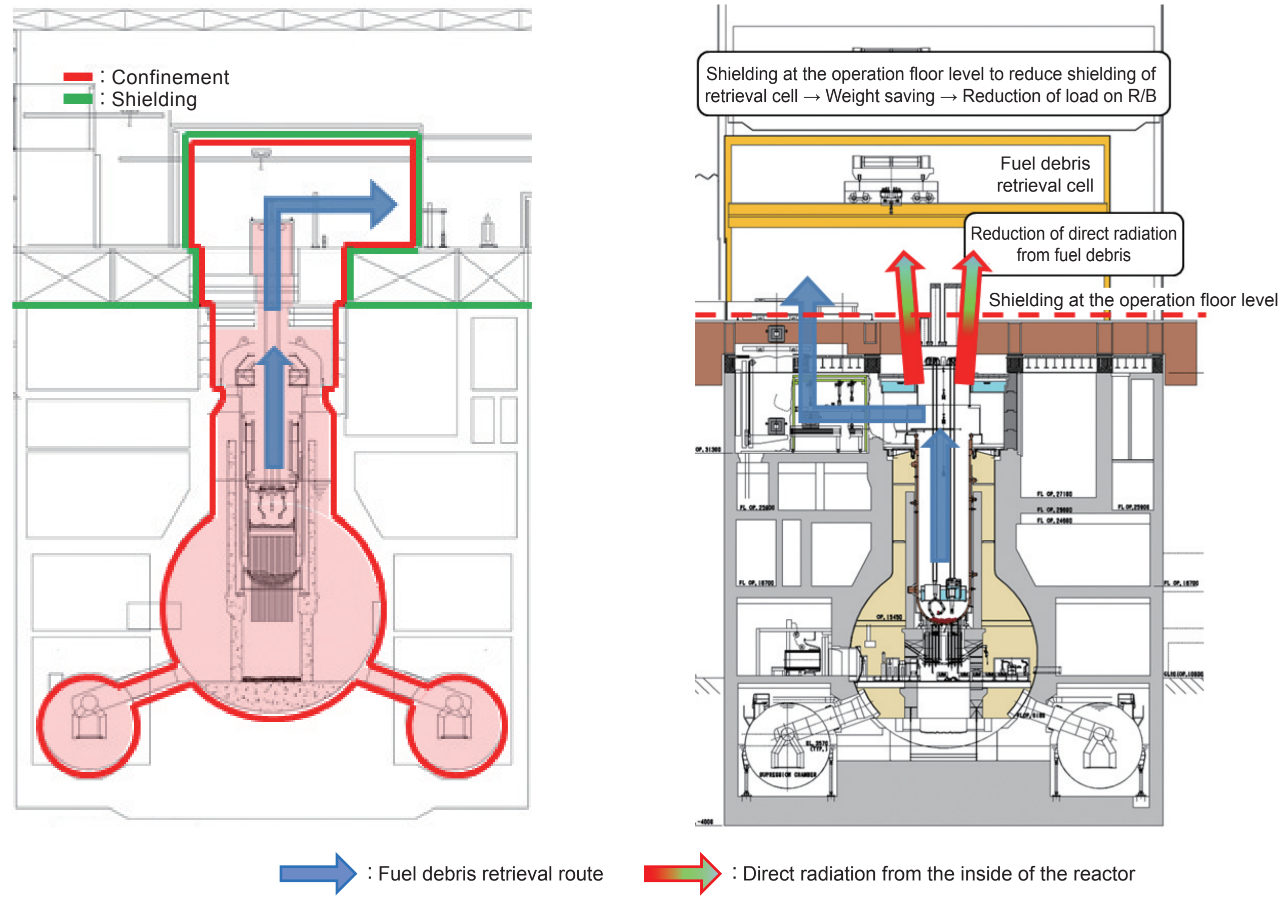
We additionally examined a safety system and an intermediate cell, considering reduction of working dose and various risks and benefits.

Based on the elemental test of the equipment to collect fuel debris after cutting, we examined the cutting conditions and actual structures (Figure 3).

We examined improvement of accuracy of the measuring devices, countermeasures against debris with various shapes such as pebbles and sand, and also a concept of sampling equipment for fuel debris inside the RPV.

Future Developments

We will standardize the multiple ideas of fuel debris retrieval method, which are currently under consideration. We will confirm feasibility of common technology required for all methods and individual technologies identified in consideration of standardization, and sharing with elemental tests, etc. For the safety system, we will confirm feasibility of the safety requirements and system function requirements examined in this fiscal year. Also for fuel debris sampling, we will prepare a scenario and verify feasibility of the technology.



[Route A]

■Shielding by the fuel debris retrieval cell on the operation floor (top floor of the reactor building)
▶Simple flow line
▶Comparably easy correspondence in emergency
▶To reduce load on the reactor building, the cell and equipment are downsized.

[Route B]

■Shielding at the operation floor (top floor of the reactor building) level
▶Reduction of load on the damaged reactor building is necessary.
▶Shielding at the operation floor level allows reduction of the shielding function of the fuel debris retrieval cell. → Weight saving
▶Shielding of direct radiation from fuel debris is allowed.
▶DSP (dryer separator pool) is utilized as a work space (for collecting in the storage canister, etc.)

Figure 1: Image of top-entry method

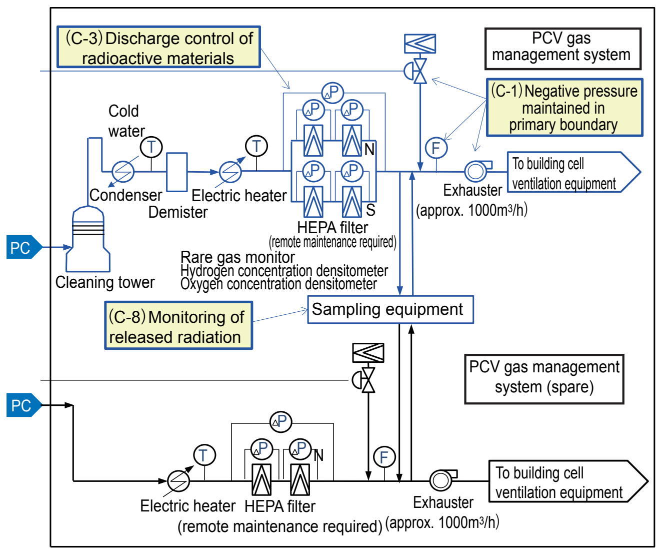


Figure 2: Gas leakage prevention system configuration (example)

The system can keep negative pressure inside the PCV, and has a high performance particle filter in the exhaust line.

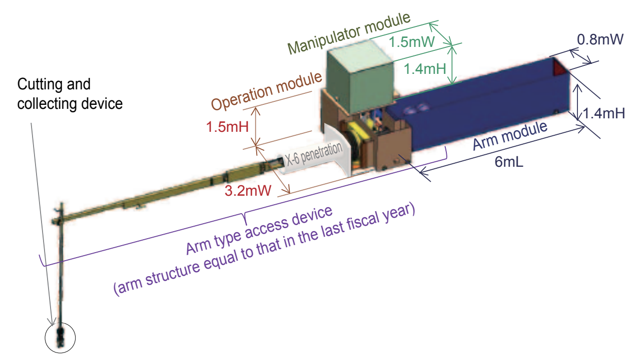


Figure 3: Example of sampling access device (arm type)

The figure above shows an example of combination of a cutting and collecting device and an access device based on sampling cutting conditions.

Main R&D Achievements Subsidized Projects
Main R&D Achievements In-house R&D Activities
Others Data Sheet

Key Challenge 2 R&D for Fuel Debris Retrieval

Development of Fundamental Technologies for Retrieval of Fuel Debris and Internal Structures

Background

It is assumed that fuel debris in the Reactor Pressure Vessel (RPV) and the Primary Containment Vessel (PCV) at the Fukushima Daiichi NPS has not currently reached criticality. However, the reactor buildings, the RPV and the PCV were damaged at the accident, and the reactors have been in unstable conditions. Therefore, it is necessary to retrieve fuel debris to maintain sub-criticality, and stabilize the reactors without diffusing radioactive materials.

Aims

We promote specific R&D for determination of fuel debris retrieval policy (around the summer of 2017), establishment of the retrieval policy for the initial unit (FY2018), and start of retrieval at the initial unit (2021). In this project, targeting three methods of fuel debris retrieval such as Submersion-Top entry method, Partial submersion-Top entry method and Partial submersion-Side entry method, we conducted an elemental test to obtain the data information required for evaluating feasibility of these methods.

Main Achievements and Approaches

1 Comprehensive adjustment of each elemental test and analysis of elemental test result

In addition to IRID, another proposer who partially engages in the project joined in the R&D. IRID examined the achievements of the entire project, including the elemental tests planned and implemented by the partial proposer.

2 Elemental test required to determine feasibility of method

In FY2016, we conducted necessary confirmation through partial trial production and partial elemental tests according to each elemental test plan formulated in FY2015.

① Prevention technology against the spread of contamination in removal of large structures

- In order to verify prevention technology against the spread of contamination, we conducted a scale model test for each work step (Figure 1).

② Prevention technology against the spread of contamination in retrieval of fuel debris inside RPV

- We conducted a test on the RPV inner sealing and lower part sealing of the access device in the RPV under the Partial submersion-Top entry method (Figure 2).

③ Technology for access to fuel debris

- A test on the hydraulic manipulator (Figure 3) was conducted.
- A test on the in-RPV access device was conducted under the Submersion-Top entry method (Figure 4).
- A test on the in-pedestal access device was conducted under the Partial submersion-Side entry method (Figure 5).

④ Remote operation technology for fuel debris retrieval

- A test on the flexible structure arm for remote operation (Figure 6) was conducted.
- A test on the handling device for the fuel debris storage canister was conducted.

⑤ Prevention technology against the spread of contamination in fuel debris retrieval

- A test on the Submersion method platform and cell (Figure 7) was conducted.
- A test on the PCV welding device used for remote seal welding related to the cell of Partial submersion-Side entry method (Figure 8) was conducted.

⑥ Dose reduction technology for workers during fuel debris retrieval

- Tests on shape following and lightweight shielding for the application of Top-entry methods were conducted.

⑦ Cutting and duct collection in fuel debris retrieval

- A test on performance of the fuel debris cutting and collection technology was conducted.

Future Developments

Among the elemental tests conducted in FY2016, we will plan and implement not only the elemental tests to be carried out continuously but also additional elemental tests to solve issues newly extracted in implementing the elemental test. We also plan to implement elemental tests to solve issues extracted from the feasibility evaluation of the methods.

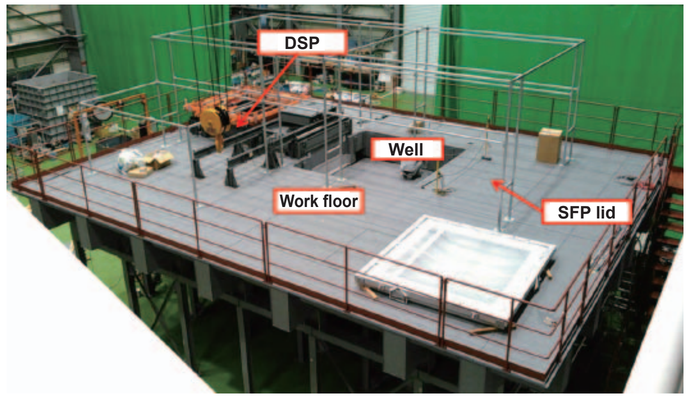


Figure 1: Scale model test facility

A scale model test was performed to study on the prevention of spreading contamination.

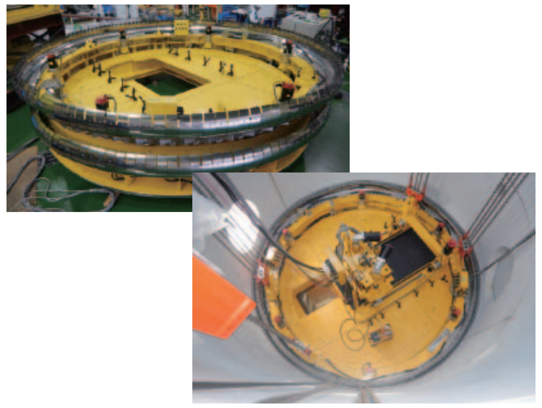


Figure 2: Access device in RPV (device inside the reactor), full-scale model

Sealing performance was verified by full-scale device model.

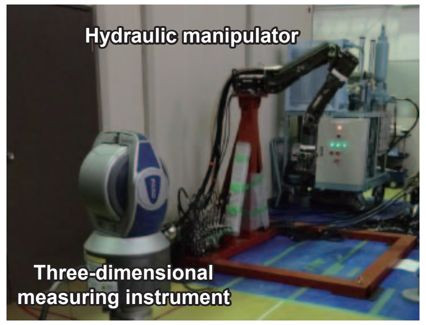


Figure 3: Hydraulic manipulator test

Pressure feedback was additionally installed to improve accuracy of control.

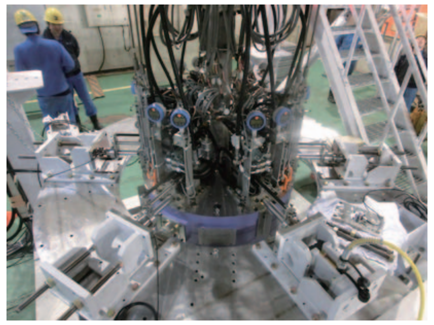


Figure 4: Access device in RPV test

The reaction force holding performance of the access device in the RPV was verified.

Figure 5: In-pedestal access device test

A test for access of the robot arm and the access rail was conducted to verify basic feasibility.

Figure 6: Prototype of flexible structure arm for remote operation

We manufactured a prototype of robot with a flexible structure arm and conducted a test of operability inside the PCV.

Figure 7: Test device for large shielding door of platform and cell

We manufactured a large shielding door inside the cell for the Submersion method, and conducted a verification test on its operability, etc.

Figure 8: Test for PCV welding device

A remotely-operate welding test was conducted to verify basic feasibility.

Key Challenge 2 R&D for Fuel Debris Retrieval

Development of Technology for Collection, Transfer and Storage of Fuel Debris

Background

According to the Mid-and-long-Term Roadmap towards the Decommissioning of Fukushima Daiichi Nuclear Power Units 1-4 (TEPCO), fuel debris retrieved from the Fukushima Daiichi NPS is expected to be moved from the reactor buildings and stored safely until the time when treatment and disposal methods are determined. For this reason, a system required for the collection, transfer, and storage of fuel debris must be established.

Aims

Based on the experience of Three Mile Island Nuclear Generating Station Unit 2 (TMI-2) in the US, and existing technologies for transportation and storage of spent fuel, we develop a fuel debris storage canister (hereinafter called "storage canister") and canister handling devices for safe and effective collection, transfer, and storage. In FY2016, we summarized storage canister design mainly for safety design.

Main Achievements and Approaches

1 Investigation and establishment of research plans for transfer and storage of damaged fuel

In order to investigate sub-criticality control technology, hydrogen management technology, damaged fuel drying technology, and characteristics of products generated by reaction with molten core and concrete (MCCI product), which were considered to contribute to the discussion of sections 2 to 5 below. We visited the Sellafield facility in the UK and the Argonne National Laboratory in the US to obtain relevant information.

2 Study on fuel debris storage system

Following FY2015, we examined the storage concept of fuel debris that was considered in FY2014 based on the latest situation of the site, information from the related projects, and the knowledge obtained from this project. We have confirmed that it is not necessary to review the storage system concept.

3 Development of safety evaluation methods

Unlike the experience of TMI-2 fuel debris, the Fukushima Daiichi NPS requires countermeasures for safe collection, transfer, and storage of fuel debris and MCCT products which are supposed to contain sea water components. In addition, for safe and effective collection, transfer, and storage of fuel debris, expansion of the inner diameter of storage canisters and remote operation are required.

Continuously from FY2015, concerning evaluation on hydrogen amount generated (countermeasure against hydrogen generation), aging degradation evaluation on materials (material selection), criticality evaluation (study on mitigation of sub-criticality conditions with water amount restriction) to reflect these requirements in the storage canister design, and study on the storage canister structure to satisfy these requirements (a buffer structure for ensuring structural integrity at the time of falling), we conducted investigation of domestic and overseas case studies, trial analysis and tests, to summarize an assessment method plan (Figures 1, 2, 3).

4 Development of technology for fuel debris collection

Based on the basic specification of storage canister designed in FY2015, we summarized an idea of storage canister for fuel debris retrieval mockup tests (Figure 4) based on the results of examination of the safety assessment method described in the section 3 above and the requirements from the related projects (lid structure for remote operation, etc.)

5 Development of technology for transfer and storage of storage canister

We formulated basic specification drafts of a storage canister hanging tool for remote operation, a storage canister lid hanging tool for remote operation and others, which would be basic equipment for the storage canister in the section 4 above (Figure 5).

Future Developments

For development of the technology, we will reflect the study results of fuel debris retrieval and collection to be conducted especially in the projects for the fuel debris retrieval method and system upgrading, and the fundamental technology in the future. Also, the final specifications of the storage canister and its device will be verified to optimize from both aspects of safety and operability.

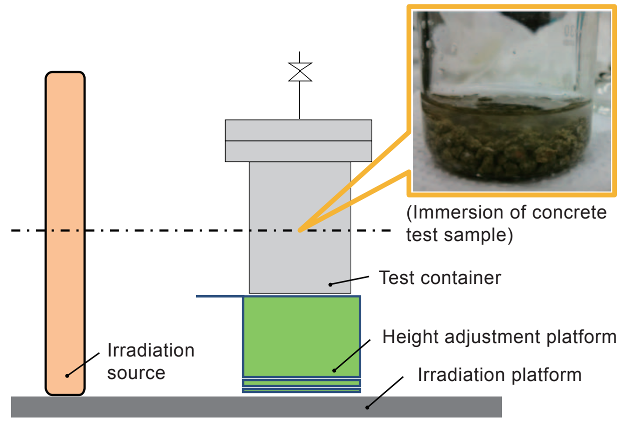


Figure 1: Elemental test for detailed investigation on hydrogen amount generated

In order to assume the amount of hydrogen generated by radiation decomposition of water remaining in fuel debris, we conducted a test to confirm the impact of the elution of concrete components into water. We will reflect this data to improve the accuracy of prediction of hydrogen amount generated.

Figure 2: Drain test for mitigation of sub-criticality condition

Increasing the inner diameter of the storage canister contributes to improved effectiveness of fuel debris retrieval and storage. We set a water amount limit value as a scenario to maintain sub-criticality even if the inner diameter is enlarged. We are conducting a test for water content reduction assessment by draining a mesh structure using oxide simulating fuel debris.

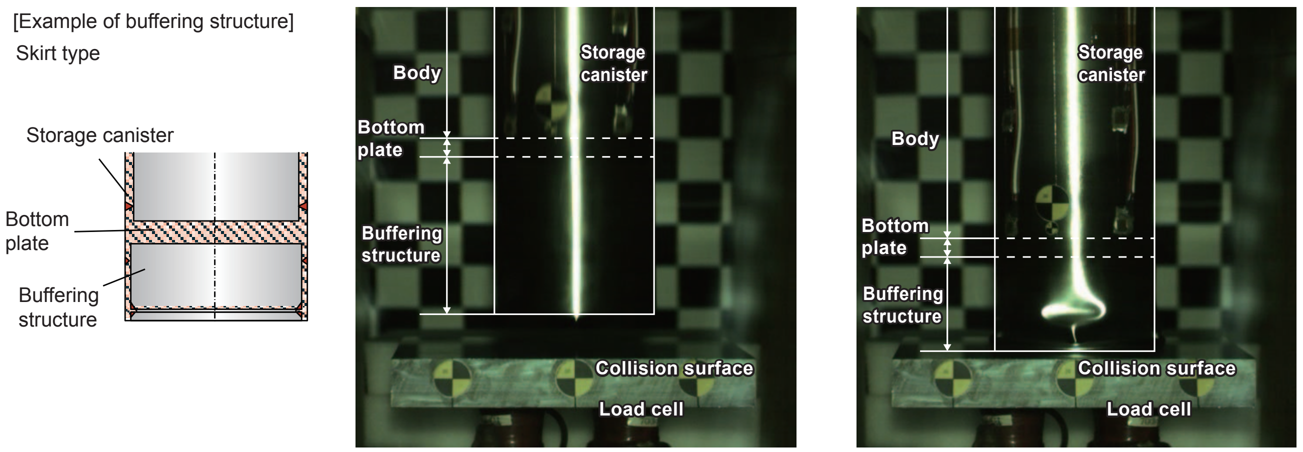


Figure 3: Elemental test for buffering structure

We performed a verification test on the function to cushion the shock applied to the storage canister in the case the storage canister would be dropped. We will reflect the result obtained into the buffering structure design and improvement of accuracy for simulated behavior analysis methods.

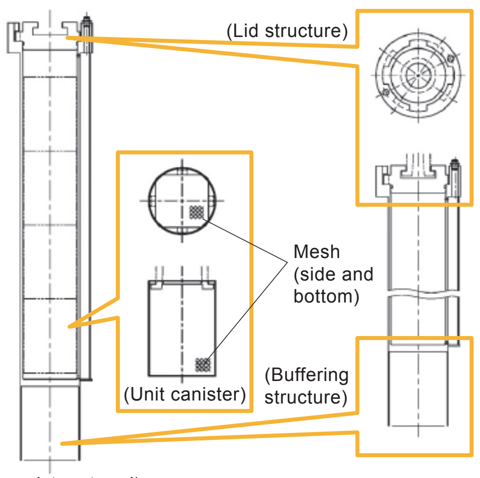


Figure 4: Idea of basic plan of storage canister

This figure shows a sample idea of storage canister form for mockup test. We will optimize the form in cooperation with the fuel debris retrieval method and system PJ.

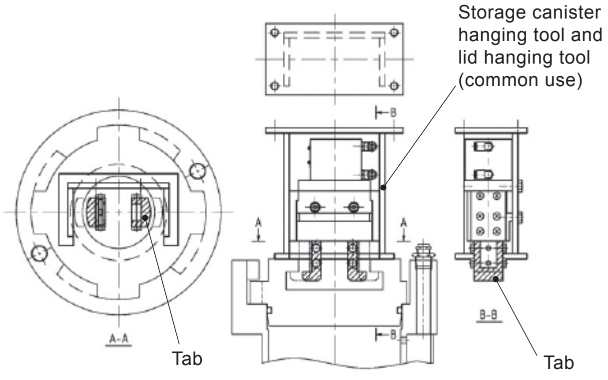


Figure 5: Example of basic design of storage canister hanging tool and lid hanging tool (common use)

To use the storage canister by remote operation, we formulated the basic specification drafts of the storage canister hanging tool and the lid hanging tool (common use). From now, we will optimize the easier handling form in cooperation with PJ of the fuel debris retrieval method and system.

Key Challenge 3 R&D for Treatment and Disposal of Radioactive Waste

R&D for Treatment and Disposal of Solid Radioactive Waste

Background

As fuel debris will be retrieved at the Fukushima Daiichi NPS, a large amount of radioactive wastes are expected to generate in the future, in addition to debris and fallen trees that had already been stored. In this situation, it is necessary to continue characterizing the radioactive wastes in parallel with the development of methods for storage, treatment, and disposal for implementation.

Aims

We develop technology for safe disposal and treatment of solid wastes generated in the accident through study on a series of storage management strategies for main radioactive wastes, update of waste management stream, analysis of wastes, characterization of wastes through inventory assessment based on the waste analysis, fundamental test of treatments, study of long-term storage strategy for secondary waste generated from contaminated water treatment, identifying of disposal concept, and study of disposal classification for accident wastes.

Main Achievements and Approaches

1 Integration of R&D achievements

Concerning the waste management stream, we established a method to narrow down waste management streams with multiple options and extracted issues to apply the method based on case studies. We also participated in a workshop for the report on the management of accident waste by the organized by OECD/NEA expert group, and cooperated in publishing the report.

2 Characterization

Rubble, secondary waste generated from contaminated water treatment, and contaminated water were transported to an off-site facility, and then radioactivity analysis was conducted. The rubble (concrete) of Unit 1 reactor building (R/B) showed results consistent with the analysis data of the rubble obtained so far (Figure 1). We examined a method of collecting cesium adsorption vessel zeolite, which was a sample of high dose, conducted conceptual design of a sampling device of the absorption vessel of the cesium adsorption apparatus, and proposed a sampling method. We also examined a method to reduce the uncertainty of evaluation for analytical inventory estimation methods, found a set of inventory data with an improved method, and made it an input for safety assessment of disposal.

3 Study on treatment of radioactive waste treatment and long-term storage method

We implemented a solidification fundamental test (Figure 2) based on the existing technology for secondary waste generated from contaminated water treatment without any experience in solidification treatment, and obtained data of availability of solidification, and integrity verification data of solidified material. We also compared the data with the requirements for technical evaluation and evaluated applicable technologies for each waste.

In order to stabilize the Advanced Liquid Processing System (ALPS) slurry, we studied a concept of device based on the operational aspects for introduction of the slurry stabilizing device on site and the verification test results (Figure 3).

An accelerated test of remaining water evaporation behavior accompanying heat generation of the cesium absorption vessel and evaluation of salinity concentration behavior were conducted to propose measures and a verification method required for long-term storage.

4 Study on disposal of waste

Based on the case studies on the overseas disposal concept (Figure 4), we organized information for formulating a disposal concept and summarized features of the existing disposal concept. In addition, we improved a safety assessment method to evaluate the disposal classification considering uncertainty.

Future Developments

For characterization, we will feed back the result of study on treatment and disposal, update the analysis plan, and reflect it into improvement of the inventory estimation model. For study on the long-term storage policy, we will investigate and examine the evaluation method of hydrogen generation and the handling method of generated hydrogen for safe and rational storage of the secondary waste generated from contaminated water treatment. When examining treatment, we will investigate possibility of long-term change in physical properties of solidified materials and formulate a test plan. When examining disposal, we will investigate the overseas disposal concepts, evaluation methods, evidence for setting, and background in detail. With the results above, we will integrate research progress and achievements comprehensively based on the waste management stream.

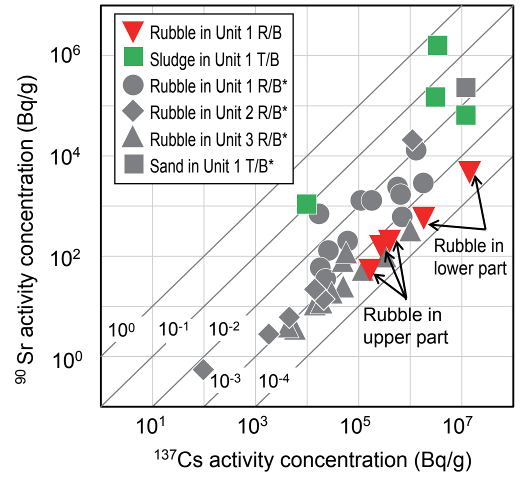


Figure 1: Ratio of ^{90}Sr/^{137}Cs in rubble

The ratio of ^{90}Sr/^{137}Cs in the rubble in Unit 1 R/B was consistent with the data obtained so far (*data from FY2014 to early FY2016).

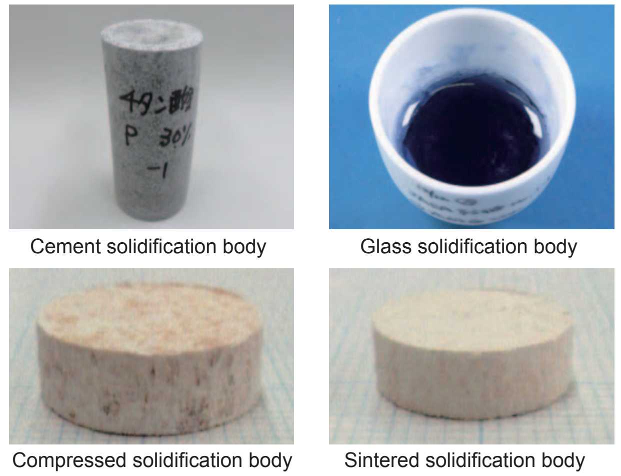


Figure 2: Various solidified body of simulated waste

We conducted a solidification fundamental test of the secondary waste generated from contaminated water treatment based on the existing technology, and collected basic data such as compressive strength, advance rate, hydrogen gas generation amount of the solidified sample.

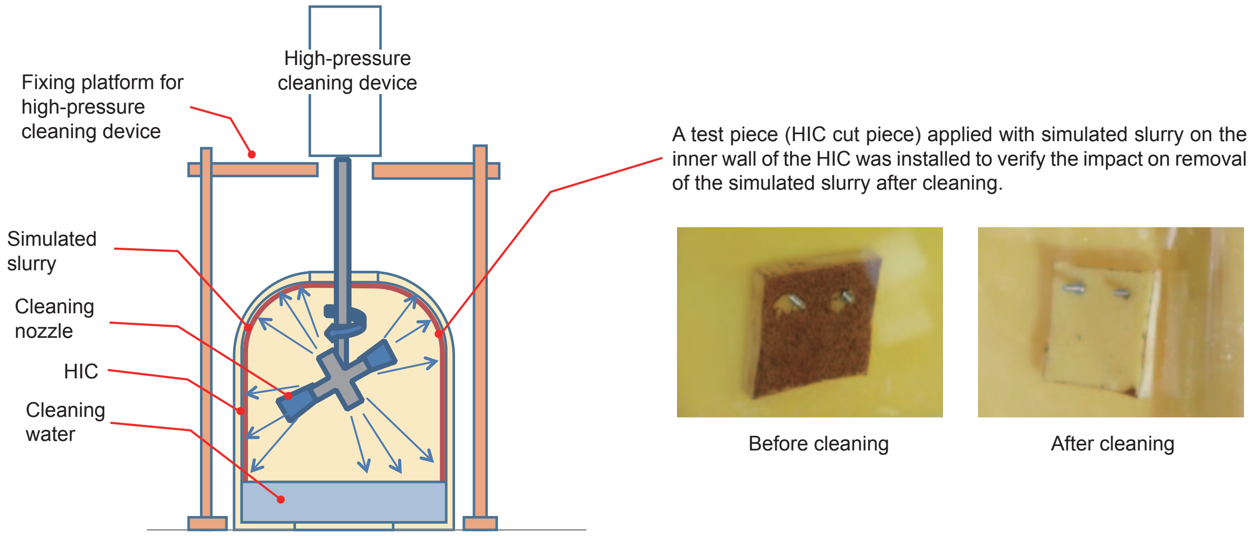
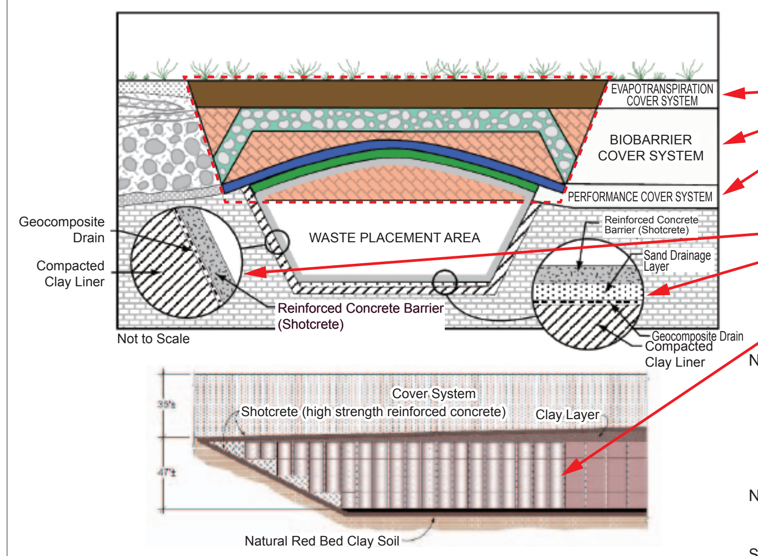


Figure 3: HIC container inner surface cleaning verification test



A three-layered cover system is installed at the top of the waste layer. (Furthermore, each cover system is configured with multiple layers with water blockage, drainage, and other functions.)

Compressed clay layer, rebar blown concrete layer, and drainage layer are placed on the bottom and sides of the waste layer. (At the bottom, a layer for withstand load is placed.)

Four-stage reinforced concrete canister (Slits are filled with sand.)

Note 1 This example shows the waste facilities (CWF) of the agreed states. In addition to this, there is a compartment of the Federal Waste Facility (FWF), and it is required not only to comply with the state regulation requirements but also to the Resource Conservation Recovery Act (RCRA), so the facility specifications and installation concept (a reinforced concrete canister of wastes solidified to the container has seven stages) are slightly different.

Note 2 The figure above shows an image created for the purpose of indication of the facility functions, and actual scale is different. The actual scaled image is shown below.

Source Application form of WCS

Figure 4: Case study example of overseas disposal concept [Conceptual diagram of the US Texas WCS (trench with liner) disposal facility]

Measurement and Evaluation of Distribution of Fuel Debris inside the Reactor at the Fukushima Daiichi NPS Unit 2

Background

In order to select an efficient fuel removal method, it is important to identify fuel debris distribution in the reactor. For this reason, we have developed technology to see through the inside of the nuclear reactor using cosmic-ray muon with the cooperation of domestic and foreign organizations. We are currently promoting a plan to utilize this technology for investigation of each unit.

Aims

Based on the muon transmission method technology developed by the High Energy Accelerator Research Organization, we intend to confirm the fuel debris distribution in the Reactor Pressure Vessel (RPV) using the improved muon transmission method for the Fukushima Daiichi NPS.

Main Achievements and Approaches

1 Measurement with compact transmission method measuring device

In the measurements of Unit 1 conducted from FY2014 to FY2015, the site operations were so congested that we could not install the equipment near the reactor building. Accordingly, we could not secure the elevation angle required to measure the lower part of the RPV. Therefore, we developed a compact device of which ground contact area was about one fifth, allowing less interference with the on-site operations, and used it in this project for the first time. In addition, in order to improve handling performance at the site, we adopted a surrounding system with lead plates at the site to shield environmental gamma rays, instead of the conventional iron box shielding body (Photo 1, 2). Since this system could be installed even in places where large cranes could not approach, it contributed to completion of the measurements.

2 Measurement results and distribution of amount of substance inside RPV

We installed a measuring device on the ground at the west side of the Unit 2 reactor building (Figure 1) and conducted measurements from March to July, 2016. Then we evaluated distribution of the amount of substance based on the measurement results (Figure 2) and confirmed a shadow which seemed to be fuel debris at the bottom of RPV.

3 Summary

- We measured Unit 2 using the more compact muon transmission method measuring device and confirmed the shadows of the main structures (high density materials).
  - ▶We confirmed the shadow of shielding concrete for the outer circumference of the containment.
  - ▶We confirmed the shadow at the position of the spent fuel pool.
  - ▶We confirmed the shadows of structures such as walls and floors of the reactor building.
- As a result of evaluating the obtained data, we have confirmed that high-density substances considered to be fuel debris exist at the bottom of the RPV.
- As a result of the evaluation by comparison with the simulation, it was estimated that there might be some high-density substances which were supposedly fuel in the lower part of the core and the core outer circumference. However, uncertainty still remains in the evaluation due to the influence of the structures of the reactor building.

Future Developments

Using the muon transmission method, we could investigate presence or absence of the high-density substances in the reactor in two measurements; the measurement at Unit 1 and Unit 2. In the future, we will consider application of this method to acquire information in a short period of time, according to the site condition and investigation needs.

(Size: Approx. 1 m × 1 m × Height 1.3 m)
Photo 1: Muon transmission method measuring device (compact type)

Photo 2: Measuring device surrounded with lead plates

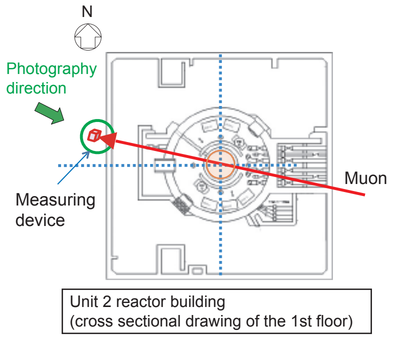


Figure 1: Measuring device installation position

※ Size of one pixel: Equivalent to approx. 25 cm at the cross section of the nuclear reactor

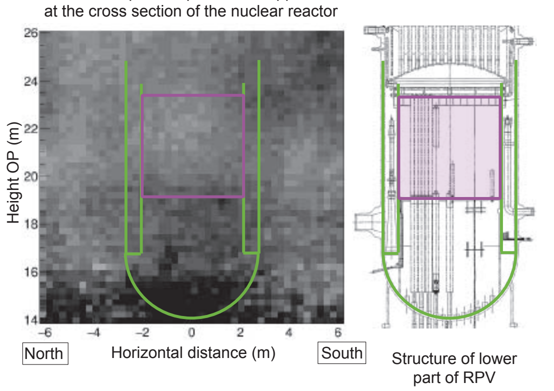


Figure 2: Distribution of amount of substance at lower part of RPV

(Measurement result: As of July 22, 2016)

Improvement of suction and blast decontamination equipment for high places

Background and aims

Since the cylinder part of the suction and blast decontamination equipment for high places developed on the supplementary budget in FY2013 was damaged during the mock-up demonstration test, we changed the design, and derived applicability of the technology. In this research, we decided to conduct a verification test for operability, which confirmed applicability of the technology.

Main Achievements and Approaches

1 Countermeasures and design change

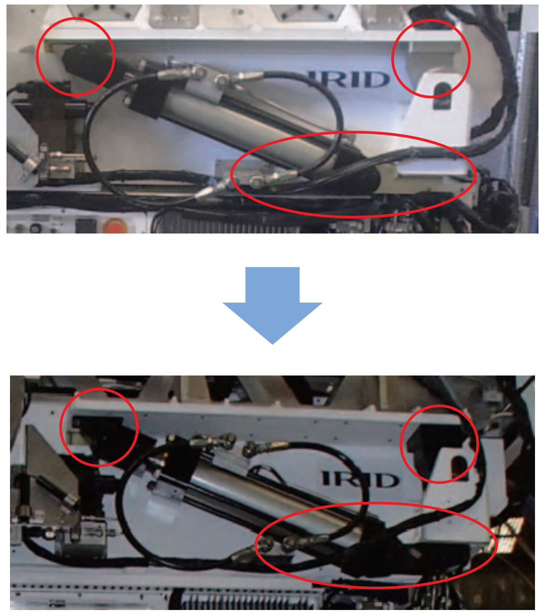
①We eliminated a stress concentration point and changed the previous material to the new one that supported stress generated by excessive load (acceleration) (Extra super duralumin ⇒ Carbon steel), and improved components.

②We installed an upper limiter for operation speed against excessive load (acceleration), and restricted its use to the range from 1.71 to 2.42 G.

2 Operation verification test result

We conducted an operation verification test with the mockup equipment after the improvement, and confirmed that it satisfied the device requirement.

We will install it according to needs at the site.



Improvement of inclined shaft hydraulic cylinder fixing bracket (both ends of cylinder) and lift part rotating shaft

Verification of effect of improved injection function of dry ice blast decontamination equipment

Background and aims

Since an event where the dry ice blast decontamination equipment for high places developed on the supplementary budget in FY2013 terminated ejecting dry ice during application to Unit 3 at the Fukushima Daiichi NPS, we confirmed reproducibility and discussed countermeasures, and derived applicability of the function. In this research, we decided to further identify a cause for countermeasures, and verified effectiveness of those countermeasures with an injection test.

Main Achievements and Approaches

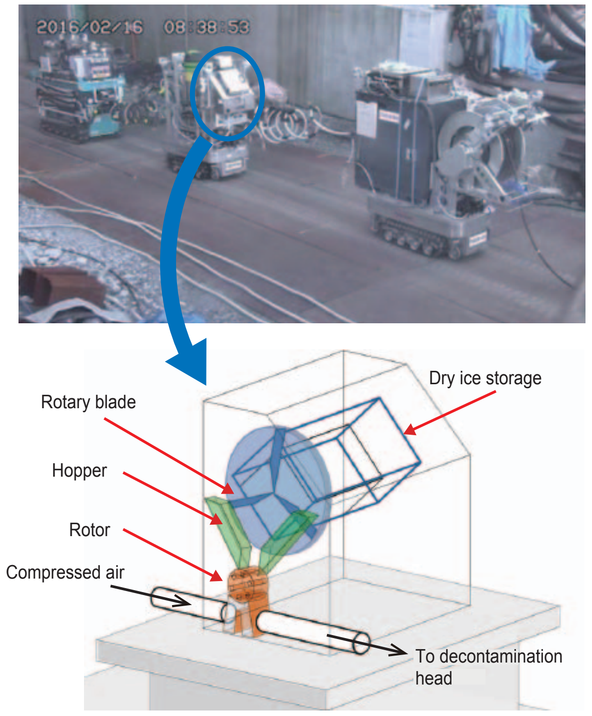
1 Identification of cause and countermeasures

We identified the cause that dry ice supply system and the injection route were clogged by intruded dry ice. We decided to confirm the situation inside the dry ice storage and clean it before and after using the equipment, and also to blow the injection route by air.

2 Injection verification test result

After taking the countermeasures mentioned above, we conducted an injection verification test and confirmed applicability of equipment without clogging of dry ice to satisfy the device requirement.

We will introduce it according to needs at the site.



Structural diagram of dry ice supply unit outline

Main Research Results in 2016

No.	Presented at/by	Date	Details
1	"Report on Decommissioning Acceleration Research Program for the Center of World Intelligence Project for Nuclear S&T and Human Resource Development (Japan-UK Collaboration Project), FY2015 "	Apr. 1, 2016	Present status in Fukushima - Robot and sensing technologies for decommissioning of Fukushima Daiichi -
2	The 1st International Forum on the Decommissioning of the Fukushima Daiichi Nuclear Power Station	Apr. 10-11, 2016	Overview of A2 investigation / B2 investigation (panel), display of static A2 investigation device, demonstration of B1 investigation device prototype
3	Lecture at Kyoto University "Nuclear power plant engineering"	Apr. 14, 2016	Current status of Fukushima Daiichi Nuclear Power Station and efforts for revitalization
4	Special Lecture on Reactor Decommissioning (Tokyo Institute of Technology)	Apr. 22, 2016	Current status of fuel debris and retrieval technology
5	International Congress on Advances in Nuclear Power Plants (ICAPP2017)	Apr. 24, 2016	Characterization of Carbonate Slurry generated from Multiple Radio-nuclides Removal System in Fukushima Daiichi Nuclear Power Station
6	Nuclear Materials and Energy (magazine)	Apr. 2016	Radioactive Contamination of Several Materials in the Accident of Fukushima Daiichi Nuclear Power Station
7	Japan Society of Mechanical Engineers	May 1, 2016	Introduction of application of muon scattering method to nuclear power
8	Journal of the Japanese Society for Non-Destructive Inspection	May 1, 2016	Development of pinhole type gamma camera to visualize gamma ray intensity distribution
9	Conference of "Conditioning and Geological Disposal of Radwastes"	May 23, 2016	JAEA R&D activities on fuel debris characterization and related research for 1F decommissioning, and other two subjects
10	Japan Society of Maintenology	May 25, 2016	Development of Remote Decontamination Technologies improving Internal Environment of Reactor Building at Fukushima Daiichi NPS
11	Exchange of opinions with UK Sellafield Ltd.	May 25, 2016	Current status of Fuel Debris Retrieval Technology Development for 1F, and other six subjects
12	Special Lecture on Reactor Decommissioning (Tokyo Institute of Technology)	May 27, 2016	Remote technology
13	Joint research group of Institute of Nuclear Materials Management / Nuclear Nonproliferation Liaison Committee of Atomic Energy Society of Japan	Jun. 1, 2016	R&D and challenges on decommissioning and fuel debris at TEPCO's Fukushima Daiichi Nuclear Power Station
14	Central Research Institute of Electric Power Industry "Annual Report 2015"	Jun. 1, 2016	Enhancement of technology base to clarify damaged and molten fuel behavior during severe accident
15	The 27th regular meeting of "Division of Water Chemistry," Atomic Energy Society of Japan	Jun. 3, 2016	Impact assessment on rust preventives for RPV / PCV on water treatment facility Overview of development of Fukushima Daiichi NPS accident waste treatment and disposal technology
16	Lecture at Nuclear Power Generation Division "Interchange seminar between professional people and students"	Jun. 4, 2016	Approach to domestic and international nuclear power
17	Nondestructive Inspection Symposium, FY2016	Jun. 7, 2016	"Measurement and visualization technology by radiation corresponding to nuclear accident" Development of pinhole type gamma camera for environmental radiation monitor
18	The 26th Joint Symposium of the Institution of Professional Engineers of Railway Technical Research Institute / the Institution of Hitachi Professional Engineers	Jun. 20, 2016	Development of pinhole type gamma camera for environmental radiation monitor
19	ICONE24 (The 24th International Conference on Nuclear Engineering)	Jun. 26, 2016	IN PLANT 3D POSITIONING SYSTEM USING POINT CLOUD DATA FOR REMOTE DECONTAMINATION MACHINE
20	WNE (World Nuclear Exhibition)	Jun. 28, 2016	Introduction of MEISTeR II and Super Giraffe
21	The 29th "Summer Seminar of Nuclear Fuel Division," Atomic Energy Society of Japan	Jul. 6, 2016	Challenges for fuel debris retrieval/ Chemical forms of seawater salt, FP in MCCI products
22	Advanced Special Lecture at Department of Sustainable Energy and Environmental Engineering, Osaka University	Jul. 8, 2016	Identifying the condition of molten fuel and criticality management at Fukushima Daiichi Nuclear Power Station
23	3rd Workshop of the OECD/NEA BSAF Project Phase 2	Jul. 8, 2016	Introduction of Japanese national projects on debris characteristics
24	"HOZENGAKU (Maintenology)" Japan Society of Maintenology	Jul. 10, 2016	Verification test of upper floor decontamination equipment
25	Journal of Nuclear Science and Technology	Jul. 12, 2016	Estimation of the Inventory of the Radioactive Wastes in Fukushima Daiichi NPS with a Radionuclide Transport Model in the Contaminated Water
26	The Social Infrastructure Maintenance Show	Jul. 20, 2016	Introduction of MEISTeR II and Super Giraffe
27	Nuclear and Radiation Subcommittee, the Institute of Professional Engineers, Japan	Jul. 22, 2016	Challenges of waste treatment and disposal for decommissioning of TEPCO Fukushima Daiichi Nuclear Power Station, and R&D status
28	The 13th academic lecture meeting of Japan Society of Maintenology	Jul. 25-27, 2016	Technology development for fuel removal towards decommissioning of Fukushima NPS, accident waste treatment and disposal technology, and other six subjects
29	Civil engineering magazine "Civil engineering construction"	Jul. 2016	Development of technology for investigation inside PCV - Response to challenges in B1 investigation for Unit 1 and investigation results -
30	Analytical Chemistry	Jul. 2016	Development of an Extraction Chromatography Method for the Analysis of 93Zr,94Nb and 93Mo in Radioactive Contaminated Water Generated at the Fukushima Daiichi NPS
31	Research Conference on Radiation Measurement for Decommissioning of the Fukushima Daiichi NPP	Aug. 4, 2016	Technology development for investigation of inside of PCV "Development light - section method"
32	Research Conference on Radiation Measurements for Decommissioning of the Fukushima Daiichi NPP (Radiation summer school for decommissioning)	Aug. 4, 2016	Overview of fuel debris measurement device (presentation)
33	IRID Symposium	Aug. 4, 2016	A2 investigation / B2 investigation (panel), Demonstration of A2, B1 investigation device prototypes Debris retrieval method and fundamental technology development (panel), exhibition of muscle robot prototype
34	The 13th Nuclear Power Generation Technology Summer Seminar	Aug. 5, 2016	Approach to domestic and international nuclear power (1)
35	Seminar at Advanced Course Program of National Institute of Technology, Fukushima College	Aug. 5, 2016	Overview of molten reaction and introduction of investigation for fuel debris characterization research
36	2016 Fall Meeting, Atomic Energy Society of Japan	Sep. 7-9, 2016	◆ Development of critical management technology for fuel debris - Critical management policy of debris retrieval, Critical behavior evaluation, Critical approach monitor, Insoluble neutron absorber, and other ten subjects
37	2016 Fall Meeting, Atomic Energy Society of Japan	Sep. 7-9, 2016	◆ Development of remote decontamination technology in reactor building - Result of verification test of upper floor decontamination equipment (blast system, high pressure water system), and other two subjects
38	2016 Fall Meeting, Atomic Energy Society of Japan	Sep. 7-9, 2016	◆ PCV integrity - Resistance evaluation of reinforced concrete structure after severe accident, and other three subjects
39	2016 Fall Meeting, Atomic Energy Society of Japan	Sep. 7-9, 2016	◆ Long-term integrity assessment of fuel assemblies removed from spent fuel pool - Three issues such as Evaluation of fuel integrity during dry storage, Evaluation of seawater component transfer by tracer, etc.
40	2016 Fall Meeting, Atomic Energy Society of Japan	Sep. 7, 2016	◆ Analysis and evaluation of assessing conditions inside reactor - Four subjects related to MAAP, six subjects related to SAMPSON, and five other subjects
41	2016 Fall Meeting, Atomic Energy Society of Japan	Sep. 7-9, 2016	◆ Fuel debris characterization - Test and evaluation on behavior of product with simulated debris, and other eight subjects
42	2016 Fall Meeting, Atomic Energy Society of Japan	Sep. 7-9, 2016	◆ R&D on treatment and disposal of solid waste - Development of treatment and disposal technology for Fukushima Daiichi NPS accident waste, Development of inventory evaluation methods, and other twelve subjects
43	2016 Fall Meeting, Atomic Energy Society of Japan	Sep. 7-9, 2016	◆ Development of repair and water leakage stoppage technology for leakage points inside the PCV - Development of water stoppage technology in S/C by filling underwater insoluble concrete
44	The 34th Annual Conference of the Robotics Society of Japan	Sep. 7, 2016	Robot for investigating inside different diameter pipes
45	EUROCORR 2016	Sep. 11, 2016	Corrosion behavior of SUS316L in nitic acid solution containing seawater components -Effect of metal ions in high active liquid waste-
46	Open Innovation Symposium, Kanagawa version	Sep. 16, 2016	Open Innovation Symposium, Technology interchange panel
47	Advanced Course for Engineering of Reactor Decommissioning Measures	Sep. 17, 2016	Characterization and treatment of fuel debris
48	Advanced Course for Engineering of Reactor Decommissioning Measures (Tohoku University)	Sep. 17, 2016	Development of robot technology for decommissioning activities and status of application on site
49	ISTP (The 27th International Symposium on Transport Phenomena)	Sep. 20, 2016	Coupled analysis on fluid structure for assisting piping closure
50	7th International Scientific and Practical Conference	Sep. 21, 2016	CHARACTERIZATION OF FUEL DEBRIS BY LARGE-SCALE SIMULATED DEBRIS EXAMINATION AT NNC KAZAKHSTAN FOR FUKUSHIMA DAIICHI NUCLEAR POWER STATIONS
51	Journal of RANDEC (Decommissioning technique) No.54	Sep. 30, 2016	Present status of solidification technology of radioactive waste using alkali active material - geopolymer -
52	13th International Conference on Probabilistic Safety Assessment and Management(PSAM13)	Oct. 2, 2016	Water Injection Influence for Accident Progression in Fukushima Daiichi Unit 1
53	HOTLAB 2016 Karlsruhe, Germany	Oct. 2-3, 2016	Approach to estimating fuel debris properties generated in Fukushima Daiichi NPS
54	PSAM 13	Oct. 2-7, 2016	Water Injection Influence for Accident Progression in Fukushima Dai-ichi Unit1
55	Fuel debris research committee, Atomic Energy Society of Japan	Oct. 4, 2016	Estimation of fuel debris distribution by analysis and evaluation
56	12th International Conference on NDE in Relation to Structural Integrity for Nuclear and Pressurized Components	Oct. 6, 2016	Overview of Revitalization for Fukushima Nuclear Accident and Study on Robotics and Inspection Technologies for Decommissioning
57	NUTHOS-11	Oct. 9-13, 2016	Analysis for the TEPCO Fukushima Daiichi unit 2 by the SAMPSON code with core support plate model
58	NUTHOS-11	Oct. 9-13, 2016	Validation of severe accident code SAMPSON debris cooling analysis module (DCA) against OLHF experiments and development of creep models
59	Japan Society of Corrosion Engineering	Oct. 17, 2016	Impact assessment on mixed phosphate of zinc and sodium carbonate affecting PCV materials in diluted artificial seawater (2)
60	Institute of Nuclear Materials Management	Oct. 17-18, 2016	Limit of detection of a bare nuclear material debris lump

Main Research Results in 2016

No.	Presented at/by	Date	Details
61	Japan Fluid Power System Society	Oct. 19-21, 2016	Development of hydraulic manipulator simulator for control system design and evaluation of cotton type model identification method and other tow issues.
62	RADIEX 2016	Oct. 20, 2016	Efforts of the Japan Atomic Energy Agency for decommissioning
63	RADIEX 2016	Oct. 20, 2016	Overview of A2 investigation / B2 investigation (panel)
64	Monthly OPTRONICS, December 2016	Oct. 27, 2016	A Robot Developed to Investigate Primary Containment Vessel (PCV) Interiors - Up to now and future
65	INES-5	Late Oct. 2016	PHITS Benchmarking for HPGe Detector Efficiency
66	ICMST 2016	Nov. 2, 2016	Development of the Remote Decontamination Robot "MHI-MEISTeR II" for an Upper Floor of Reactor Building in Fukushima Daiichi NPP
67	Science AGORA	Nov. 5-6, 2016	A2 investigation (panel), exhibition of static device
68	Nuclear Materials Conference (NuMat2016)	Nov. 7, 2016	Sic issues such as Phases and morphology in the simulated MCCI products prepared by arc melting method, etc.
69	Annual Meeting of the International Network of Laboratories for Nuclear Waste Characterization (LABONET)	Nov. 7, 2016	Characterization of Carbonate Slurry generated from Multi-Radionuclide Removal System in Fukushima Daiichi Nuclear Power Station
70	Annual Meeting of the International Network of Laboratories for Nuclear Waste Characterization (LABONET)	Nov. 7, 2016	Study on H2 gas produced by radiolysis of the waste of carbonate slurry generated from Multi-Radionuclide Removal System
71	LS-DYNA & JSTAMP Forum 2016	Nov. 8, 2016	Coupled analysis on fluid structure for assisting piping closure
72	The 2nd ITA-ACCELERATE Symposium	Nov. 10, 2016	Large structural imaging using cosmic rays
73	Fuel debris research committee, Atomic Energy Society of Japan	Nov. 16, 2016	Investigation result on fuel debris characterization - Large-scale MCCI test, MOX simulated debris, U-containing simulated debris, comparison with TMI-2
74	ROBOT FESTA FUKUSHIMA	Nov. 19, 2016	Overview of A2 investigation / B2 investigation (panel), display of static A2 investigation device, demonstration of B1 investigation device prototype
75	Electronic journal of Japan Society of Maintenology	Nov. 25, 2016	High-pressure Water Jet Decontamination Apparatus for Upper Part of 1st Floor of Reactor Buildings for Fukushima Daiichi Nuclear Power Station, and other two subjects
76	International Conference on the Safety of Radioactive Waste Management	Nov. 25, 2016	Inventory Estimation for Accident Waste Generated at the Fukushima Daiichi NPS
77	International Conference on the Safety of Radioactive Waste Management	Nov. 25, 2016	Approaches of Selection of Adequate Conditioning Methods for Various Radioactive Wastes in Fukushima Daiichi NPS
78	NTHAS 10	Nov. 27-30, 2016	Validation of Debris Freezing Model in a Penetration Tube for the SAMPSON Code against the GEYSER experiments
79	Technical Committee on Laser Processing Technology, the Institute of Electrical Engineers of Japan	Nov. 30, 2016	Laser application for fuel debris retrieval at Fukushima Daiichi NPS
80	The Society of Instrument and Control Engineers	Dec. 15, 2016	Construction and basic experiment of RSNP remote control system of multiple crawler robots
81	"Fuel debris research committee," Division of Fuel Debris, Atomic Energy Society of Japan	Jan. 26, 2017	Overview of development of fuel debris criticality management technology
82	Health Physics	Feb. 2, 2017	Radiochemical analysis of debris samples collected at Fukushima Daiichi NPS
83	IAEA "First Coordinated Research Meeting and Consultancy Meeting on Management of Severely Damaged Spent Fuel	Feb. 14, 2017	Characterization to Estimate Fuel Debris properties Generated in Fukushima Daiichi NPS
84	WM2017(Waste Management Symposia)	Mar. 5, 2017	Long Reach Manipulator for PCV Repair at Fukushima Daiichi and other two issues.
85	Annual Spring Meeting 2017, Atomic Energy Society of Japan	Mar. 27-29, 2017	Development of analysis method for Pd-107 in radioactive waste collected at Fukushima Daiichi NPS
86	Annual Spring Meeting 2017, Atomic Energy Society of Japan	Mar. 27-29, 2017	Analysis and evaluation of assessing conditions inside reactor of the TEPCO Fukushima Daiichi NPS, seven subjects related to SAMPSON, and another one subject

List of Joint Researches / Contract Researches

No.	Project Nameresearch	Category	Subject	Partner	Period
1	Upgrading for Identifying Comprehensive Conditions inside the Reactor	Contract research	Inverse problem evaluation with virtual reactor	University of Tokyo	Aug. 1, 2016 - Feb. 28, 2017
2	Upgrading for Identifying Comprehensive Conditions inside the Reactor	Contract research	Analysis of pipe deformation during core material slumping	University of Tokyo	Aug. 1, 2016 - Feb. 28, 2017
3	Upgrading for Identifying Comprehensive Conditions inside the Reactor	Contract research	MELCOR analysis related to transition of events during core slumping	Waseda University	Aug. 1, 2016 - Feb. 28, 2017
4	Upgrading for Identifying Comprehensive Conditions inside the Reactor	Contract research	Segregation analysis of MCCI reactants at the time of melting and solidification	Tohoku University	Oct. 17, 2016 - Feb. 28, 2017
5	Upgrading for Identifying Comprehensive Conditions inside the Reactor	Contract research	Research on supplementary reaction and diffusion behavior of cesium	Osaka University	Oct. 17, 2016 - Feb. 28, 2017
6	Development of Technology for Criticality Control in Fuel Debris	Contract research	Verification test of critical approach detection system and neutron absorber to be applied to systems containing fuel debris	Kyoto University	Dec. 26, 2016 - Feb. 28, 2017
7	Development of Fundamental Technologies for Retrieval of Fuel Debris and Reactor Internals	Contract research	Investigation on discontinuous large load tip control	Osaka University	Jun. 24, 2016 - Feb. 28, 2017
8	R&D for Treatment and Disposal of Solid Radioactive Waste	Contract research	Research on criticality safety in geological disposal of damaged fuel at Fukushima Daiichi Nuclear Power Plant	University of California, Berkeley, U.S.A	Apr. 1, 2016 - Mar. 31, 2017
9	R&D for Treatment and Disposal of Solid Radioactive Waste	Contract research	R&D on radioactive waste management methods using gamma ray measurement (2)	Tokyo Institute of Technology	Apr. 21, 2016 - Feb. 28, 2017
10	R&D for Treatment and Disposal of Solid Radioactive Waste	Contract research	Research on estimation/assessment technology of radioactive waste inventory (Phase 2)	Central Research Institute of Electric Power Industry	May 20, 2016 - Jan. 31, 2017

Main R&D Installations / Equipment

Over 1 million yen

No.	Project Name	Details
1	Development of remotely operated decontamination technology	Suction/blast decontamination equipment for high places
2	Development of remotely operated decontamination technology	Suction/blast decontamination equipment for high places Testing device
3	Development of remotely operated decontamination technology	Dry ice blast decontamination equipment for high places
4	Development of remotely operated decontamination technology	Dry ice blast decontamination equipment for high places Testing device
5	Development of remotely operated decontamination technology	High-pressure water jet decontamination equipment
6	Development of remotely operated decontamination technology	High-pressure water jet decontamination equipment Testing device
7	Development of remotely operated decontamination technology	Decontamination equipment for upper floors (Work cart, Suction/Blast decontamination unit)
8	Development of remotely operated decontamination technology	Decontamination equipment for upper floors (Relay cart, Cable winder, Dry ice blast decontamination unit)
9	Development of remotely operated decontamination technology	Decontamination equipment for upper floors (Transportation card, Support card, High-pressure water jet decontamination unit)
10	Development of remotely operated decontamination technology	Decontamination equipment for upper floors Testing device
11	Development of remotely operated decontamination technology	Cart positioning measurement software
12	Development of remotely operated decontamination technology	Interference verification software
13	Development of remotely operated decontamination technology	Crawler cart for cooperation control verification
14	Development of remotely operated decontamination technology	Durability test device for cable hose
15	Full-scale test of repair and water stoppage technology for leakage points inside the Primary Containment Vessel	Heating / Feed water equipment
16	Full-scale test of repair and water stoppage technology for leakage points inside the Primary Containment Vessel	Turbid water treatment equipment
17	Full-scale test of repair and water stoppage technology for leakage points inside the Primary Containment Vessel	Work floor
18	Full-scale test of repair and water stoppage technology for leakage points inside the Primary Containment Vessel	Mock-up transfer rail
19	Full-scale test of repair and water stoppage technology for leakage points inside the Primary Containment Vessel	Full-scale mock-up
20	Development of technology for investigation inside the PCV	B1 investigation device
21	Development of technology for investigation inside the PCV	Dispersion prevention equipment for B1 investigation device
22	Development of technology for investigation inside the PCV	Subsidiary equipment for B1 investigation device
23	Development of technology for investigation inside the PCV	B1 investigation device Simulated device for mock-up
24	Development of technology for investigation inside the PCV	Shielding block removal equipment
25	Development of technology for investigation inside the PCV	Fuel debris measurement equipment Equipment for elemental test
26	Development of technology for investigation inside the PCV	Fuel debris measurement equipment Equipment for elemental test
27	Development of technology for investigation inside the PCV	A2 investigation equipment (including chambers and guide pipes)
28	Development of technology for investigation inside the PCV	Set of X-6 penetration hole boring device
29	Development of technology for investigation inside the PCV	Set of previously confirming equipment inside the penetration
30	Development of technology for investigation inside the PCV	Set of deposits removal equipment (including chambers)
31	Development of technology for investigation inside the PCV	Set of subsidiary equipment for A2 investigation
32	Development of technology for investigation inside the PCV	Set of pre-investigation equipment inside the pedestal
33	Development of technology for investigation inside the PCV	Set of simulated mock-up structure in the PCV
34	Development of technology for investigation inside the PCV	Set of elemental test device for A3 investigation
35	Development of technology for investigation inside the PCV	Set of relevant equipment for hatch opening device
36	Development of technology for investigation inside the PCV	B1 investigation device
37	Development of technology for investigation inside the PCV	Dispersion prevention equipment for B1 investigation device
38	Development of technology for investigation inside the PCV	Subsidiary equipment for B1 investigation device
39	Development of technology for investigation inside the PCV	B1 investigation device Simulated device for mock-up
40	Development of technology for investigation inside the PCV	Set of prototype underwater swimming device
41	Development of technology for investigation inside the PCV	Set of remote X-6 penetration hole boring device
42	Development of technology for detection of duel debris in the reactor	Shielding material of the measurement equipment for transmission method
43	Development of technology for detection of duel debris in the reactor	Small measurement equipment for transmission method
44	Development of technology for detection of duel debris in the reactor	Small muon tracking system for scattering method
45	Development of technology for detection of duel debris in the reactor	Muon tracking system for scattering method to be used at the Fukushima Daiichi NPS
46	Fuel debris characterization	Large capacity of thermogravimetric balance and simultaneous thermal analysis equipment
47	Fuel debris characterization	Piezoelectric crystal four-component cutting dynamometer
48	Fuel debris characterization	Elemental analysis system for SEM
49	Fuel debris characterization	Automatic hydraulic embedding machine
50	Fuel debris characterization	Inverted metallurgical microscope

Over 1 million yen

No.	Project Name	Details
51	Fuel debris characterization	Carbon coater
52	Fuel debris characterization	Vacuum displacement arc melting furnace
53	Fuel debris characterization	Fuel debris compression test device
54	Fuel debris characterization	Fuel debris sonic speed measuring device
55	Fuel debris characterization	Metallographic image analysis device
56	Fuel debris characterization	Dynamic micro hardness tester
57	Fuel debris characterization	Simultaneous thermal analysis system
58	Fuel debris characterization	Gas piping valve heater
59	Fuel debris characterization	Sample cutting machine
60	Fuel debris characterization	Sample polisher
61	Fuel debris characterization	Core collecting device
62	Fuel debris characterization	Laser diffraction/scattering particle size distribution measuring device
63	Fuel debris characterization	Dry automatic density meter
64	Fuel debris characterization	Heating furnace for thermal analyzer
65	Development of Fundamental Technologies for Retrieval of Fuel Debris and Reactor Internals	Full-scale test device
66	Development of Fundamental Technologies for Retrieval of Fuel Debris and Reactor Internals	Full-scale test equipment
67	Development of Fundamental Technologies for Retrieval of Fuel Debris and Reactor Internals	1/4-scale test device
68	Development of Fundamental Technologies for Retrieval of Fuel Debris and Reactor Internals	1/4-scale test equipment
69	Development of Fundamental Technologies for Retrieval of Fuel Debris and Reactor Internals	1/4-model test equipment
70	Development of Fundamental Technologies for Retrieval of Fuel Debris and Reactor Internals	1/4-scale reaction force holding mechanism combination test body
71	Development of Fundamental Technologies for Retrieval of Fuel Debris and Reactor Internals	Test body for water shielding of upper part of RPV
72	Development of Fundamental Technologies for Retrieval of Fuel Debris and Reactor Internals	Set of flexible structure arm
73	Development of Fundamental Technologies for Retrieval of Fuel Debris and Reactor Internals	Flexible structure arm control device
74	Development of Fundamental Technologies for Retrieval of Fuel Debris and Reactor Internals	Equipment for mock-up in PCV
75	Development of Fundamental Technologies for Retrieval of Fuel Debris and Reactor Internals	Equipment for hatch carrying-in test equipment
76	Development of Fundamental Technologies for Retrieval of Fuel Debris and Reactor Internals	1/1-scale hydraulic reaction force holding mechanism
77	Development of Fundamental Technologies for Retrieval of Fuel Debris and Reactor Internals	1/1-scale electric motor-driven reaction force holding mechanism
78	Development of Fundamental Technologies for Retrieval of Fuel Debris and Reactor Internals	Laser gouging power measurement unit
79	Development of Fundamental Technologies for Retrieval of Fuel Debris and Reactor Internals	Laser gouging head
80	Development of Fundamental Technologies for Retrieval of Fuel Debris and Reactor Internals	Equipment for access device elemental test
81	Development of Fundamental Technologies for Retrieval of Fuel Debris and Reactor Internals	Robot arm
82	Development of Fundamental Technologies for Retrieval of Fuel Debris and Reactor Internals	Access rail
83	Development of Fundamental Technologies for Retrieval of Fuel Debris and Reactor Internals	PCV welding device
84	R&D on treatment and disposal of solid water	Chamber for alpha nuclide analysis
85	R&D on treatment and disposal of solid water	Digital spectrometer
86	R&D on treatment and disposal of solid water	Efficiency calculation program for gamma-ray measurement
87	R&D on treatment and disposal of solid water	Aerosol transition observing device
88	R&D on treatment and disposal of solid water	Well type Ge detector
89	R&D on treatment and disposal of solid water	Core sampling device

Main R&D Achievements Subsidized Projects
Main R&D Achievements In-house R&D Activities
Others Data Sheet