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IRID Annual Research Report 2015

IRID International Research Institute for Nuclear Decommissioning

Message from the President

It has been almost five years 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 revised last June, the latest milestones are clarified, and consistent achievements are required for R&D technology toward decommissioning more than ever.

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 Fukushima Daiichi NPS since its start in 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 R&D technology, we committed to gathering expertise from around the world under an integrated management which encourages mutual coordination among projects, overlooking the whole R&D projects, and conducted fifteen subsidized projects and one in-house R&D project in FY 2015.

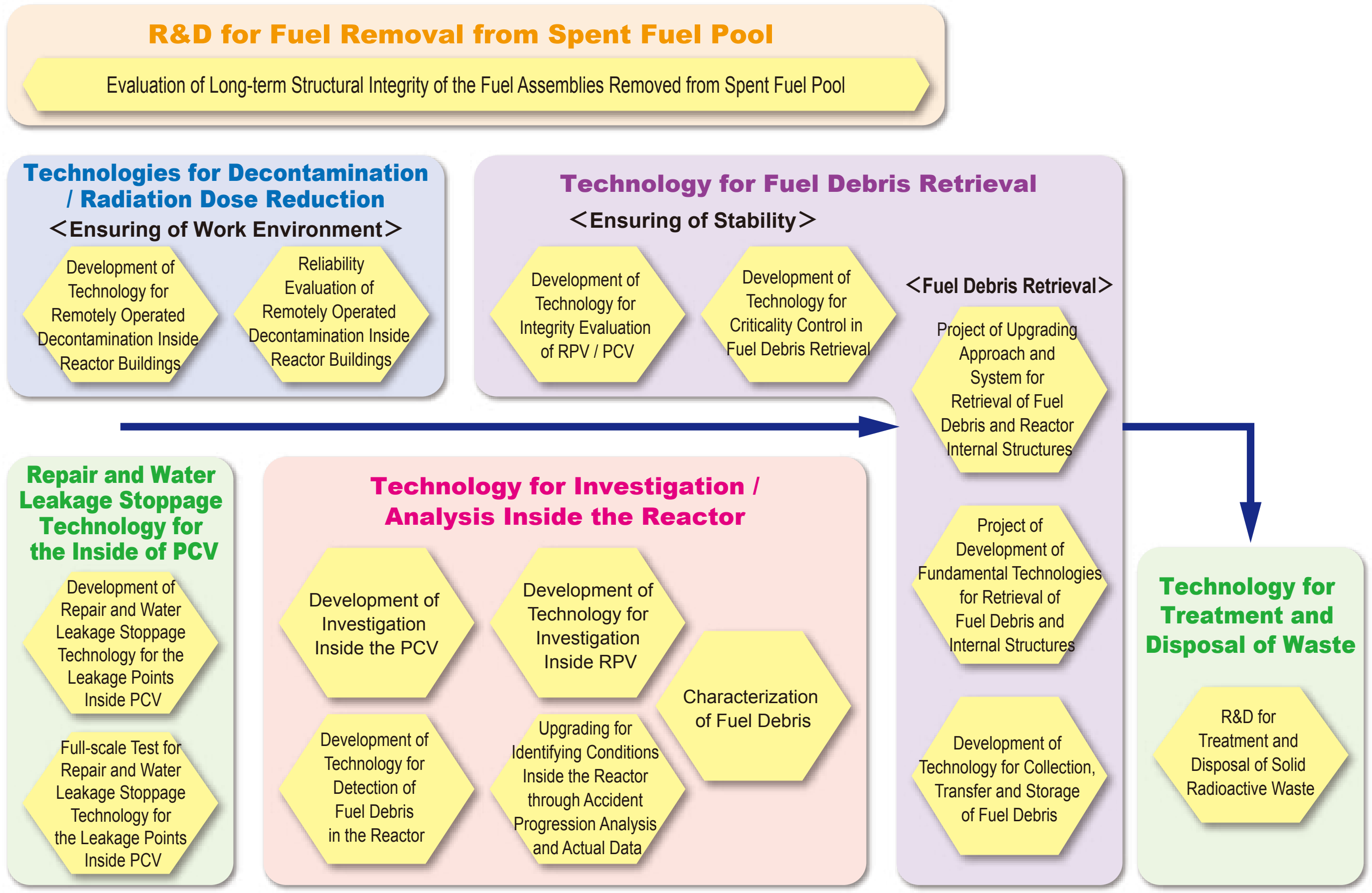
As a result, robotic technologies were developed. Remotely operated robots for the decontamination of the reactor building were tested on-site, and the feasibility of the technologies verified. Robots for the investigation inside the Primary Containment Vessel (PCV) brought us effective information inside the PCV. Furthermore, Muon tomography using permeation method was developed and revealed that fuel debris hardly exists in the central region of the Reactor Pressure Vessel (RPV) at Unit 1. We have achieved success in the development of technologies that 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 2015. 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 2016

Hirofumi Kenda
President, International Research Institute for Nuclear Decommissioning

IRID Research and Development Project, FY 2015 (Overview)



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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 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 in future, it will be necessary to accurately evaluate the effects of these conditions on fuel assembly component materials and the integrity of the fuel assemblies for long-term storage.

Aims

In order to evaluate whether fuel assemblies retrieved from the spent fuel pools at Units 1-4 at the Fukushima NPS can be stored safely or not, it will be necessary to perform corrosion tests and investigate actual fuel in an environment that simulates real storage conditions. The impact on the fuel integrity during dry storage will be evaluated by a simulated test.

Main Achievements and Approaches

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

a- Evaluation on the deposits on the surface of the fuel assemblies

In order to analyze white deposits which had been observed on the surface of the spent fuel stored in the common pool at Units 1-4 at the Fukushima Daiichi NPS, preparations were made for transport of fuel components and a post irradiation test in the test facility.

For a smooth transportation of fuel components, loading procedures for transport casks and collecting lock nut were examined. A mockup for loading lock nut into transport casks was conducted.

In the preparation for testing in the post irradiation test facility, a test guideline for analysis of white deposits was created as well as electrochemical test using lock nut was studied. It is confirmed that tests are enabled to be conducted in the post irradiation test facility.

b- Evaluation of Integrity of fuel in dry storage conditions

When fuel assemblies retrieved from the spent fuel pool at Units 1-4 at the Fukushima Daiichi NPS are stored in dry condition, it is necessary to evaluate the impact on the integrity of fuel assemblies caused by scratches on fuel cladding due to falling rubbles or seawater due to seawater injection.

Therefore, a low-ductility test (Hydride deposition behavior evaluation test) and a high strain test (creep test) were conducted. Test conditions in 2015 confirmed that there was a small impact on integrity of fuel assemblies caused by scratches on fuel cladding due to rubbles or seawater by seawater injection (Fig. 1 and 2). We will continue to study the impact on the integrity of fuel assemblies caused by specific factors under dry storage conditions at Units 1-4 at the Fukushima Daiichi NPS.

2 Basic tests for long-term structural integrity

In order to evaluate quantitatively the seawater components transfer behavior in a crevice structure of fuel components (upper end plug), tests using tracer were conducted.

The test result proved that concentration of chloride ion decreased in accordance with fluid concentration by improving the quality of the solution and was not enriched in the crevice structure, although chloride irons were taken into the upper end plug crevice structure by seawater injection (Fig. 3 and 4).

Future Developments

Efficient data for the study of corrosion occurrence during storage in the common pool will be acquired through analysis and identification for white deposits on the surface of fuel assemblies.

Furthermore, we will continue to study the effect on the integrity of fuel assemblies in dry storage caused by specific factors at Units 1-4 at the Fukushima Daiichi NPS.

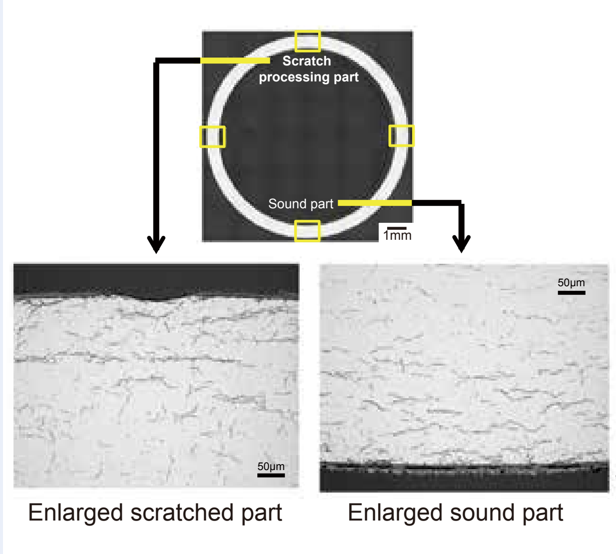


Fig. 1: Example of a verification test result for hydride precipitation behavior(Irradiated 9x9 fuel cladding, 300℃, cooling velocity 0.3℃/h, stress in a circumferential direction 70MPa, with scratches)

The direction of hydride precipitation behavior in a scratched area is an almost circumferential direction as well as sound parts. There is no effect on hydride precipitation behavior with scratches.

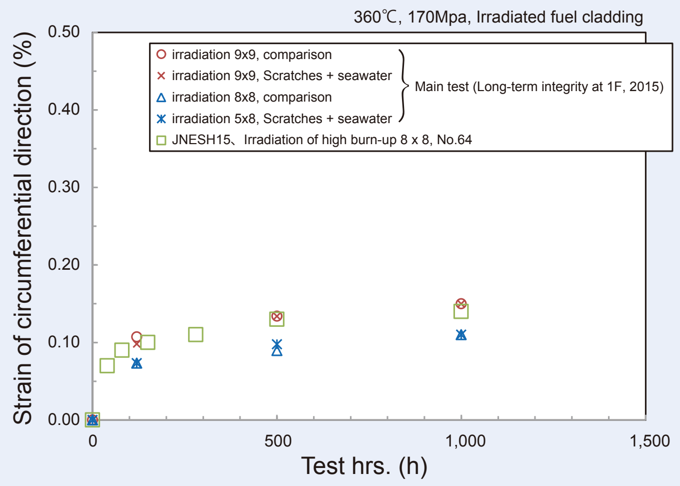


Fig. 2: Example of a study result (attached scratches + seawater) that can affect the creep speed

There is no clear difference due to existence (of scratches + seawater) by the 1,000-hr. long test results. Behavior continues to be verified.

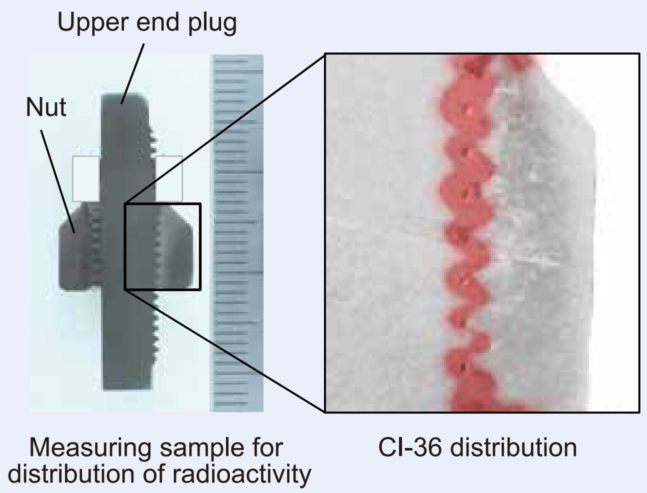


Fig. 3: Distribution of Cl-36 in upper end plug crevice structure (80℃, 2x dilution of artificial seawater after 50 hrs. of immersion)

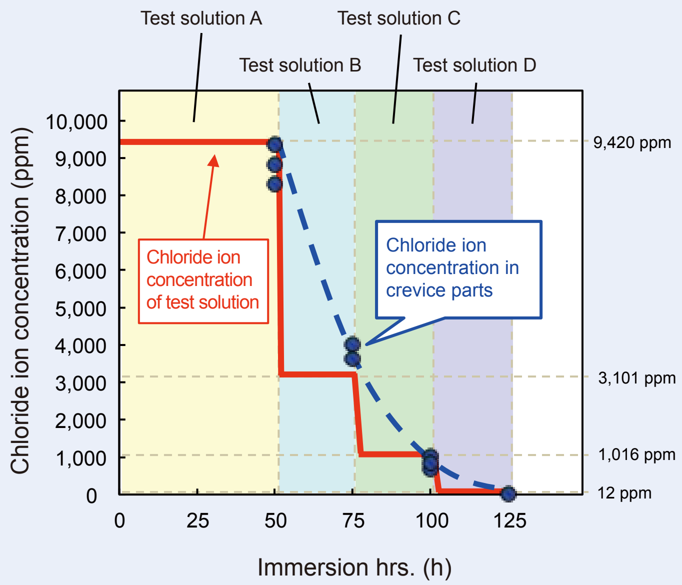


Fig. 4: Evaluation result of chloride ion concentration in the upper end plug crevice structure

According to the results of Fig. 3 and 4, it is confirmed that seawater components transfer to the crevice structure of upper end plug by immersing in a 2x dilution of saltwater; however, concentration decreased afterwards as water quality improves.

Key Challenge 2 – R&D for Preparation of Fuel Debris Retrieval

Development of Technology for Remotely Operated Decontamination Inside Reactor Buildings

Background

In order to retrieve fuel debris from the Reactor Pressure Vessel (RPV) and the Primary Containment Vessel (PCV) at Units 1-3 at the Fukushima Daiichi NPS, various tasks are planned inside the reactor buildings. For smooth execution, it is necessary to improve the working environment and reduce overall radiation levels in combination with decontamination, shielding and removal of radioactive sources.

Aims

Based on R&D achievements up to FY 2014, decontamination equipment for high places will be improved and verified. Decontamination equipment of upper floors will be manufactured and verified, and conceptual study will be conducted for decontamination of basement floors.

According to the above steps, remotely controlled decontamination techniques, one of the important technologies required for the smooth implementation of surveys, repairs and water leak blockage inside the reactor buildings will be established.

Main Achievements and Approaches

Remotely operated decontamination equipment is mainly used in lower parts of the first floor, upper parts of the first floor, upper floors and basement areas. By FY 2016, development of decontamination equipment for lower parts of the first floor has completed. Decontamination equipment for high places was manufactured and verified by tests and that for upper floors was designed and manufactured. Based on the aforementioned results, the following tasks were executed in FY 2015.

1 Development of decontamination equipment for high places

a- Improvements of decontamination equipment

Based on the results of verification tests in FY 2014, various improvements were made. High-pressure water jet decontamination equipment was improved for visibility and easy-maintenance; dry ice blast decontamination equipment was advanced for operability during travelling or decontamination work and suction/blast decontamination equipment was improved for visibility and collecting operability in case of emergency (Fig. 1).

b- Verification test for decontamination equipment

In a mock-up test facility in which upper parts of the first floor in the reactor building were simulated, verification re-tests were conducted to mainly improve features of high-pressure water jet decontamination equipment, dry ice blast decontamination equipment and suction/blast decontamination equipment such as decontamination performance, remote travelling, operability and safety function. The results showed that it would be applicable in actual use.

2 Development of decontamination equipment for upper floors

a- Manufacturing of decontamination equipment

Following FY 2014, decontamination equipment for upper floors was manufactured. Decontamination equipment is designed to access upper floors of the reactor buildings (the second and the third floors) by using an elevating work cart for decontamination of the floor and wall surface (up to approx. 2 meters). Equipment is comprised of a work cart, a transport cart, a support cart and a relay cart. Mounted in each cart are the decontamination units: high-pressure water jet decontamination unit, suction/blast decontamination unit and dry ice blast decontamination unit (Fig. 2, 3, 4).

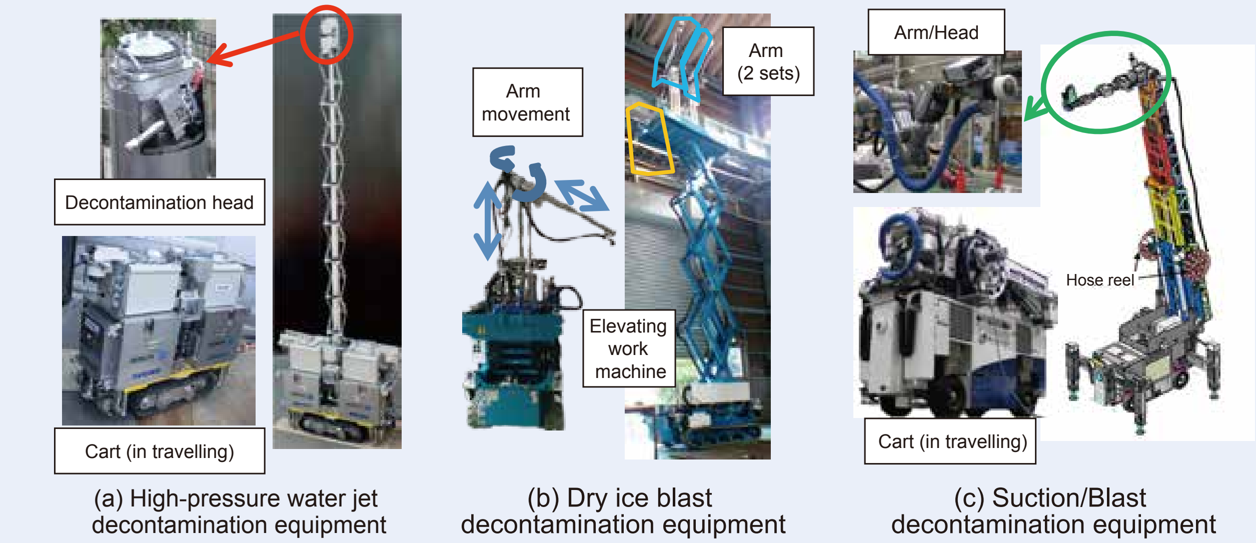
b- Verification test for decontamination equipment

A mock-up test facility that simulates the second floor of the reactor building and elevating test facilities were created, and verification tests were conducted. Various safety tests conducted for decontamination performance of each decontamination system such as remote travelling performance, operability, accessibility to upper floors, and safety function on collection, etc., during emergency. The results showed that it would be applicable in actual use.

3 Conceptual study on decontamination equipment for basement areas

A conceptual study was performed on stagnant groundwater that had been considered as a radiation source on the basement floors of the turbine building and sludge accumulated in the bottom as well as treatment for highly contaminated water accumulated in the main condenser (primary stagnant water). The study showed that existing technologies would be applicable.

In order to confirm the feasibility of conceptual study on the basement of the reactor building, the necessary information on the contamination contributing to radiation dose was organized.



(a) High-pressure water jet decontamination equipment
(b) Dry ice blast decontamination equipment
(c) Suction/Blast decontamination equipment

Fig. 1: Decontamination equipment for high places

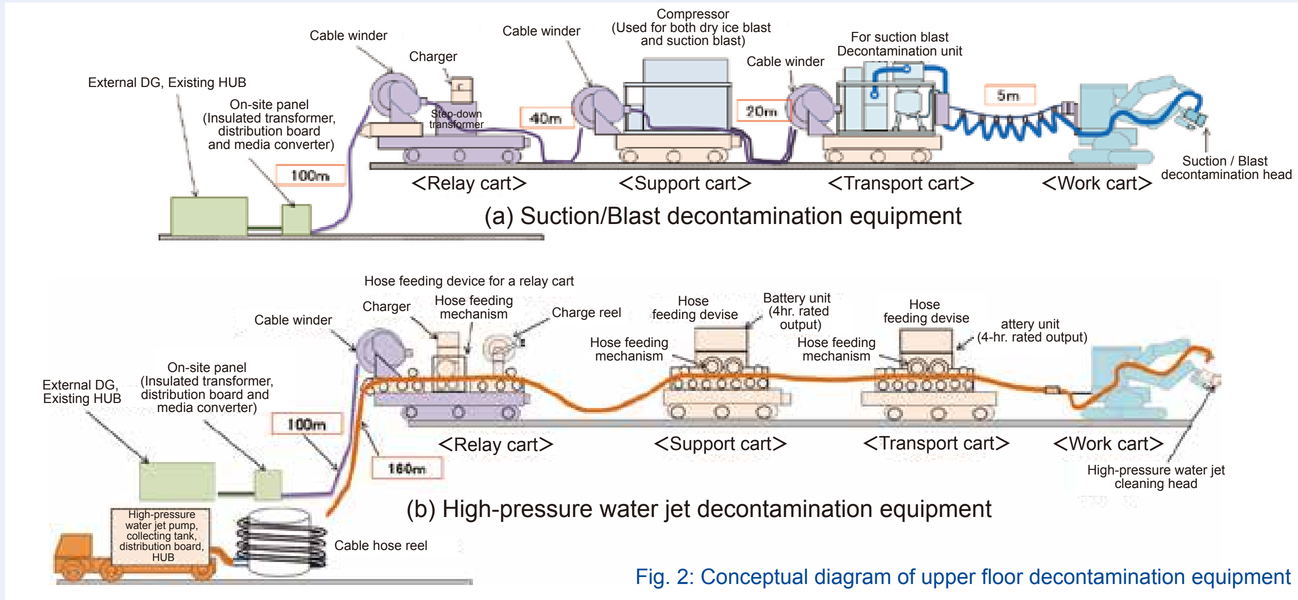


Fig. 2: Conceptual diagram of upper floor decontamination equipment

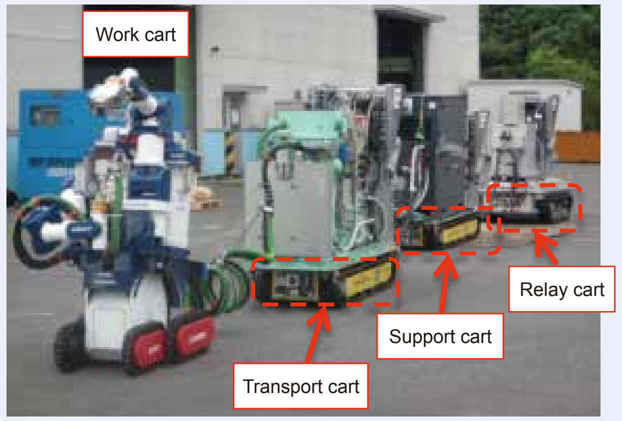


Fig. 3: Upper floor decontamination equipment (Showing mounted suction/blast equipment)

Fig. 4: Upper floor decontamination equipment (Elevating)

Future Developments

Development of the decontamination equipment for high places and upper floors has been completed in FY 2015. Dry ice blast equipment for high places was applied in actual use in the second semester of FY 2015. On the other hand, other decontamination equipment will be utilized according to on-site requirements and timing for start of construction.

In regards to the basement floor of the turbine building, decontamination will be conducted accordingly as water level decreases. The results from the conceptual study and on-site investigation will be used as reference. These results will be utilized since on-site investigation requires clarifying needs for the basement of the reactor building.

Key Challenge 2 – R&D for Preparation of Fuel Debris Retrieval

Development of Repair and Water Leakage Stoppage 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, the PCV is planned to be submerged. Therefore, it is necessary to ensure that water leakage is prevented from the PCV.

Aims

Toward the realization of the submersion method for fuel debris retrieval, this project is intended to establish technology for repair and water leakage stoppage from leakage points in the PCV.

Main Achievements and Approaches

1 Technology for strengthening the suppression chamber (S/C) support columns

On the technology for strengthening the suppression chamber (S/C) support columns (Fig. 1), tests in FY 2013 proved that the intensity of a reinforcement material decreased due to a long distance travel after placing. In order to solve this problem, further test confirmed that the improved material is enabled to maintain sufficient strength. Furthermore, the performance target was secured for long-distance fluidity, evenly balanced behavior (Fig. 2) and effects on surmounting obstacles.

2 Water stoppage technology involving injection of sealing material into the vent pipe

a- Water stoppage technology inside of the vent pipe facility

A full-scale deployment test for an inflatable sealing bag (a cloth bag), which will be installed at the tip of vent pipe, was conducted. Challenges were identified. In addition to dealing with the challenges, an elemental test for sub-inflatable bag that contains materials to help water stoppage (a clogging prevention material for a gap between a vent pipe and an inflatable sealing bag) will be conducted. Candidate materials will be selected for actual use.

b- Water stoppage technology involving filling inside of the suppression chamber (S/C)

Material compounding of water stoppage materials were determined. Downcomer water stoppage test, quencher water stoppage test and strainer water stoppage test were conducted. Additionally, basic performance for water stoppage by using the materials was confirmed through tests: a long distance fluidity test at the merging point of the two-way flows and a vacuum break valve water stoppage test (Fig. 3).

c- Water stoppage technology involving injection of sealing material into the vacuum break line

Considering the vacuum break line that is not accessible from right above the first floor of the reactor building at Unit 1, a full-scale test for installation of a flexible guide pipe and a water stoppage plug that can be accessed from oblique direction was conducted. In the result, workability and feasibility were confirmed through identifying pressure resistance, installation of water stoppage plug, opening a hole of the vacuum break line and setting a guide pipe (Fig. 4).

3 Water stoppage technology for leakage from the seal section and of vent pipe bellow

As equipment hatch is not accessible due to high radiation dose, conceptual study of remotely operated welding method has commenced (Fig. 5).

4 Boundary construction technology for PCV connector pipes

In order to isolate connector pipes from the PCV, the study of water stoppage proposals is ongoing. Atmospheric Control System (AC System), Radioactive Waste System (RW System) and Reactor Component Cooling Water System (RCW) were selected. Water stoppage tests were conducted after clarifying the required performance of water stoppage materials.

Future Developments

In order to establish water stoppage technology for each part to be repaired (water stoppage), various tests continue to be conducted for practical use, and challenges will be reviewed, analyzed and evaluated. In addition, considering the applicability in actual use, the required performance will be reflected in the equipment and the study of long-term water stoppage maintenance will proceed further.

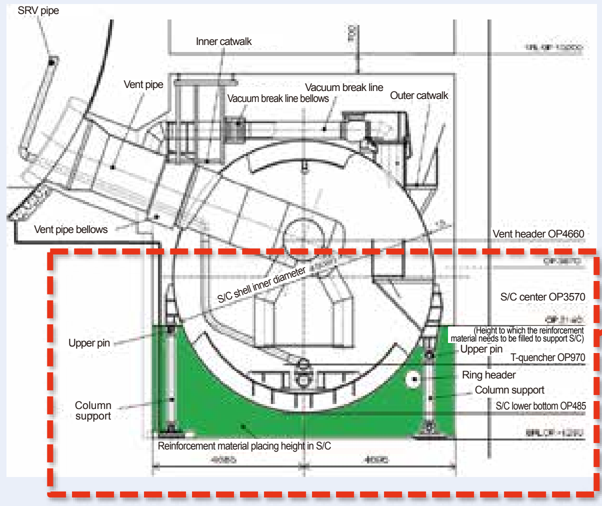


Fig. 1: Reinforcement material placing height in S/C (green area)

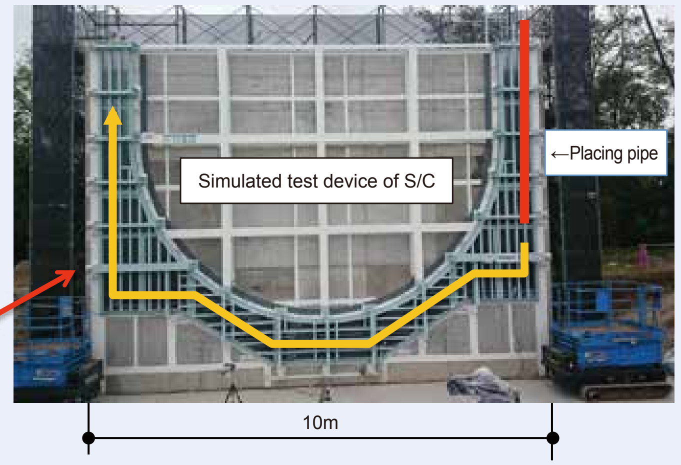


Fig. 2: Verification test for evenly balanced flow behavior of strengthening S/C support column (1/1 scale: simulated Unit 2,3)

It was confirmed whether reinforcement materials poured into a placing pipe flowed evenly toward the opposite side through under S/C.

Fig. 3: Status of down-comer long-distance fluidity test

Water stoppage materials were placed from both ends of a 24-m water tank. Capacity for repletion/water leak blockage in the interior of the down-comer around the center of the merging area from both ends was confirmed.

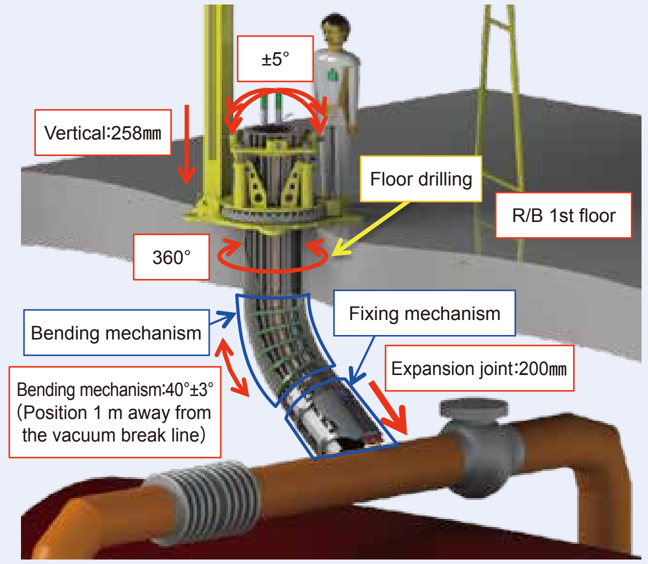
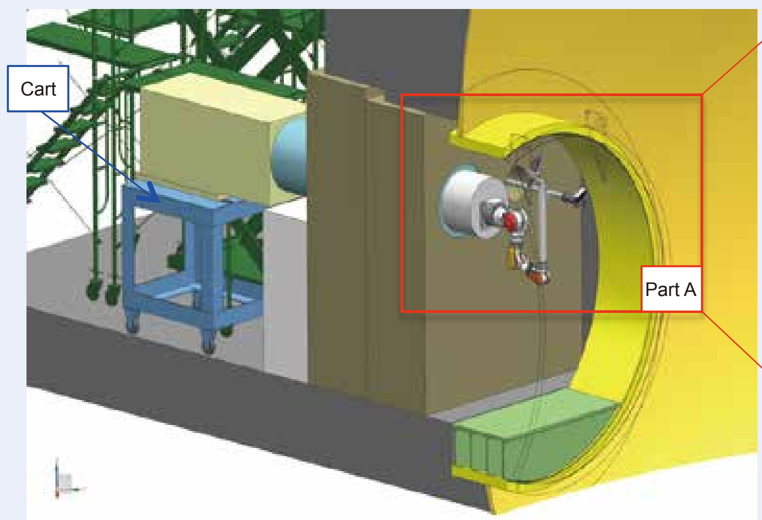
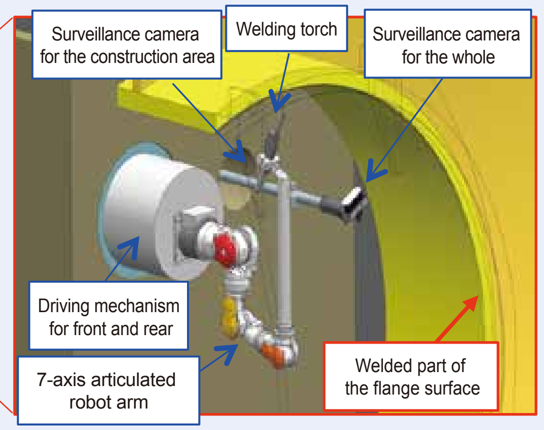


Fig. 4: Flexible guide pipe overview

A guide pipe was developed to be accessed from oblique position as it is not able to drill holes right above the vacuum break line due to an obstruction at the drilling position of the 1st floor.





Water leakage stoppage method for equipment hatch has been studied through methods involving injection of cement-based materials; however, it proved difficult to completely stop water from leaking. Therefore, a remote welding water stoppage method is now being considered.

Fig. 5: Water leakage stoppage technology for the sealing parts (water stoppage for the equipment hatch)

Key Challenge 2 – R&D for Preparation of Fuel Debris Retrieval

Full-scale Test for Repair and Water Leakage Stoppage Technology for Leakage Points Inside the Primary Containment Vessel

Background

Toward the decommissioning of the Fukushima Daiichi NPS, it is essential that repair and water leakage stoppage technology for leakage points inside the Primary Containment Vessel (PCV) is established for fuel debris retrieval by full submersion (Submersion method).

Aims

This project is intended to perform a full-scale test for the verification of developed technologies and operational training toward the application of repair and water leakage stoppage technology on-site (construction methods, remotely controlled equipment, etc.).

Testing is planned at the Naraha Remote Technology Development Center as a joint proposal with the Japan Atomic Energy Agency (JAEA).

Main Achievements and Approaches

1 Full-scale mock-up test of equipment to repair and stop leakage in the lower part of PCV

In order to confirm the operability of reinforcement equipment for suppression chamber (S/C) support reinforcement developed in the project entitled “Development of Repair and Water Leakage Stoppage Technology for Leakage Points Inside the PCV,” a full-scale mock-up test using water has commenced (Photo 1).

In addition, necessary tests and confirmation items were identified for the study of actual construction and were thoroughly examined for further full-scale test.

2 Design and production of full-scale mock-up

Continuing from FY 2014, a full-scale mock-up facility simulating the lower part of the PCV at the Fukushima Daiichi NPS Unit 2 was designed, manufactured and installed. This year saw the completion of installation for a full-scale mockup facility in the Naraha Remote Technology Development Center (Photo 2).

The main specifications of the full-scale mockup are as follows:

- A full-scale model is a one-eighth sector simulating the lower parts of the PCV (the suppression chamber [S/C], the vent pipes [excluding the bellows], vent head, downcomer and surface of torus room wall) at Unit 2 at the Fukushima Daiichi NPS.

3 Study/design/production/installation of required equipment including plumbing and turbid water treatment systems

In a full-scale testing, we simulated actual conditions at Units 1-3 at the Fukushima Daiichi NPS as much as possible such as size and temperature. Continuing from FY 2014, the following equipment was studied, designed, manufactured and installed. Installation of necessary equipment has completed as scheduled in the Naraha Remote Technology Development Center.

[Heating/Feed water equipment]

Heated water, which simulated actual stagnant water at the Fukushima Daiichi NPS, is planned to be used for a full-scale mock-up test. Installed equipment is capable to heat water and supply heated water at the required volume.

[Turbid water treatment equipment]

Water stoppage materials (grouting material [including cement]) is used for stopping water at the lower part of the PCV, therefore, water containing cement generates after the test. Turbid water treatment equipment was installed to drain water outside the facility.

[Work floor]

The height of the first floor of the basement at Unit 1 differs from the height at Units 2 and 3 at the Fukushima Daiichi NPS. As the water stoppage equipment is expected to be installed from the first floor in actual operation, work floors were installed to change the floor level for each unit when a full-scale test is conducted.

[Transfer rail for test device]

Since the weight of the mock-up is very heavy (approx. 5,400t), it is difficult to move once it is assembled. Therefore, a transfer rail that enables to move the mock-up was installed.

Photo 1: Full-scale testing

Upper left: Placement device, Upper right: Surveillance camera image of the placement device tip, Lower left: Lowering nozzle of the placement device

Photo 2: Assemblies of full-scale test device

Lower left: Assemblies for S/C shell, Upper right: Assemblies for vent pipes, Lower right: Simulated torus room wall (panoramic view)

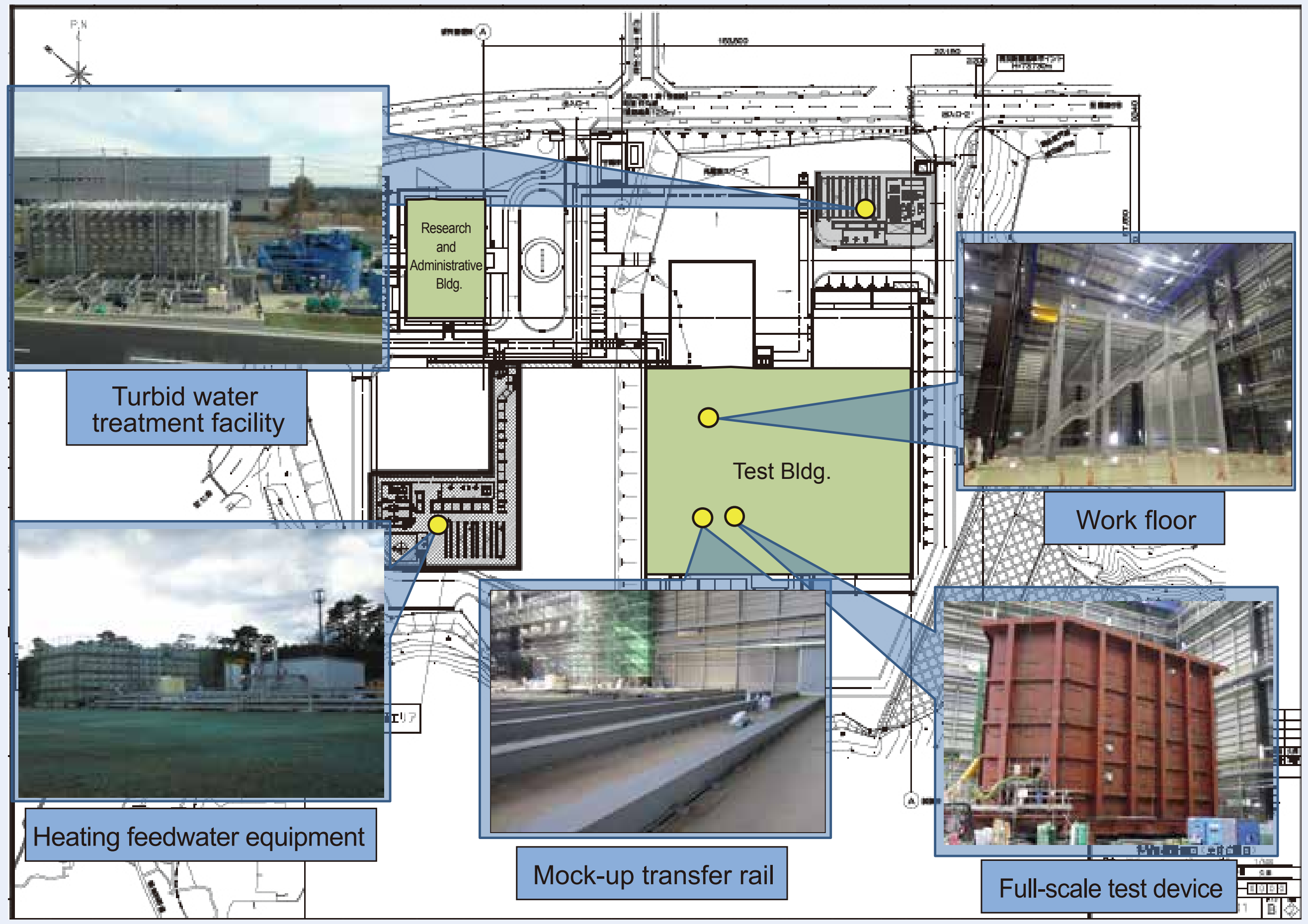


Fig.1: Preparation of a test device facility

Future Developments

A full-scale mock-up test will be conducted accordingly to utilize the achievements of the development activities for repair and water stoppage technologies.

Considering the actual construction, a guideline will be created and procedures will be verified. In addition, performance validation will be performed for repair and water leakage stoppage materials through observing the conditions of filling for the materials. These results will be the criterion to determine the methods of fuel debris retrieval.

Key Challenge 2 – R&D for Preparation of Fuel Debris Retrieval

Development of Technology for Investigation Inside the Primary Containment Vessel

Background

It is considered that reactor cores at Units 1-3 have melted, and fuel has partially fallen with reactor internals into the Reactor Pressure Vessel (RPV) and Primary Containment Vessel (PCV). It is assumed that fuel debris has fallen from the RPV bottom into the pedestal, which supports the RPV, then spread out of the opening of the pedestal bottom into outside of the pedestal and remained in the PCV bottom. However, actual conditions have not been identified.

Aims

With regard to the conditions of the PCV internal, information such as remotely visual image, radiation dose and temperature in the vicinity of the penetration were obtained through access to the PCV internals from X-100B Penetration at Unit 1 and X-53 at Unit 2. Based on the results, it is confirmed that the visibility is limited due to the presence of stream and stagnant water in the dark, in addition to the harsh, highly radioactive and extremely humid environment. Moreover, there is the possibility of unforeseen obstructions generating from the accident. In order to overcome these challenges, this project aims to develop technology for the investigation inside the PCV.

Main Achievements and Approaches

1 Formulation of a plan for investigation inside the PCV and development of investigation equipment

Prior to formation of the plan, requirements were organized and analyzed in cooperation with other related projects. Moreover, we followed up the discussion on the requirements and studied further investigation plan based on the results of the investigation on the grating zone (B1 investigation).

2 Development of investigation devices, etc.

a- Technology for accessing inside the pedestal

Equipment for the investigation on the platform inside the pedestal (A2 investigation) was improved as reflected in test results. Preparation has been completed for on-site verification at Unit 2 (Photo 1).

Moreover, for further investigation of the inside pedestal (A3 investigation), an element test on research techniques was conducted in parallel with test production/testing for equipment to free a hatch of X-6 penetration.

b- Technology for removal of the shielding block

Remotely operated equipment for removal of the shielding block placed in front of the X-6 penetration at Unit 2 (equipment for removing the X-6 shielding block) has been developed. An on-site verification test was conducted from June to July 2015. Results showed that 128 out of 135 shielding blocks and two iron plates in the back of three plates were removed. It was confirmed that remaining blocks and a plate were strongly adhered. Installation structure was different from what had been expected; therefore, an end effector that would remove the adhesion was additionally developed (Photo 2).

c- Technology for accessing the the outside of the pedestal

In order to obtain information e.g. visual image, radiation dose and temperature access through investigation device working on the grating of the first floor inside the PCV at Unit 1 (B1 investigation), an investigation was conducted in April 2015. Through this investigation, it showed that there was no extensive damage on the existing facilities inside the PCV. Data on the radiation dose and temperature in the 3/4 range of the grating on the 1st floor were obtained (Photo 3).

In addition, conceptual study, elemental and manufacturing tests were conducted. Investigation methods were established for further investigation (investigation on the basement floor [B2 investigation]) following B1 investigation (Fig. 1).

d- Technology for measurement of fuel debris

Fuel debris measurement equipment, which identifies the position and distribution of melted materials that are suspected fuel debris, under extremely harsh environments – darkness, high level of radiation, raindrops and fog – was designed and prototyped. Taking into account the severe environment inside the PCV, optical cutting method was applied. Development is ongoing towards verification tests after FY 2016.

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Photo 1: A2 investigation device (inside the pedestal)

Photo 3: B1 investigation device (outside the pedestal)

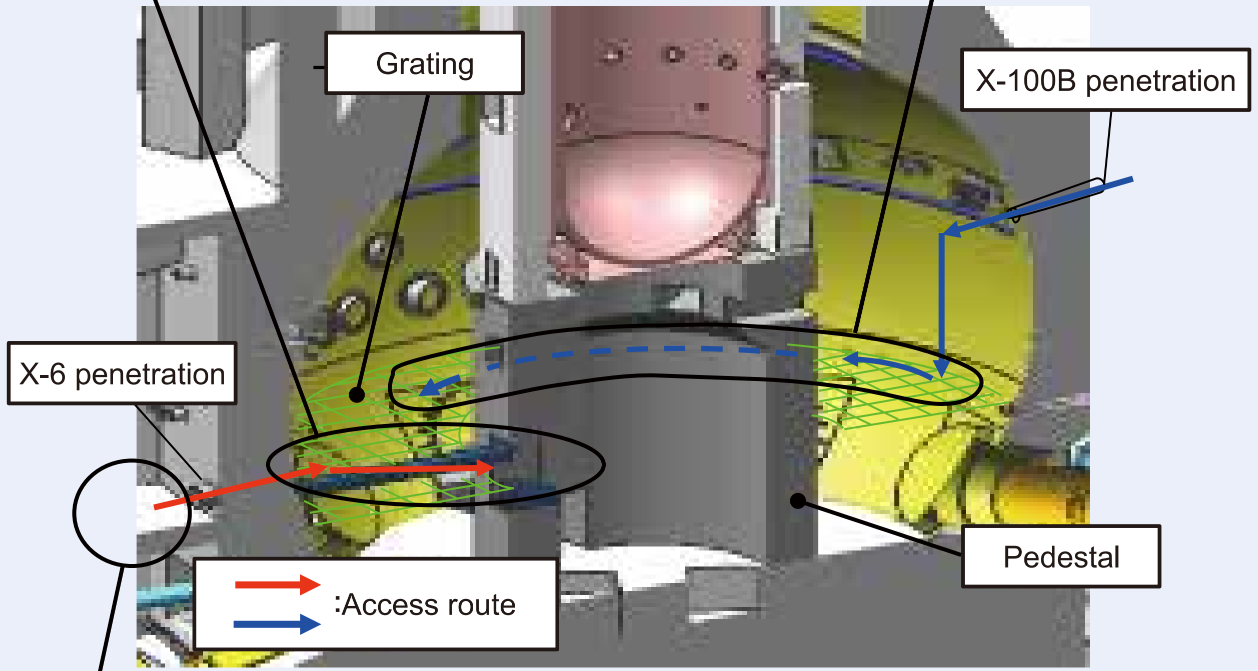


Photo 2: Shielding block removal equipment (in front of X-6 penetration in Unit 2)

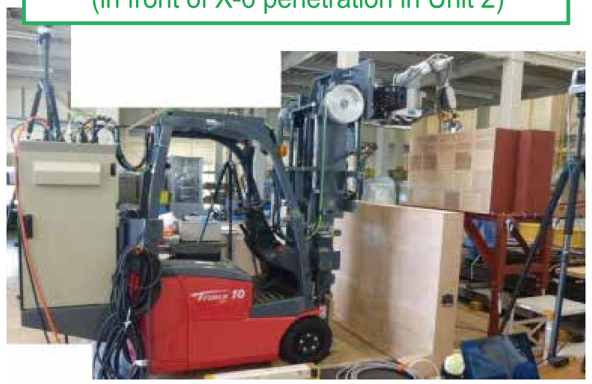
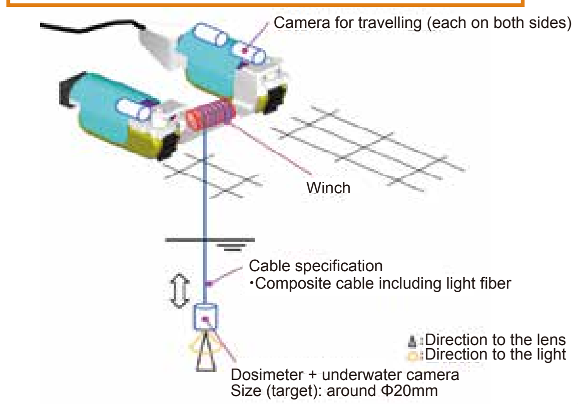


Fig. 1: B2 investigation (Image)* (Basement outside of Unit 1 pedestal)



Devices currently being developed and their applications

*Fig. 1 is a current idea based on the concept examined by IRID. This image might be changed due to further improvements depending on the future development and achievements of element tests.

Future Developments

Development continues toward the verification test for B2 investigation at Unit 1 and A2 investigation at Unit 2 which are scheduled to be conducted in FY 2016. In addition, development of equipment also continues for further investigation both into the interior and exterior of the pedestal inside the PCVs.

Key Challenge 2 – R&D for Preparation of Fuel Debris Retrieval

Development of Technology for Investigation Inside the Reactor Pressure Vessel

Background

In order to develop equipment for retrieving fuel debris and core internals, information about the location, shape and conditions must be understood in advance. However, it is currently a difficult situation to acquire directly information about the inside of the Reactor Pressure Vessel (RPV) due to extremely high radiation level, in addition to the complicated reactor internal structure.

Aims

Based on the compiled needs from related projects and the construction site, the development plan and investigation plan established in FY 2014 will be updated. In addition, the feasibility of the system and technologies will be confirmed through elemental test of the system that maintains the boundary functions, which is a primary technology for the investigation of the RPV internals, access technology for the reactor core and sampling technology for fuel debris.

Main Achievements and Approaches

❶ Formulation and updating of plan for investigation/development

The information needs that had to be obtained through the investigation of the RPV internals were collected from related projects on fuel debris retrieval and on-site parties. Moreover, information, particularly high necessity items like radiation dose level inside of the RPV, situation of molten fuel remains and distribution and composition of fuel debris were analyzed, evaluated and extracted. In order to obtain the information, investigation items such as visual checking in the reactor cores and sampling methods for fuel debris were updated and clarified. In addition, based on the evaluation results for the feasibility of the technologies as shown below (❷, ❸), the development plan was updated after FY 2016.

❷ Elemental testing and feasibility assessment for technology of investigation by drilling holes at the top of the reactor

a- Conceptual study on the system for maintaining boundary function

The conceptual study was conducted for an alternative boundary facility (guide pipe) that will be placed on the head of the PCV (Fig. 1) as a prevention system against the scattering of radioactive material caused by the drilling of holes for the building of an investigation route. In addition, an elemental test was conducted for sealing properties between the alternative boundary and the connecting part of the PCV head. Through the test, the feasibility of boundary function was verified.

b- Conceptual study on the technology for accessing the reactor core

In order to build an investigation route to the reactor core, the conceptual study on the technology of drilling holes, remotely controlled in the complex reactor internals (steam dryer, steam separator, upper grid pate and etc.), was conducted. Taking into account the difficulty of supporting the reaction force via remote operating, contactless cutting methods such as razor cutting/plasma cutting/water jet cutting (abrasive cutting) were considered before fundamental test was conducted. Elemental test was also conducted for drilling holes using a partially simulated test device of the reactor internals. Also, the feasibility of the investigation route was verified (Fig. 2).

❸ Investigation of the technology for fuel debris sampling and feasibility assessment

As a result of the fuel debris sampling required from related projects, composition of fuel debris and mechanical properties (hardness) were mainly chosen. In order to determine design conditions of the sampling system, sampling items such as measurement, number and position were organized.

In addition, the entire flow was studied from the sampling of fuel debris to its analysis (Fig. 3). Conceptual design (including elemental test) was performed for cutting and collecting devices as the main component device, access device (entry from the side of the PCV), measuring device for uranium/plutonium, sampling cell, etc. The feasibility of the sampling technology, which is under development, was verified. In order to analyze fuel debris, necessary evaluation items were clarified when transporting the sampling of fuel debris outside the Fukushima Daiichi NPS.

Future Developments

Based on results from the needs investigation, conceptual study and elemental test conducted in FY 2015, development of technology for investigation inside the RPV continues. In regards to the technology of investigation for drilling holes at the upper part of the reactor, trial manufacturing of the investigation system and securing the boundary system and mock-up test also will proceed. Regarding the technology of fuel debris sampling, test production of cutting and collecting devices and performance test will be conducted. Upgrading of access device and cell will be studied as well.

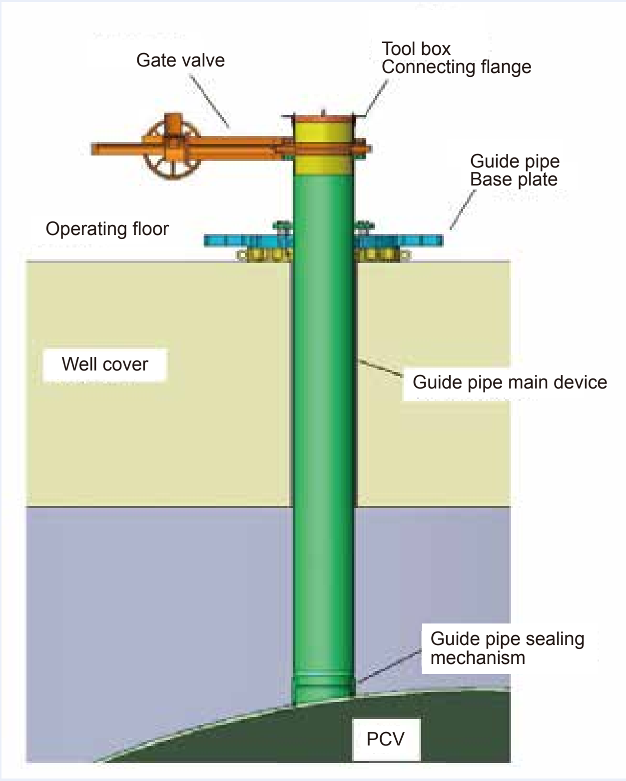


Fig. 1: Image of boundary function maintaining device

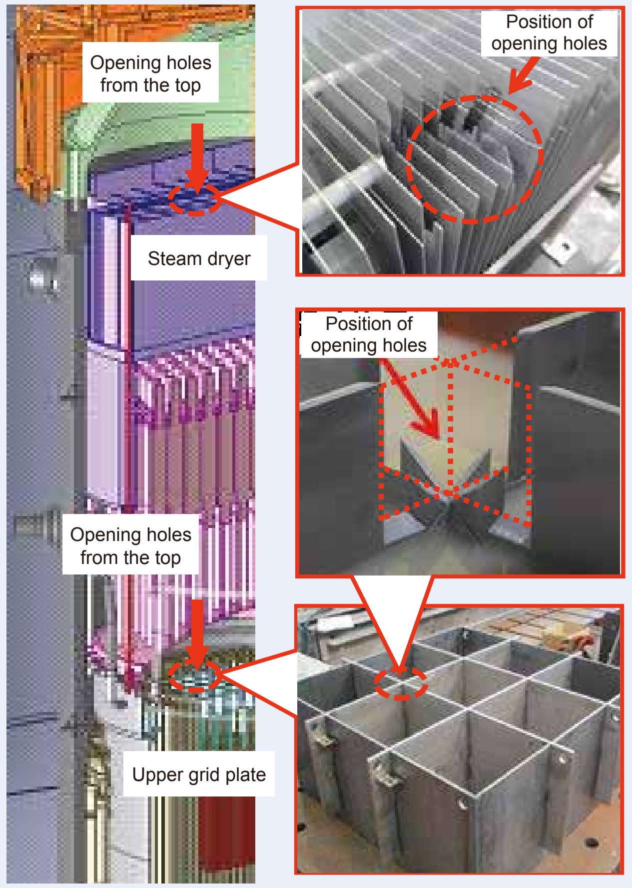


Fig. 2: Elemental analysis for opening holes using partial simulated test device of the reactor internals

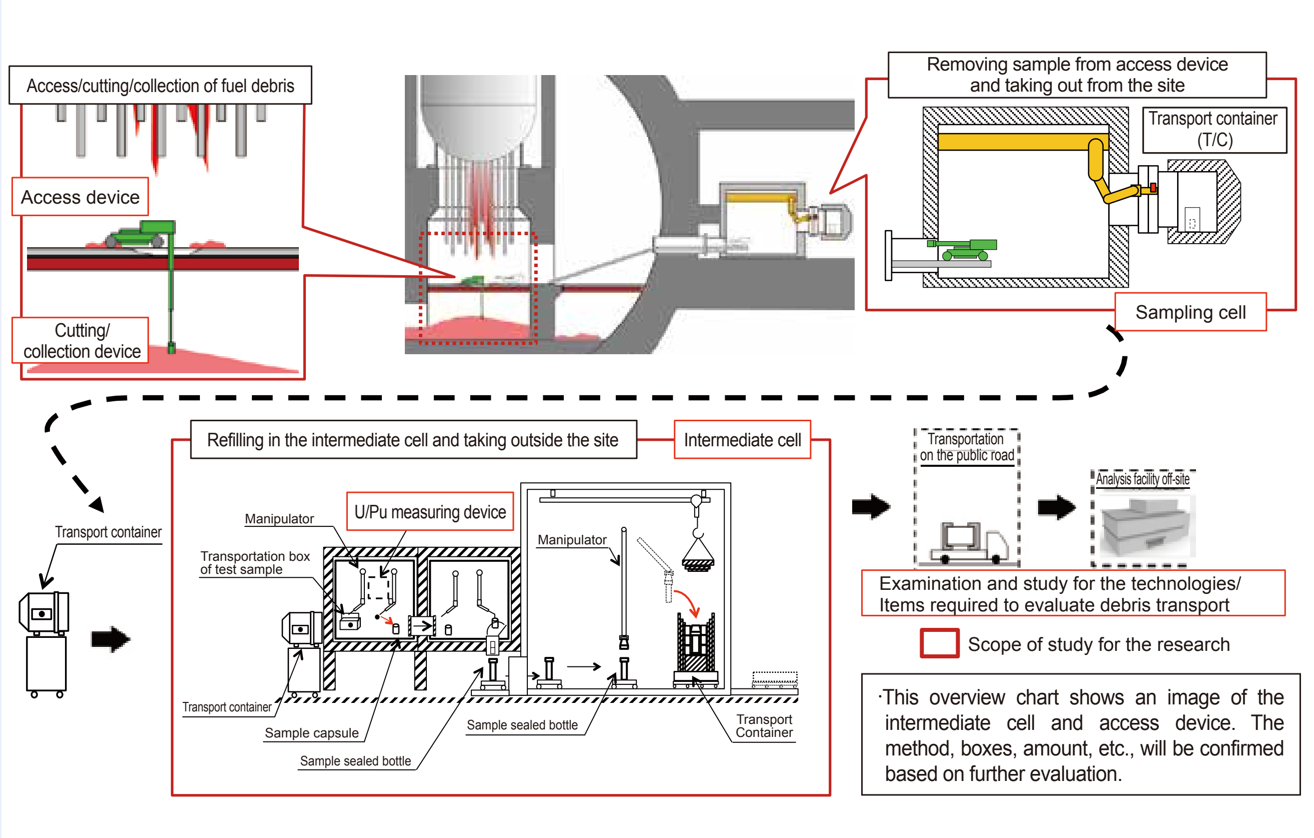


Fig. 3: Flow chart of the whole process from fuel debris sampling to analysis (Image)

Key Challenge 2 – R&D for Preparation of Fuel Debris Retrieval

Upgrading for Identifying Conditions Inside the Reactor through Accident Progression Analysis and Actual Data

Background

It is essential to identify conditions inside reactors for planning the method for fuel debris retrieval and developing safety measures. However, it is difficult to directly investigate the inside of Units 1-3 at the Fukushima Daiichi NPS due to extremely high radiation levels.

Aims

This project is intended to upgrade accident progression analysis codes (MAAP, SAMPSON) and promote decommissioning by providing necessary information to identify conditions inside the reactor, conducting comprehensive analysis and evaluation utilizing accident progression analysis and actual data. This is a joint project with the Institute of Applied Energy.

Main Achievements and Approaches

1 Improvement and updating of accident progression analysis

In accordance with the upgraded specifications established in FY 2014, improvement was made on a physical phenomenon model for fission product transition and fuel debris behavior of accident progression analysis code (MAAP and SAMPSON) (Fig. 1). Also, technology for accident progression analysis was enhanced.

- MAAP (Modular Accident Analysis Program): Comprehensive analysis code that enables the consistent evaluation of thermal-hydraulic phenomena/fission product behavior in the Reactor Pressure Vessel (RPV) and the Primary Containment Vessel (PCV) at the time of severe nuclear power plant accidents.
- SAMPSON (Severe Accident analysis code with Mechanistic, Parallelized Simulations Oriented toward Nuclear fields): A model, based on a multidimensional numerical formula presenting a physical phenomenon and theory, that is suited to analyze the dispersion and properties of fuel debris.

2 Assumption and evaluation in identifying inside the reactor utilizing accident progressing analysis

Accident progressing analysis and sensitivity analysis were conducted through improved analysis code. Existing position, quantity and composition of the fuel debris and fission product were evaluated (Fig. 2). In addition, analysis code was developed to evaluate the Molten Core-Concrete Interaction (MCCI) and fuel debris in detail, and fuel debris behavior that had fallen in the reactor estimated (Fig. 3). Furthermore, tests were conducted to verify the conditions of the damaged in-core instrumentation by using real-size penetrating pipes and actual corium in Korea Atomic Energy Research Institute (KAERI) (Fig. 4).

Additionally, the three-year Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station (BSAF) Phase-2 Project has been executed. Its results have been utilized for evaluating conditions inside the reactor.

3 Development of comprehensive analysis and assessment database

The comprehensive analysis and assessment database for the interior of the reactor was developed to collect and organize data information obtained from other R&D projects, on-site operation and related information of accident progressing analysis. Results were utilized for comprehensive analysis and assessment.

4 Comprehensive analysis/assessment based on results of actual data and other projects

Comprehensive analysis and assessment (see below) were performed utilizing the data information obtained from the results of accident progressing analysis, actual investigation. Existing location, quantity and composition of the fuel debris and fission product were evaluated. As a result, it is estimated that most of the fuel debris had fallen from the Reactor Pressure Vessel (RPV) into the Primary Containment Vessel (PCV) at Units 1 and 3. On the other hand, a certain portion of the fuel debris was assumed to exist in both of the reactors at Unit 2.

Items	Unit 1	Unit 2	Unit 3
Accident progression analysis	Most of fuel debris has fallen down to PCV.	Distribution of fuel debris largely depends on the amount of injected seawater by fire engines.	Most of fuel debris has fallen down to PCV.
Evaluation of heat balance method	Heat source is few in RPV.	A certain portion of fuel debris exists both in RPV and PCV.	A certain portion of fuel debris exists both in RPV and PCV.
Muon Radiography	High density structures (fuel debris) hardly exist in the reactor core.	A large mass of fuel debris hardly exist in the reactor core.※	Not measured
Investigation of inside of PCV	The investigation shows no large-scale damage to PCV wall(in the past researches.)	No large-scale damage in RPV lower head.	The investigation shows no large-scale damage in the structures (in PCV in the past researches)
Comprehensive evaluation	Most of fuel debris has fallen down to PCV.	A certain portion of fuel debris exists both in RPV and PCV.	Most of fuel debris has fallen down to PCV.

※Utilized joint research results by TOSHIBA and Nagoya University.

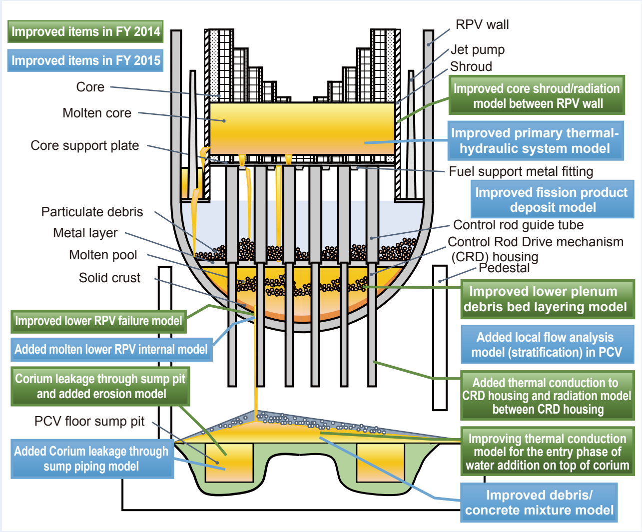


Fig. 1: Improvement / upgrading for MAAP Analysis Model

Improvements were made on a physical phenomenon model on the transfer of fuel debris behavior and fission products. Technology for accident progression analysis was upgraded.

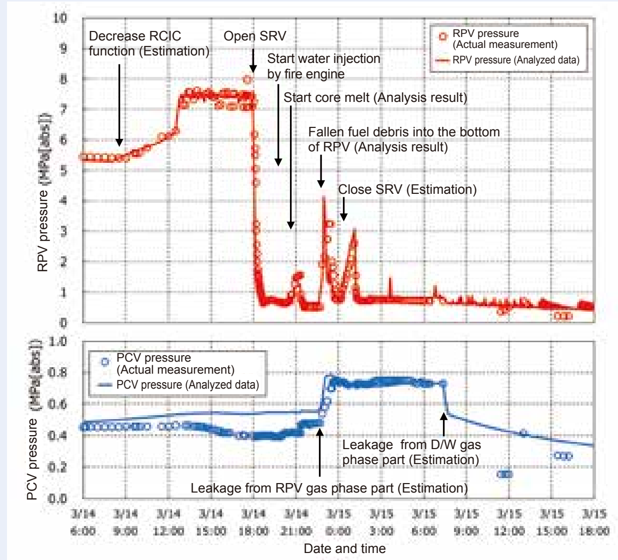


Fig. 2: Example of MAAP code accident progression analysis (Unit 2)

Analysis results were obtained from the improved code and sensitivity analysis which was almost the same as those of the actual measurement in Unit 2 with complicated accident progression.

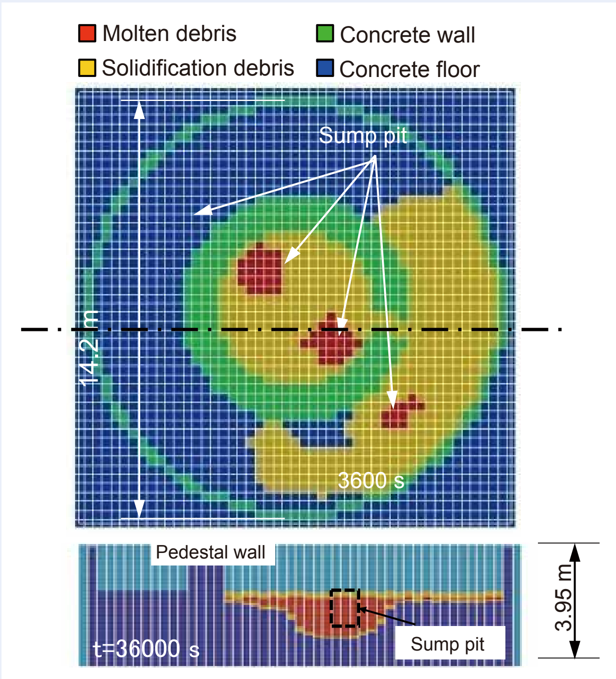


Fig. 3: Example of Fuel debris behavior evaluation in PCV (Unit 1)

Analysis code was developed to evaluate Molten Core Concrete Interaction (MCCI) in detail and estimate the behavior of fuel debris fallen into PCV.

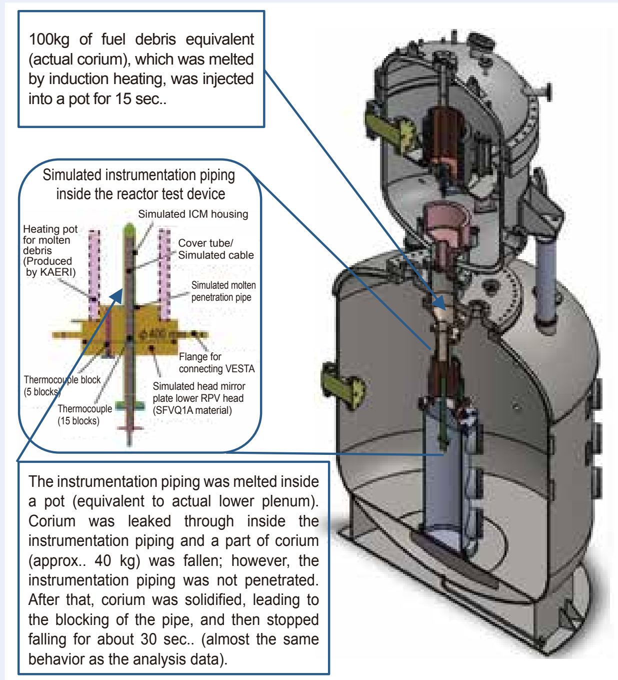


Fig. 4: PCV penetration pipe molten failure test (Simulated instrumentation piping inside the reactor test)

In the Korea Atomic Energy Research Institute (KAERI), a test was conducted using a full-scale penetration pipe and actual corium, which is equivalent to fuel debris; the damaged situation for the instrumentation piping inside the reactor was also confirmed.

Future Developments

The reliability of comprehensive analysis and assessment will be advanced by improving the accuracy of prediction for progressing events through sensitivity analysis based on actual data information. Additionally, BSAF Phase-2 project and database continue to be managed and organized. Upgrading for identifying conditions inside the reactor will be promoted.

Key Challenge 2 – R&D for Preparation of Fuel Debris Retrieval

Development of Technology for Detection of Fuel Debris in the Reactor

Background

In order to select effective methods of fuel retrieval, it is important to understand conditions inside the reactors including the distribution of fuel debris.

However, it is difficult to verify the conditions directly since the radiation level inside the RPV is too high for workers to access the debris location. Therefore, the tomography using cosmic-ray muon is expected to provide us with information inside the reactor.

Aims

The Muon tomography consists of transmission and scattering methods. The transmission method is implemented readily at the site and enables to identify fuel debris at Unit 1 at the Fukushima Daiichi NPS at the spatial resolution of about 1 m. Furthermore, a measurement system that uses the Muon scattering method will be developed to confirm the location of the remaining fuel debris in the reactor at the spatial resolution of about 30 cm.

Main Achievements and Approaches

1 Small-scale verification test by the transmission method at Unit 1 (at the spatial resolution of about 1 m)

a- Improvement of a detection system for the transmission method to be applied on-site

In order to measure muon even in high radioactive environment (0.4 mSv/h), the detection system that was used at Unit 1 was stored in an enclosure with iron shielding. The detection system is comprised of three units to reduce the effects of gamma radiation by conducting coincidence counting (Fig. 1). On the other hand, for easy handling and less spatial restriction on-site, the compact system, which sizes a fourth of the current whole system, was developed by improving the surface of the detection system without reducing the resolution (Fig. 2).

b- Measurement and assessment by the transmission method at Unit 1

In order to verify the capability to detect the remaining fuel debris in the reactor or spent fuel stored in the spent fuel pool at the spatial resolution of about 1 m, measurement was conducted for a period of three months at three locations in Unit 1 (Fig. 3). As a result, spatial resolution of about 1 m was ensured through assessment of spent fuel in the pool. On the other hand, high density structures such as the Reactor Pressure Vessel (RPV) and the Primary Containment Vessel (PCV) were identified through measurement of data, however high density large materials were not identified in the reactor core where fuels were supposed to remain. Therefore, it is concluded that there are few fuels remaining in the reactor core (Fig. 4 & 5).

2 Design and manufacturing for detection system through scattering method (at the spatial resolution of about 30 cm)

a- Development of a muon measurement system under high radiation (less than 50 μSv/h)

Since the detectors for the scattering method are large, an electric circuit incorporating gamma ray-eliminating logic in signal processing was developed to reduce the amount of shielding materials. The 7 x 7 m muon tracking device system that incorporates this electric circuit was installed. A measurement test was conducted using two muon tracking device (upper/lower) placed to surround the simulated core of the Fukushima Daiichi NPS – lead block (simulated fuel), concrete block (biological shield, building), and polyethylene block (simulated water) (Fig. 6). As a result, it was confirmed that uranium can be visualized at the muon scattering angle not affected by the lead structure. Additionally, functionality of the algorithm to adjust the position of the detector was confirmed by the measurement results of the system.

b- Development of an estimation technique of fuel debris distribution inside the reactor by muon tracking

Improved technique was developed as an estimation method of fuel debris distribution by adding a unique noise suppression technology to 3-D reconstruction algorithm that is used for CT technology in the medical field. The improvement ensured the capability to estimate fuel debris distribution with better detection rate (rate of determining the fuel debris in areas where fuel debris is inexistent), even though measuring time was shortened (Fig. 7). In addition, configurations of fuel debris that were deposited in the lower part of the RPV were clearly confirmed, applying the technique to remove scattered components of the building structure (Fig. 8).

Fig. 1: Detection system for the transmission method used for measurement in Unit 1

Three units of detectors are housed in an enclosure with iron walls 10 cm thick. The shape is approx. 2.5 m in length, 2.0 m in width and 2.1 m in height.

a) Surface of two-layered scintillator

b) Assembled small detector

Fig. 2: Overview of a small detector for the transmission method

Position resolution enables to quadruple by shifting a cubic foot of scintillator elements 5 mm the direction toward both of x and y to make two-layer structure; therefore, the resolution can be obtained same as the original even if size of a detector is half of its length and width and distance between detector units is lessened by half.

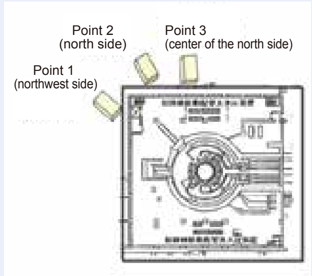


Fig. 3: Position of detection system for transmission method

Although the place to be measured was limited due to on-site conditions, measurement was carried out at three locations in the northwestern area of the Unit1 bldg.

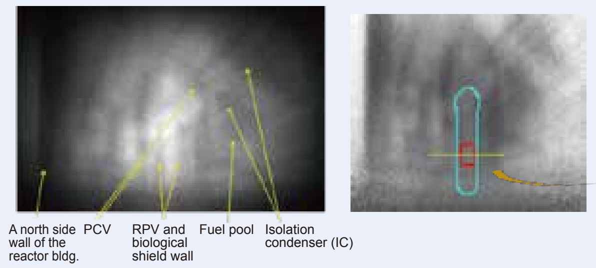


Fig. 4: Image of observation data (left fig.) and distribution of transmissivity (right fig.)

The black area shows a small transmission amount of Muon while the white part has a large transmission amount. The observation data represents a distribution of transmissivity inside the reactor.

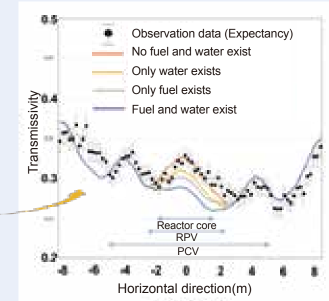


Fig. 5: Comparison with simulated transmissivity measurement

The transmission amount of Muon, which passed through inside the bldg., was predicted by simulation and was compared with the observation data. It is concluded that fuel hardly existed in the reactor core.

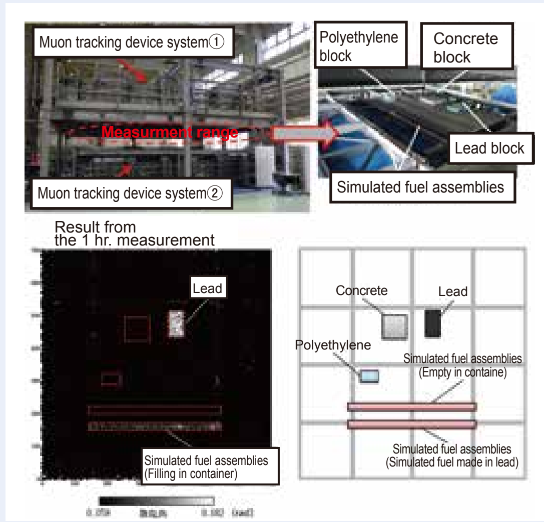


Fig. 6: Measurement results of the transmission method by the developed system

The figures are layout for two Muon tracking device systems (upper/lower) and simulated materials of the reactor core: a lead block (simulated fuel), a concrete block (biological shield and the bldg..) and a polyethylene block (simulated water) at the Fukushima Daiichi NPS. The measurement of about 1 hr. reveals the distribution of scattering angle in the layout.

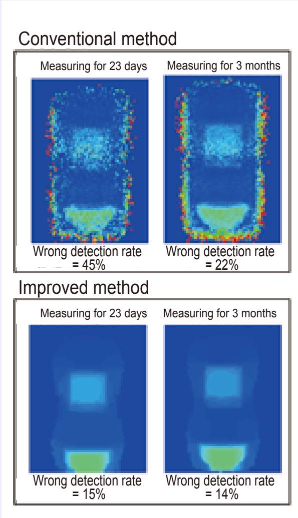


Fig. 7: Shortening of measuring time

The figures show the comparison among simulation results of debris distribution between the conventional method and the improved method applied to ML-EM method used in the medical field. The measuring period is approx. 3 months and 23 days, which is equivalent to1/4 of 3 months.

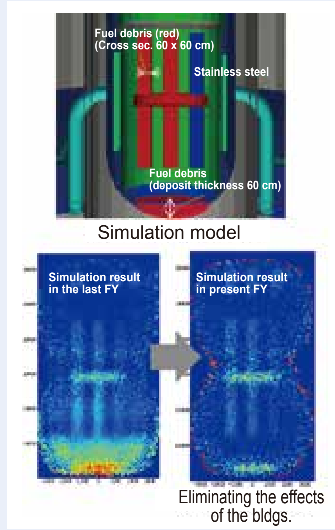


Fig. 8: Effect reduction in the bldg.

The figure shows simulation result of the simulation model which has arranged stainless steel to compare with fuel debris in RPV. Compared with the simulation result in the last FY, this FY's simulation result confirms the deposits of fuel debris in the lower RPV with higher contrast by reducing effects of bldgs.

Future Developments

According to the measurement results at Unit 1, the transmission method confirmed its applicability for the measurement technology of fuel debris distribution under high radiation environment. On the scattering method, a large detection system has been completed for implementation and confirmed to identify fuel debris with high detection capability. Both systems of the transmission method and scattering method will be applied to on-site use accordingly, considering the site situation and progress of identifying conditions inside the reactor.

Key Challenge 2 – R&D for Preparation of Fuel Debris Retrieval

Development of Fundamental Technology for Retrieval of Fuel Debris/Reactor Internals

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, reactor buildings, the RPV and the PCV suffered damage at the accident, and the reactors are in unstable conditions. Therefore, it is necessary that fuel debris be retrieved to maintain subcritical conditions and become stable condition by preventing diffusion of radioactive materials.

Aims

The feasible R&D continues toward the determination of the fuel debris retrieval methods (Summer 2017), finalization of retrieval methods from the unit to be firstly retrieved (the first half of 2018) and commencement of fuel debris retrieval from the unit to be firstly retrieved (around December 2021). In order to verify the feasibility of the fuel debris retrieval methods, element tests will be conducted to obtain the necessary information/data for three methods: Submersion-top entry method, Partial submersion-top entry method and Partial submersion-side entry method under this project.

Main Achievements and Approaches

1 Coordinating and analyzing results for each element test

Another proposer who partially engages in the operation joins in the R&D aside from IRID for this project. IRID will coordinate the whole project including element testing that the proposer plans and conducts. Currently, IRID is organizing test plans including plans by the proposer.

2 Element tests required for the feasibility of the methods

Test plan was established for each element test in FY 2015 (includes tests under planning). Based on the test plan, feasibility of the methods is verified through test manufacturing and element tests accordingly.

a- Prevention Technology against the spread of contamination in retrieving contaminated large structure

- In order to verify the prevention technology against the spread of contamination, a small model test is being conducted for each working step (Fig. 1).

b- Prevention technology against the spread of contamination in retrieving fuel debris in the Reactor Pressure Vessel (RPV)

- Tests are being conducted on the sealing in the access device and in the lower part of the device inside the RPV under submersion-side entry method (Fig. 2).

c- Technology for accessing fuel debris

- A test is being conducted for the hydraulic manipulator (Fig. 3).
- Test plan of the access device inside the RPV is formulated, and an elemental test is being planned under submersion-top entry method (Fig. 4).
- Test plan of the access device inside the pedestal is formulated and a mock-up test is being planned under partial submersion-side entry method (Fig. 5).

d- Technology for remotely operated retrieval of fuel debris

- Test of remotely operated flexible structure arm is being conducted (Fig. 6).
- Test plan of handling device of fuel debris canister is formulated, and an element test is being planned (Fig. 7).

e- Prevention technology against spread of contamination in retrieving fuel debris

- Test plan of the platform and cell by Submersion method is formulated, and an element test is being planned (Fig. 8).
- For the purpose of remotely seal welding for the cell by partial submersion-side entry method, a test plan and an element test of the PCV welding device are being formulated (Fig. 9).

f- Technology for reducing exposure of workers in retrieving fuel debris

- Test plan of the lightweight shielding and shape to be applied for top entry method is formulated, and an elemental test is being planned.

g- Cutting/dust collection for fuel debris retrieval

- Performance test of cutting/dust collection technology for fuel debris retrieval is being conducted

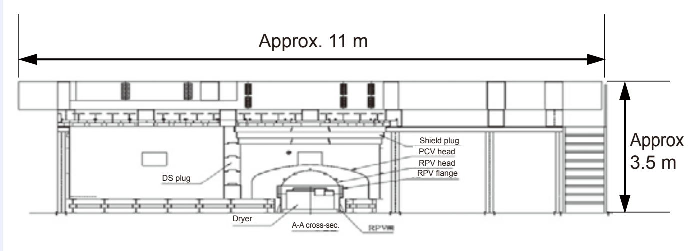


Fig. 1: Conceptual diagram of a plan for the scale model testing facility

The study to prevent the spread of contamination will be proceeded by a scale model test.

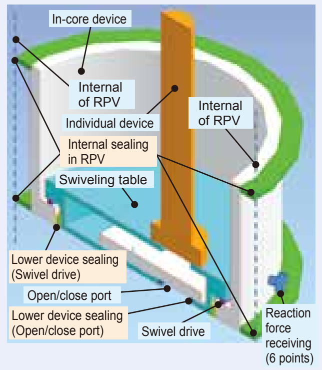


Fig. 2: Conceptual diagram of the access device inside RPV (In-core device)

Issues will be clarified during actual application such as confirming the sealing performance through a partial model device.

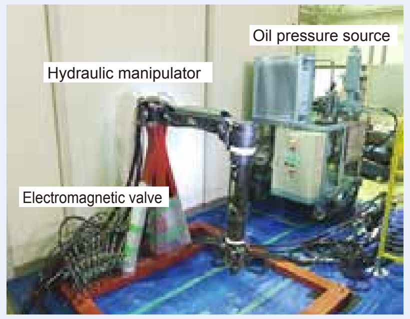


Fig. 3: Photo of test device for hydraulic manipulator

In accordance with a simulating actual remote distance, the test device checks the operational error.

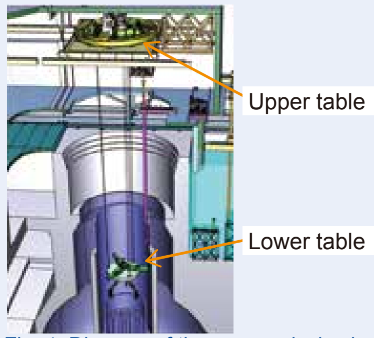


Fig. 4: Diagram of the access device inside RPV (plan)

Comparison study is performed for the supporting method to a horizontal direction for a lower table that is hung with a wire.

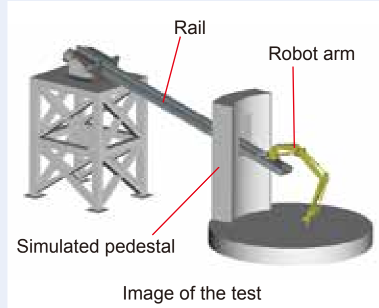


Fig. 5: Mock-up test facility for the access device inside the pedestal and test condition (Image)

The feasibility of Partial submersion-side entry method will be verified by simulating an actual arrangement.

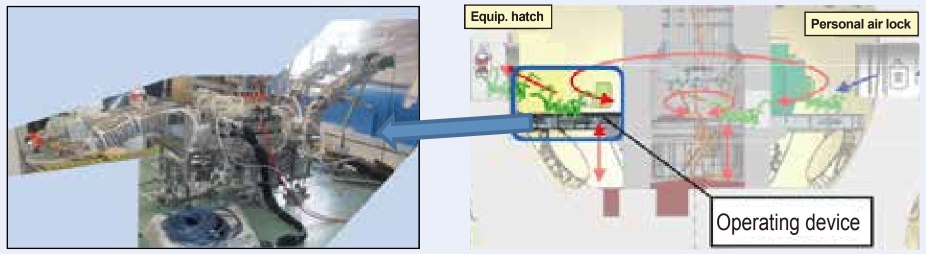


Fig. 6: Remotely operated flexible structure arm in actual use (Image)

It is possible that flexible structure arm can establish a remotely operated device with a variety of shapes/functions similar to toy blocks.

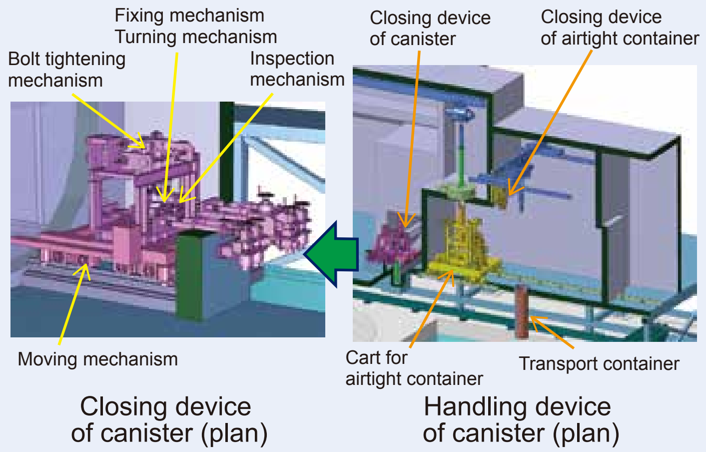


Fig. 7: Handling device of fuel debris canister (Image)

Basic design of the device and elemental test items of the closing device will be clarified.

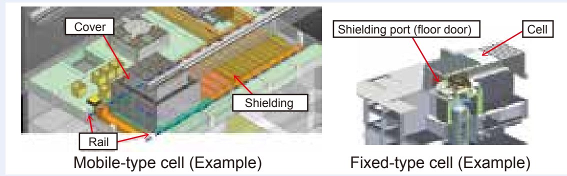


Fig. 8: Mobile-type/fixed-type cell (Image)

Mobile-type cell and fixed-type cell will be compared and evaluated from the point of required function and workability among others.

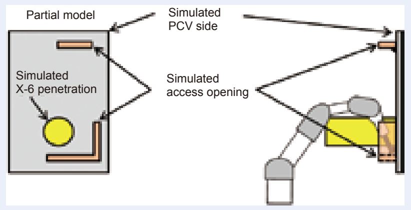


Fig. 9: Mock test for PCV welding device (Image)

A welding test to be operated remotely will be conducted under the same conditions in actual use; solutions and feasibility will be confirmed.

Future Developments

In order to evaluate the feasibility of the methods for the three fuel debris retrieval methods, Submersion-top entry method, Partial submersion-top entry method and Partial submersion-side entry method, elemental tests will be conducted until the end of FY 2016. Through testing, we will identify and examine further issues, and then discuss action plans.

Key Challenge 2 – R&D for Preparation of Fuel Debris Retrieval

Upgrading Approach and System for the 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 subcriticality and become in a stable condition by preventing diffusion of radioactive materials.

Aims

In order to retrieve fuel debris, three retrieval methods are chosen; Submersion-top entry method, Partial submersion-top entry method and Partial submersion-side entry method. Toward finalizing the retrieval methods of fuel debris and reactor internals, plant-related information will be examined, retrieval methods for fuel debris and reactor internals, system and equipment studied, feasibility of the retrieval methods and the whole system verified. Also, development plan of retrieval system and equipment will be formulated.

Main Achievements and Approaches

1 Clarifying plant information to determine the retrieval policy of fuel debris/reactor internals

Plant data and development results of related projects were organized, and vital information was examined to determine the retrieval methods for fuel debris.

2 Study of retrieval methods for fuel debris/reactor internals, system and equipment

a- Study of the feasibility of the methods

The retrieval methods were classified into 12 types in combination with the water level that fuel debris would be retrieved from (roughly divided into Submersion method and Partial submersion method) and the direction that fuel debris would be accessed from (entry from top, side and bottom). In order to prove technical challenges from the three representative methods (Submersion-top entry method, Partial submersion-top entry method and Partial submersion-side entry method), a flow chart (Fig. 1 & 2) and a working step chart summarizing the changing conditions of the plant, the flow of each working step and treatment of retrieved fuel debris and wastes.

After identifying the issues to be solved, assessment of the feasibility and development plans for resolution will be designed.

In alignment with related projects such as the development of canister, scope of each study was confirmed, latest information was shared, the common challenges were clarified, and preconditions/hypothetical conditions were set up.

b- Conceptual study of the system

In order to ensure safety for fuel debris retrieval, it is necessary to install systems such as cooling equipment to maintain the circulating and cooling water for fuel debris and the equipment to prevent dust scattering including radioactive particles that would generate from cutting fuel debris. Safety criteria and requirements for the system are reviewed.

3 Formulation of development plans of device/system for retrieval of fuel debris/reactor internals

In order to formulate development plans for the retrieval system of fuel debris and reactor internals and the equipment, technical challenges are being clarified.

Future Developments

Following FY 2015, we continue to study for promotion of plans in response to the changing situation, new challenges and requirements.

This achievement will serve as useful information to determine the retrieval policy of fuel debris, finalize the retrieval method and commence retrieval for the unit which will be firstly retrieved from, based on the Mid-and-Long Term Road Map (revised in June 2015).

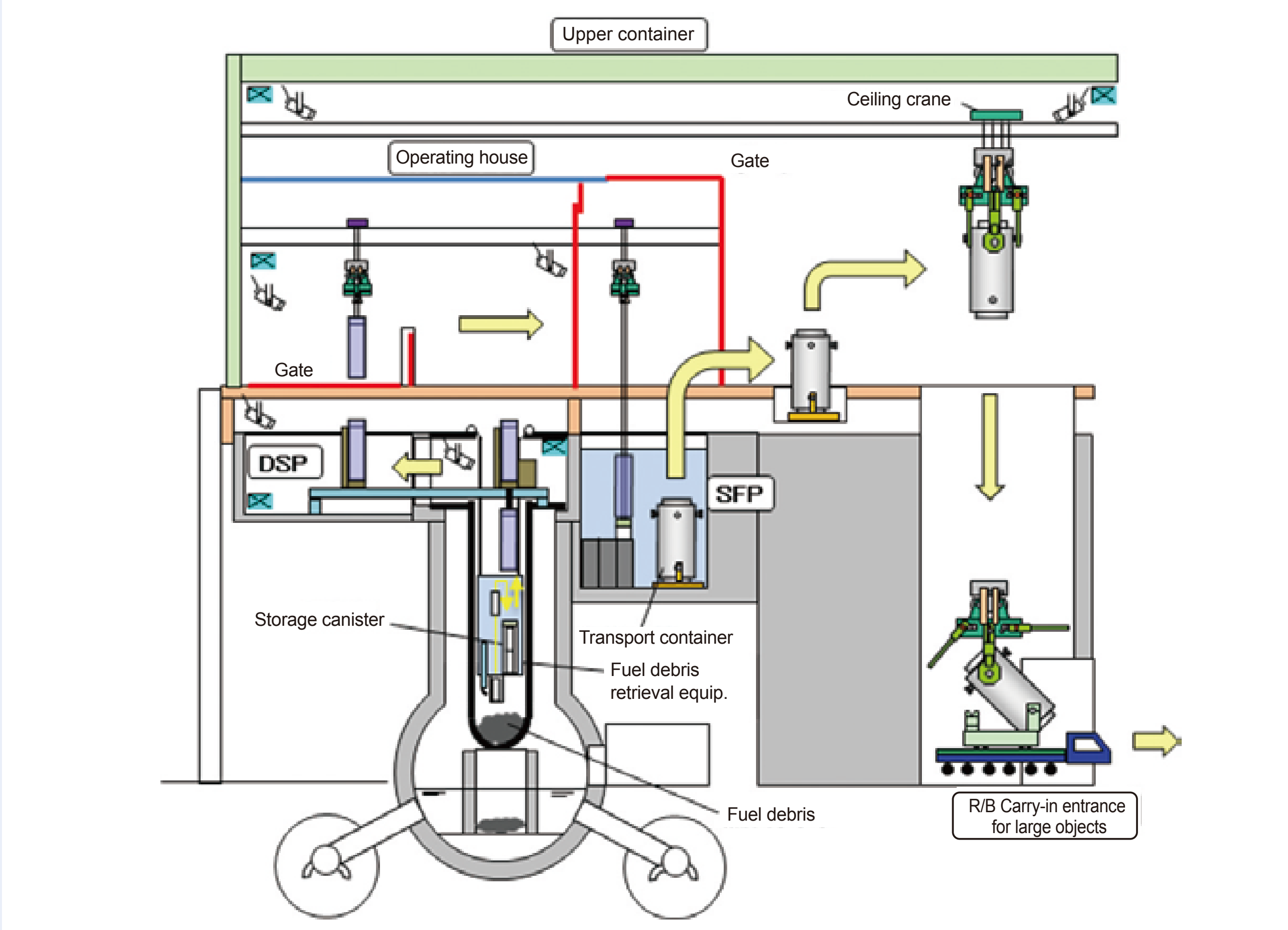


Fig. 1: Flow chart of fuel debris retrieval by Partial submersion-top entry method (Image)

This chart shows the flow of Partial submersion-Top entry method; fuel debris is retrieved from the RPV bottom, collected in a canister and loaded into a vehicle to transfer outside of the reactor building (R/B).

The flow chart details how to deal with actual fuel debris and extracts technical challenges to be analyzed.

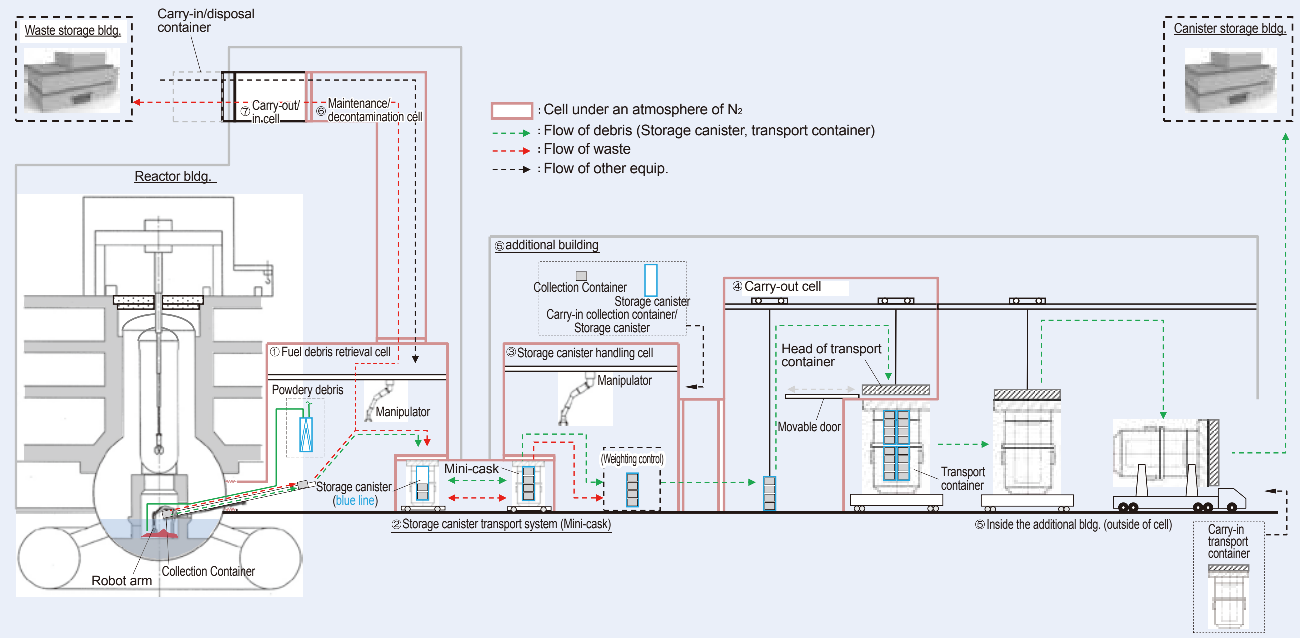


Fig. 2: Flow chart of fuel debris retrieval by Partial submersion-side entry method (Image)

The chart shows a flow of fuel debris transfer by Partial submersion-side entry method; fuel debris in the PCV bottom is transferred to the side, stored in the canister and loaded into vehicle for transferring. Fuel debris is safely transferred in the order shown inside the red square frames. In order to realize the flow, technical challenges and issues will be clarified and analyzed.

Key Challenge 2 – R&D for Preparation of Fuel Debris Retrieval

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

Background

Fuel debris retrieved from the Fukushima Daiichi NPS is expected to be carried out in the reactor buildings and stored safely until the time when the method of final disposal is determined. For this reason, system required for the collection, transfer and storage of fuel debris must be established.

Aims

Based on existing technologies for transportation and storage of spent fuel, fuel debris storage canisters (hereinafter called “storage canister”) and canister handling device for safe collection will be developed. Materials and structure of storage canisters will be studied, and basic specifications of the canister will be created in FY 2015.

Main Achievements and Approaches

1 Investigation and establishment of a research plan for the transfer and storage of damaged fuel

In order to investigate for the design of storage canisters, subcriticality evaluation technology, evaluation technology for hydrogen generation, sludge collection technology, drying technology for vacuum and technical information were obtained from a visit to Paks Nuclear Power Plant in Hungary, the Pacific North West National Laboratory (PNNL) and the Hanford Site in the United States (Fig. 1 & 2).

2 Study on the fuel debris storage system

Based on the knowledge obtained from the latest situation on-site, information from other R&D projects and this project – concept of fuel debris storage that was studied in FY 2014 – was verified. Results confirmed that the current storage concept was not necessary to review.

3 Development of safety evaluation methods

Fuel debris in the Fukushima Daiichi NPS consists of products generated from the Molten Core Concrete Interaction (MCCI) and seawater substances, unlike the actual fuel debris from the Three-Mile Island Unit 2 (TMI-2) Reactor. In addition, for the effective and safe collection, transport and storage of large amount of fuel debris, it is necessary to enlarge the internal diameter of storage canisters, and handle them through remote operation. In order to reflect these matters in the design of storage canisters, reviewing evaluation methods through investigation of documents, survey of overseas case, tentative analysis and conducting tests, items for detail study were determined on evaluation for the amount of hydrogen generated (countermeasures for hydrogen generation), evaluation on material aging degradation (material selection of the storage canister), criticality evaluation (subcriticality design) and structural strength evaluation of storage canisters (Fig. 3).

4 Development of technology for fuel debris storage

Based on the knowledge obtained from R&D of the ongoing project, we have examined requirements for the design of storage canisters and reviewed basic specifications of the storage canister. Future challenges were also clarified (Fig. 4).

5 Development of technology for transfer and storage of the canister

In order to study the equipment handling storage canisters, the report entitled “Flow of the Collection, Transfer and Storage for the Fuel Debris Storage Canister” formulated in FY 2014 was updated based on the latest situation on-site and knowledge obtained from the project (Fig. 5).

Future Developments

The feasibility of basic specifications for the storage canister will be verified in cooperation with other R&D projects based on achievements in FY 2015. Furthermore, in order to solve clarified issues, the basic design of storage canisters will be performed through analyses, evaluations and element tests.

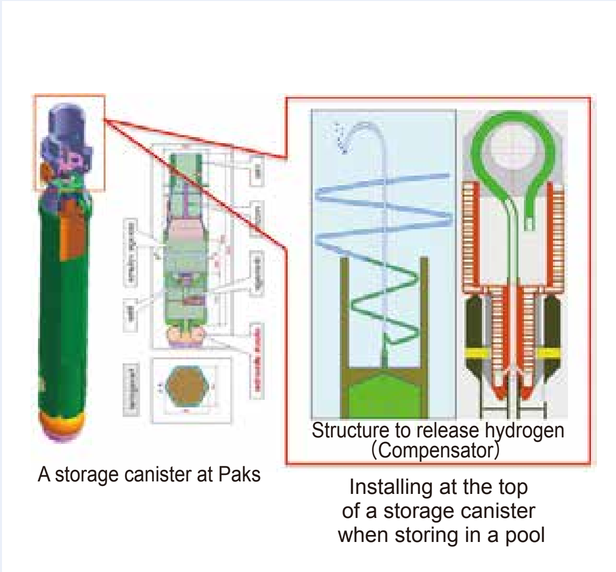


Fig. 1: A storage canister used at Paks NPS

Technical information was obtained for collection, transfer and storage of damaged fuel at Paks NPS.

Ref.:L. Szőke, Managing of the damaged fuel at Paks NPP, Meeting in Vienna on Management of Severely Damaged Spent Fuel And Corium 2013

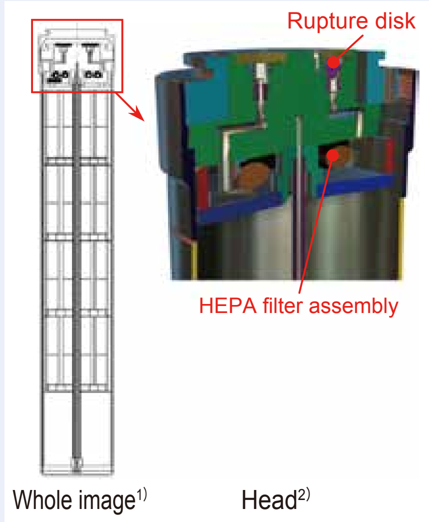


Fig. 2: Storage canister used at Hanford Site in U.S. (Ref.)

Technical information was obtained for collection, transfer and storage of metal uranium damaged fuel caused by decommissioning of pool at Hanford Site.

Ref.: 1)R.McCormack,Multi-Canister Overpack, Nuclear Waste Technical Review Board, 2014

2)N Reactor (U-metal) Fuel Characteristics for Disposal Criticality Analysis, DOE/SNF/REP-056, Rev0, 2000

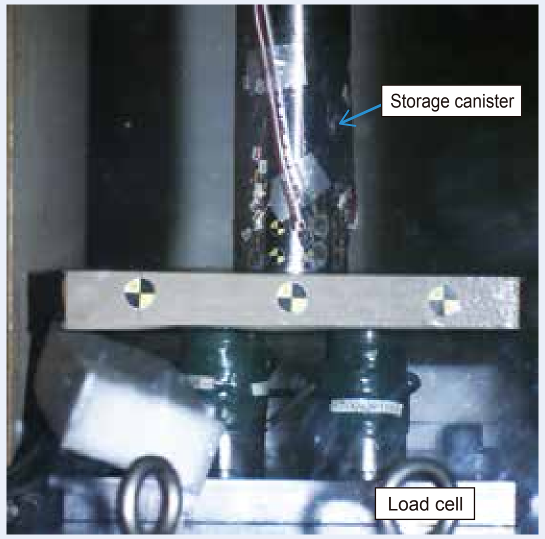


Fig. 3: Example of a storage canister drop test in the vertical direction

Posited on an occurrence of an accident during the handling of storage canister, a drop test in the vertical direction was conducted by structure evaluation method.

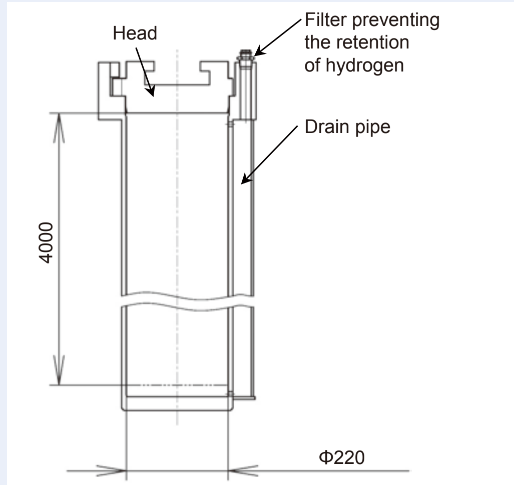


Fig. 4: Basic design of a storage canister (Example)

Based on the information and requirements acquired, the basic design of a storage canister was examined. The spec. and shape are continuously updated depending on the results of the safety evaluation and the study on the handling of storage canister when retrieving fuel debris.

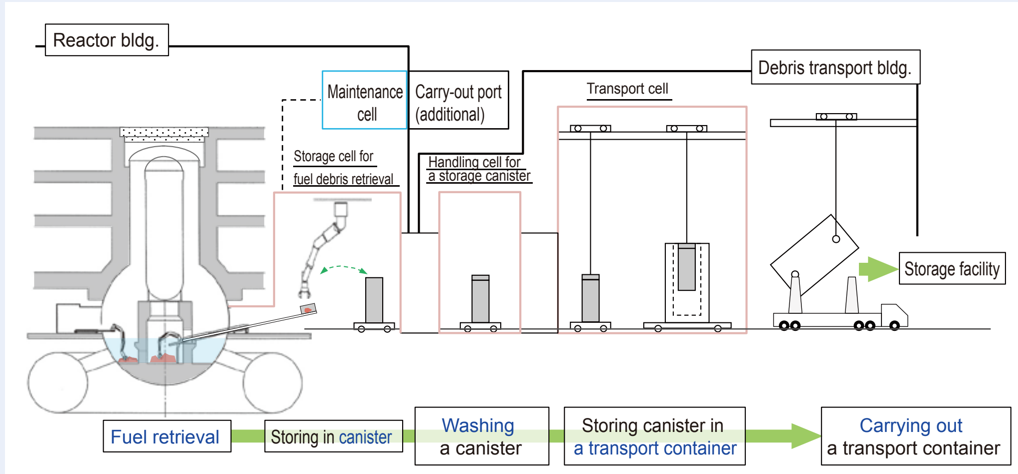


Fig. 5: Flow chart for the collection, transfer and storage for a storage canister (Fig. shows non-submerged from side method from the side of the reactor.)

In preparation for the study on required equipment for transfer, the flow of handling canister in FY 2014 was reviewed. More updates will follow.

Key Challenge 2 – R&D for Preparation of Fuel Debris Retrieval

Development of Technology for Integrity Evaluation of the Reactor Pressure Vessel/Primary Containment Vessel

Background

Due to the devastating Great East Japan Earthquake, the deterioration of the structural materials at the Reactor Pressure Vessel (RPV) and the Primary Containment Vessel (PCV) at the Fukushima Daiichi NPS, which was caused by high temperatures, injection of seawater and fallen debris, became the concern. Action plan is necessary to maintain the structural integrity of the PCV and the RPV over the long period of retrieving fuel debris from the reactor core.

Aims

In order to evaluate the feasibility of Submersion method, urgent seismic-resistance evaluations are required accordingly in response to assumed various reactor situations. In addition, considering seismic-resistance evaluations for corrosion thinning amount for a long period of time, the integrity of the PCV/RPV will be evaluated. Corrosion control measures will be also studied to maintain the integrity.

Main Achievements and Approaches

1 Evaluation of submersion method feasibility based on the seismic integrity of the PCV/RPV

The feasibility of dry/full submersion methods of fuel debris retrieval (entry from top) was examined through a seismic-resistance evaluation for the PCV/RPV equipment, which reflected assumed reactor conditions. For areas that showed severe result on the evaluation, a detailed evaluation of seismic resistance was conducted. Additionally, assuming that the suppression chamber (S/C) support is not reinforced, a detailed evaluation of seismic resistance was conducted (Fig. 1).

2 Simple evaluation of the seismic resistance of equipment considering repairs (water leakage stoppage) and water level rise in the PCV

Seismic response analysis was conducted using the selected parameters (water level in the PCV) that impact on seismic response analysis. Response comparison of seismic load due to parameters variation was examined as well. Furthermore, a simple evaluation method was developed for the seismic resistance of the equipment in combination with various methods. Results of the evaluations as compared with those of the simple evaluation method and regular active analysis were almost same, and the validity of the simple evaluation method was confirmed.

3 Development of anti-corrosion measures

Results from tests evaluating rust inhibiting under high levels of radiation exposure after selecting rust-preventive agents and sufficient information was obtained to judge their applicability on-site. Evaluation tests of the rust-preventive agents on the effect on the function of water treatment equipment were conducted, and technical issues were clarified (Fig. 2).

4 Upgrading predictability of long-term corrosion thinning rate

In order to improve accuracy prediction of corrosion thinning rate, a long-term corrosion test for 10,000 hours was conducted. Additionally, new knowledge was obtained through investigation of corrosion impact and leaching components from fuel debris and concrete inside the reactor.

5 Evaluation of pedestal erosion impact

Tests such as concrete strength test and rebar corrosion test were conducted under high temperature heating and air/underwater exposure conditions to obtain various data on cylindrical specimen, small-scaled mock-up test body and block specimen among others. Through these tests, effective information was obtained for further study (Fig. 3).

Future Developments

In order to implement on-site, it is necessary to select a rust inhibitor that would not cause leakage due to local corrosion such as crevice corrosion. Since an immersion test for 500 hours is not able to evaluate the applicability, it is necessary to evaluate a local corrosion occurrence and its progression through electrochemical measurements that do not depend on the evaluation time. Furthermore, since there is a concern that precipitation film type of rust inhibitor (phosphate) would adhere on high temperature area of the fuel debris surface, its impact requires to be evaluated.

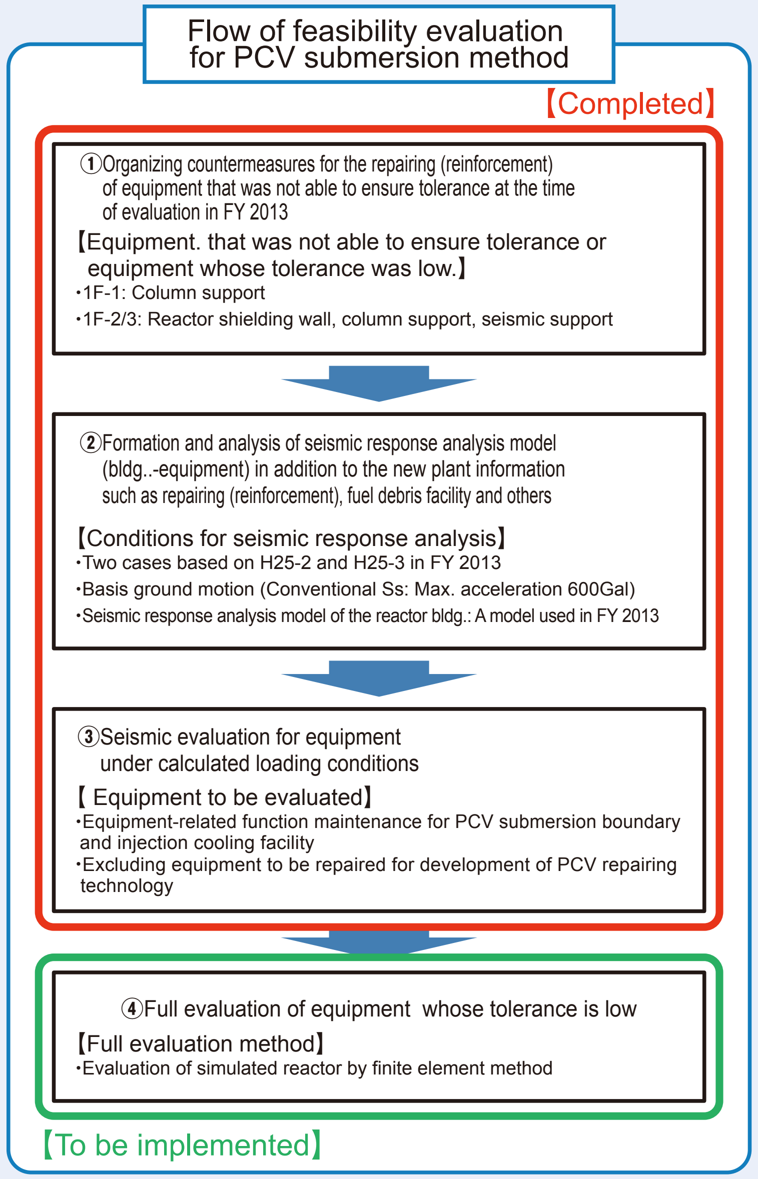
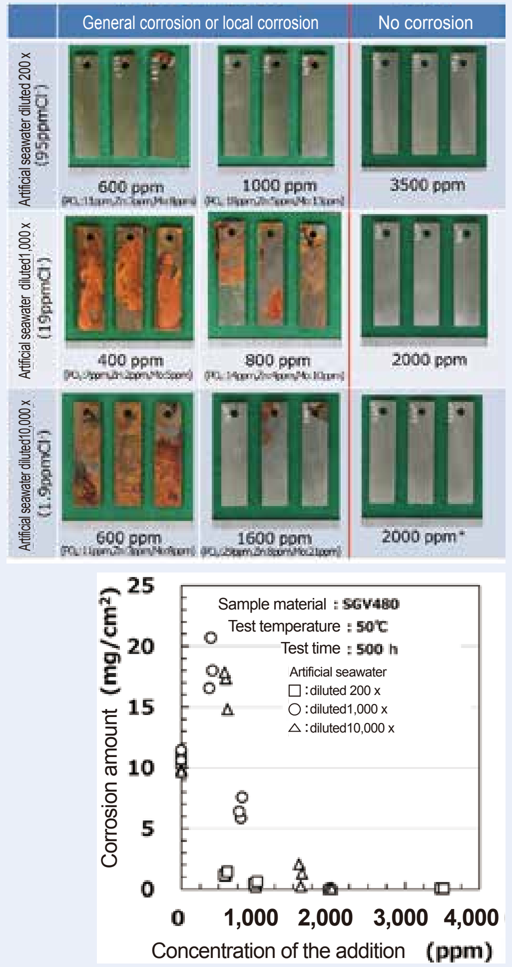


Fig. 1: Flow of PCV submersion method feasibility evaluation



·Test specimen presented a metallic gloss by adding 3,500 ppm seawater diluted 200 x and 2,000 ppm seawater diluted 1,000/10,000 x. Corrosion amount was almost 0 mg/cm^2
(Confirmed anti-corrosion effects on carbon steel)

Fig. 2 Example of validation test for anti-corrosion effects of rust-preventive agents
(Test result of adding zinc / sodium molybdate mixing phosphate)

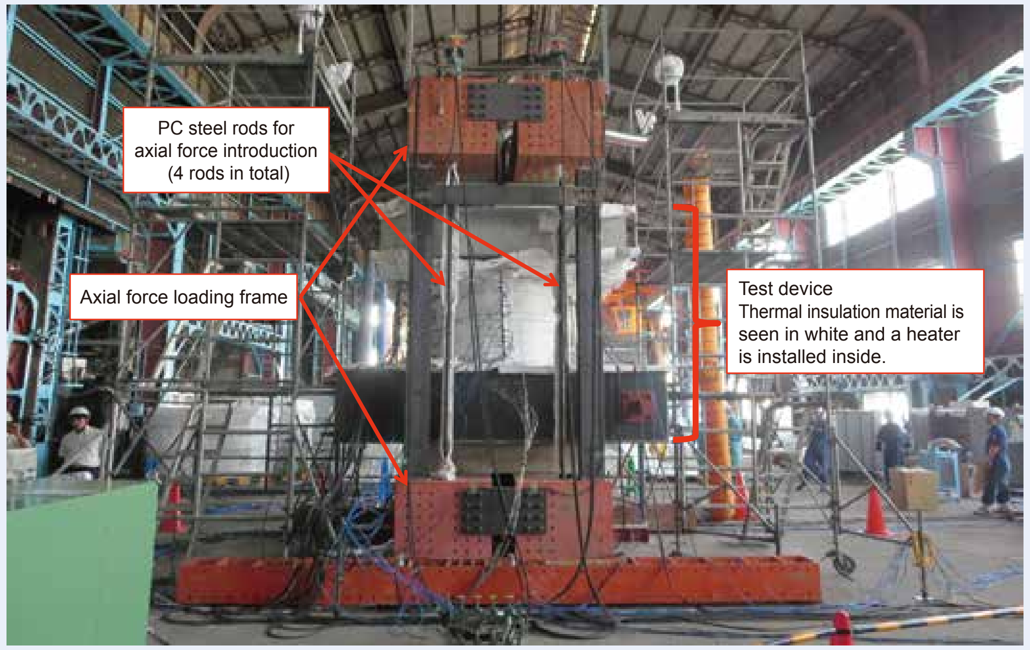


Fig. 3: Proof stress evaluation test of the RPV pedestal (Installation loading frame)

A cylindrical test device simulating the RPV pedestal, which was injected with cooling water after being exposed to high temperature at the accident, was loaded to confirm tolerance of concrete structures, which deteriorated due to high temperature, and the impact on the destructiveness. (After drying, the cylindrical test device is heated at 400℃ and 800℃ for prolonged time and immersed in water.)
Outside diameter: 1,240 mm, Inner diameter: 834 mm, thickness: 200 mm

Key Challenge 2 – R&D for Preparation of 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. However, since the shape of fuel debris and water level may change during retrieval of fuel debris in the future, it is necessary to develop criticality scenarios, criticality evaluation, monitoring technology and criticality prevention technology accordingly.

Aims

Criticality scenarios for multiple methods (i.e. Full submersion, Partial submersion, Partial dry and Dry) will be clarified in FY 2015. In addition, preventing technology for criticality using neutron absorption materials and monitoring technology such as criticality detection will be developed; criticality control will be established to maintain subcriticality, and recriticality detection technologies will be developed to prevent excessive exposure, even if criticality occurs.

Main Achievements and Approaches

1 Evaluation of Criticality

Criticality scenarios, which reflected the latest information concerning various construction methods, were reviewed. Criticality risks were evaluated for each unit and part (Fig. 1). Criticality behavior evaluation and exposure assessment on the remaining fuel in the reactor core, which has an increasing criticality risk, were conducted by submerging the upper level of the Primary Containment Vessel (PCV). During the evaluation, a practical assessment was conducted using detailed fuel compositions of the fuel rod unit after considering the operating history for each unit. In the result, it showed that submerging method using pure water is deemed applicable, if remaining fuel in the reactor core is within the expected range. In addition, criticality analysis was conducted considering the proportion of Gadolinium (Gd) in fuel debris by using the actual data and was studied to streamline criticality evaluation (Table 1).

Basic idea of criticality control for the fuel retrieval methods was studied. The idea of criticality safety control was studied based on defense in depth at the time of incident/abnormal events. Requirements for the system of fuel debris retrieval were examined (Table 2).

2 Technology for reactor recriticality detection

"Gas sampling fission product gamma radiation detector system," which detect recriticality, was confirmed to be applicable in multiple methods.

Additionally, in order to confirm the feasibility of the system, a verification test plan for subcriticality estimation algorithm in actual use was designed.

"Neutron detection system" was studied for its application as a backup method.

In regards to the "in-core criticality approach detection system," which detect abnormalities before criticality is reached, concept of the detection system was established in combination of several methods (Fig. 2). In order to confirm the feasibility of the detection system, a test plan that used a storage facility of spent fuel and critical assembly was developed.

3 Technology for criticality prevention

"Non-soluble neutron absorbing material" is a technology that prevents criticality by supporting soluble neutron absorbing materials through direct application on the debris. Continuing with FY 2014, in order to select candidate materials, irradiation tests for new candidate materials and improved candidate materials were conducted. In addition, required input of "non-soluble neutron absorbing material" was evaluated while the conceptual design of applicable equipment conducted.

"Soluble neutron absorbing material" will be widely used to prevent criticality between the PCV submersion and retrieval of fuel debris. In order to confirm the effect the application of this material will have on the integrity of the PCV and the Reactor Pressure Vessel (RPV), corrosion testing using sodium pentaborate in the low concentration region was performed. Result confirmed that significant corrosion was not observed with a concentration of over 2,000 ppmB, and scope of applicability was clarified with tests in the high concentration region (Fig. 3). Additionally, regarding the applicability of the "soluble neutron absorbing material" to an actual plant, the conceptual study for water quality control facility, taking into account the impact on the nuclide removal system, was performed.

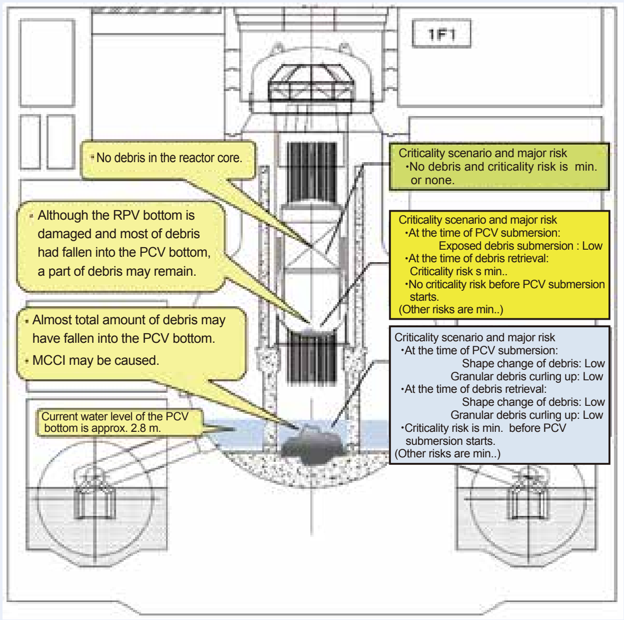


Fig. 1: Result of criticality risk evaluation at Unit

The criticality risk was evaluated based on the estimated conditions inside the reactor of Unit 1.

As a result, it proved that the criticality risk is low at the time of PCV submersion when there is almost no fuel debris in the reactor core and the RPV bottom.

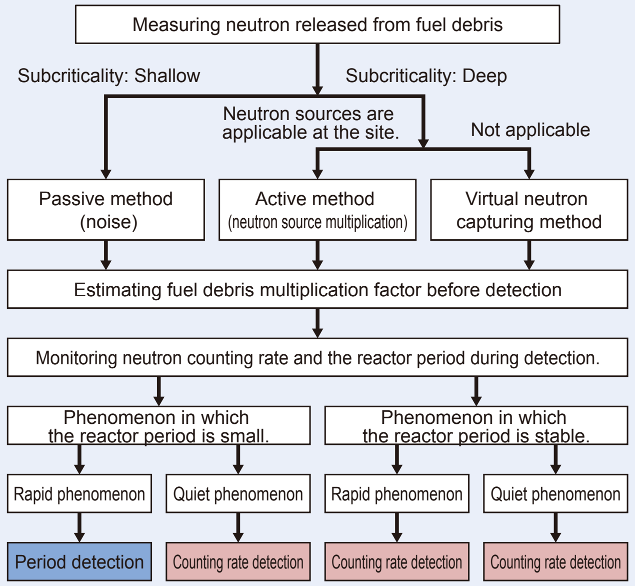


Fig. 2: Detection process of criticality approach detection system inside the reactor core

Fuel debris multiplication factor is measured by Passive method based on the measurement of neutron released from fuel debris.

When difficult to measure by Passive method due to a deep subcriticality, a multiplication factor is estimated by virtual neutron capturing method. Active method is optional in accord to the current situation. Based on estimated multiplication factor, neutron counting rate and the limited value of the reactor period (equivalent to the change rate of the counting rate) are defined, and the counting rate and the period are monitored during detection.

		Concentration degree	Gd concentration	Required boron conc. (ppmB)
Unit 1	Min. burnup fuel assemblies ave.	No Gd	8,500
		Ave. in the reactor core	6,400
Unit 2 (Unit 3)	Min. burnup fuel assemblies ave. (evaluated in Unit 2)	No Gd	10,100
		Ave. in the reactor core (Unit 3)	5,150
		Ave. in the reactor core ((Unit 2)	2,550

Table 1: Required boron concentration considering Gd

When analysis was conducted for fuel with low burnup in both cases where Gd did not remain on fuel originally containing Gd, and where Gd remains average, it proved that required boron concentration can decrease approx. 6,000 ppmB in order to maintain subcriticality.

Besides, fuel in Unit 2 was evaluated as a representative as fuel concentration in Unit 3 is lower than in Unit 2.

	PCV submersion		Debris retrieval
	Pure water	Boric acid solution	
Criticality control methods	Submerging gradually while confirming subcriticality condition	Same as left	Retrieving under subcriticality
Criticality prevention system (PS)	Submerging gradually at limited speed while confirming subcriticality condition	•Criticality prevention due to boric acid	•Monitoring subcriticality of criticality approach detection system nearby debris •Criticality prevention by soluble/insoluble neutron absorbing material
Mitigation system of the impact at the time of criticality(MS)	Detection by gas sampling fission product system FP gamma radiation detector system and terminating criticality by injecting boric acid solution		Same as lef

Table 2: Basic idea of criticality control

Two cases are considered for the submersing of PCV: one is to submerge gradually injecting pure water while confirming subcriticality condition and the other is to submerge using boric acid solution while prevent criticality.

Neutron absorbent will be used accordingly as criticality approach detection system is operated during fuel debris retrieval.

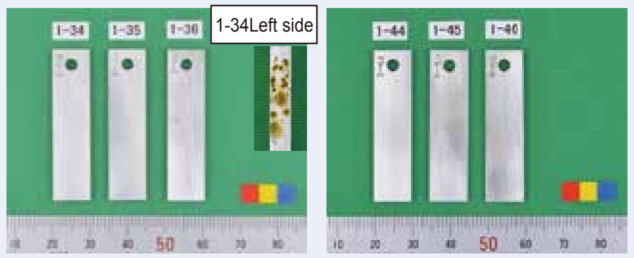


Fig. 3: Soluble neutron absorption material low concentration corrosion test result (example)

Left photo shows results of a 100-hr. test on 1,000 ppmB of sodium pentaborate for non-coated base material and seawater diluted 1,000x.

Right photo shows test results on 2,000ppmB of sodium pentaborate; other conditions were same.

Low concentration of 1,000 ppmB of sodium pentaborate partly caused corrosion; however, it was confirmed that 2,000 ppmB did not cause corrosion.

Future Developments

In regard to criticality evaluation for the multiple methods of fuel debris retrieval examined in FY 2015, criticality control methods during the retrieval of fuel debris will be studied while formulating updates and elaboration showing the latest expertise.

Verification tests of the feasibility for the essential technologies such as the "in-core criticality approach detector system" and "non-soluble neutron absorbing materials" will be conducted. Criticality control methods for the retrieval methods continue to develop, in addition to the establishment of the essential technology for criticality control. The technologies will be utilized for the finalization of fuel debris retrieval policy in the summer of FY 2017.

Key Challenge 2 – R&D for Preparation of Fuel Debris Retrieval

Fuel Debris Characterization

Background

To ensure safe and steady implementation for retrieval, collection, transfer and storage of fuel debris toward decommissioning, fuel debris mechanical properties such as hardness, thermal properties and moisture characteristics must be fully understood through analysis, investigation, assumption and evaluation.

Aims

In order to obtain and share the necessary information for the study of fuel debris retrieval methods, development of collection, transfer and storage of fuel debris, and fuel debris characteristics will be estimated by testing and analysis using simulated debris. Furthermore, analysis scenario will be studied, and elemental analysis technologies will be developed to achieve fuel debris analysis.

Main Achievements and Approaches

1 Assumption of fuel debris characterization inside reactor

Based on the survey of the project needs for the preparation of fuel debris retrieval, fuel debris properties for evaluation were clarified; micro-properties/macro-properties of fuel debris were estimated, and the data compiled (Fig. 1).

2 Evaluation of the characteristics using simulated debris

a- Understanding the data on debris characteristics

In order to expand the data on mechanical fuel debris properties, Zr (O) (solid solution of oxygen and zirconium), using the amount of oxygen as a parameter, was fabricated. Mechanical properties (hardness, elasticity ratio and fracture toughness) were measured and evaluated. In addition, (U, Zr) O_2, which was mixed into the Stainless Steel (SUS) components and fission product (FP) elements to form a solid solution, was fabricated; the formation layer and its mechanical properties were measured and evaluated. Then, the impact of the elements such as FP that can be dissolved to create solid solution was clarified, and necessary data on oxide-based fuel debris characteristics were compiled. Furthermore, in order to evaluate element behavior like FP elements of molten core concrete interaction (MCCI) products, simulated MCCI products in homogeneous melting were produced by arc melting. To evaluate the formation layer, structure and mechanical properties, simulated MCCI products, having heterogeneous (layered) structure under temperature gradient through locally focused radiation melting, were also produced to clarify the characteristics of the behavior of FP elements (Fig. 2).

Test focusing on the hydrous/dry properties that impact on hydrogen generation at the time of collection/storage of fuel debris, was conducted using UO_2, ZrO_2 and cement paste. Additionally, property change due to oxidizing velocity/oxidation of fuel debris under the conditions of the atmosphere, which easily progress oxidation, was evaluated by using simulated MOX debris.

b- Comparison with TMI-2 debris

In order to compare and examine the data between the characteristics of the obtained fuel debris mentioned in the above 2.a and the characteristics of fuel debris collected from the Three Mile Island Unit 2 (TMI-2) accident, metal structure was observed, and Vickers hardness was measured using three kinds of TMI-2 debris including accumulated debris on the lower head (Fig. 3). In addition, it is confirmed that TMI-2 debris in the pool can be melted through alkali fusion technique and ammonium salt fusion technique that are considered to be effective melting methods for fuel debris.

c- Evaluation on unevenness of properties

In cooperation with France's Alternative Energies and Atomic Commission (CEA), we continued to collect data on the characteristics of chemical compositions, hardness, and porosity by using MCCI test products that CEA had produced in the past. Furthermore, metal ceramic solidified products containing UO_2 were produced in cooperation with National Nuclear Center (NNC) of the Republic of Kazakhstan (Fig. 4),obtaining knowledge on hardness, interfacial morphology between metals and ceramics and particle distribution, which contribute to criticality control and development of fuel debris retrieval equipment.

3 Development of elemental technologies for fuel debris analysis

In regards to analysis work of fuel debris, a work flow from sample collection to analysis was examined, and a development plan was formulated to extract factors of technological development. In addition to the study for quantitative analysis methods of element through ICP-AES (Inductively Coupled Plasma-Atomic Emission Spectrometry) for fuel debris that was melted through alkali fusion technique, the effect of spectral interference and ionization interference by a coexistent element was evaluated. A transport study including fuel debris derived from MOX fuel was performed.

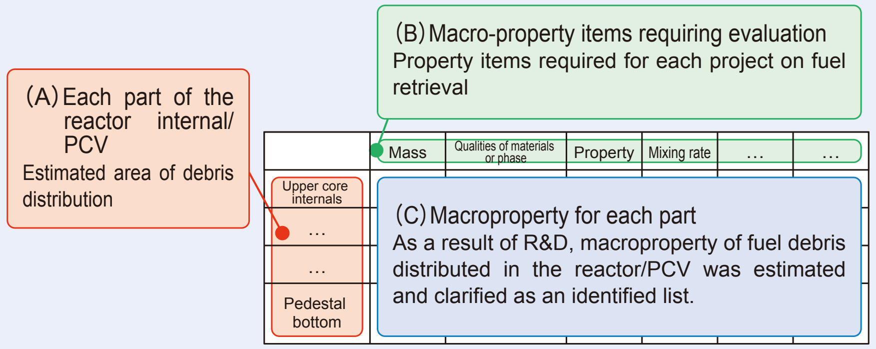


Fig. 1: Diagram of fuel debris characterization (sectional)

Based on the needs for each project on the fuel debris retrieval, required property item (B) was set up. Items based on the property data for estimated area of debris distribution (A) will be completed. Property data is estimated from the data obtained in experiment, thermodynamic analysis results, existent information, etc.

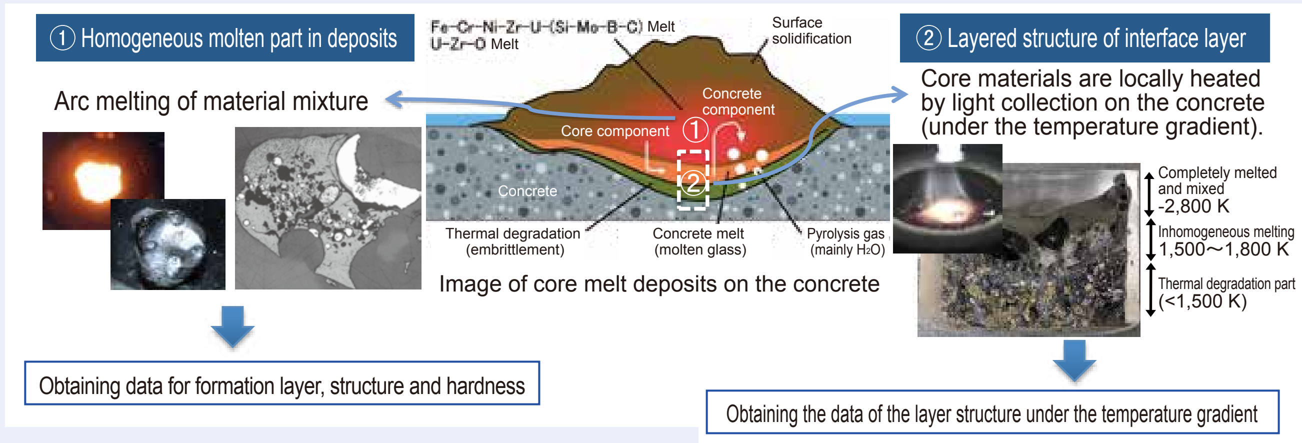


Fig. 2: Production method for two kinds of simulated MCCI products according to the test purpose

MCCI products are generated from reaction of molten core fallen to the RCV bottom and concrete of the reactor. The property data of formation layer (Fig.①) produced the simulated MCCI product through homogeneous melting by using arc melting and obtained the data such as a structure and hardness. In addition, the data of inhomogeneous layer structure under the temperature gradient (Fig.②) produced the simulated MCCI product by the local collecting/heating and obtained the behavior data such as Gd and FP elements.

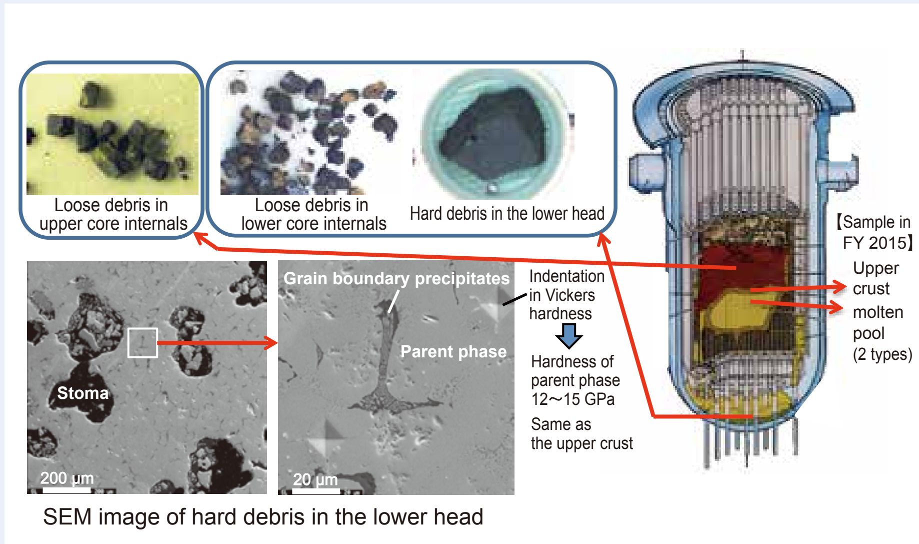


Fig. 3: SEM observation results of TMI-2 debris appearance/ cross-section

Three kinds of debris among TMI-2 debris stored in JAEA; appearance of loose debris in upper core internals, loose debris in lower core internals and hard debris in the lower head were observed, their hardness measured through SEM/EPMA analysis. According to the results of SEM/EPMA analysis, it was verified that parent phase of the lower head debris was mainly formed cubic (U, Zr, Fe, Cr, O_2), Fe-,Cr-,Al-,Ni- and O-based oxide was deposited on the grain boundary and hardness was almost same as the upper crust.

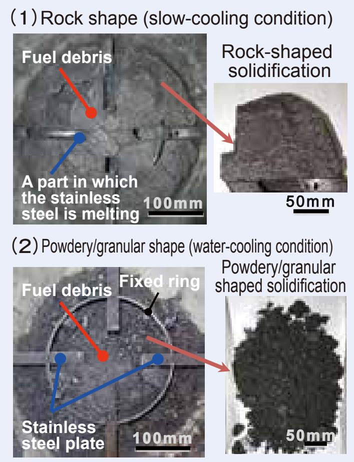


Fig. 4: Production of metal ceramics melt-solidification using UO_2 in the NNC, Kazakhstan.

Large-scale melt solidification test for UO_2+Zr+B_4C, which simulated fuel debris components in the Fukushima Daiichi NPS, was conducted in NNC Kazakhstan. Rock-shaped melt solidification, mixing with metal and ceramics, and powdery/granular solidification were firstly produced. In addition, the obtained physical property data will be utilized for the development of device and the criticality control for fuel debris retrieval.

Future Developments

We continue our international cooperative efforts with CEA to evaluate the characteristics of MCCI products and expand the characteristics data on MCCI products for which we have little information. Hydrous/dry properties of fuel debris required for the development of technologies for the collection, transfer and storage are further evaluated by tests and investigation, and property data of fuel debris related to storage canister are continuously compiled until the end of FY 2016. Furthermore, analysis elements technology for fuel debris will be developed based on the development plan formulated in FY 2015.

R&D for Treatment and Disposal of 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 to be implemented.

Aims

Technology for safe disposal and treatment of accident-generated solid wastes will be developed through a study on a series of storage management strategies for main radioactive wastes (waste management stream), analysis of wastes, characterization of wastes through inventory assessment based on waste product analysis, fundamental test of treatments, study of long-term storage strategy for secondary wastes stemming from treatment of contaminated water, understanding of disposal concept and the study of disposal classification for accident-generated waste.

Main Achievements and Approaches

1 Integration of R&D achievement (Suggestion of basic concepts on disposal/treatment)

After gathering the current knowledge and information for the waste management stream, change of the amount of materials and inventory for waste management stream will be examined in accordance with the waste management stream and evaluation items at a turning point. In addition, information management tool to support R&D was studied; and the relationship between the data and workflow about characterization, disposal and treatment was organized. Furthermore, we participated in the investigation committee by a specialist group established by OECD/NEA and cooperated to organize a report on the management of accident-generated wastes.

2 Characterization

Along with the formulation of a mid-and-long-term plan on waste analysis, debris, secondary waste stemming from treatment of contaminated water treatment and contaminated water were transported to an off-site facility, then radioactivity analysis was carried out. Results showed that the slurry generated from the Advanced Liquid Processing System (ALPS) mainly consists of ^{90}Sr and also includes Pu (Fig. 1). Analysis model for inventory assessment was improved to reduce uncertainty according to the analysis result. Waste products generated from the secondary cesium adsorption apparatus were evaluated using the analysis data such as treated water.

3 Study on treatment of radioactive wastes treatment and long-term storage methods

In order to narrow down the packaging technologies for radioactive waste, flow of technology assessment and assessment requirements were studied and examined. In addition, applicability tests on cement solidification for secondary waste stemming from the treatment of contaminated water such as the slurry generated from ALPS and absorbent, geopolymer solidification, sintering, and compaction (Fig. 2) were conducted while selection requirements for the treatment equipment were examined.

In regards to the long-term storage of cesium absorption vessels, a measurement test for the distribution of Cesium absorption using full-scale absorption vessel of the secondary cesium absorption apparatus (Fig. 3) was conducted, and applicability for absorption code of nuclides was verified based on the test results.

4 Study on disposal of the waste

With the establishment of a provisional disposal classification based on the inventory data setting for safety assessment of the waste, safety assessment techniques (scenarios, models, parameters and analysis case) were reviewed (Fig. 4). As safety assessment for the provisional disposal classification was being conducted, information for safety improvement was also examined.

Future Developments

Aside from the continuous analyzing of accident-generated waste characteristics, inventory data setting will also be reviewed. In regards to the study of long-term storage strategies, processing equipment for the stabilization of slurry generated from ALPS will be selected and studied for implementation on-site.

A plan to expand the technical information and data on fundamental tests for the processing study is being formulated, and treatment and disposal technologies will be evaluated.

Candidate methods of applicable disposal concept and safety evaluation will be narrowed down for the study of waste disposal. The above results for processing and disposal methods will be reflected to further promote waste management.

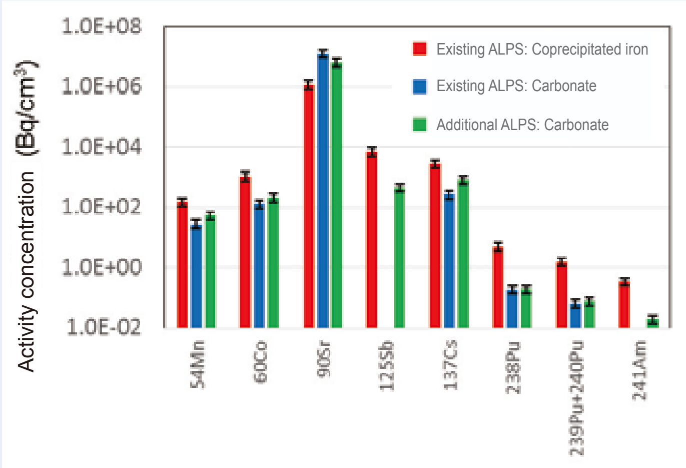


Fig. 1: Analysis data of radioactive waste (slurry) generated by ALPS operation

Fig. 2: Pressure filtering test device to examine the dehydration treatment for the slurry from ALPS

Fig. 3: Appearance of a full-scale absorber vessel to be used for various tests for absorbing, etc.

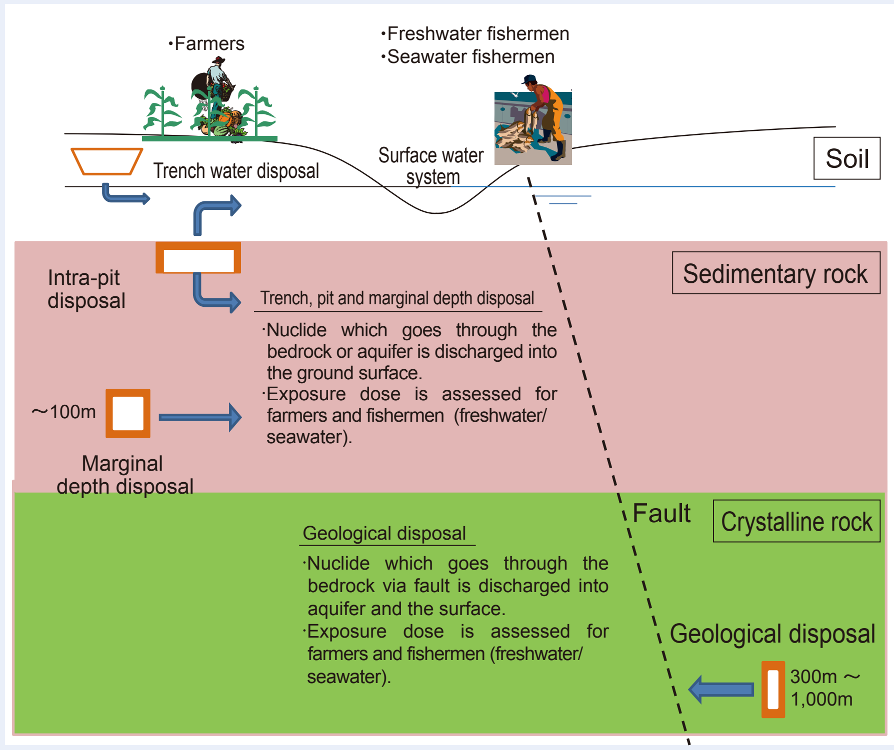


Fig. 4: Conceptual model for a transition of the groundwater

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

Reliability Evaluation of Technologies for Remotely Operated Decontamination Inside Reactor Buildings

Background

The decontamination work was firstly conducted for low places on the first floor of the reactor building at the Fukushima Daiichi NPS. On the other hand, according to the investigation of the contamination level in the reactor building, the radiation dose in high places (duct, cable tray, piping, etc.) is the highest – around 70%. Reduction of radiation dose in high places is expected.

Aims

Upon conducting the decontamination work for high places on the first floor of the southwest area in Unit 3 at the Fukushima Daiichi NPS, dry ice blast decontamination equipment, which was developed in FY 2014-2015 for high places, will be mounted to the decontamination work carts and support carts, used since FY 2013 for low places. Dependable capacity and decontamination performance for the equipment will be confirmed as well.

Main Achievements and Approaches

1 Combination test on dry ice blast decontamination equipment for high/low places

In addition to the maintenance of dry ice blast decontamination equipment for low places which had been stored at the Fukushima Daiichi NPS, a combination test for high places will be conducted. Result confirmed that there will be no problem on the decontamination of high places in the reactor building at Unit 3 (Photo 1).

2 Measurement of radiation dose rate for high places

Using a dry ice blast decontamination equipment for high places that is connected to support cart, in which a dry ice blast decontamination equipment for low places was mounted, the radiation dose (β ray and γ ray) was measured in 19 areas in the southwest of the first floor in the reactor building at Unit 3 from December 23, 2015 to January 22, 2016 (Photo 2).

3 Decontamination for high places (suction and dry ice blast decontamination)

Based on the measurement results on the high radiation dose rate, the decontamination (suction/dry ice blast) was performed in priority to high radiation rate area (β ray) in the southwest of the first floor in the reactor building at Unit 3.

Future Developments

We will evaluate and study the results of the decontamination work and consult with the Tokyo Electric Power (TEPCO) for future on-site implementation.

Photo 1: Scene of a combination test for the decontamination equipment for high places and low places

Photo 2: Scene of radiation dose rate measurement for high places (The decontamination equipment comprises two of the equipment for high places (left) and the equipment for low places (right).

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Main Research Results Announced/Published in 2015

No.	Presented at/by	Date	Details
1	Radiation Science Society	Apr. 17, 2015	Reactor imaging through scattering method of cosmic ray muon
2	Young Researchers Conference, Atomic Energy Society of Japan	Apr. 17, 2015	Reaction products at interface between molten core and concrete
3	Japanese-German Symposium on Technological and Educational Resources for the Decommissioning of Nuclear Facilities	Apr. 21, 2015	R&D on Treatment and Disposal of Radioactive Waste resulting from Accident at Fukushima Daiichi NPS
4	2015 International Congress on Advances in Nuclear Power Plants (ICAPP 2015)	May 3, 2015	Improvement of Molten Core-Concrete Interaction Model of the Debris Spreading Analysis Module in the SAMPSON Code
5	Non-Destructive Inspection International Conference	May 19, 2015	Fukushima Daiichi Muon Imaging
6	23rd International Conference on Nuclear Engineering (ICONE)	May 19, 2015	Fukushima Daiichi Muon Imaging
7	23rd International Conference on Nuclear Engineering (ICONE)	May 19, 2015	Mechanical properties of fuel debris for defueling toward decommissioning
8	23rd International Conference on Nuclear Engineering (ICONE)	May 19, 2015	Study of Treatment Scenario for Fuel Debris Removed from Fukushima Daiichi NPS
9	5th International Conference on the Chemistry and Physics of the Transactinide Elements	May 25, 2015	Current Status and JAEA's Challenges for Fukushima Dai-ichi Nucleaer Power Stations
10	EPRI D&D Workshop and International Low-level Waste Conference and Exhibition	Jun. 14, 2015	Fukushima Inspection Manipulator (FIM)
11	Water Chemistry Division, Atomic Energy Society of Japan	Jun. 15, 2015	Environmental remediation at the Fukushima Daiichi NPS and development of technologies
12	National Symposium on Power and Energy Systems	Jun. 18, 2015	Approaches and achievements for remotely operated technologies
13	Summer School in Radiation Detection and Measurements	Jun. 18, 2015	Issues of Radiation Measurement for Fukushima-Daiichi Restoration Activities-Visualization and Non-Invasive Inspection
14	Nuclear Fuel Division, Materials Science and Technology Division, Water Chemistry Division, Atomic Energy Society of Japan	Jul. 9, 2015	Data acquisition for estimating property of fuel debris retrieval
15	Annual Conference, Japan Society of Maintenology	Jul. 14, 2015	Development of the decontaminating equipment for low places, Suction / blast decontamination equipment, Dry ice blast decontamination equipment, High pressure water jet decontamination equipment
16	Annual Conference, Japan Society of Maintenology	Jul. 14, 2015	Investigation of the reactor through the Muon transmission method at the Fukushima Daiichi NPS
17	Annual Conference, Japan Society of Maintenology	Jul. 14, 2015	Improvement for severe accident code MAAP and accident analysis
18	Annual Conference, Japan Society of Maintenology	Jul. 14, 2015	Overview of the research on fuel debris characterization
19	Back-end Summer Seminar, Atomic Energy Society of Japan	Aug. 5, 2015	R&D for treatment and disposal of radioactive waste
20	17th International Conference on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors	Aug. 9-13, 2015	Research results for corrosion evaluation technology contributing evaluation for the integrity of the RPV/PCV structure
21	Japan-Korea Joint Summer School, Atomic Energy Society of Japan	Aug.t 16, 2015	Overview of development of criticality control technology for fuel retrieval
22	imPACT Tough Robotics Challenge	Aug. 23, 2015	Status of development of robotics technology for decommissioning
23	10th International Conference NUCLEAR AND RADIATION PHYSICS 2015	Sep. 8, 2015	Characterization of fuel debris by large-scale simulated debris examination for Fukushima daiichi nuclear Power stations
24	TIA-ACCELERATE Symposium	Sep. 9, 2015	Development of non-destructive detection system of large-scale structure through cosmic-ray muons
25	2015 Fall Meeting, Atomic Energy Society of Japan	Sep. 9-11, 2015	Evaluation of long-term structural integrity of fuel assemblies removed from the spent fuel pool
26	2015 Fall Meeting, Atomic Energy Society of Japan	Sep. 9-11, 2015	Verification test of decontamination equipment for high places: Suction/blast decontamination equipment, Dry ice blast decontamination equipment, High pressure water jet decontamination equipment
27	2015 Fall Meeting, Atomic Energy Society of Japan	Sep. 9-11, 2015	Research for identifying conditions inside of the reactor through the MAAP severe accident analysis code
28	2015 Fall Meeting, Atomic Energy Society of Japan	Sep. 9-11, 2015	Impact assessment on seawater for identifying conditions inside of the reactor during severe accident
29	2015 Fall Meeting, Atomic Energy Society of Japan	Sep. 9-11, 2015	Development of jet breakup phenomenon prediction method for the complicated structure of the BWR lower plenum

Main Research Results Announced/Published in 2015

No.	Presented at/by	Date	Details
30	2015 Fall Meeting, Atomic Energy Society of Japan	Sep. 9-11, 2015	Development of criticality control technology for fuel debris
31	2015 Fall Meeting, Atomic Energy Society of Japan	Sep. 9-11, 2015	Investigation of prosperity for the secondary wastes generated from contaminated water treatment at the accident at the Fukushima Daiichi NPS
32	2015 Fall Meeting, Atomic Energy Society of Japan	Sep. 9-11, 2015	Study of analysis method for the secondary wastes generated from multi-nuclide removal system
33	2015 Fall Meeting, Atomic Energy Society of Japan	Sep. 9-11, 2015	Development of inventory evaluation method for the wastes generated at the accident at the Fukushima Daiichi NPS
34	2015 Fall Meeting, Atomic Energy Society of Japan	Sep. 9-11, 2015	Study of conditioning technology for the secondary wastes generated from contaminated water treatment
35	2015 Fall Meeting, Atomic Energy Society of Japan	Sep. 9-11, 2015	Structural integrity of the Reactor Pressure Vessel / Primary Containment Vessel
36	2015 Fall Meeting, Atomic Energy Society of Japan	Sep. 9-11, 2015	Characterization of fuel debris
37	TopFuel 2015	Sep. 13-17, 2015	Evaluation of long-term integrity structural integrity of fuel assemblies removed from the spent fuel pool
38	Open symposium, Japan Energy Association	Sep. 17, 2015	High pressure water jet decontamination equipment / Investigation of the torus room in Unit 2 / Investigation inside of the Primary Containment Vessel in Unit 1
39	Global2015	Sep. 20-24, 2015	Comparative study of Sr adsorbents for radioactive contaminated water on severe accident
40	Safety seminar for light water reactor, Tohoku University	Sep. 28, 2015	Development of robotics technology for decommissioning and the status of application on site
41	Institute of Nuclear Materials Management	Sep. 29, 2015	Application of Muon Transmission method to reactor/ nuclear security
42	Takasaki Advanced Radiation Research Symposium	Oct. 8-9, 2015	Evaluation of hydrogen gas for inorganic solidification which is simulated wastes generated from ALPS during gamma radiation
43	Society of Muon and Meson Science of Japan	Autumn 2015	Imaging measurement for the reactor through muon transmission method
44	Academic meeting, Laser Society of Japan	Oct. 15, 2015	Application of laser for fuel debris retrieval at the Fukushima Daiichi NPS
45	Workshop on Radiation Measurement	Oct. 17, 2015	Development of visualization technique inside the reactor using cosmic-ray muon
46	American Physical Society	Oct. 29, 2015	Muon Reactor Imaging
47	Workshop on the environment and materials, Japan Society of Corrosion Engineering	Nov. 26-28, 2015	Study results of corrosion evaluation technology contributing structural integrity evaluation of the RPV/PCV
48	Annual Meeting of the International Network of Laboratories for Nuclear Waste Characterization (LABONET)	Nov. 10-13, 2015	Inventory evaluation based on the analysis data of contaminated water generated from the contaminated water treatment system at the Fukushima Daiichi NPS
49	Workshop on dose reduction cases for decommissioning work at the Fukushima Daiichi NPS	Nov. 10, 2015	Radiation dose reduction due to the decommissioning work at the reactor building at Unit 2, Fukushima Daiichi NPS
50	Japan Joint Automatic Control Conference	Nov. 14, 2015	R&D on technology for improving remotely operability of robots
51	International Robot Exhibition	Dec. 2-5, 2015	Decommissioning system for high places, “Super Giraffe”
52	Radiation Measurement Forum in Fukushima	Dec. 7, 2015	Announcement of the investigation results at the southern area on the 1st floor in Unit 1, Fukushima Daiichi NPS
53	Human Resource Development Seminar in Kobe	Dec. 12, 2015	A robot working for decommissioning at the Fukushima Daiichi NPS
54	Radiation Measurement Forum in Fukushima	Dec. 15, 2015	Development of radiation measurement technologies by HITACHI that support the restoration and revitalization of Fukushima
55	Power-generating technology, Japan Power Engineering and Inspection Corporation	Dec. 15, 2015	Status of development of a robotic technology for decommissioning and on-site application
56	Basic Education Series Seminar, Nuclear Workshop, Tokyo Institute of Technology	Dec. 22, 2015	Development of decommissioning equipment for high places, Dose measurement by gamma camera, Development and investigation robots: water swimming robot / floor surface traveling robot / Shape change robot
57	2015 International Chemical Congress of Pacific Basin Societies (Pacifichem)	Dec. 2015	Basic study on the solidification of wastes generated from the contaminated water treatment facility at the Fukushima Daiichi NPS
58	Pioneering R&D Committee, Japan Society for the Promotion of Science	Jan. 15, 2016	Development of robotics technology for decommissioning work and on-site application
59	Session on Structural performance of the reactor building, Decommissioning Investigation Committee, Atomic Energy Society of Japan	Jan. 15, 2016	Progress report of the integrity evaluation test / Analysis of 1 FRPV pedestal reinforced concrete

Main Research Results Announced/Published in 2015

No.	Presented at/by	Date	Details
60	Nuclear Technologies Seminar, Japan Power Engineering and Inspection Corporation	Jan. 28, 2016	Development of robotic technology for decommissioning and on-site application
61	Briefing session on Fukushima, Japan Atomic Energy Agency Sector of Fukushima Research and Development	Jan. 27, 2016	Fuel debris retrieval and the further approach – R&D policy for fuel debris retrieval and response to the policy – and others
62	Society of Engineering and Research in Nuclear System	Feb. 16, 2016	Overview of investigation inside the Reactor Containment Vessel
63	National Meeting, The Institute of Electrical Engineers of Japan	Mar. 16-18, 2016	Nuclear technology and its trends for decommissioning of the Fukushima Daiichi NPS – Technology trends for measurement/analysis technology
64	National Meeting, The Institute of Electrical Engineers of Japan	Mar. 16-18, 2016	Development of robotic technology for decommissioning and on-site application
65	Annual Spring Meeting 2016, Atomic Energy Society of Japan	Mar. 26-28, 2016	Improvement of robustness for the muon scattering method based on EM algorithm
66	Annual Spring Meeting 2016, Atomic Energy Society of Japan	Mar. 26-28, 2016	Study on long-term storage method for spent zeolite vessels (Reports no.10 & 11)
67	Annual Spring Meeting 2016, Atomic Energy Society of Japan	Mar. 26-28, 2016	Investigation of properties for secondary waste generated from contaminated water treatment at the Fukushima Daiichi NPS accident
68	Annual Spring Meeting 2016, Atomic Energy Society of Japan	Mar. 26-28, 2016	Hydrogen generation due to radiolysis of carbonate slurry generated from the multi-nuclide removal system
69	Annual Spring Meeting 2016, Atomic Energy Society of Japan	Mar. 26-28, 2016	Development of the eroded concrete advection-diffusion model due to MCCI analysis

List of Joint Researches/Contract Researches

No.	Project Name	Category	Subject	Partner	Period
1	Development of technology for the detection of fuel debris in the reactor	Contract research	Measurement of fuel debris in the reactor through muon transmission method at the Fukushima Daiichi NPS	High Energy Accelerator Research Organization (KEK)	Jun. 2014- Dec. 2015
2	Development of technology for detection of fuel debris in the reactor	Contract research	Development of supporting algorithm for identifying fuel debris through muon transmission method	Los Alamos National Laboratory (LANL), U.S.A	Jul. 2014- Nov. 2015
3	R&D on treatment/disposal of solid radioactive waste	Contract research	Research on estimation/assessment technology of radioactive waste inventory	Central Research Institute of Electric Power Industry	May 22, 2015- Jan. 29, 2016
4	R&D on treatment/disposal of solid radioactive waste	Contract research	Study on properties of criticality of accident-generated waste	University of California, Berkeley, U.S.A.	Sep. 30, 2015- Jan. 29, 2016
5	R&D on treatment/disposal of solid radioactive waste	Contract research	R&D on radioactive waste management methods using gamma ray measurement	Tokyo Institute of Technology	Nov. 26, 2015- Feb. 29, 2016
6	Characterization of fuel debris	Contract research	Physical evaluation of melt-solidification fuel debris (2)	Central Research Institute of Electric Power Industry	Oct. 2015- Feb. 2016
7	Development of the remotely operated decontamination equipment	Contract research	Research on camera calibration method for simulated overhead image generation system installed in a remotely operated robot	University of Tokyo	Dec. 15, 2014- Aug. 31, 2015
8	Development of the remotely operated decontamination equipment	Contract research	Research on technology on robots for identifying three dimensions	University of Tsukuba	Dec. 15, 2014- Aug. 31, 2015
9	Development of the remotely operated decontamination equipment	Contract research	Research on improvement for remotely operability of manipulator	Kobe University	Dec. 15, 2014- Aug. 31, 2015
10	Development of the remotely operated decontamination equipment	Contract research	Research on coordinate movement control of multiple carts	Shibaura Institute of Technology	Jan. 20, 2016- Mar. 9, 2016

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 for high places 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 (Transport cart, Support cart, 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 position measuring 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 leakage stoppage technology for leakage points inside the Primary Containment Vessel	Heating/Feed water equipment
16	Full-scale test of repair and water leakage stoppage technology for leakage points inside the Primary Containment Vessel	Turbid water treatment equipment
17	Full-scale test of repair and water leakage stoppage technology for leakage points inside the Primary Containment Vessel	Work floor
18	Full-scale test of repair and water leakage stoppage technology for leakage points inside the Primary Containment Vessel	Mock-up transfer rail
19	Full-scale test of repair and water leakage stoppage technology for leakage points inside the Primary Containment Vessel	Full-scale mock-up
20	Development of technology for investigation inside the Primary Containment Vessel	B1 investigation device
21	Development of technology for investigation inside the Primary Containment Vessel	Dispersion prevention equipment for B1 investigation device
22	Development of technology for investigation inside the Primary Containment Vessel	Subsidiary equipment for B1 investigation device
23	Development of technology for investigation inside the Primary Containment Vessel	B1 investigation device Simulated device for mock-up
24	Development of technology for investigation inside the Primary Containment Vessel	Shielding block removal equipment
25	Development of technology for investigation inside the Primary Containment Vessel	Fuel debris measurement equipment
26	Development of technology for investigation inside the Primary Containment Vessel	Fuel debris measurement equipment Equipment for an elemental test
27	Development of technology for investigation inside the Primary Containment Vessel	A2 investigation equipment (Including chambers and guide pipes)
28	Development of technology for investigation inside the Primary Containment Vessel	Set of X-6 penetration hole boring device
29	Development of technology for investigation inside the Primary Containment Vessel	Set of previously confirming equipment inside the penetration
30	Development of technology for investigation inside the Primary Containment Vessel	Set of deposits removal equipment (Including chambers)
31	Development of technology for investigation inside the Primary Containment Vessel	Set of subsidiary equipment for A2 investigation

Main R&D Installations / Equipment

Over 1 million Yen

No.	Project Name	Details
32	Development of technology for investigation inside the Primary Containment Vessel	Set of previously confirming equipment inside the pedestal
33	Development of technology for investigation inside the Primary Containment Vessel	Set of simulated mock-up structure in the PCV
34	Development of technology for investigation inside the Primary Containment Vessel	Set of elemental testing device for A3 investigation
35	Development of technology for investigation inside the Primary Containment Vessel	Set of relevant equipment for a hatch opening device
36	Development of technology for detection of fuel debris in the reactor	Shielding materials of the measurement equipment for transmission method
37	Development of technology for detection of fuel debris in the reactor	Small measurement equipment for transmission method
38	Development of technology for detection of fuel debris in the reactor	Small muon tracking device system for scattering method
39	Development of technology for detection of fuel debris in the reactor	Muon tracking device for scattering method to be used at the Fukushima Daiichi NPS
40	Fuel debris characterization	Large capacity of thermogravimetric balance and simultaneous thermal analysis equipment
41	Fuel debris characterization	Piezoelectric crystal four-component cutting dynamometer
42	Fuel debris characterization	Elemental analysis system for SEM
43	Fuel debris characterization	Automatic hydraulic embedding machine
44	Fuel debris characterization	Inverted metallurgical microscope
45	Fuel debris characterization	Carbon coater
46	Fuel debris characterization	Vacuum displacement arc melting furnace
47	Fuel debris characterization	Fuel debris compression test device
48	Fuel debris characterization	Fuel debris sonic speed measuring device
49	Fuel debris characterization	Metallographic image analysis device
50	Fuel debris characterization	Dynamic Micro Hardness tester
51	Fuel debris characterization	Simultaneous thermal analysis system
52	Fuel debris characterization	Gas piping valve heater
53	Fuel debris characterization	Sample cutting machine
54	Fuel debris characterization	Sample polisher
55	Fuel debris characterization	Core collecting device
56	R&D on treatment and disposal of solid waste	Chamber for alpha nuclide analysis
57	R&D on treatment and disposal of solid waste	Digital spectrometer
58	R&D on treatment and disposal of solid waste	Gamma-efficiency calculation program
59	R&D on treatment and disposal of solid waste	Aerosol transition observing device