IRID International Research Institute for Nuclear Decommissioning

VEGETABLE OIL INK
Printed on recycled paper.
GREEN PRINTING JFPI
P-B10221
Published: March 2015

IRID Annual Research Report 2014

IRID International Research Institute for Nuclear Decommissioning

Greetings

This report presents a collection of the results of the Research & Development (R&D) projects which the International Research Institute for Nuclear Decommissioning (IRID) conducted during FY2014. I would like to take this opportunity to sincerely thank all those who extend their understanding and support to IRID's activities on a daily basis.

Since its establishment, IRID has been addressing the pressing R&D challenges presented by the decommissioning of the Tokyo Electric Power Company's (TEPCO) Fukushima Daiichi Nuclear Power Station (NPS) based on the 'Mid-and-Long-Term Roadmap towards the Decommissioning of TEPCO's Fukushima Daiichi NPS Units 1-4' ('Mid-and-Long-Term Roadmap'). IRID completed 17 subsidized projects and one in-house R&D project during FY 2014 as part of this effort.

IRID has been involved with the three following R&D project categories, promoting interlinkage between projects in an integrated and comprehensive manner, and collaborating with worldwide organizations:

(1) Fuel Removal from Spent Fuel Pool
(2) Preparation of Fuel Debris Retrieval
(3) Treatment and Disposal of Radioactive Waste

As an example, IRID developed a remotely operated robot which enabled us to verify the effectiveness of techniques for conducting decontamination work and surveying contaminated water leakage points inside reactor buildings at the Fukushima Daiichi NPS. Furthermore, in order to determine the location and state of the debris inside the reactors, IRID has been developing devices to use cosmic ray muon detection technology and robots to look inside the primary containment vessel.

In order to efficiently and steadily undertake the unprecedented and extremely difficult – even by global standards – of decommissioning the damaged Fukushima Daiichi Reactors, IRID is ready to play its part in the area of R&D, working in cooperation with the government, the Nuclear Damage Compensation and Decommissioning Facilitation Corporation, and TEPCO.

IRID will continue to steadily push forward with R&D efforts without losing a sense of urgency and with a positive attitude, we will strive to make visible progress in decommissioning efforts, enabling those people in Fukushima forced to live as evacuees to return home at the earliest opportunity, and ensure peace of mind for the society as a whole.

I strongly hope that this report helps a wider audience gain a better understanding of IRID's R&D achievements, and ask you for your continuous guidance and support in our activities.

March 2015

Hirofumi Kenda
President, International Research Institute for Nuclear Decommissioning

Contents

Main R&D Achievements: Subsidized Projects

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

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

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

Main Study Results: In-house R&D Activities

Others – Data Sheet

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

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

Background

Fuel assemblies in spent fuel pools at Units 1-4 at the Fukushima Nuclear Power Station (NPS) have been stored in an environment with water quality different from normal conditions due to the injection of seawater and addition of rubble. 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 determine if fuel assemblies retrieved from the spent fuel pools in 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. Through these tests, the structural integrity of fuel assembly components during long-term storage stored in a common pool can be assessed.

Main Achievements

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

a. Development of technology to assess long-term integrity

As a large load is placed upon the upper and lower sections of fuel assemblies when they are handled, the integrity of these parts is extremely important. To make sure no problems would arise due to the spent fuel pool water quality or rubble, immersion tests (up to for 8,000 hours) and strength tests using a test specimen were conducted, and the integrity of the parts was evaluated. Results showed that there was no corrosion that may affect fuel assembly integrity or degradation due to mechanical stress in the threaded sections and cladding subject to evaluation (Figure 1).

b. Survey of condition of spent fuel stored in the common pool

In order to confirm corrosion conditions under long term storage in the common pool, regarding fuel assemblies retrieved from the spent fuel pool in Fukushima Daiichi NPS Unit 4, surface observations of the fuel assembly, observations of the inner threaded portion surface in crevice areas and measurement of the thickness of the cladding oxide layer were performed. Results of surface observations showed that while a white deposit had formed on spent fuel, there were no issues with external appearance (Figure 2). Furthermore, corrosion was found to have not occurred on the inner surface of the threaded section, and measurements of the thickness of the cladding oxide layer showed that when compared with oxide layer thickness in existing fuel stored in the common pool, no increase in thickness had taken place.

c. Evaluation of integrity of fuel in dry storage

Tests were conducted to assess hydride precipitation behavior inside cladding material when scratches due to rubble exists and the effect on fuel integrity of the moisture contained in rubble which has penetrated crevice areas under the assumption that spent fuel will be stored in a dry environment. Test results showed that these had only a minimal impact on fuel assemblies in dry storage. Investigations will continue on the aging of materials in the assessment of the integrity of fuel in dry storage.

2. Basic tests related to long-term structural integrity

a. Evaluation of seawater component transfer behavior on fuel components

In order to assess the seawater component transfer behavior on fuel components, tests were conducted using simulated scaling (crud) and a tracer. Results showed that the amount of seawater component transferring to the crud and material surface was minimal. In evaluating whether there is an uptake in seawater components to the oxide film that is formed on the fuel assembly cladding surface or not, confirmation tests that take the heat generated inside the fuel assemblies into account were conducted, and it was found that there was no significant uptake that occurred.

b. Evaluation of effect of corrosion derived from seawater or rubble components under radiation environment

The results of an evaluation of fuel material taken from new fuel assemblies that were in the Fukushima Daiichi NPS Unit 4 spent fuel pool showed no signs of corrosion or loss of integrity. Additionally, results of electrochemical tests conducted under radioactive conditions confirmed that in room temperature water that contains a chloride ion content of less than 100ppm, repassivation potential for crevice corrosion was greater than steady-state corrosion potential, and that crevice corrosion did not occur. (Figure 4) It was also confirmed that even after 1,500 hours of corrosion testing, no signs of corrosion were observed.

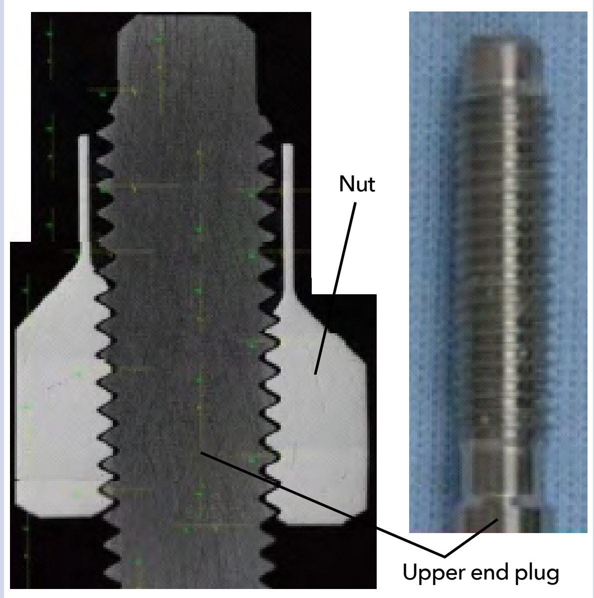


Figure 1 Results of cross-sectional (left) and external (right) observation of fuel assembly upper section test specimen (The test specimen was immersed for 1,000 hours at a temperature 90°C and a chloride ion concentration of 2,500ppm. After that, it was further immersed for 7,409 hours at a temperature 60°C and chloride ion concentration of 100ppm.)

This test simulated an environment in which concrete rubble was assumed to be presented. No corrosion was observed in upper-end plugs (zircaloy) and nuts (stainless steel).

Figure 2 Survey of condition of fuels stored in common pool
External observation (example) of spent fuel from Fukushima Daiichi NPS Unit 4 (upper tie plate of fuel mixed with rubble (burn-up=49.3GWd/t))

A white deposit was noted on the upper tie plate and lock nut surface, but there were no issues in appearance.

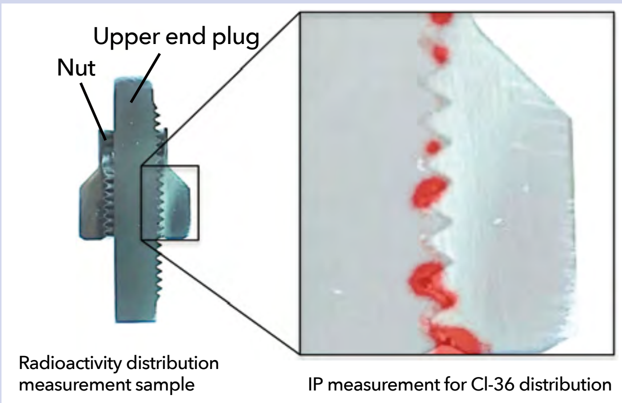


Figure 3 Distribution of Cl-36 in upper end plug screw structure (80°C, two-fold dilution of artificial seawater, after 50 hours immersion)

By immersing the specimen in artificial seawater with a two-fold dilution of chloride, it was confirmed that seawater components transfer to the screw structure, but concentration decreases afterwards as water quality improves.

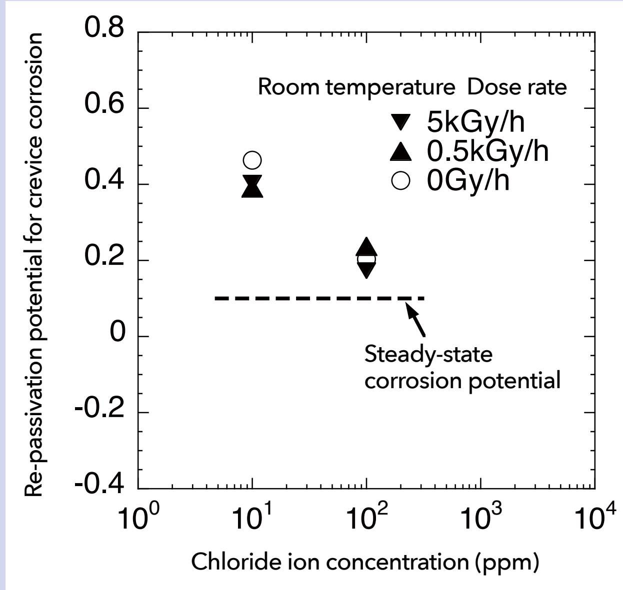


Figure 4 Relationship diagram between re-passivation potential for crevice corrosion in diluted artificial seawater and chloride ion concentration obtained through electrochemical test using crevice corrosion test specimen (testing using crevice corrosion specimen made of a combination of Zr and SUS304L).

In room temperature water with a chloride ion concentration of less than 100ppm and exposure to gamma radiation at 0-5kGy/h, it was confirmed that crevice corrosion would not occur as re-passivation potential for crevice corrosion was higher than steady-state corrosion potential.

Future Developments

In order to ensure that fuel assemblies retrieved from spent fuel storage pools in Fukushima Daiichi NPS Units 1-4 can be safely stored, material taken from spent fuel in Unit 4 will be transported to a "post-irradiation test facility" and studies of white deposits observed in surface observations and so on will be performed. Additionally, tests are planned to be performed on applicable fuel assemblies that take combined environmental factors into consideration in order to evaluate the integrity of fuel assembly components during dry storage.

Study of Methods to Process Damaged Fuel Removed from the Spent Fuel Pool

Background

In addition to having been contaminated with impurities such as seawater and concrete, spent fuel in Units 1-4 may have also been damaged due to falling debris. In order to determine how this damaged fuel is to be dealt with, it is necessary to confirm whether reprocessing the fuel is technically feasible or not as one of the treatment methods.

Aims

In addition to identifying the technical challenges regarding the reprocessing of damaged fuel, indicators must be developed to determine the feasibility of reprocessing the fuel. The aim of this project is to assess the effect on the chemical treatment process and so on, and extract and organize the expected impact at domestic reprocessing plants (RRP/TRP*).

* RRP: Rokkasho Reprocessing Plant; TRP: Tokai Reprocessing Plant

Main Achievements

1. Evaluation of corrosive influence of impure substances on re-processing equipment

Corrosion tests (immersion, electro chemistry) was conducted on materials in high-level concentrated waste liquid storage tanks and high-level liquid waste storage tanks, using simulated high-level liquid waste solutions that took into account impurities and fission products, etc. Under all conditions, results showed that while there was uniform corrosion in the form of intergranular corrosion, pitting was not observed (Figure 1 (1)). Furthermore, an increase in chloride ion concentration was accompanied by a decrease in corrosion (Figure 1 (2)).

2. Evaluation of in-process behavior of impurities

In order to understand the effect of the transition of impurities to uranium/plutonium (U/Pu) products and the impact of impurities on the extraction of U/Pu, solvent extraction testing took place using a simulated solution with added impurities and fission products in addition to a simulated solution to which U/Pu had been added. The results of these tests confirmed that impurities had a low distribution ratio of 10^{-2} -10^{-3} and they were hard to be extracted into the solvent, and that the distribution ratio of U/Pu was not dependent on the presence of impurities (Figure 2).

3. Evaluation of the influence of impurities on waste body

To determine the composition of waste solutions assumed to be handled at domestic reprocessing facilities (pressurized water reactor with 45,000 MWd/tU burn-up)*, a glass specimen was created from a formulation of powder materials with the major components of seawater and mortar added as impurities, and an evaluation of homogeneity and of glass property values was then performed. The result was that, under all conditions, there was no phase separation precipitation, and vitrification occurred, and there was no effect of impurities on the physical property data of the glass (Figure 3).

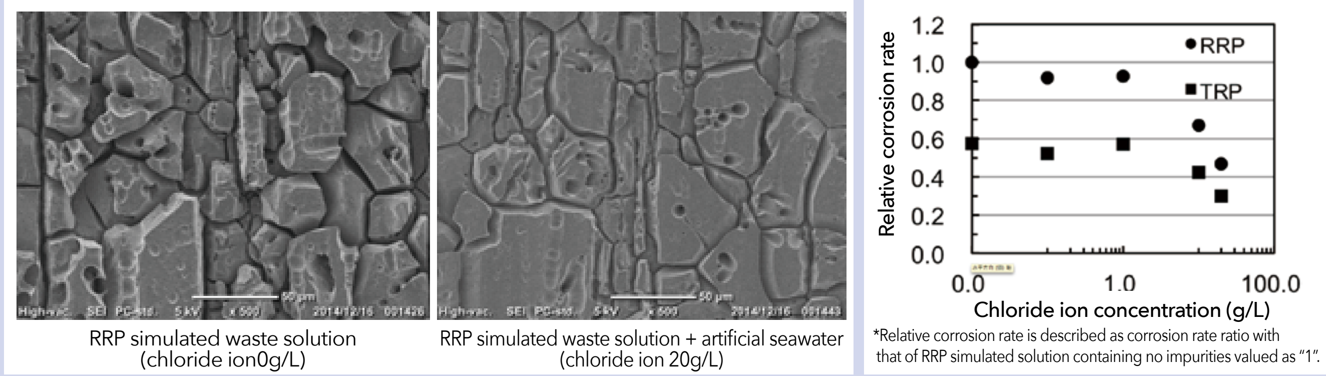
* This composition was used in order to make comparisons with previous tests. It was previously confirmed that there is no difference between the results of this test and in the case found at the Fukushima Daiichi NPS.

4. Identifying and outlining of other influences

Influences expected at reprocessing facilities at the time of processing damaged fuel were comprehensively identified and the presence or absence of required elements of research is clarified. The result was that there were no new research elements outside of the existing evaluation items; evaluating the handling of used fuel, evaluating the corrosive impact on reprocessing equipment due to impurities, evaluating in-process behavior, or waste impact assessment.

Future Developments

Through evaluation tests to date, it has been possible to obtain key data (if impurities will impact on the chemical treatment process, etc.) necessary to determine if reprocessing is feasible. The necessity of investigating issues related to the handling of damaged fuel and so on will be determined in light of the results of confirmation of the state of fuel in the spent fuel pool of Unit 3 that is planned for future retrieval.

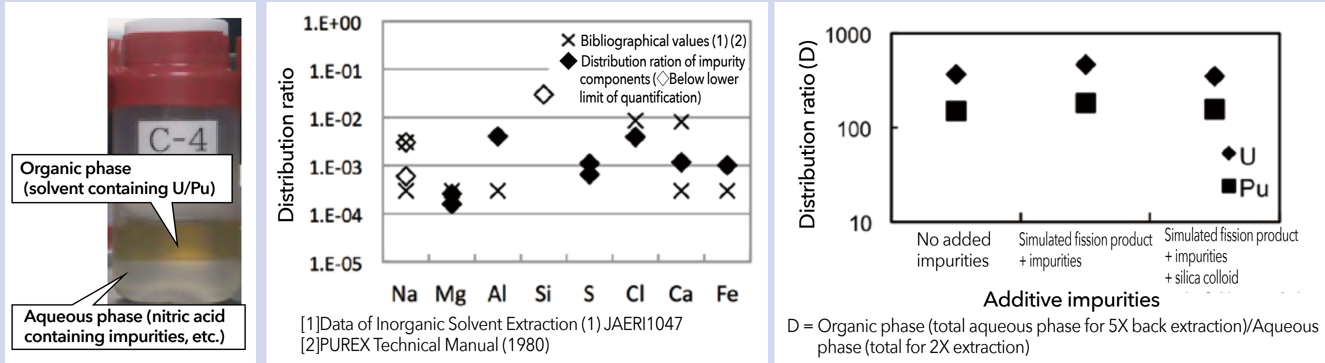


(1) Surface observations of immersion test piece

(2) Relative corrosion rate and chloride ion concentration (artificial seawater)

Figure 1 High-level liquid waste storage tank immersion test results (Immersion time: 960h)

In immersion tests of high-level liquid waste storage tank material (SUS316L) using a simulated solution to which impurities (artificial seawater, etc.) have been added in a simulated waste solution containing a simulated fission product component, etc., pitting due to chloride ions within impurities was not observed, while intergranular corrosion growth was shown to have occurred under conditions without the addition of impurities, as surface observations indicated. A decrease in the corrosion rate was also observed due to an increase in the chloride ion concentration, and it was determined that there was no acceleration in equipment corrosion due to the presence of impurities.



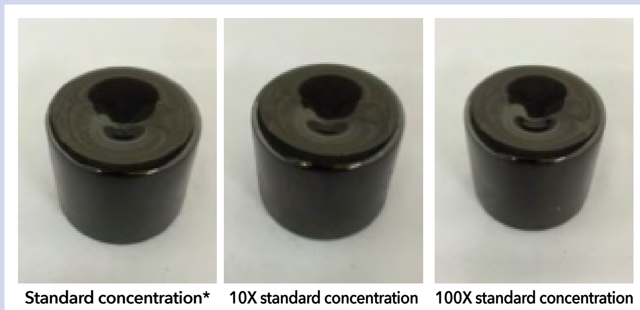
(1) Samples left to stand after extraction

(2) Distribution ratio of impurity components during coexistence state with simulated fission products

(3) Distribution ratio of U/Pu during coexistence state with impurity components

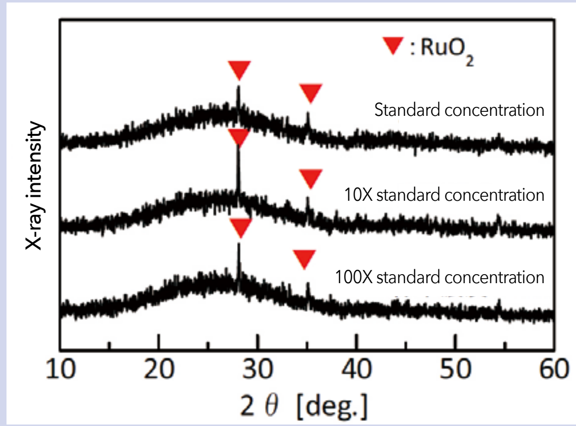
Figure 2 Outcome of evaluation of the in-process behavior of impurities

In solvent extraction testing using a simulated solution obtained by adding impurity components to simulated fission product components, the distribution ratio of impurities was of the order of a low 10^{-2} to 10^{-3}, making them difficult to extract from the solvent (30% TBP-n-dodecane) as the graph above shows. Furthermore, in solvent extraction testing using impurity components in a U/Pu solution and simulated solutions to which simulated fission product components had been added, the U/Pu distribution ratio confirms that there was no significant change in the presence/absence of impurities. Based on these findings, it was determined that there was no effect on the transition of impure substances to the U/Pu products or on extraction of U/Pu due to impurities.



(1) Visual observation

*Standard concentration: Estimated mix concentration to vitreous waste through conservative estimate of amount of adhered/entrained impurities (seawater/mortar components).



(2) X-ray diffraction (XRD) measurement results.

Figure 3 Evaluation of homogeneity results using glass specimen containing seawater/mortar components.

In an evaluation of homogeneity of glass specimens created adding major components of seawater and mortar to act as an impurity in a standard waste liquid composition, no precipitation of phase separation objects was visually observed and as with standard glass in XRD measurements, no other peaks aside from RuO_2 derived from standard waste liquid were observed.

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

Development of Technology for Remotely Operated Decontamination Inside Reactor Buildings

Background

In the lead-up to retrieving fuel debris from Units 1-3 at Fukushima Daiichi, various tasks are planned to take place inside the reactor buildings. In order to perform these tasks smoothly, improving the working environment is essential. An overall reduction in radiation levels is sought through a combination of decontamination, shielding, and removal of radioactive sources.

Aims

Improvements/verification tests of equipment for decontamination of high places and production/ verification tests of equipment for decontamination of upper floors based on R&D achievements obtained up to FY2013 is to be conducted. Through these steps, remotely controlled decontamination techniques required for the smooth implementation of surveys and repairs planned to take place inside reactor buildings will be established.

Main Achievements

Remotely operated decontamination equipment is used in (1) lower parts of the first floor, (2) upper parts of the first floor, (3) upper floors, and (4) basement areas (Figure 1). Prior to FY2013, (1) completing development for lower parts of the first floor, (2) production/element testing of decontamination equipment for upper parts of the first floor and (3) design of decontamination equipment for upper floors were conducted. Based on the results of these actions, the following tasks were undertaken during FY2014:

1. Development of decontamination equipment for high places

a. Improvement of decontamination equipment

Improvements were made to the three types of decontamination equipment for high places produced during FY2013 (high pressure water jet, dry ice blast, and suction/blast) in preparation for their actual application. These improvements included the production of a transfer unit for the high pressure water jet equipment and improvements to the operability of dry ice blast during traveling or decontamination. In addition, there was a reduction in the size of the hose reel section in the suction/blast decontamination equipment (Figure 2).

b. Verification testing of decontamination equipment

A mockup test facility was produced (Figure 3) and verification tests took place.

Equipment to be evaluated during actual operations was identified, and mockup equipment was produced taking the diversity and priority of applicability into consideration. Decontamination performance, traveling performance/operability when remotely controlled and safety functions were evaluated during verification tests to confirm required performance.

Test results confirmed the effective range of decontamination of high places using the high pressure water jet and dry ice blast equipment. The workability of the suction/blast decontamination equipment on the primary construction target of wall surfaces after obstacles have been removed was also confirmed. Through these tests, the goal of confirming applicability to the actual plant was obtained.

2. Development of decontamination equipment for upper floors

Plans are in place to introduce an elevating work cart from the open equipment hatch to access the upper floors of the reactor buildings (2nd and 3rd floors) where decontamination of the floor and wall surface (approximately 2 m in height) will be performed. At the present time, work is progressing on developing ways to utilize the same carts for decontamination technology developed for the decontamination of low areas (high pressure water jet, dry ice blast, and suction/blast).

Specifically, production of equipment based on methods of accessing upper floors and designs considered during FY2013 commenced. Equipment is comprised of a shared cart (work cart/transport cart/support cart/relay cart) and each decontamination unit (Figure 4). Production of the equipment will be completed during FY2015, and will then go through a verification testing process.

3. Conceptual study on decontamination equipment for basement areas

In future, a drop in groundwater levels around the reactor buildings due to the operation of the bypass system and the ground freezing method is expected, and the stagnant water level inside the reactor buildings must therefore also be lowered in response. However, this may result in a rise in dust and in radiation levels inside the buildings. A conceptual study was therefore conducted on dust diffusion reduction measures for basement areas and decontamination work when required in basements.

Future Developments

In addition to identifying challenges through verification tests and implementing steps to deal with these, equipment for the decontamination of high places will be utilized at the Fukushima Daiichi NPS according to the required purpose and timing. Equipment for the decontamination of upper floors will be completed during FY2015 and verification testing on mock-ups conducted. Additionally, consideration of the necessity of developing new technologies for the decontamination of basement areas will continue.

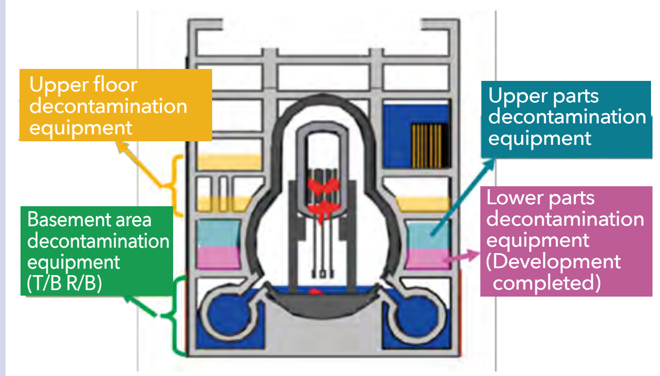


Figure 1 Locations where each type of decontamination equipment is to be placed.

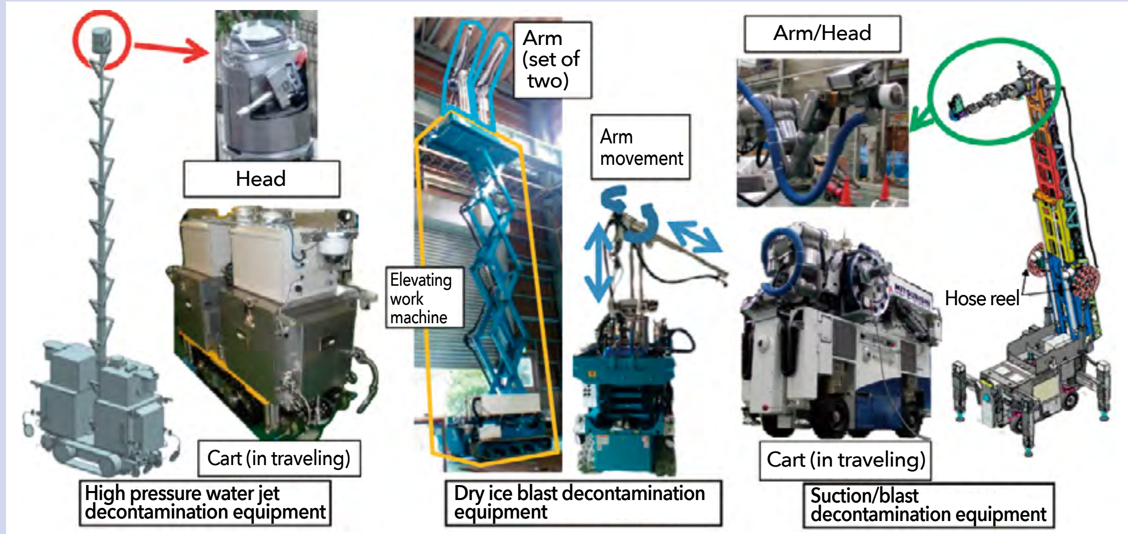


Figure 2 Improvements to equipment for decontamination of upper parts conducted during FY2014.

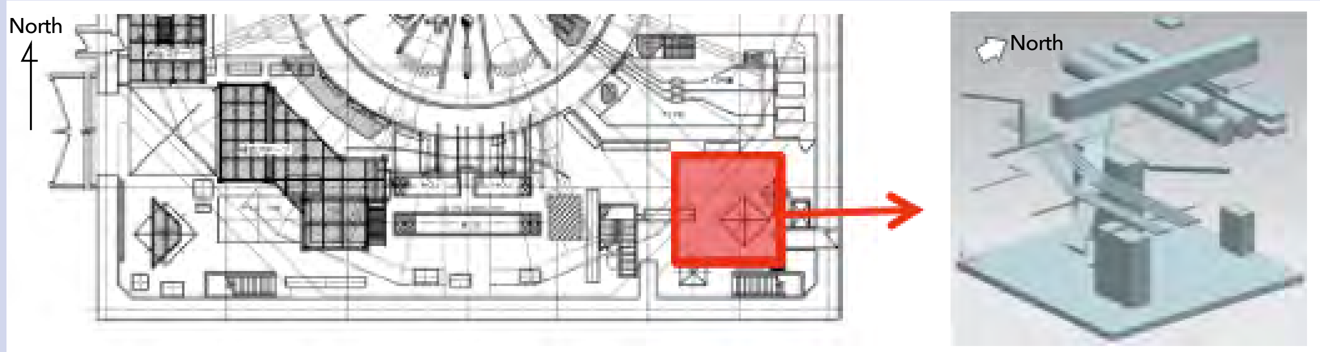


Figure 3 Mockup facilities simulating actual plant.

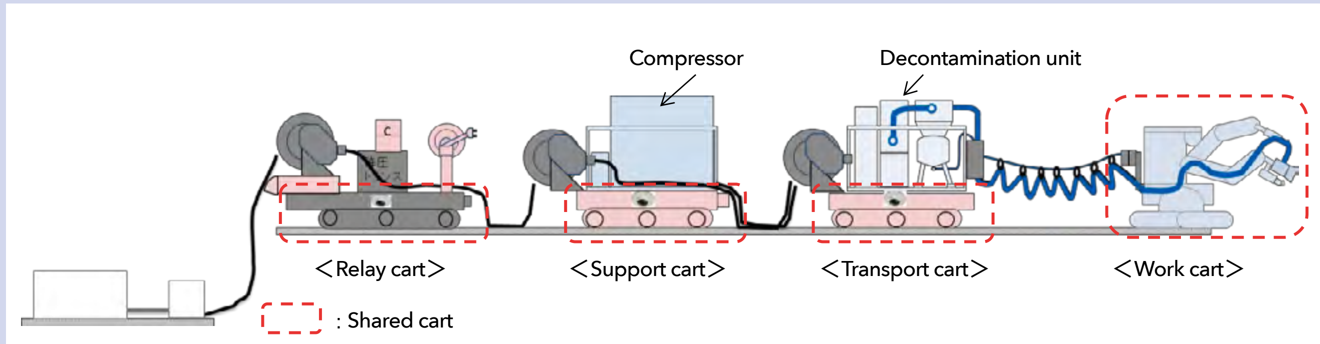


Figure 4 Conceptual diagram of upper floor decontamination equipment (shows mounted blast equipment as an example).

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). A plan exists to submerge the PCV in order to retrieve the fuel debris; however, it is necessary to ensure that there is no water leakage from the PCV prior to this step.

Aims

This plan aims to establish technology that enables the repair of water leakage points in the PCV that will allow for the retrieval of fuel debris in a submerged state.

Main Achievements

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

In conjunction with the "Development of Technologies for Assessment of the RPV/PCV Integrity" R&D project, the reinforcement material placing height (height to which the material needs to be filled to support the S/C in the torus room) was set to the upper support column pins and target compression strength was established (Figure 1). Looking ahead, casting tests and developments made in order to execute construction of the reinforcements will continue.

2. Study on the circulation cooling system

A conceptual study of PCV circulation cooling equipment and water intake was conducted. Based on this study, a conceptual diagram outlining construction steps will be created (Figure 2) in conjunction with an examination of the repair of each leakage point (stopping water leakage).

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

Tests conducted based on an auxiliary sealing material deployment improvement plan showed that there is a likelihood of further improvements to the deployment of the material. A one-half scale model was used for water stoppage testing, and successful patterns of achieving a reduction in leakage are being verified (Figure 3). Looking ahead, water stoppage tests and developments to execute construction of the reinforcement will continue. In addition, element testing for filling the S/C to prevent water leaks (stopping water from quencher/ strainer/downcomer) is being performed sequentially. Following this, tests using a full-scale model were conducted in preparation for the execution of actual construction.

4. Water stoppage technology involving injection of sealing material into the vacuum brake line

A study and sequential testing of methods to improve the stoppage of water leakage using cloth packing with mortar filling and silicon-based material is underway. A flexible type of guide pipe is currently being considered (Figure 4).

5. Water stoppage technology for leakage from the seal section and vent pipe bellow

Based on the findings and achievements of the preceding fiscal year, test plans for stopping water leakage from the equipment hatch and penetrating upper part of the PCV, and sealing the personnel access lock room, are being studied. Water stoppage tests are underway, and developments to execute these plans are progressing.

6. Boundary construction technology for PCV connector pipes

For this study, AC, RW, and RCW systems were selected. Consideration of water stoppage proposals for each system is ongoing. In addition, water stoppage testing and developments based on study findings are proceeding towards actual construction.

7. Water stoppage technology involving injection of sealing material into clearance between the torus room wall and penetrating pipes.

Applicability was confirmed based on previous test results. Taking into account the actual environment and the current progress of each project, plans are underway to arrange relevant penetration parts taking conditions at the time of construction into consideration, construction feasibility, selection of water stoppage materials, and the necessity of element testing in order to prevent water leakage between rooms. Water stoppage testing will continue towards actual construction.

8. Drywell (D/W) shell repair technology

Assumptions about the damage caused by such a severe accident are being made, and water stoppage material selection and water stoppage methods are being studied. The feasibility of these methods will be studied sequentially.

9. Establishment of a plan in lead-up to filling the PCV with water

Considering system configuration at time of submersion, the processes required for planning in the lead-up to filling the PCV with water are being examined. Scenarios at each Unit are being collated, and submergence plans clarified.

Future Developments

In order to establish water stoppage technology for each repair/water leak location, ongoing tests will be performed and challenges will be reviewed, analyzed and evaluated. A strategy to resolve these challenges that emerge in testing will be developed, and steady R&D efforts will continue to ensure there are no issues left unresolved. Additionally, PCV submersion scenarios will be studied for each unit separately and the establishment of water stoppage methods to be executed at the actual plant sought.

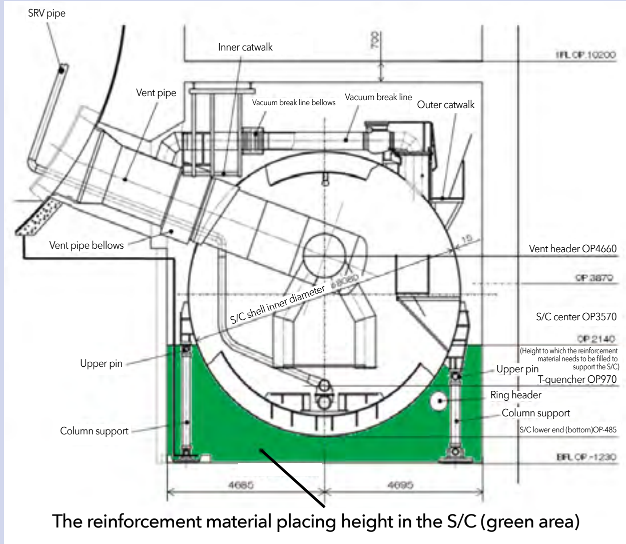


Figure1 The studied reinforcement material placing height in the suppression chamber (S/C)

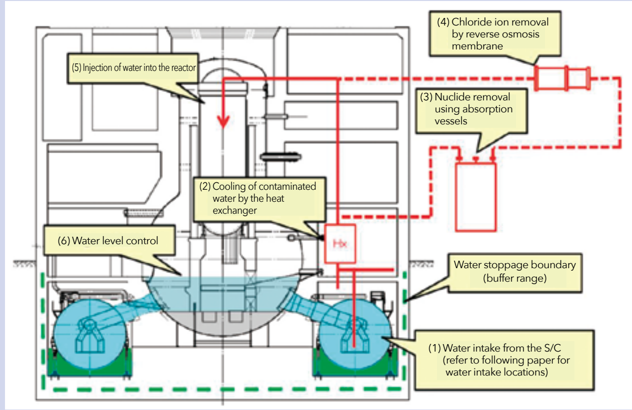


Figure 2 Concept of the PCV circulation cooling system


Figure 3 Water stoppage testing using a one-half model of the bent pipe

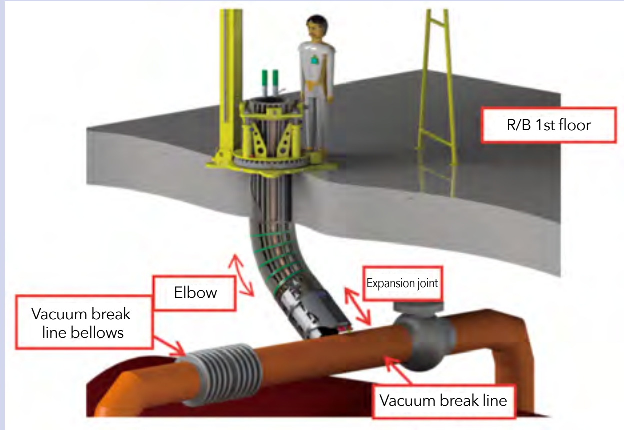


Figure 4 Flexible guide pipe overview

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

Background

In order to prepare for the removal of fuel debris after submerging it in water (submersion method) as part of the steady decommissioning of the Fukushima Daiichi NPS, the establishment of technology to repair and stop PCV leaks is required.

Aims

Development of repair and water leakage stoppage technology for leakage points inside the PCV in preparation for the implementation of the submersion method is progressing. This project involves full-scale testing for the verification of the developed technologies (construction method, remotely controlled equipment, etc.) and the operational training applicable on site. Development is planned to take place at the Naraha Remote Technology Development Center through a joint proposal with the Japan Atomic Energy Agency.

Main Achievements

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

In order to confirm the operability of placement equipment for suppression chamber support reinforcement developed as part of the 'Suppression chamber support reinforcement technology' component of the 'Development of Repair and Water Leakage Stoppage Technology for Leakage Points Inside Primary Containment Vessels' (development of water leak repair technology) project, this plan involves the use of water in full-scale mock-up tests. To determine test facility specifications this financial year, the knowledge derived from the development of water leak repair technology was incorporated. Figure 1 shows a draft conceptual design of the full-scale test facility.

2. Design/production of full-scale mock-up

In a plan for the design, production, and on-site installation of a full-scale mock-up of the lower part of the PCV at the Fukushima Daiichi NPS Unit 2, this year saw the commencement of a compilation of specifications and detailed design take place in preparation for actual production of the mock-up.

The primary specifications of the full-scale mock-up are as follows:

- The mock is a one-eighth scale model of the suppression chamber including the vent pipes (excluding the bellows), vent head, downcomer and surface of torus room wall at Fukushima Daiichi Unit 2.

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

As full-scale testing requires the simulation of the actual environment in the Fukushima Daiichi Units 1-3 to the fullest extent possible, including dimensions and temperature conditions, the study/design/production/installation of the following equipment is planned. During this financial year, specifications were collated in preparation for production and detailed designs were embarked upon (Figure 2).

- Heating/Feed water equipment: In order to provide the heating function to produce heated water that simulates stagnant water in the Fukushima Daiichi NPS in a full-scale mock-up test, system designs and a deployment plan for equipment that supplies heated water at the required volume were formulated, and a boring survey at the proposed construction site required for the construction application was conducted (Photos 1 and 2).
- Turbid water treatment equipment: As stopping water leakage at the lower part of the PCV is expected to involve the use of grouting material (including cement), water that contains cement is generated after conducting tests. A system design and deployment plan for purification equipment outside the plant that processes the turbid water to a quality that allows for drainage was performed.
- Work floor: As the height of the first floor of the basement in Unit 1 and Units 2 and 3 at Fukushima Daiichi differs, the design and examination of methods of changing the height of the work floor at each Unit (first floor of reactor building) took place.
- Mock-up transfer rail: As the weight of the mock-up is very heavy (approximately 5,400t), moving it using a crane after it has been assembled is difficult. The design/examination of the installation of a mechanism to move the mock-up (mock-up transfer rail) was performed.

Future Developments

The design, production and installation of the test facility will continue in parallel with the project to develop repair and water leakage stoppage technology in the lead up to implementing full-scale mock-up tests. In addition to evaluating the applicability to Unit 2 of the status of implementation and results of tests performed after the test facility is completed, the facility will be used for training personnel and is aimed to establish repair and water stoppage technologies for PCV leakage points towards implementation of the submersion method.

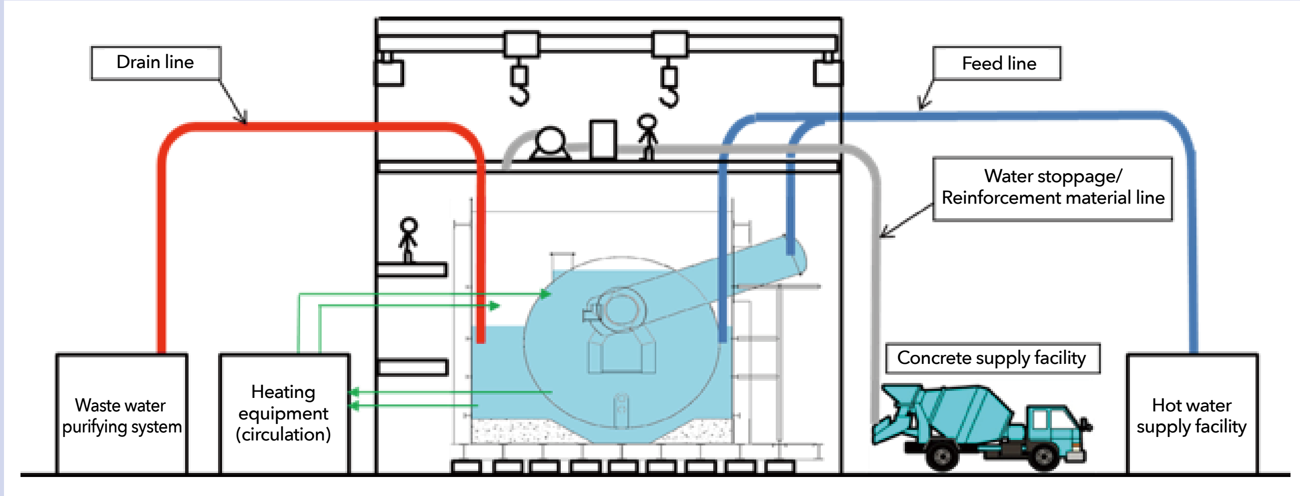


Figure 1 Conceptual design of the full-scale test facility

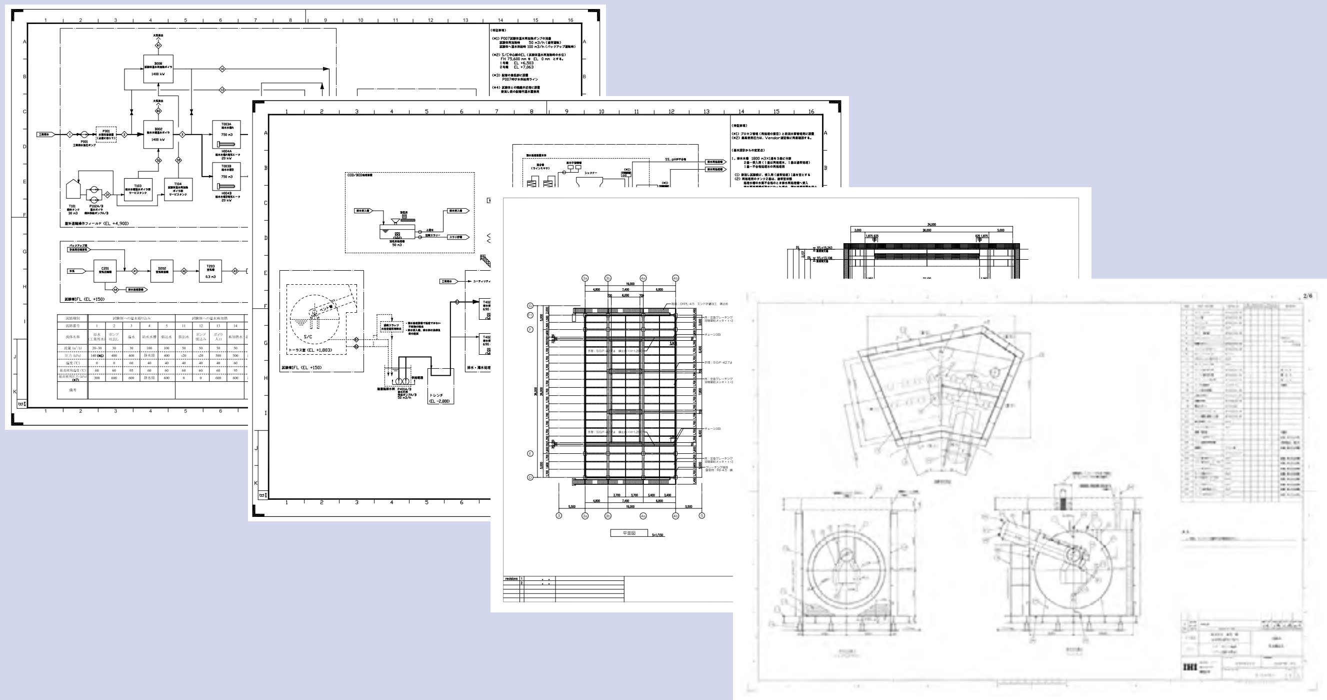


Figure 2 Designing various testing equipment

Photo 1 Boring survey at the site

Photo 2 Soil sample taken during boring survey

Development of Technology for Investigation Inside the Primary Containment Vessel

Background

It is estimated that reactors cores in Units 1-3 have melted and fuel has partially fallen with reactor internals into the RPV and PCV. In particular, it is possible that after fuel debris melted through the bottom of the PCV in Unit 1, they emerged from the inside of the pedestal (which supports the PCV) and spread out of the pedestal opening. However, the actual condition has not been confirmed.

Aims

Video footage, information about radiation dose and temperature have confirmed that, in addition to being a harsh, highly radioactive and extremely humid environment, visibility inside the PCV is limited due to the presence of steam and stagnant water and to it being very dark. Moreover there is the possibility of unforeseen obstructions that were generated from the accident. In order to overcome these challenges, this project aims to develop technology for the investigation inside of the PCV.

Main Achievements

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

Needs analysis were conducted for other related projects (Development of Technology for Fuel Debris Retrieval/Reactor Internals, Development of Repair and Water Leakage Stoppage Technology for Leakage Points inside the PCV, etc.) and the results were summarized. To respond to these needs, the outline of PCV access methods for internal investigation and investigation methods were studied. Prospects emerging from the results of these studies and plans were shared with other related projects, and adjustments were made accordingly.

2. Development of investigation devices, etc.

a. Technology for accessing inside the pedestal

Development of devices that enter the PCV in the Fukushima Daiichi Unit 2 and then penetrate into the pedestal to conduct investigations ('Investigation on platform inside the pedestal' (A2 Investigation)) is currently underway (Photo 1).

A self-propelled miniature robot will enter via the opening (diameter about 115 mm) of the X-6 penetration (PCV penetrating part), and after traveling through a guide pipe inserted into the PCV, traverse the CRD rail into the pedestal interior. Upon arrival in the pedestal, it is planned that onboard measurement devices (camera, etc.) will gather various types of data. Verification tests are expected to commence in the first half of FY2015.

b. Technology for removal of the shielding block

Prior to installation of the development unit required for the A2 Investigation, removal of the shielding block placed in front of the X-6 penetration via remote control is required. Technology for this purpose (Equipment for removing the X-6 shielding block) is currently being developed (Photo 2).

Preparations are currently underway in preparation for verification testing at Unit 2 in the first half of FY2015.

c. Technology for accessing the pedestal exterior

A shape-changing robot to investigate the pedestal exterior inside the Unit 1 PCV (an investigation device that works on the grating outside of the pedestal ('B1 Investigation') was developed (Photo 3).

This robot takes a tubular form as it travels along the existing guide pipe (inner diameter about 100 mm) placed at the X-100B penetration, and changes into a U shape once landing on the grating* on the 1st floor inside the PCV in order to achieve stable maneuvering. It is expected that this will be the first robot to investigate the inside of the PCV of Unit 1, and verification testing is planned for the first half of FY2015.

* Grating: grid-like floor

d. Technology for measurement of fuel debris

Development of fuel debris measurement equipment is underway to understand the position and distribution of molten materials assumed to be fuel debris in extremely harsh environments; darkness, high radiation, rain drops and fog. Taking this severe environment inside the PCV into account, this equipment will employ an optical cutting method and development is progressing towards verification tests after FY2016.

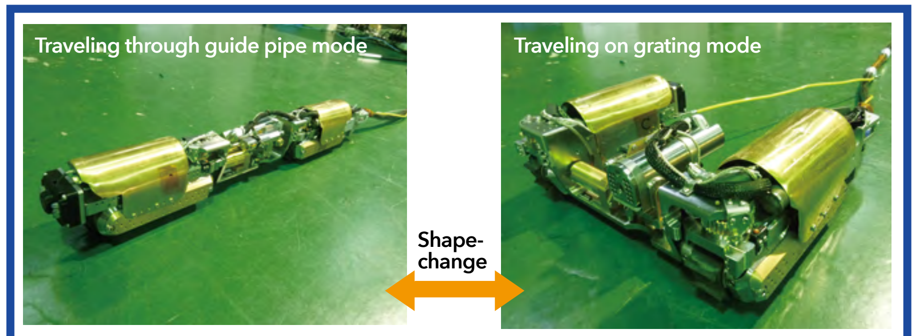
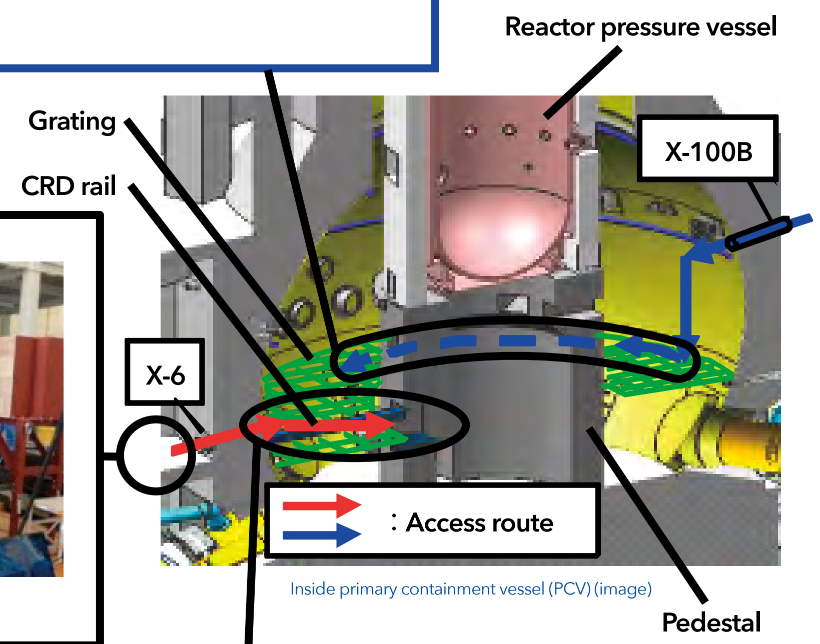


Photo 3 B1 investigation device (shape-changing robot)
To be used outside Unit 1 pedestal.

Photo 2 Shielding block removal equipment
To be used in front of X-6 penetration in Unit 2.



Inside primary containment vessel (PCV) (image)

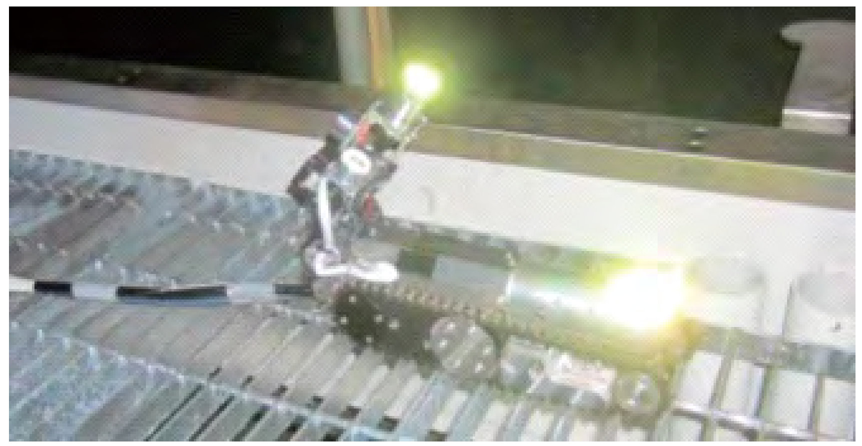


Photo 1 A2 investigation device
To be used inside Unit 2 pedestal.

Figure Devices currently being developed and where they will be used.

Future Developments

Development will continue in the lead up to verification tests from the X-100B penetration in Unit 1 and the X-6 penetration in Unit 2 scheduled for the first half of FY2015. In addition, device development to further investigate both the interior and exterior of the pedestal inside PCVs will be ongoing.

Development of Technology for Investigation Inside the Reactor Pressure Vessel

Background

In order to develop equipment for their retrieval, information regarding the location, shape and condition of fuel debris and core internals must be understood in advance. However, in addition to the complex reactor internal structure, radiation levels are extremely high, creating a situation that makes it currently difficult to directly obtain information from the inside of the RPV.

Aims

The overall plan for the survey of the interior of the RPV will be optimized based on related projects and after compiling needs on the construction site. Specifically, in addition to considering at an early stage survey methods in order to access the RPV, the development of technology of survey equipment and devices will take place.

Main Achievements

1. Formulation of plan for investigation inside the RPV

Based on related projects such as "Development of technologies for retrieving fuel debris and core internals" and the collating of needs (including fuel debris sampling) from the site (TEPCO), survey locations and items, and the time required for and necessity of surveys were identified.

As a result, survey needs (amount and scope of fuel debris located in the core or lower part of the reactor, etc.) were considered based on the progress of research and development for each project. In addition, the selection of the method of fuel debris retrieval from the site and survey needs (temperature inside the RPV/cooled state of fuel debris, etc.) towards each construction method and equipment design milestone were clarified.

In regard to the sampling of fuel debris, the timing of implementing this task and its purpose and positioning were set out and coordinated with the 'Analysis of Debris Properties' project, etc., along with a clarification of the division of roles.

2. Study on investigation methods and Formation of development plan for investigation equipment

A survey of technology that can be applied to investigation inside the RPV stemming from technology developed in the 'Development of Technology for Investigation inside the PCV' project and from a FY2013 IRID Request for Information (RFI) was conducted.

Based on this survey, a request for technologies as below that can be applied to investigations inside the RPV was made both within Japan and internationally, and the feasibility of these technologies considered:

(1) Transfer technology of investigation equipment (small diameter pipe expansion/transfer technology)
(2) Investigation support technology (radio communication in highly radioactive environments)

3. Development of investigation devices/equipment

a. Survey technology that utilizes existing routes (pipes, etc.)

Prototype testing of survey devices using existing large-diameter pipes took place. During this financial year, verification of accessibility was conducted by studying travel motion (running devices in horizontal/vertical/elbow situations or through pipes of various diameter), ability to grip surfaces (maintenance of position and posture), and handling of direction change (running through T-junctions, directional control). By assuming that devices will operate in an environment that will expose them to up to 1000 Gy/h, movement mechanisms of a hydraulic (Figure 1) or electric (Figure 2) type was studied.

b. Survey technology that utilizes new routes (opening of holes, etc.)

Element testing took place to verify the feasibility of drilling holes via remote control for the purpose of accessing the inside of the RPV by opening a hole at the top of the reactor containment vessel (Figures 3 and 4).

Future Developments

Based on findings from survey of onsite needs in FY2014, development of technology will continue in the lead-up to early investigation of the reactor well in FY2016 and survey of the RPV interior in and after FY2018.

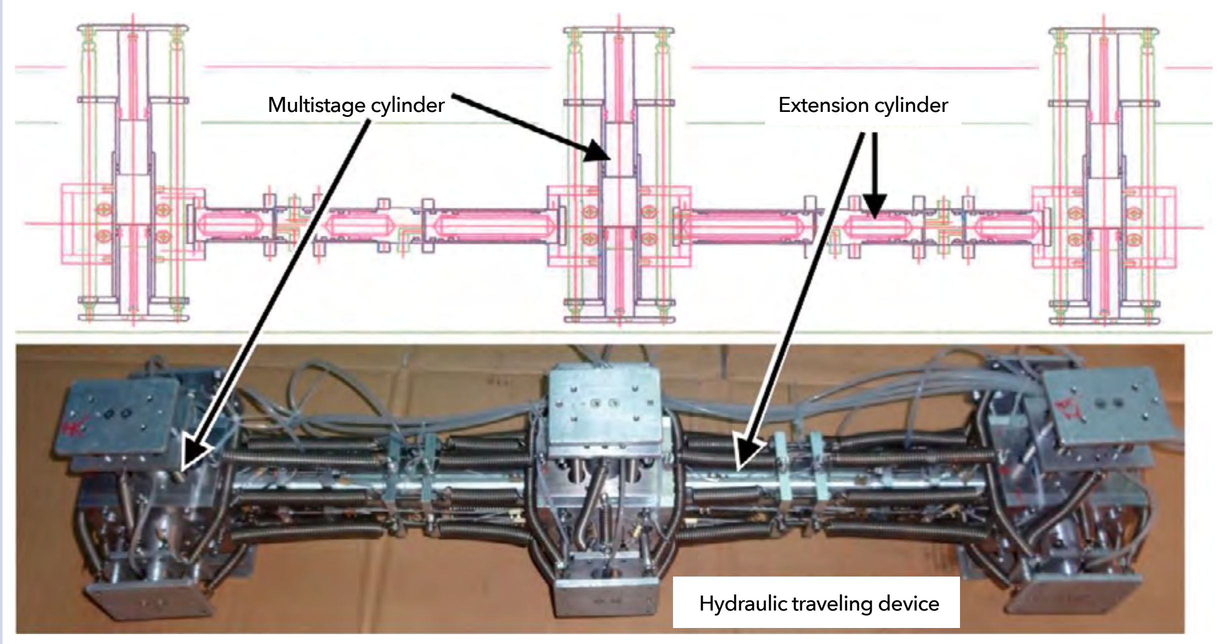


Figure 1 Hydraulic traveling device

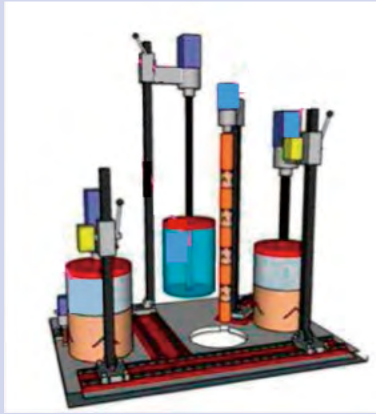


Figure 3 Shield plug hole boring device (image)

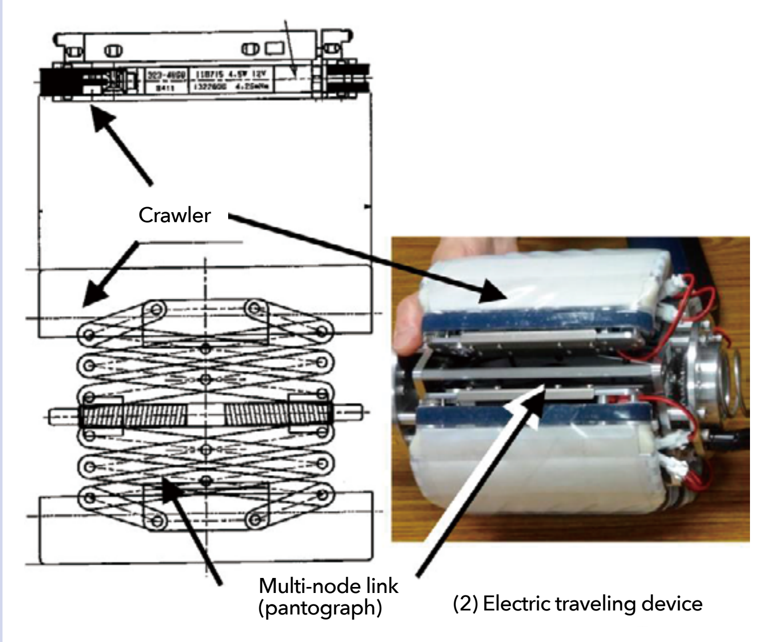


Figure 2 Electric traveling device

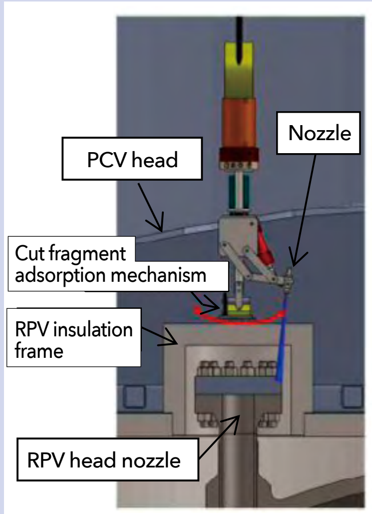


Figure 4 RPV head hole boring device (image)

Identifying Conditions Inside the Reactor through Application of Severe Accident Analysis Code

Background

Identifying conditions inside reactors is essential in the process of formulating measures for fuel debris retrieval and developing safety plans. However, it is extremely difficult to investigate the inside of Units 1-3 at the Fukushima Daiichi NPS directly due to the extremely high radiation levels present.

Aims

This project aims to enhance severe accident analysis codes (MAAP, SAMPSON) in order to elucidate accident progression and promote decommissioning efforts through the sharing of information required for identifying conditions inside reactors that are gained from accident progression analysis, testing, and so on.

Main Achievements

1. Improvement of the MAAP code and accident analysis

MAAP(Modular Accident Analysis Program)is an analysis code that enables the consistent evaluation of thermal-hydraulic phenomena/fission product behavior in the RPV or PCV during severe nuclear power plant accidents.

Improvements were made on a physical phenomenon model assuming that the molten core flowed down and was deposited in the RPV lower plenum after the core was damaged, based on accident progression at the Fukushima Daiichi NPS and analyses of on-site surveys. Then, an accident progress analysis was conducted (Figure 1). From the results of this analysis and on-site investigations, an estimate was made of the distribution of fuel debris in the RPV and PCV (Figure 2).

2. Improvement of the SAMPSON code and accident analysis

SAMPSON (Severe Accident analysis code with Mechanistic, Parallelized Simulations Oriented towards Nuclear fields) is a model constructed using multi-dimensional mechanistic models and theoretical-based equations that is suited to analyzing the dispersion and properties of fuel debris.

In order to improve the accuracy of estimates regarding the dispersion of debris contained in the lower part of the RPV, functional improvements to models that combine the RPV and PCV, and improvements to the model analyzing thermal-hydraulic behavior were made. An analysis using a molten core relocation analysis module yielded prediction results that suggest that, after water in the bottom of the RPV evaporates, in-core monitor housing is the first site of damage (Figure 2). In order to verify this, tests using real-size penetrating pipes and real corium is planned for the next financial year.

3. Analysis and evaluation of conditions Inside the RPV and the PCV

A 3D analysis model that enables analysis of debris lateral erosion inside the concrete floor like sump pit was developed, using as a reference the Molten Core Concrete Interaction (MCCI) analysis module in the SAMPSON code and also through reviewing the fluidic cell boundary condition setting model. CCI test analyses from the OECD/MCCI projects were performed in order to verify the developed model. The results of an analysis of the depth and shape of concrete erosion match test results well, as Figure 3 shows, and confirmed the predictive performance of the developed model.

4. Tests to simulate the conditions inside the reactor during the severe accident

Testing was done to determine the effect that injecting seawater into a reactor, as took place in the Fukushima Daiichi NPS accident, has on thermal-hydraulic behavior. As an example test result, it was found that the difference between the wall temperature (corresponding to cladding tube surface temperature) and fluid temperature grew with an increase in the amount of heat per unit area (heat flux), as Figure 4 shows. It was also found that this difference grew in relation to the increase in concentration of a sodium chloride solution.

5. International cooperation concerning identifying the condition inside the reactor

The Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station (BSAF) Project commenced in 2012. A project report containing summarized results of an analysis conducted by the eight participating countries that included a study of the location and amount of fuel debris was released this year (Figure 2). The scope of the BSAF Project will be expanded to include an analysis of the migration behavior of fission products in Phase 2, which will also see 12 countries participating.

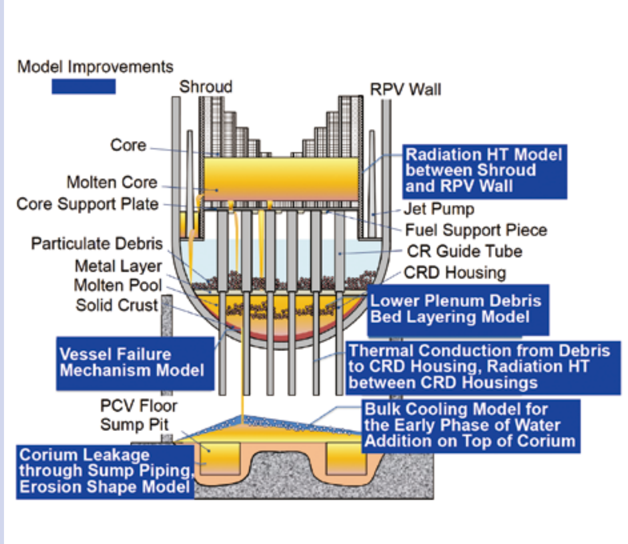
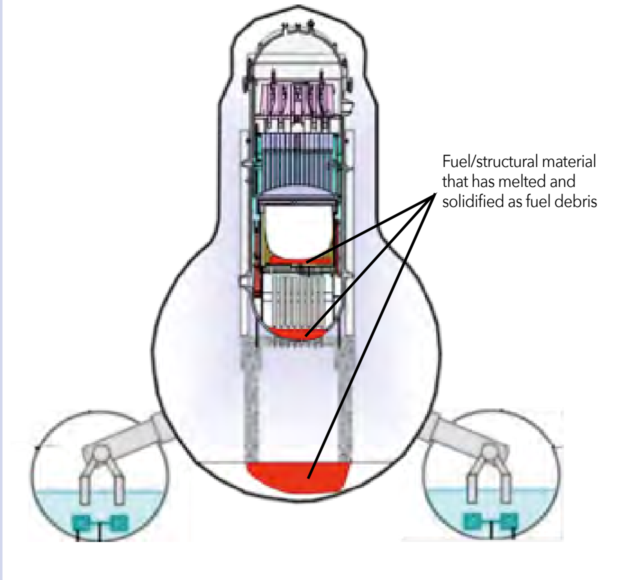


Figure 1 Improvement of MAAP Analysis Model

Improvements were made to a physical phenomenon model that treated the behavior of molten core flowed down, deposited in the RPV lower plenum, and released to outside of the RPV after the core was damaged.



Weight distribution of fuel debris

	Core (%)	RPV bottom (%)	PCV (%)
MAAP	0	Approx. 10	Approx. 90
SAMPSON	0	0	100
BSAF (9 organizations)	0~2	0~5	70~100

Figure2 Results of estimation of conditions inside Unit 1

It is estimated that most of the fuel debris in Unit 1 has fallen down to the floor of the PCV. (Values shown represent ratio compared to the weight of all structural material present in the reactor core prior to the accident. The sum of all percentages of less than 100% indicates that a certain proportion did not become debris and remains intact.)

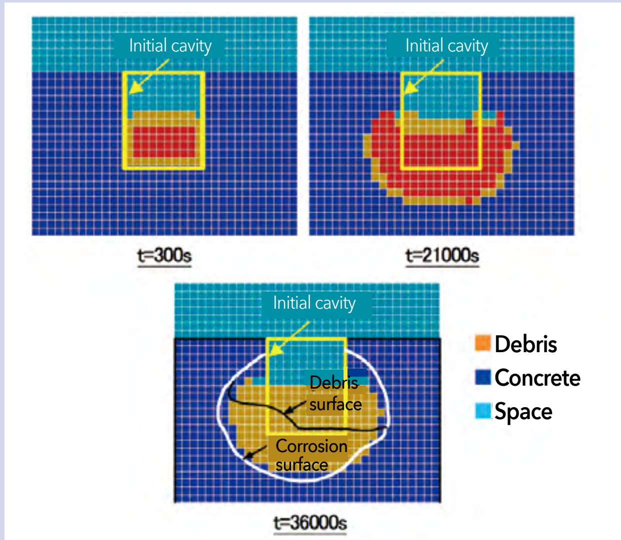


Figure 3 Pit erosion behavior evaluation example

Results of an analysis of concrete erosion depth and shape agree with the findings of the OECD/MCCI project CCI-2 test, and confirmed the prediction performance of sophisticated models.

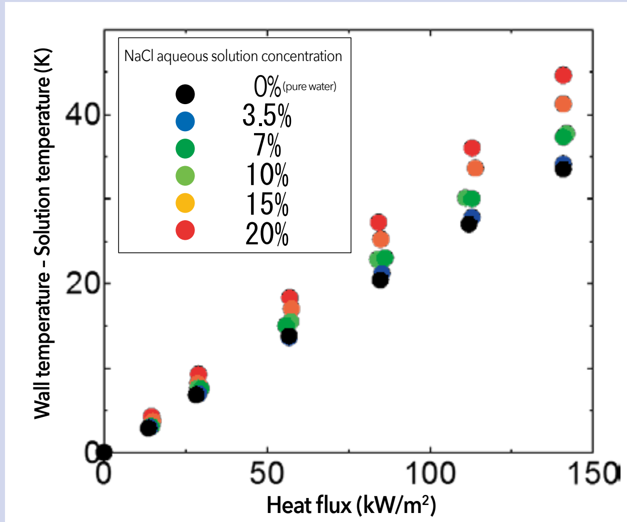


Figure4 Figure 4 Example of test results using sodium chloride solution (Temperature difference between wall and solution)

As heat flux was raised, wall temperature increased and the temperature difference with the solution became greater. In addition, the temperature difference between the wall and the solution also rose as the concentration of the sodium chloride solution increased.

Future Developments

The sophistication of severe analysis codes will be further advanced, and an improvement in accuracy of predicting the location of fuel debris and migration of fission product nuclides will be sought. In addition, utilizing data obtained from the actual reactors, comprehensive analysis and evaluation of conditions within reactors will be executed, the results of which will be used to confirm and implement methods of fuel debris retrieval at each unit.

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, as the radiation level inside the RPV is extremely high, it is very difficult to verify the condition directly.

Aims

This project uses cosmic-ray muons to image and map the distribution of fuel debris inside the RPV from outside the reactor buildings. Using the muon transmission method, it is hoped that fuel debris at Unit 1 can be identified at the spatial resolution of about 1m and the data obtained will be used to study methods of how fuel debris can be retrieved. A measurement system that uses the muon scattering method will be used to confirm the location of fuel and debris remained in the reactor at Unit 2 (the spatial resolution of about 30 cm).

Main Achievements

1. Development and measurement of a detection system for the transmission method under high radiation

Testing confirmed appropriate shielding thickness that enables the effective measurement of muons even in radioactive environments (0.4mSv/h). In addition, a measurement system that can reduce the effects of background gamma radiation by conducting coincidence counting using three-layered 2D resolution scintillation detectors to determine muon transmission position was developed. This measurement system was placed outside the Unit 1 reactor building and the measurement to image inside the reactor was conducted. Furthermore, an evaluation of the transmission method measurement performance was conducted (Figures 1 and 2).

2. Development of a large muon tracker under high radiation

Large-scale muon tracking devices with 7 x 7 m^2 active sensor areas that use the muon scattering method to image the inside of the RPV from outside the reactor building were developed. The muon trackers are comprised of 3.5-meter drift tube detectors that are joined in two end-to-end to make chambers of 7 m in length. There are 140 of these tube chambers that are arranged to form a 7 x 7 m^2 active sensor area in a six-layered stack. The gas encapsulated in the detectors is of a type that does not contain hydrocarbons that can be broken down by gamma rays, ensuring resistance to radiation during the period of measurement (Figure 3).

Two of the muon trackers were placed apart, one above the other, in order to confirm measurement performance (Figure 4).

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

As detectors that utilize the scattering method are large in scale, time-coincidence logic incorporated into the electronic circuit to remove high background gamma rays was developed. A combination of coincidence counting in more than four layers and linear reaction patterns allows muon measurements without shielding even at the dose rate of approximately 50 μSv/h. Additionally, in terms of measurement performance, time resolution of less than 2ns (position resolution of less than 0.1 mm) and a coincidence counting gate width of less than 1μ were achieved.

4. Development of an estimation technique of fuel debris distribution inside the reactor by muon tracking

A method of estimating the distribution of fuel debris from measurement values taken at Unit 2 by muon tracking devices was developed. Specifically, taking angular resolution, position resolution and the effect of shielding material into account, the muon scatter angle is evaluated, and fuel debris density contrast images are created by the relationship between the structure composition and the scattering angle of muons as they pass through the structure. In order to evaluate the accuracy of this development approach, the simulation test took place using a 3D model of the reactor; results were then used to predict measurements of the actual site (Figure 6).

Future Developments

While expanding the scope of measurements using the transmission method at Unit 1, detection of fuel debris at Unit 2 (lower part of the RPV and the reactor core) will be conducted using the scattering method. Furthermore, by improving processing algorithms and measurement techniques, the visual range will be expanded, measurement time reduced, and more detailed information provided by resolution improvement. This will contribute to the establishment of measurement technology for the distribution of fuel debris inside the reactors and the selection of efficient methods of fuel removal.

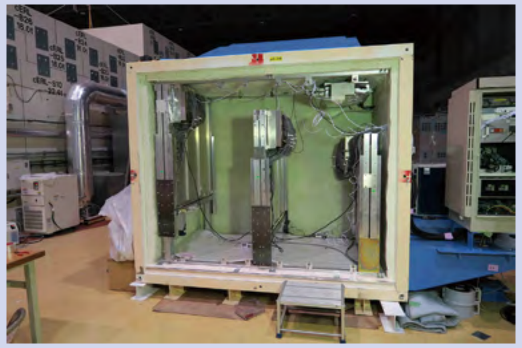


Figure 1 Muon detector which uses transmission method to be installed at the Fukushima Daiichi NPS Unit 1

Three-layered scintillation detector is housed in an enclosure with iron walls approximately 10cm thick to shield against background gamma rays. Temperatures are kept at a constant level inside the enclosure through an air conditioning unit.

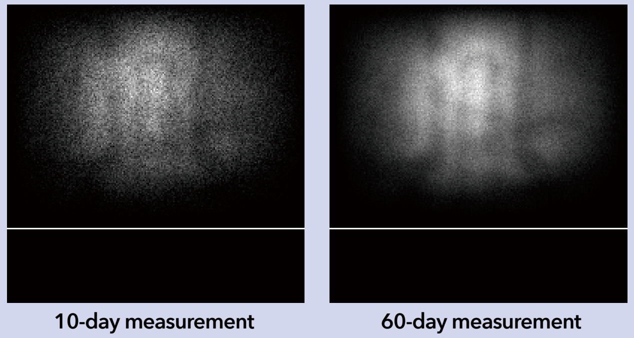


Figure 2 Result of transmission method simulation

Using the Monte Carlo method, measurement days and image clarity were examined. As the image contrast is proportional to the density of muon traveling length, the longer the measurement time, the clearer the image contrast becomes.

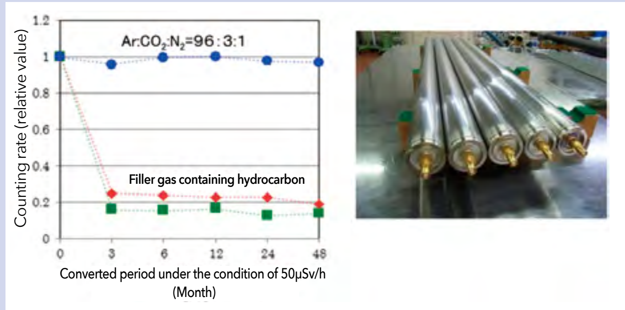


Figure 3 Production of highly radiation-resistant drift tube detectors.

Manufacturing of the drift tube detectors (effective length of 3.5m, as shown at right) employing an encapsulated gas mixture ($Ar:CO_2:N_2$=96:3:1) that enables operation without any reduced sensitivity, even after exposure to gamma radiation over the equivalent of a four-year period at a maximum design dose rate of 50μSv/h.

Figure 4 Muon tracking device system performance test platform

Muon tracking devices with a 7 x 7m^2 active sensor area comprised of 3,360 drift tube detectors (3.5m long) were constructed and placed one above the other. Lead and other materials were placed between them in order to evaluate system performance.

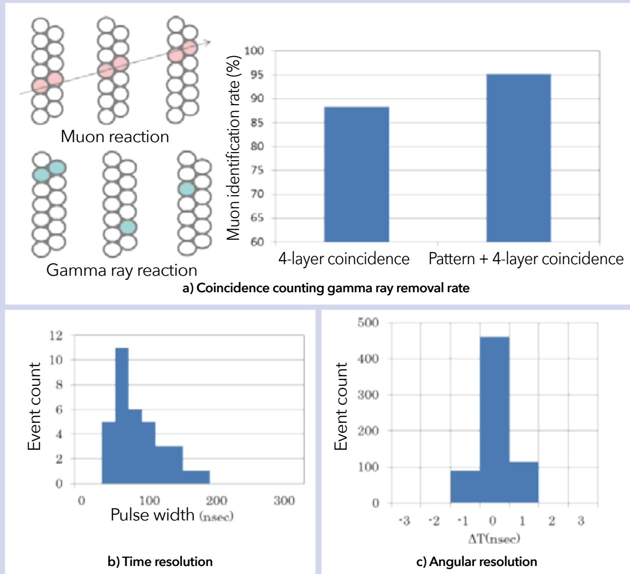


Figure 5 Results of circuit system performance evaluation

a) Even under a gamma radiation field of 50μSv/h, it was possible to detect over 90% of muons using coincidence counting and linear reaction patterns integrated into the circuitry.
b) With a 200ns pulse width, was achieved the gate width of less than 1μs required to remove gamma radiation.
c) A time resolution (FWHNM) of under 1ns was confirmed, against the target lower than 2ns.

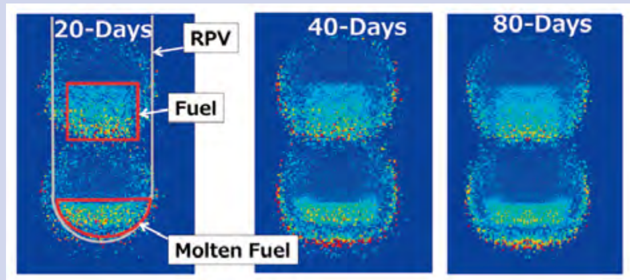


Figure 6 Results of scattering method simulation

Predicted results from 7m x 7m muon tracking devices placed above one another in production level conditions to place them outside turbine or reactor buildings (measurements at 20, 40 and 80 days)

Development of Technology for Non-destructive Detection of Radioactive Materials Accumulated in the Suppression Chamber

Background

In order to repair and stop water leakage in the suppression chamber (S/C), it is necessary to determine the condition of radioactive material that has accumulated inside. However, a method of evaluating conditions has not yet been established. The optimum solution would be a non-destructive examination technique, but it is not clear if deposits of radioactive material that need to be detected can be evaluated using a non-destructive method. Another important issue is to establish approaches that can determine the amount deposited throughout the entire area, including the S/C.

Aims

In order to gather information required for repairs and water leakage stoppage in the S/C, in addition to estimating the condition of accumulated radioactive material present inside the S/C, a method of measurement will be developed.

Main Achievements

1. Formulation of development plan

Developments and work items required for the detection of radioactive materials believed to be indispensable were extracted (Table 1) and development plans were formulated.

2. Radioactive materials migration scenarios

Consideration was given to scenarios involving the migration of radioactive materials into the S/C and torus room (Figure 2). As it is thought that the likelihood of an inflow of radioactive debris that exceeded the acceptable limit was low, by taking measurements of the lower part of the S/C and the region around the sand cushion drain pipe outlet where there is a relatively high probability of radioactive materials having been deposited, it can be confirmed that there are no radioactive materials that exceed the acceptable limit.

3. Evaluation of impact of radioactive substances on water stoppage material

Among the influence factors arising from remaining radioactive materials, there is a concern that even the smallest deposit may heat water to a temperature of 80°C, and the heat generated within water stoppage materials may cause cement deterioration. Calculations were made based on a conservative estimate of the weight of uranium in this scenario.

4. Development of technology for detection of radioactive material

a. Evaluation on nuclide composition and distribution of radioactive material

Nuclides originating from fuel (nuclides selected for measurement (Cm-244, Eu-154, etc.), background nuclides, and nuclides in shielding material) were determined using the ORIGEN code. The mix ratio of nuclides originating from fuel and structural materials has been set based on MAAP code analysis results. The S/C and torus room calculation model (Figure 1) is a 3D 1/16 scale representation of the S/C and torus room; this model was used to evaluate neutron and gamma ray flux in the area around the bottom of the S/C (Figure 2).

b. Evaluation of background radiation

An assessment of the effect of background gamma radiation (Cs-134, Cs-137) on stagnant water was conducted (Figure 3).

c. Selection of optimum techniques for detection of radioactive substances

To measure acceptable background radiation level and radio-sensitivity, a B-10 neutron detector and CdTe/$LaBr_3$(Ce) gamma-ray detectors were selected as the best choice for the task and their responses evaluated at the locations in a. and b. above.

d. Study on method of estimating the amount deposited in the S/C

A comparison between measurements done in the vicinity of locations with a high probability of radioactive material deposits and threshold levels set through the analysis in a. will decide the presence or absence of radioactive material that exceeds acceptable levels.

Future Developments

The possibility of radioactive material that exceeds acceptable levels having flowed into the S/C or torus room is considered low, and it was confirmed that it is technically feasible under the assumed conditions to verify the presence or absence of this material using non-destructive methods. The implementation of measurement systems or design/production of devices to access these areas will be decided based on the results of developments of methods to repair and stop water leakage.

No.	Content
1	Design/production of measurement systems
2	Design/production of access devices
3	Combination performance testing
4	Design and production of perforating apparatus
5	Verification tests
6	Non-destructive detection work

Table 1 Development/Work items

In the formulation of development plans, development/work items required for the detection of radioactive materials were identified in advance.

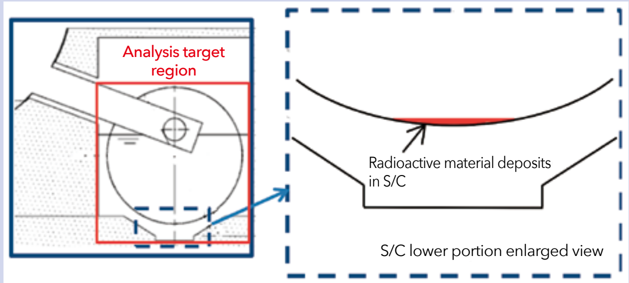


Figure 1 Computational scheme

Form	S/C	Torus Room
Melted radioactive substances	Radioactive substances in the drywell reach the S/C vent pipe entrance and flow into the S/C.	Radioactive substances in the drywell erode the PCV shell and sand cushion and flow in via the drain pipe.
Pulverized radioactive substances	Material moves due to the flow resulting from the injection of coolant water and flows in via the S/C vent pipe.	Material moves due to the flow resulting from the injection of coolant water and flows in via the sand cushion drain pipe.
Aerosols	Material moves due to the flow of gases, and flows in via the SR piping or S/C vent pipe, etc.	Material moves due to the flow of gases, and traversing the S/C, flows in via damaged vacuum break line, etc.

Table 2 Scenario where there is a relatively high probability of an inflow of highly radioactive material

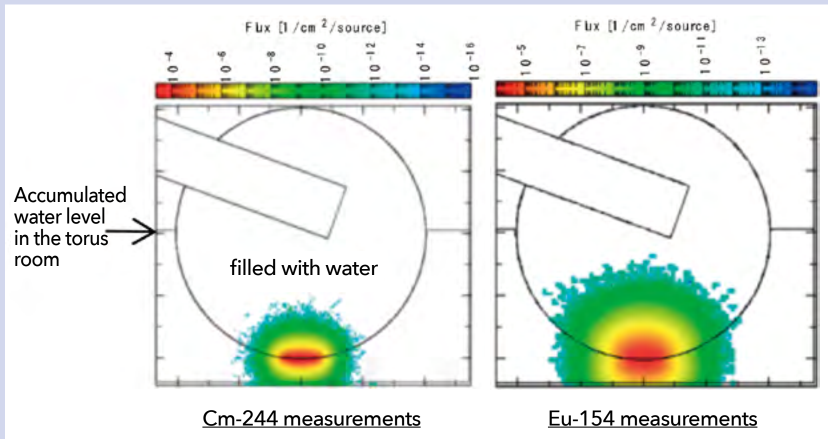


Figure 2 Distribution of radiation originating from fuel in the vicinity of the S/C (Unit 1)

With the radiation source established as fuel debris, neutron flux and gamma ray flux at the bottom of the S/C were evaluated. It was confirmed that, even in the background environment (Figure 3), Cm-244 (neutrons) and Eu-154 (gamma radiation) could be measured.

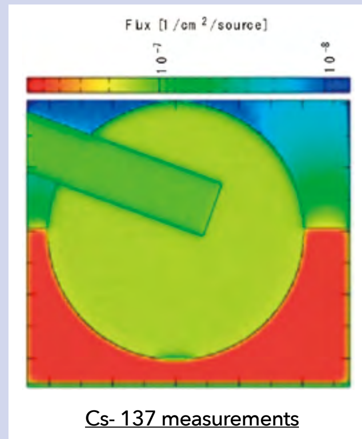


Figure 3 Background radiation (Unit 1)

After evaluating background gamma radiation in stagnant water, it was found that radiation levels were high (red zone: around 10^{-7}/cm²/source) around the stagnant water in the torus room. It was also confirmed that gamma ray flux was almost of a uniform level in both the S/C and torus room.

Development of Technology for Retrieval of Fuel Debris and Reactor Internals

Background

It is believed that fuel debris in the RPV and PCV at the Fukushima Daiichi NPS has not reached criticality at the present time. However, reactor buildings, RPV and PCV suffered damage during the accident, and the plant itself remains in an unstable condition. It is therefore vital that re-criticality is prevented while retrieving fuel debris, and the plant is kept stable to avoid the dispersion of radioactive materials.

Aims

In the lead-up to the commencement of the mock-up test of fuel debris retrieval methods (planned for FY2018) based on the Mid-and-Long-Term Roadmap (June 2013 revision), specific R&D will be undertaken. Under this project, plans for retrieval methods at each plant will be drawn up, and development plans for required equipment and element tests created. In addition, element tests of common technology will be conducted and included in the FY2015 plan.

Main Achievements

1. Setting of conditions for deciding on fuel debris/reactor internals (hereinafter, 'debris') retrieval technology

Consideration of the method of debris retrieval must be based on the situation at each plant and the information obtained from other related projects. Therefore, questions as to what kind of requirements or plant-related information existed and by when these would be clarified were set out. Additionally, in preparation for the commencement of actual fuel debris retrieval in December 2021, the required timing and accuracy of this information was clarified.

2. Designing a plan for finalizing debris retrieval methods

In reviewing debris retrieval, methods were classified into twelve types in terms of the PCV water level and the direction that debris would be accessed from; issues were identified and organized for three representative methods thought to be the most practical (Figure 1). Decision items at the time when debris retrieval methods are used at each plant were pointed out, and each corresponding situation arranged (Table 1).

3. Survey and review of existing technologies

In addition to the technical catalogs*, the applicable technologies for debris retrieval were extracted from the types of equipment used at TMI-2, Sellafield and Paks, and existing technologies and proven technologies from outside the nuclear energy field (e.g., medicine).

4. Formulation of development plans for related element technology and equipment

Element testing for technical issues that the twelve scenarios/methods have in common was implemented.

(1) Evaluation test for cutting considering fuel debris collection.
(2) Prototyping of simulated debris for use in cutting evaluation tests, etc.
(3) Position control characteristics evaluation test for access equipment for use in remote controlled work.
(4) The material selection and handling tests of isolating sheet for preventing expansion of contamination (Figure 2).
(5) Prototyping and operational testing of remotely operated arm for supplementary work under very high radiation (Figure 3).

In addition, measures to address issues identified in '2. Designing a plan for finalizing debris retrieval methods' were reviewed and a development plan was formulated.

Future Developments

Centered on the important issues of shielding/preventing spread of contamination and remote automation, technology development for the following items will be conducted, looking towards the presentation of a debris retrieval method (procedure) plan in late FY2016.

(1) Setting of conditions towards fuel debris retrieval method policy decision

(2) Formulation of development plans for related element technologies and equipment
  i. Investigation/study of existing technologies
  ii. Formulation of development plans for related element technologies and equipment
  iii. Element test/technology development
  iv. Feasibility evaluation on the actual unit/mock-up

(3) Study of fuel debris retrieval method/system/equipment

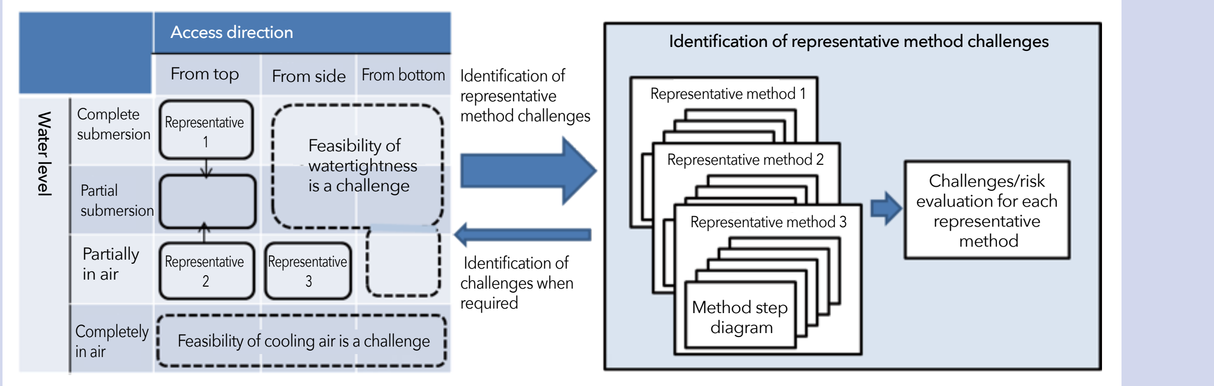


Figure 1 Representative methods and scheme of challenges extraction

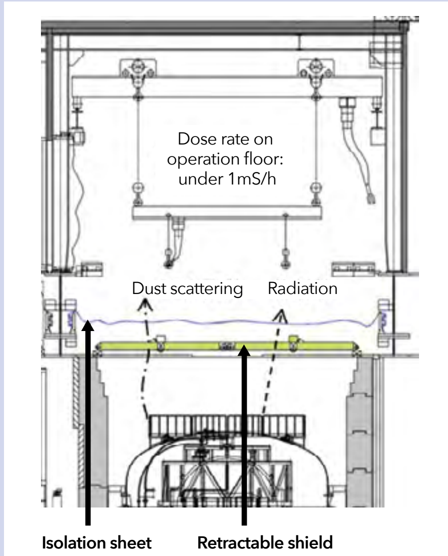


Figure 2 Use of isolation sheet for preventing the spread of contamination

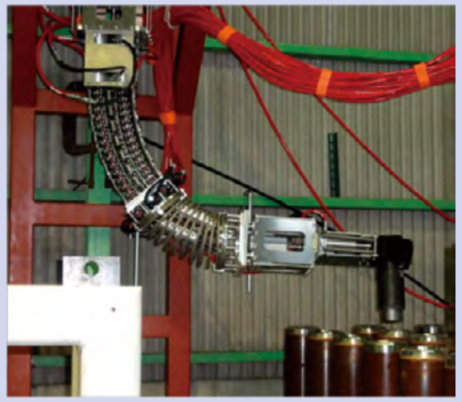
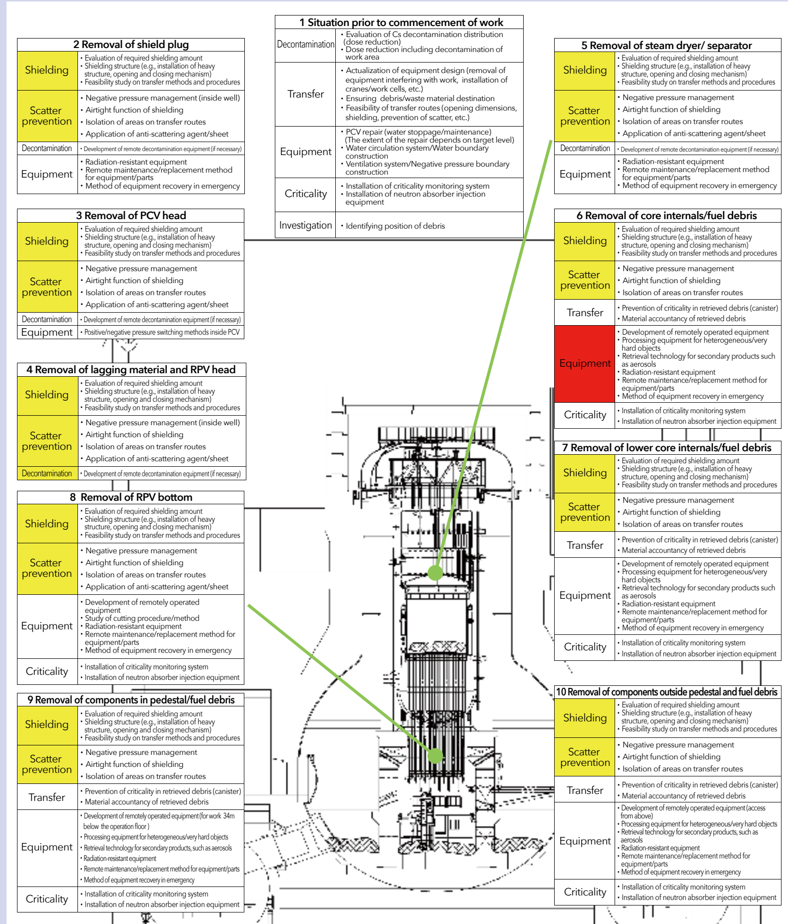


Figure 3 Handling test on remotely operated working arm (artificial muscle robot)



No	Challenges and ways to proceed in development
1	Shielding and scattering (spread of contamination) prevention are issues common to all steps and of high priority. -> The issues are related to processing devices, lifting equipment, work cells and negative air pressure control equipment used in each work step. Review of entire work process is required. (Will be effective if carried out confirming method feasibility, identifying problems, and points to improve at each work step and overall, using a model, etc.)
2	Compared to other devices, the highly radioactive environment around the reactor core presents requirements of remote automated devices that are more severe than with other devices and a higher degree of difficulty. There is a need to develop essential technology, etc., through partial mock-up testing, and to improve device feasibility at an early stage. (Preparation for actual devices is, from looking at method and device feasibility prospects, without waste and efficient.)
3	Preparation for mock-ups needs to proceed as planned for the purpose of device testing and worker training.
4	Development of safety-related design criteria, examination of safety equipment, and safety evaluations (accident assumption, risk evaluation) are required to be undertaken.

Table 1 Important issues and development procedures for each work step towards retrieval of fuel debris.

Required work processes for the methods of fuel debris retrieval (three representative methods) were divided into ten steps and challenges identified. Measures to resolve these challenges were examined and organized, and ways forward combined into four categories.

* The Agency for Natural Resources and Energy will conduct surveys and call for technical proposals publically for the FY2011 'Technical catalog for the development of equipment to prepare for fuel debris removal towards the Decommissioning of the TEPCO Fukushima Daiichi Nuclear Power Station' as a project related to the Subsidy for Development of the Technological Foundation for Decommissioning and Safety of Nuclear Reactors for Power Generation.

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 of reactor buildings and stored until the time when the method of final disposal is determined. For this reason, technology required for the collection, transfer and storage of fuel debris must be established at an early stage.

Aims

Development of storage canisters for the housing of retrieved fuel debris will be based on the established spent fuel transportation and storage technologies. In FY2014, development requirements for storage canisters at the Fukushima Daiichi NPS will be organized, and canister design approaches and basic structural design concepts will be derived.

Main Achievements

1. Study on a fuel debris collection, transfer and storage system

Scenarios for the collection, transfer and storage of retrieved fuel debris studied were as follows: Collection (Submerged/Non-submerged from above/Non-submerged from side) -> Transfer (Wet/Dry/Half-dry) -> Storage (Wet/Dry/Half-dry). In addition to confirming that all scenarios can be feasible, scenarios that are expected to be advantageous were selected in terms of safety, work efficiency and feasibility, etc., for each individual step (Figure 1).

2. Establishment of a storage canister design concept

After studying the work involved in each step of the scenarios described above, it was found that as the collection of fuel debris would take place in highly radioactive environments or in narrow spaces, emphasis would need to be placed on operability in the design process. For this reason, it was thought that the safety function of the storage canisters should be minimum, and the safety would be guaranteed by the surrounding equipment and installations instead. In future, adjustments will progress taking surrounding equipment and installations into consideration (Table 1 and 2, Figure 2).

3. Development of safety evaluation methods

Evaluation methods for determining the shape/confirming applicability of storage canisters were examined. In principle, the design methods for spent fuel transportation casks or the plant design can be used. However, it was found that for part of the safety evaluation, verification data would need to be further expanded. Additionally, it was seen that there are many items dependent on the characteristics of the fuel debris itself (temperature limits, etc.) (Table3).

4. Investigation on the transfer and storage of damaged fuel

In order to carry out 1-3 above, technical information related to the transfer and storage of fuel debris from Three-Mile Island Reactor 2 (TMI-2) in the US were collected, and sub-criticality evaluation technology, drying technology and technology to deal with hydrogen gas were studied during a visit to U.S Idaho National Laboratory (INL) where fuel debris from TM-2 are stored.

Future Developments

In order to solve issues which may become problems in developing the basic design of storage canisters, various analyses, evaluations and element tests will be performed. Achievements in other projects will be reflected to fuel debris collection/transfer/storage scenarios and storage canister design.

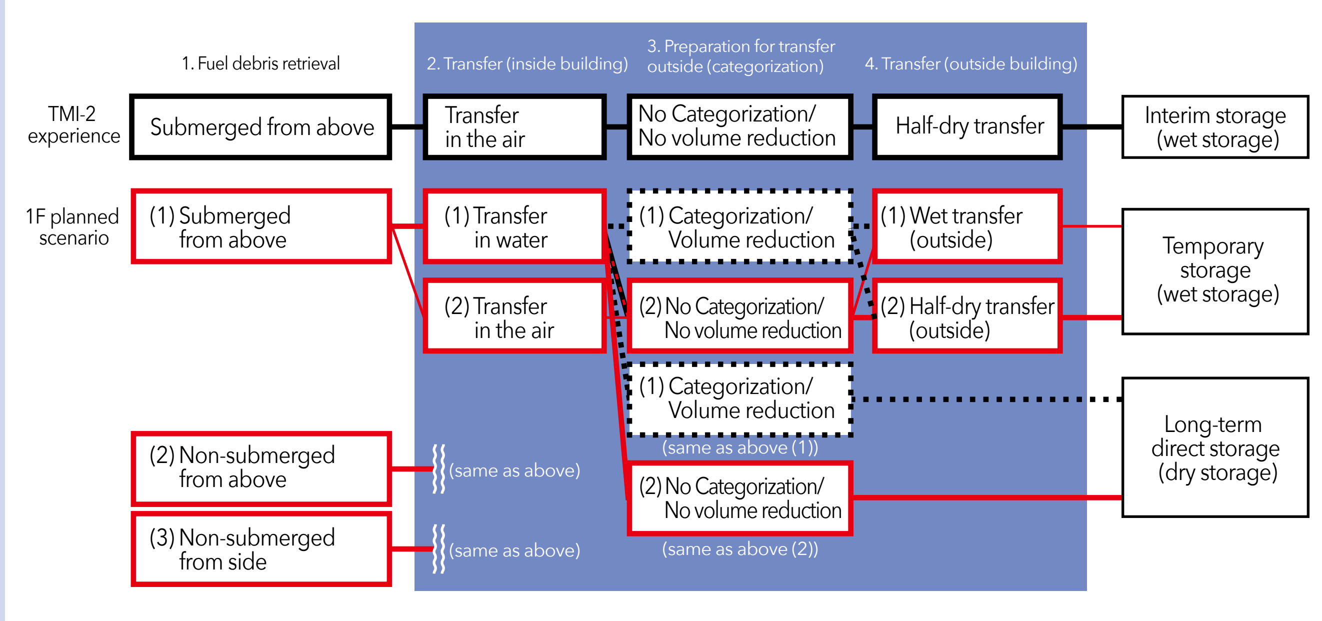


Figure 1 Scenario for collection, transfer, and storage of retrieved fuel debris (partial)

The measures in the red boxes represent an example of scenarios that are expected to be of merit in terms of safety, work efficiency, and feasibility. However, there are many unknowns about fuel debris characteristics and as knowledge increases, it is expected that the degree of difficulty and merits of each scenario may change. It is for this reason that technology development will involve taking technical elements in common without narrowing them down, and reviewing the order of priority.

Safety function	Basic storage canister concept	Relevant considerations
Heat removal	Simple natural heat radiation	Adjustments to be made to cask baskets and facility air conditioning, etc., in order to ensure temperature environment surrounding fuel debris canisters.
Structures	Weight reduction focused on minimum safety functions to maintain sub-criticality	Where required, make adjustments to utilize cushioning material, etc., to avoid large stress on canisters in the event of them falling, etc.
Shielding	Weight reduction through non-expectation of shielding function.	Adjustments to minimize worker exposure using peripheral equipment and facilities.
Confinement	Sealed construction not employed from the perspective of remote lid closure.	Suppression of radioactive materials into the environment managed through improvement of airtight performance of transfer cask and use of filtering on exhaust vents at storage facility.
Sub-criticality	It is assumed that sub-criticality will be maintained in isolation, and dimensions were kept large to the extent possible from a receiving perspective.	Sub-criticality at time of arraying storage casks to be secured through racks and transfer container basket.
Hydrogen	Reduction of hydrogen amount using catalyst process and adoption of hydrogen release mechanism using filters, etc.	Consider dealing with hydrogen outside storage canisters through use of catalysts or vent in the transfer containers.
Material property	Application of materials/corrosion inhibitors suited to expected environment.	Consider materials/corrosion inhibitors. Where necessary, exercise option to manage water quality on environment side, etc.
Fire prevention	Prevention through water injection or use of inert atmosphere (nitrogen, etc.)	Adjustment by injecting water or deploying gas supply facility for creation of inert atmosphere.

Table1 Storage canister basic concept

Assuming that the storage canisters will be used in the collection (→transfer→temporary storage) →transfer→ storage process, each step was first examined from a work environment safety perspective before establishing a reasonable storage canister concept.

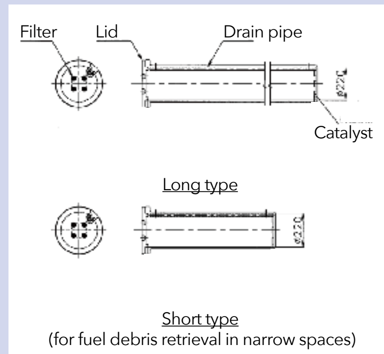


Figure 2 Storage canister basic concept (example)

A stylized example concept based on Table 1. Reviewed based on handling capability in construction process.

Issue		Future plan
Parts optimization	Flange Structure	Formation of specific plan that links with construction side for remote closing of lid, etc.
	Filter	Determined based on amount of radioactive material that is released through the filter outside canisters and structural integrity to handle canister accidents, etc.
	Drain	Formation of specific design plan that links with remote water drainage and fuel debris removal.
Internal structure		Formation of specific plan of internal structure that adapts to shape of fuel debris and retrieval method.
Dimensions etc.		Determination of each dimension based on safety evaluation.
Materials		Design of materials based on study of corrosion inhibitors.

Table2 Proposed way forward for basic storage canister design (partial)

Safety evaluation items	Future plan
(1) Evaluation of functions to maintain subcritical state.	Consider prerequisites, such as determining the nuclear fuel material that clouds pool water, etc.
(2) Evaluation of structural strength	Adjustments to prerequisites related to hypothetical accidents, etc. Complex storage canister components verified through detailed analysis and drop tests.
(3) Evaluation of aging degradation of materials (including corrosion)	Selected based on collected case examples and testing of rust prevention, etc., through a combination of stainless-steel and metal.
(4) Evaluation concerning management of hydrogen gas	Close examination of water absorption effects through testing, etc. Evaluation of catalyst recombination effect.
(5) Evaluation of radiation shielding function	No particular challenges.
(6) Evaluation of decay heat removal function	Reflect latest findings regarding fuel debris temperature limitation by linking with other projects, etc.
(7) Evaluation of confinement function	Reflect latest findings regarding amount of radiation emitted from fuel debris by linking with other projects, etc.

Table3 Proposed way forward for safety evaluation of transfer/storage (partial)

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

Background

The deterioration of the RPV and PCV structural materials at the Fukushima Daiichi NPS resulting from the high temperatures, injection of seawater and falling debris that occurred due to the severe nature of the Great East Japan Earthquake is a cause for concern. In the period leading up to the retrieval of fuel debris from the reactor core, a plan for maintaining the structural integrity of the PCV/RPV over the long term is required.

Aims

Based on seismic evaluations that take the impact of age-related deterioration due to corrosion and fallen fuel debris into consideration, an evaluation of the structural integrity of the PCV/RPV will be made. Corrosion control measures will be studied considering methods of fuel debris retrieval and PCV repair/water leak prevention from a seismic intensity perspective and used in maintaining the structural integrity of the PCV/RPV.

Main Achievements

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

A seismic response analysis model was developed for reactor building and large-sized equipment interaction that took both in air (current water level) and complete submersion methods of fuel debris retrieval into consideration under the plant conditions reflecting latest plans such as repairs (Table 1). Utilizing this model, the seismic load (shear and moment) of the PCV/RPV regions for evaluation is calculated and, assessment of fatigue strength at each region will take place and the feasibility of the submersion method etc., examined.

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

Based on the conditions of reactor buildings and PCV/RPV following the accident and on the methods of building repair and fuel debris retrieval, the necessity of changes to the seismic response analysis model or to weight was examined, and relationships between plant conditions and seismic response analysis parameters was compiled. In future, calculation of seismic load variation coefficients and consideration of methods to combine these will be made, and development of a simple method of evaluation will continue.

3. Development of anti-corrosion measures

In addition to rust-preventive agents (sodium tungstate, sodium pentaborate, etc.) identified previously, phosphates (phosphate mixed with zinc, sodium molybdate/sodium zinc oxide mixed phosphate) and sodium metavanadate were also selected as candidates, and validation tests, including those to evaluate rust inhibiting effects under high levels of radiation exposure, commenced (Table 2). After selection of an effective rust-preventive agent, an evaluation as to whether there are any secondary effects, such as adverse radiolysis effects or negative effects on the function of water treatment equipment, will be carried out.

4. Advancement of predictability of long-term corrosion thinning amount

A long-term corrosion test (target of 10,000 hours) was initiated in order to build a corrosion thinning amount prediction model (Figure 1). Based on the data obtained from this test, a corrosion thinning amount prediction model equation will be constructed and the corrosion thinning amount in the PCV/RPV up to the time of fuel debris retrieval will be calculated. These findings will be reflected in seismic strength evaluation.

5. Evaluation of pedestal erosion impact

Testing samples were manufactured in order to conduct concrete and rebar basic materials tests under high temperature heating/underwater exposure conditions, simulated block tests with the same thickness as found in actual RPV pedestals, and the tolerance tests with the small-scaled mock up. Tests and analyses will be performed following the RPV pedestal examination flow (Figure 2), and the integrity of RPV pedestals will be evaluated.

Future Developments

In addition to accelerate development of corrosion prevention measures, simple methods of evaluating seismic strength developed as part of this project will continue to be leveraged. Furthermore, the examination of an evaluation model equivalent to the method to retrieve fuel debris in air from the side and the effects of boric acid injection on the long-term integrity of equipment will be properly evaluated.

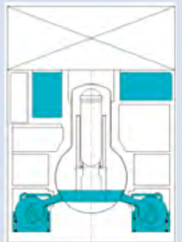
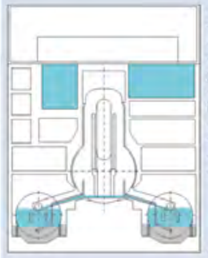
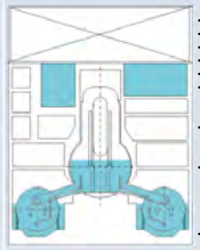
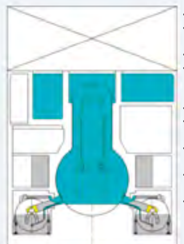
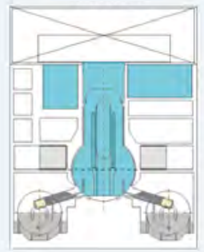
<table>
<tr><th>Plant/Case</th><th>1F-1</th><th>1F-2</th><th>1F-3</th></tr>
<tr><td>H26-1
(In air (current water level))</td><td>
• Assumed future state: Parameter (in 10 years, 15 years, 40 years)
• Building damage model
• D/W water level: approx. 2.9m
• S/C interior: Filled with water
• Torus room: OP3680
• Vent pipes: Filled with water
• Vacuum break line: Filled with water
• Additional equipment for operations floor: Parameter (None, approx. 5100t, approx. 6100t)
• Injection of the water stoppage material into individual rooms: None
• Attenuation constant: Parameter
(1) Concrete 7%, steel 4% (regulatory guide)
(2) Concrete 5%, steel 1% (for design during construction)
• Seismic waves: Current Ss</td><td>
• Assumed future state: Parameter (in 15 years, 40 years)
• Building damage model
• D/W water level: approx. 0.6m
• S/C interior: Concrete OP-1050 Water level OP3100
• Torus room: Concrete (column support upper pin position: OP-100)
• Vent pipe: Running water in lower portion
• Additional equipment for operations floor: approx. 4710t
• Injection of the water stoppage material into individual rooms: None
• Attenuation constant: Parameter
(1) Concrete 7%, steel 4% (regulatory guide)
(2) Concrete 5%, steel 1% (for design during construction)
• Seismic waves: Current Ss</td><td>
• Assumed future state: Parameter (in 10 years, 15 years, 40 years)
• Building damage model
• D/W water level: approx. 6.5m
• S/C interior: Filled with water
• Torus room: OP3200
• Vent pipes: Filled with water
• Additional equipment for operations floor: Parameter (None, approx. 4710t)
• Injection of the water stoppage material into individual rooms: None
• Attenuation constant: Parameter
(1) Concrete 7%, steel 4% (regulatory guide)
(2) Concrete 5%, steel 1% (for design during construction)
• Seismic waves: Current Ss</td></tr>
<tr><td>H26-2
(Full submersion)</td><td>
• Assumed future state: Parameter (in 15 years, 40 years)
• Building damage model
• D/W water level: Well filled with water
• S/C interior: Concrete OP3570
• Torus room: Concrete (column support upper pin position: OP2140)
• Vent pipe: Considering repair
• Vacuum break line: Considering repair
• Additional equipment for operations floor: approx. 6100t
• Injection of the water stoppage material into individual room: Yes
• Attenuation constant: Parameter
(1) Concrete 7%, metals 4% (regulatory guide)
(2) Concrete 5%, metals 1% (for design during construction)
• Seismic waves: Current Ss</td><td colspan="2">1F-3 representative

• Assumed future state: Parameter (in 15 years, 40 years)
• Building damage model
• D/W water level: Well filled with water (approx. 35m, OP39920)
• S/C interior: Concrete OP1900
• Torus room: Concrete (column support upper pin position: OP-100)
• Vent pipe: Considering repair
• Vacuum break line: Considering repair
• Additional equipment for operations floor: approx. 4710t
• Injection of the water stoppage material into individual room: Yes
• Attenuation constant: Parameter
(1) Concrete 7%, steel 4% (regulatory guide)
(2) Concrete 5%, steel 1% (for design during construction)
• Seismic waves: Current Ss</td></tr>
</table>

Table 1 Seismic response analysis cases and conditions

Rust inhibitor	Sodium tungstate	Sodium molybdate	Sodium pentaborate	sodium nitrite	Phosphate		Sodium metavanadate
					Zinc phosphate mix	Zinc/ molybdenum acid sodium mixed phosphate	
Rust inhibitor effect confirmation test results	Completed in FY2013	Completed in FY2013	Completed in FY2013	Completed in FY2013	Due to be completed in FY2014-15	Due to be completed in FY2014-15	Due to be completed in FY2014-15
Anticorrosion coating types	Oxide film type	Oxide film type	Oxide film type	Oxide film type	Precipitation film type	Oxide film type + Precipitation film type	Oxide film type
Input required for anticorrosion effect	Medium	Large	Large	Small	Small	Medium	Being confirmed

Table 2 Rust inhibitor selection result

Figure 1 Long-term corrosion testing

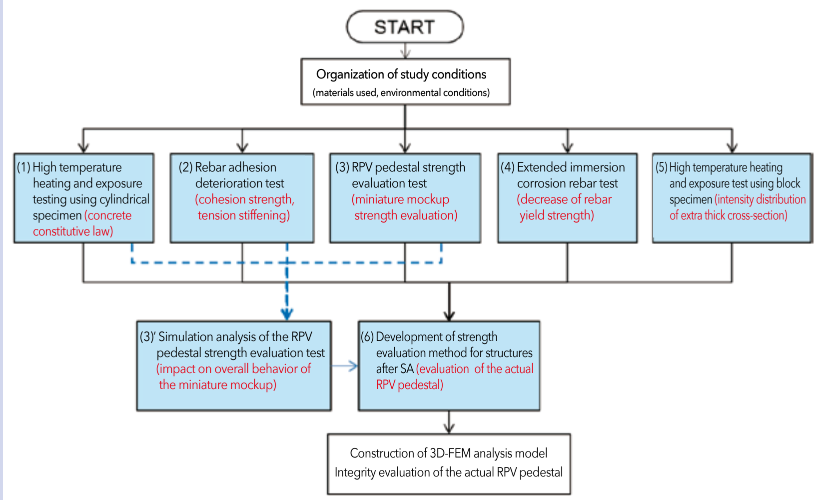


Figure 2 RPV investigation workflow

Development of Technology for Criticality Control in Fuel Debris Retrieval

Background

It is assumed that fuel debris is not currently in a critical state. However, as it is posited that changes in the shape of fuel debris or in water level may occur during the process of fuel debris retrieval in future, the development of criticality scenarios, criticality evaluation, monitoring technology, and technology to prevent criticality to prepare for such a situation is required.

Aims

In the lead-up to fuel debris retrieval, in addition to preventing re-criticality in each process, criticality safety controls to prevent excessive exposure to the general public or workers on site are to be established. In conjunction to developing criticality safety controls for the PCV submerging process, neutron absorption materials and fuel debris re-criticality detection and monitoring technology will be developed.

Main Achievements

1. Evaluation of criticality

The most recent findings regarding conditions inside the reactor were reflected in criticality scenarios for each process between PCV submersion and the retrieval of fuel debris. Additionally, evaluation of criticality of fuel debris attached to CRD piping was added, and the risk of criticality for each part, such as fuel debris at the RPV lower head and the effect of submerging fuel remaining in the reactor core, was evaluated.

After an examination of criticality controls at the time of PCV submersion based on this evaluation, it was confirmed that in criticality prevention where sodium pentaborate is used as a soluble neutron absorption material, the required boron concentration was approximately 12,000 ppm. However, when the highly enriched ^{235}U in fuel debris is set at approximately 4.0 wt.%, higher than the average value in the fuel assemblies, and in light of fuel combustion efficiency and the effectiveness of neutron absorption material such as gadolinia, the required concentration is expected to decrease.

Additionally, even in the event that criticality occurs, if the speed of water submergence is closely monitored, it can be expected that the total fission number can be suppressed to stay within the target range (Figure 1).

2. Technology for reactor re-criticality detection

The applicability of gas sampling fission product gamma radiation detector systems to actual reactors to detect re-criticality was confirmed. Specifically, a sub-criticality estimation algorithm that uses the differences in the yield of fission product nuclides produced in nuclear fission was evaluated.

Furthermore, detectors were placed in the vicinity of fuel debris retrieval positions and development commenced of an in-core criticality approach detection system to detect abnormalities before criticality is reached. In addition to extracting candidate criticality approach detection methods and making evaluations (Table 1), a study of test methods for system verification was conducted.

3. Technology for criticality prevention

Development of non-soluble neutron absorbing material is currently in progress. This technology prevents criticality when it is possible to access the fuel debris by supporting soluble neutron absorbing material through the direct application on debris. Irradiation tests that compared materials to candidate materials chosen in the previous fiscal year were executed, and candidates were selected based on elution characteristic radiation-resistance performance (Figure 2). In addition, an applicability evaluation of the fundamental physical properties of newly-devised candidate materials (underwater curing epoxy/Gd_2O_3) was carried out, and the required functions of non-soluble neutron absorbing material and input pathway for its application, etc., were examined.

Soluble neutron absorption material will be widely used to prevent criticality between PCV submersion and the retrieval of fuel debris. In order to confirm the effect the application of this material will have on the integrity of RPV/PCV, corrosion testing using a boric acid + pH adjuster and sodium pentaborate was performed. Results showed that significant corrosion was not observed with the concentration required for the suppression of criticality (Figure 3) and applicability for the prevention of criticality during PCV submersion was confirmed together with the results of a rough evaluation of insertion equipment.

Future Developments

In regard to criticality controls for the PCV submersion process examined during this year, in addition to achieving refinement by reflecting the latest findings (on debris location and so forth), further development of criticality control methods will take place while multiple methods of fuel debris retrieval involving extraction in air and in water are considered. The feasibility of the in-core criticality approach detector system applicable to this process will be confirmed through the performance of verification tests. Furthermore, collection and selection of candidate non-soluble neutron absorbing materials based on confirmation of nuclear characteristics will take place and methods of material application will be examined. The plan is to utilize these technologies in the process of selecting safe and efficient methods of fuel retrieval.

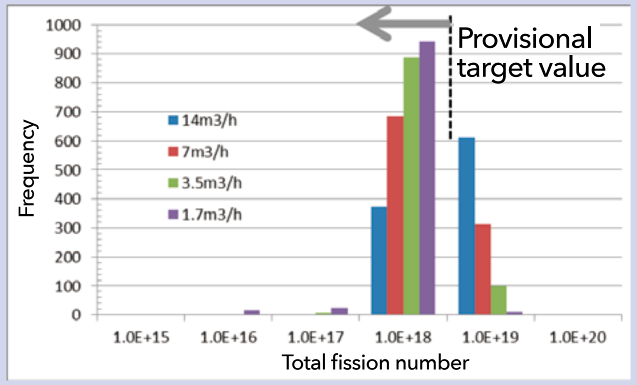


Figure 1 Evaluation of total fission number on the assumption of a criticality accident occurring at the time of PCV submersion

Using calculations that simulate behavior in the period between when criticality is reached and when it is detected and criticality suppression commences, the total fission number that affects exposure level is evaluated. Frequency distribution was sought with unknown conditions set as parameters. By limiting submersion speed (blue -> purple distribution), the total fission number can be kept under 10^{19} times, a figure that keeps the exposure level under an allowable range.

Detection method	Characteristics	Applicability
Neutron source multiplication method	- Active method that uses external neutron sources. - Neutron multiplication factor is estimated based on intensity of the neutron counting rate.	Evaluating detection efficiency of debris is a challenge.
Feynman-α method	- Passive method that does not require an external neutron source. - Neutron multiplication factor is estimated based on neutron fluctuating signal (noise) statistics.	Applicable to shallower sub-criticality with a neutron multiplication factor of 0.7 or more.
Time Interval Distribution (TID) method	- Passive method that does not require an external neutron source. - Neutron multiplication factor is estimated based on neutron fluctuating signal (noise) statistics.	Through preliminary investigations, a prospect to distinguish between spontaneous fission and fission multiplication was obtained. Detailed examination of neutron multiplication factor estimation is required for the next step.

Table 1 Criticality approach detection technique candidates

Focusing on proven conventional methods, applicability based on conditions at the time of fuel debris retrieval was examined and candidate methods were selected. In future, selection will be made based on applicability to multiple retrieval methods, including in-air and under water, and confirmation of feasibility through verification testing.

Category	Neutron absorber candidates
Solid	B_4C/Sintered metal material
	B or Gd-containing glass material
	Gd_2O_3 particle
Liquid -> Solid	Cement/Gd_2O_3 particle
	Water glass/Gd_2O_3 particle
	Underwater curing epoxy/Gd_2O_3
Liquid	Slurry/ Gd_2O_3

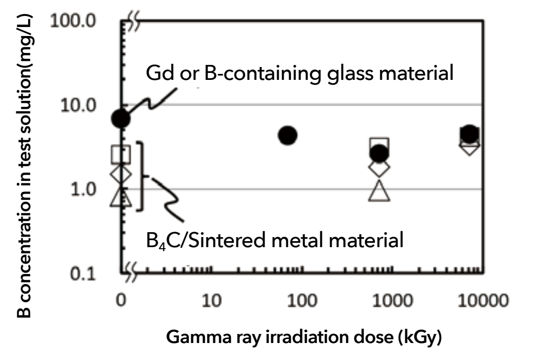


Figure 2 Non-soluble neutron absorption material dissolution test result (example)

Candidate materials that were chosen based on confirmation of basic physical properties in the previous year were singled out by confirming their characteristics under irradiation.

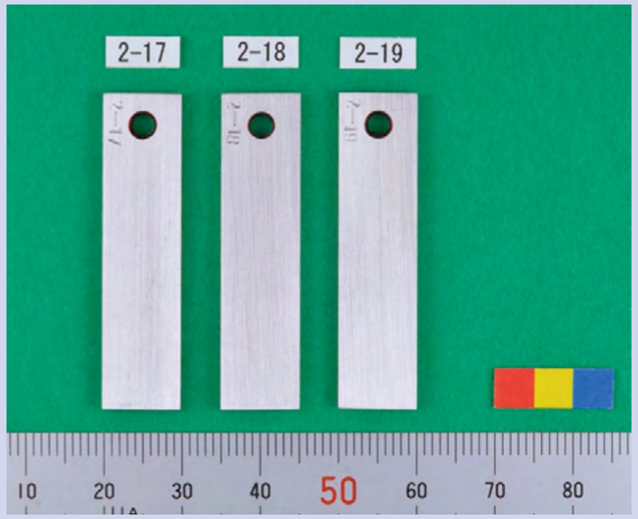


Figure 3 Soluble neutron absorption material corrosion test result (example)

Image shows results of 500-hour test on sodium pentaborate of 10,000 ppm for non-coated base material.
Stable passivation film was formed and it was confirmed that no significant corrosion was observed. Likewise, corrosion did not occur even when conditions were changed, such as testing in the presence or absence of boric acid + pH adjuster or of rust inhibitors.

Development of Technology for Fuel Debris Characterization and Treatment

Background

In order to develop, design and manufacture equipment, tools and containers for the retrieval and storage of fuel debris, the underlying data that is comprised of fuel debris mechanical properties, such as hardness, thermal properties, chemical stability or moisture characteristics must be fully understood.

Aims

In order to examine fuel debris retrieval methods or obtain/share the data or information required for examination in other projects such as that related to development of equipment or devices for retrieval and technology for collection, transfer and storage of fuel debris, testing and analysis using simulated debris will be conducted and estimates made of fuel debris characteristics.

Main Achievements

1. Analysis of the characteristics using simulated debris

a. Understanding debris characteristics data

(U,Zr)O_2, in tetragonal and monoclinic phases, and that includes ZrO_2-stabilizer, was fabricated. Hardness, elasticity ratio and fracture toughness as mechanical properties were measured and evaluated, and data on oxide-based fuel debris characteristics was expanded. At the same time, Fe_2 (Zr,U) was fabricated and evaluation of metallic characteristics commenced. In addition, boring tests on multiple cold materials with differing characteristics were performed, clarifying the impact each property had on boring performance.

U-Zr-O phase relations under conditions where hyper oxides in U-Zr-O-based oxides occur were investigated experimentally, and data was expanded in order to predict the chemical form of U. Using two kinds of melting methods – arc melting and focused radiation melting (Figure 1) – basic data related to reactions with concrete was obtained. Properties data on fine debris generated after immersion in cooling water for an extended period was acquired. In addition, solidified oxide melt debris were produced and by going through the melting process, the impact on debris characteristics could be clarified. Calculations of physical property data for systems that assume the presence of debris generated by fuel containing Gd or composite systems containing structural material (Fe) were performed.

International cooperative efforts related to molten core concrete interaction (MCCI) products enabled the use of MCCI test products stored by the French Nuclear International Agency (CEA) (Figure 2) to obtain properties data such as chemical composition and hardness. Additionally, knowledge concerning the interfacial morphology between metals and ceramics was obtained by studying metallic/ceramic solidified products using UO_2 produced by the Kazakhstan National Nuclear Energy Center (NNC) (Figure 3).

b. Comparison with TMI-2 debris

In order to evaluate the characteristics of fuel debris from the Three Mile Island Unit 2 (TMI-2) accident and compare and examine it against data obtained in the step 'Understanding debris characteristics data' above, sample processing and ceramography were conducted (Figure 4) and Vickers hardness measured using three different samples of TMI-2 debris held by the Japan Atomic Energy Agency (JAEA). In addition, in order to establish specific properties analysis techniques using actual debris samples, an alkali fusion technique was applied and analyzed. Assuming that existing facilities can be effectively utilized for further actual debris sample analysis in future, a transport study using TMI-2 debris as an example was conducted.

2. Development of debris treatment technology

While sharing information with other projects related to the collection, transfer and storage of fuel debris, a research and development plan based on future technology development needs was formulated. Additionally, in regard to hydrous/drying property data required for estimation of hydrogen generation by water radiolysis (problematic from a safety perspective), testing using porous ceramics was undertaken and factors that affect drying properties set out. Further to this, simulated MOX fuel debris consisting of UO_2, ZrO_2 and PuO_2 was manufactured, and redox behavior and properties changes in redox process experimentally were investigated.

Future Developments

In regard to the mechanical properties of fuel debris particular to the Fukushima Daiichi NPS, in addition to evaluating simulated debris, we will continue international cooperative efforts with France' s CEA to evaluate the characteristics of MCCI products and cooperate with Kazakhstan's NNC to produce solidified products containing UO_2, and compile data related to fuel debris characteristics by the end of FY2015.

In the development of treatment technologies, based on a plan formulated in FY2014, evaluation of hydrous/drying properties and so on required for the development of retrieval and storage technology will be performed.

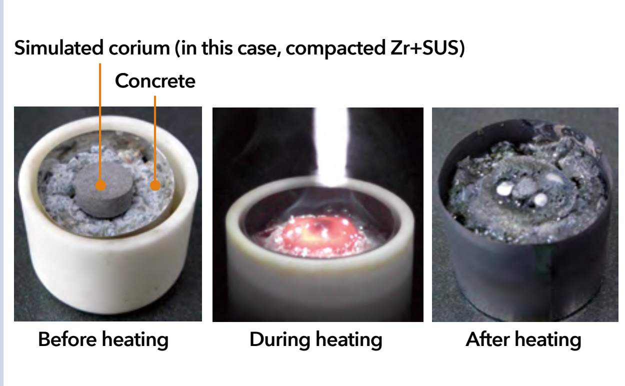


Figure 1 State of simulated MCCI products manufactured through light condensing and heating

MCCI phenomenon was simulated on a small lab-scale in the following method: , a simulated corium disk (photos show compacted Zr+SUS) was placed on a piece of concrete and the simulated corium only was melted by heating only the disk using a light condensing and heating device. Using this method, data that focused on characteristics of heterogeneous parts of MCCI that is thought to have occurred in the Fukushima Daiichi NPS accident was acquired.

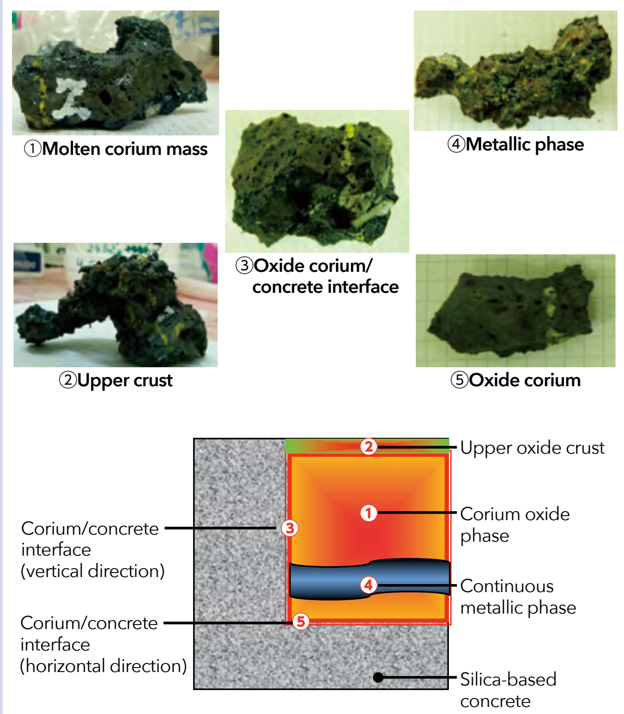


Figure 2 Sample selection of the MCCI test product from the French Nuclear International Agency (CEA)

A few campaigns were selected from previously formed MCCI test products held by the CEA, based on the criteria that conditions of the campaign should closely reflect concrete components and the ratio of fuel and structural materials found after the accident at the Fukushima Daiichi NPS. Five samples from these test products were taken and subjected to chemical composition and hardness tests. As a result, data that could be used for predicting the general characteristics of MCCI products in a short period was obtained.

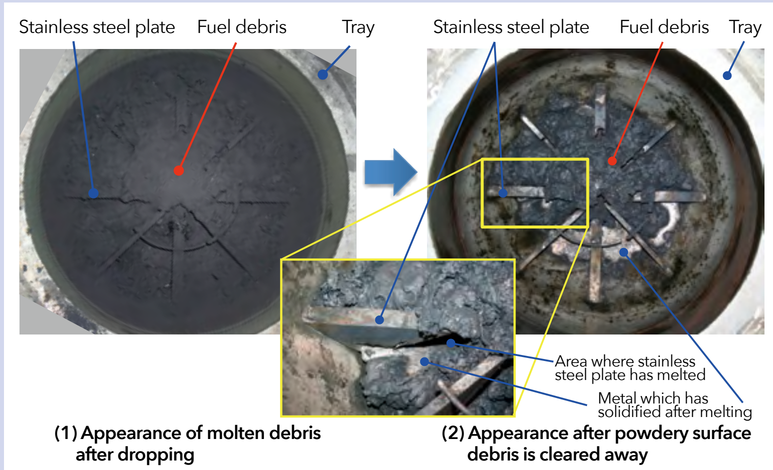


Figure 3 Result of fabrication of metal and ceramic solidified products produced at the Kazakhstan National Nuclear Energy Center (NNC)

In cooperation with Kazakhstan's NNC, a UO_2 + Zr + B_4C mixture of the order of 10kg was heated and melted before dropping into a pan placed on stainless steel of the same type used in reactor construction in order to fabricate large simulated debris. As a result of the removal of powdery surface debris, the metallic and ceramic solidified product could be confirmed (Figure 3 (2)), and knowledge of the metallic and ceramic mixture state and interface form, and of product physical properties, could be obtained.

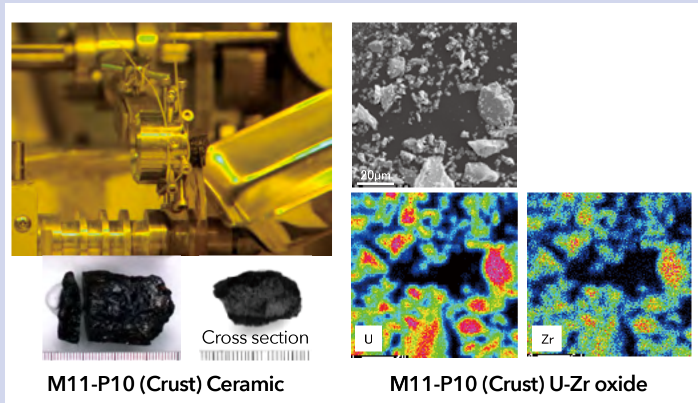


Figure 4 Cutting of TMI-2 debris sample and SEM/EPMA analysis of cutting dust

Photo at upper left shows pretreatment (cutting) of TMI-2 debris held by the JAEA before being subjected to hardness testing and analysis before treatment testing. The photo at lower left is an example of crust portion from the TMI-2 debris. The results of SEM observations and EPMA analysis of external appearance and cutting dust (right) showed that debris taken from the crust portion was primarily an oxide consisting of U and Zr.

Development of Technology for Analysis of Debris Properties

Background

Determining the properties of debris is crucial to understanding the situation inside the reactor core, stability management, and ensuring that retrieval and storage operations proceed safely and steadily. In order to provide useful information without delays, analysis-related technologies, including pre-analysis processing and wastewater treatment technologies, must be established in advance so as to handle a wide variety of material properties.

Aims

The final goal of this project is to analyze Fukushima debris and determine their physical properties in order to deploy and use the obtained information in other projects. For that purpose, preparation is to be made in an examination of the overall analysis workflow, the formulation of an analysis and measurement technology development plan, and development of individual analysis and measurement technologies.

Main Achievements

1. Establishment of development plan for fuel debris analysis and measurement technology

Analysis items and the overall workflow of debris analysis were examined in line with the needs of other projects and technical challenges were identified. Items that include basic analysis such as observation of debris shape, element/nuclide quantitative analysis for composition evaluation, and mechanical/thermal property measurements have been selected (Figure 1). The establishment of procedures for dissolving debris, the influence of flux components in quantitative analysis, and the actualization of test measurement methods have been identified as the main technical challenges. These potential challenges were collated and future development plans have been formulated.

2. Development of analysis and measurement technology

a. Development of dissolution technique using fusion method, etc.

A fusion method used as a pre-analysis processing method is being examined for the purpose of elemental analysis of fuel debris samples. In addition to Zr cladding and reactor core internal structural materials being mixed together, it is thought that MCCI products, debris reacted at high temperatures with concrete, exist and are extremely insoluble. In response to this, it was confirmed that by using sodium peroxide, simulated debris could be completely dissolved in nitric acid through the "alkaline fusion method," known as a method for the dissolution of low solubility samples (Figure 2).

b. Study on chemical state analysis method

To investigate the composition of solid debris (including chemical state), simulated solid products from assumed components were produced and a quantitative analysis using SEM/WDX solid state analysis methods was studied. The quantitative analysis method involved the measurement of composition area ratio using SEM/WDX reflection electron imaging and using spot analysis and so on as a method of identifying chemical states (Figure 3). In testing of simulated U+UO_2 solidified product, almost identical results were seen in quantitative results yielded from powder X-ray diffraction (Rietveld analysis).

c. Maintenance of debris analysis equipment

High-dose rate debris is required to be handled within a cell. For the purpose of analysis of debris, analysis equipment (SEM/EDX/WDX) and the cell to contain them were configured, and cell specifications and the required modification of equipment were examined. From a safety and operability standpoint, all of the equipment is required to be placed within the cell, which was required to be 2.5 m X 2.8 m X 2.8 m in size as a result. In regard to the modification of the equipment, locations that required modification and steps involved were clarified in terms of operability, reduction of the impact of debris radiation, and maintenance (Figure 4).

3. Study on properties analysis for the proper treatment and disposal of debris

The fuel debris produced at the Fukushima Daiichi NPS is of a type that contains internal structural components. If the appropriate analysis and classification can be employed at the time the debris is retrieved, the subsequent storage process and selection of storage location can be streamlined. Classification of debris was therefore examined in terms of 'nuclear material management', 'streamlining of storage' and 'safety at time of disposal'. In addition, a proposed sorting method for the measurement of nuclear material (Figure 5) was examined, and the sorting performance (lower detection limit) of devices using the active neutron method was analytically studied.

Future Developments

The technically-challenging fuel debris dissolution procedures will be established and technology to deal with flux components in quantitative analysis will be developed. These outputs will be reflected in the development of the overall workflow for fuel debris analysis. Working closely with other projects, in addition to further clarifying required physical property values and content, test and quantitative evaluation methods that can produce appropriate property values will be explored. Furthermore, in preparation for early analysis, containers for transportation of fuel debris for analysis will also be studied.

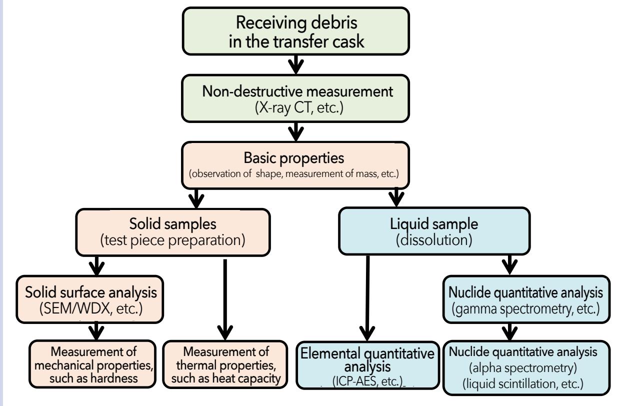


Figure 1 Overall debris analysis workflow plan (outline)

The above figure shows an outline of the overall analysis workflow considered to enable the efficient implementation of debris analysis. Under this workflow, values of debris with unknown structural components are roughly analyzed by the non-destructive method and then a work schedule of overall analyses is created.

Figure 2 Debris dissolution test

The photo shows situation immediately after removal from electric furnace during alkaline fusion. The test involved reacting simulated debris with sodium peroxide at 850°C and then, after cooling, dissolving the debris using concentrated nitric acid. No solids were observed in the resulting solution, and it was confirmed that complete dissolution had occurred.

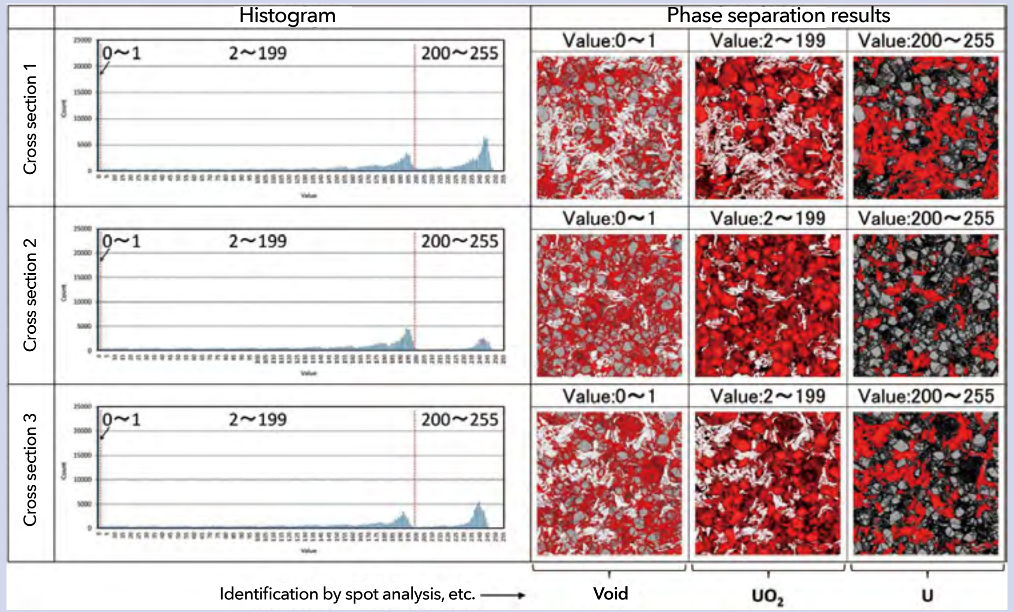


Figure 3 Results of phase analysis using backscattered electron images and gradation distribution taken by SEM/WDX

The proportion of each phase of the simulated solidified product (U+UO_2) was measured using gradation distribution of backscattered electron images taken by SEM/WDX. The chemical form of the phase was identified through SEM/WDX spot analysis.

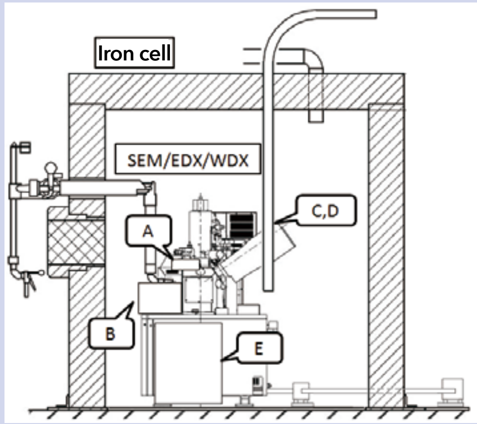


Figure 4 Cell composition and equipment modification points

The required modifications are as below:
- Automation of objective aperture (A)
- Stroke extension of sample introduction part (B)
- Addition of EDX drive unit (C)
- Installation of WDX shield (D)
- Installation of control device (E)

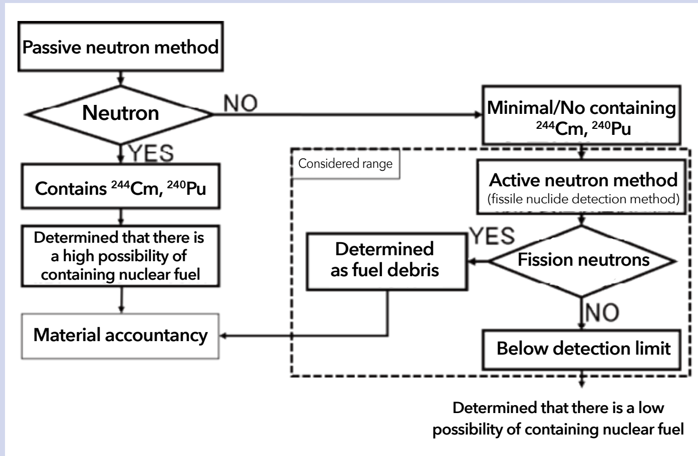


Figure 5 Proposed sorting method related to the presence or absence of nuclear material

When a significant amount of neutrons are measured using the passive neutron method, material is classified as containing nuclear fuel. When there is less than a meaningful quantity, it is dealt with using the active neutron method. When a significant amount of fission neutrons are detected, material is classified as containing nuclear fuel. When they are not detected, it is assumed the material has a low possibility of containing nuclear fuel.

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

Development of Technology for Treatment and Disposal of Accident-generated Waste

Background

Waste generated through the accident at the Fukushima Daiichi NPS contains radionuclides derived from fuel, may contain seawater components, is highly radioactive, wide-ranging in contamination level, and high in volume, making it different from radioactive waste generated at typical nuclear power plants.

Aims

Through characterization by inventory assessment based on waste product analysis and analytical techniques, the study of long-term storage strategies for stable management up to processing and disposal, investigations and basic tests related to technologies for processing and packaging waste, and research into and organization of existing disposal concepts and safety assessment techniques, technologies for the safe treatment of accident-generated waste and for its disposal will be developed.

Main Achievements

1. Characterization of accident-generated waste

Radioactivity level readings are being sought through the analysis of a variety of accident-generated waste. Contaminated water, plants and rubble (Figure 1) have been analyzed and in regard to the analysis of standing trees, ^{3}H (tritium) and ^{14}C have been detected along with ^{137}Cs and ^{90}Sr, revealing the distribution of radioactivity at the power station site. At the same time, development of analytical techniques intended for nuclides difficult to analyze at low concentrations, collation of data of the characteristics of secondary waste stemming from treatment of contaminated water (zeolite for Cs absorption, etc.), and evaluation of radioactivity levels (inventory) for waste currently difficult to analyze were executed.

2. Study on long-term storage method of accident-generated waste

In regard to the long-term storage of cesium absorption vessels used to separate and recover cesium from contaminated water, a simulation test using a 1/4 scale mock-up was used to determine conditions inside the vessels (Figure 3). Observations showed that insoluble components containing chlorine are transferred to the part that generates heat by absorbing Cs. Based on this knowledge, estimations were made by calculating temporal changes inside the vessels. Additionally, experiments also sought to determine absorption vessels material corrosion conditions.

Slurry dehydration/drying tests were conducted in order to stabilize iron hydroxide and carbonate slurry generated by the Advanced Liquid Processing System (ALPS). In addition, the amount of hydrogen gas expelled by stable products due to exposure to radiation was also evaluated.

3. Study on treatment of accident-generated waste

Basic testing and the creation of a technical catalog were conducted in the initial selection of candidate treatment and conditioning techniques. Targeting the slurry generated by the ALPS and waste absorbents, solidification tests were performed by using various types of solid-type materials and the curing process and waste characteristics were explored. Additionally, resin-based absorbent volume reduction testing and radioactive material gas phase migration experiments using high-temperature processing were conducted. In addition to making a technical catalog that includes a waste treatment, conditioning technique technical outline, applicable waste, process workflow, processing capacity, and technology readiness (Figure 4), organization of technical requirements commenced in the process of narrowing down initial candidate technologies.

4. Study on disposal of accident-generated waste

Based on the results of surveys on existing disposal concepts and safety assessment techniques (scenarios, models, and parameters), basic concepts and characteristics were arranged. In addition, safety evaluation method was developed provisionally considering characteristics of accident-generated waste assuming that existing disposal concepts would be applied to accident-generated waste.

Furthermore, analysis cases for assumed scenarios were created, and in addition to estimating the safety of each waste product, sensitivity analysis to determine parameter variation was implemented and information that contributes to the investigation of disposal classification and relates to important nuclides that affect disposal safety were extracted.

5. Study on prerequisites for R&D

In addition to classifying accident-generated waste characteristics and contamination history, a series of examples (waste management stream) looking at how storage, processing and disposal processes are to be handled was drawn up (Figure 5). Information items (waste amount, radioactive concentration, storage conditions, etc.) were also organized together with this action. Related water analysis result data was published in the FY2013 edition of the Database on Analyses of Contaminated Water (JAEA-Data/Code 2014-016). Additionally, the relationship between main work items and information items in the development of processing and disposal technologies was organized as a step towards the development of tools designed for information management.

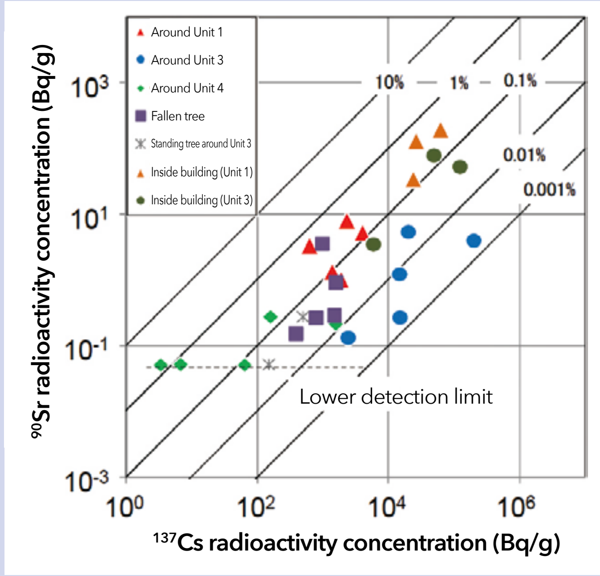


Figure 1 Relationship in concentration of ^{90}Sr and ^{137}Cs detected in rubble and plants (felled and standing trees) taken inside and outside reactor buildings

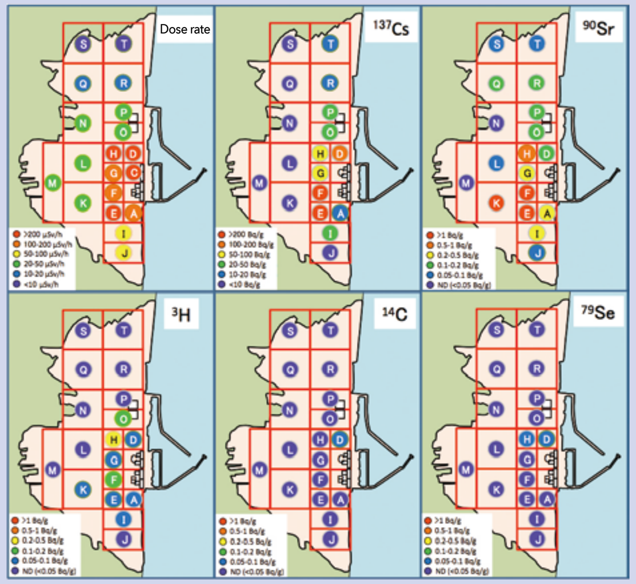
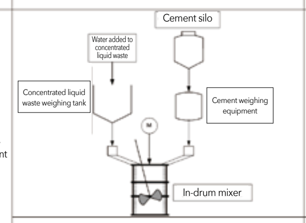


Figure 2 Distribution of radioactivity in standing trees (branches and leaves) at the Fukushima Daiichi NPS

Figure 3 Simulated tests to determine conditions inside cesium absorption apparatus

Technology No. 1	Cement kneading solidification
Process overview	This is a technique to solidify and encapsulate waste on the basis of a property of cement material to be hardened by hydration reaction.
Applicable waste material	Spent ion exchange resin, sludge, incinerated ash, etc.
Process flow and equipment overview	 The above diagram represents in-drum mixing method: the placing of weighed concentrated liquid waste together with cement and kneading water, etc., in a waste canister and kneading conducted to generate waste materials. In addition to the in-drum mixing method, out-drum mixing and vacuum injection methods are also available, See Chart 1-1 (Appendix) for more details.
Special remarks	Domestic licenses (normal Portland cement, high early strength Portland cement, ultra high early strength Portland cement, moderate heat Portland cement, sulphate resisting Portland cement, blast furnace slag cement (A-type, B-type, C-type) Admixture sometimes added during kneading (See Chart 1-2 in Appendix.)

Pretreatment	Incinerated ash: Substances that generate hydrogen (aluminium, etc.) are required to be removed beforehand. Waste liquid containing borate: Adding Ca^{2+}, borate concentration must be kept below 5wt% Waste liquid containing sulphuric acid: SO_4^{2-} concentration must be reduced by adding $Ca(OH)_2$. 8% $3CaO/Al_2O_3$ cement: Use SO_4^{2-} concentration of 150-1,000ppm 5% $3CaO/Al_2O_3$ cement: Use SO_4^{2-} concentration of greater than 1,000ppm
Processing ability and waste characteristics	(1) Processing conditions
	Temperature: Room temperature Pressure: Atmospheric pressure
	(2) Processing conditions
	Filling rate: 2-66 wt% Domestic: - International: See Chart 1-3 in Appendix
	(3) Waste characteristics
	Compressive strength: 3.2-70 MPa Density of solidified waste: 1.5-2.3 ton/m³ Radiation resistance: No change to mechanical properties at irradiation of around 10^{8}Gy. $G(H_2)$: 0.05-0.13 Heat resistance: Resistant to temperatures of below 200°C during long-term storage See Chart 1-4 in Appendix for heat reaction. Nuclide leaching rate: See Chart 1-5 in Appendix. Partition coefficient: 21 cm³/g(Cs), 11 cm³/g(Sr)
Applicability evaluation	(1) Technology maturity (Actual, etc.)
	Domestic: Many results at nuclear power stations (See Chart 1-1 in Appendix.) International: Many results at nuclear power stations or research facilities (See Chart 1-3 in Appendix.)
	(2) Operational safety
	High degree of safety as operations carried out at room temperature and dangerous material not used.
	(3) Serviceability
	Serviceability is high as operations are not required for heating etc. and device is simple.
	(4) Generated amount of secondary waste
	-
	(5) Applicability to waste matter generated by accident (organic matter, etc)
	As CN progresses from decontamination equipment sludge, it is necessary to perform pre-treatment and the decomposition of CN.

Figure 4 Example of technical catalog

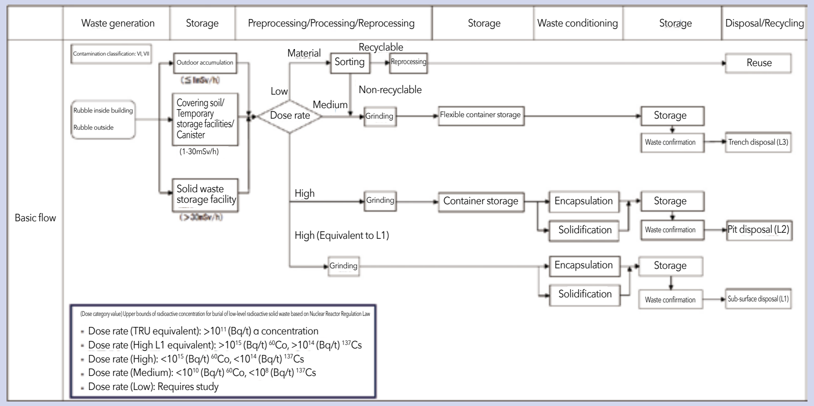


Figure 5 Example of series of handling processes for storage, processing and disposal of waste

Future Developments

In addition to continuing to analyze accident-generated waste characteristics, inventory assessment using analytical techniques will be studied. In regard to the study of long-term storage strategies, engineering tests related to the stabilization of slurry generated by the ALPS will be conducted to evaluate the entire system. Data related to the processing and packaging of waste will continued to be obtained in the study of processing techniques. Meanwhile in the study of waste disposal, existing evaluation methods will be reviewed and classification evaluation and indication of challenges regarding accident-generated waste will be undertaken. Finally, in the examination of prerequisites for R&D, organization of waste management stream candidates and information on accident-generated waste will be ongoing.

Analysis of Contaminated Objects Sampled from Reactor Buildings

Background

In order to develop remote decontamination devices, obtaining detailed data on contamination conditions inside the Unit 1-3 reactor buildings is essential. An understanding of the contamination situation and acquisition of basic data through analysis of contaminated samples from each reactor building is therefore required.

Aims

Contaminated samples taken from high does rate areas in reactor buildings (Units 1-2) will be analyzed to understand in detail the type of radioactive materials present and their distribution (particularly their permeation into concrete walls exposed to a steam environment). Steps will be taken to expand basic data that will contribute to the development of remotely controlled decontamination equipment and the formulation of a decontamination plan.

Main Achievements

1. Analysis of contaminated samples taken from the operation floor of Unit 2

In a sample taken from the central part of the well plug, the surface of the curing sheet covering the floor showed fixed contamination of ^{137}Cs radioactive material, etc., at 2.6×10^6Bq/cm^2, and fixed contamination of 1.4×10^2Bq/cm^2 was also found on the epoxy coating surface. While there were some signs of degradation of the thin epoxy coating surface, contaminants were found to have not permeated into the epoxy coating and concrete, and it was confirmed that the epoxy coating suppressed the penetration of the contaminants into the concrete (Figure 1). Fixed contamination of 1.0×10^4Bq/cm^2 was found on the thinly-layered surface coating on the ceiling of the upper part of the well plug (deck plate), and by organizing cut test pieces and performing simple decontamination tests, the effectiveness of decontamination techniques were confirmed.

2. Analysis of contaminated samples taken at the southern side of the first floor of Unit 1

In a sample taken from the area in between the X-6 penetration and piping, fixed contamination on the surface of the epoxy coating covering the floor was found to be 1.4×10^2Bq/cm^2, but was not found to have permeated and caused internal contamination. Samples taken from the vicinity of water stains at the AC pipe base showed fixed contamination on the surface of the epoxy coating covering the floor at 2.7×10^2Bq/cm^2. While permeation into the coating was not found, in places where cracks had formed in the coating, penetrating contamination in the lower concrete layer was observed to be around 1.7×10^2Bq/cm^2. It was assumed to be caused by moisture included ^{137}Cs that had passed through the cracks in the epoxy coating and had permeated into the concrete below.

Future Developments

Basic data obtained through this research on the contamination situation at each Unit will be used in the selection of decontamination technology for decontamination work at the south side on the first floor at Unit 1 and the operation floor at Unit 2, and will be used for such things as the improvement of remotely controlled decontamination equipment in other projects and will be reflected in decontamination plans.

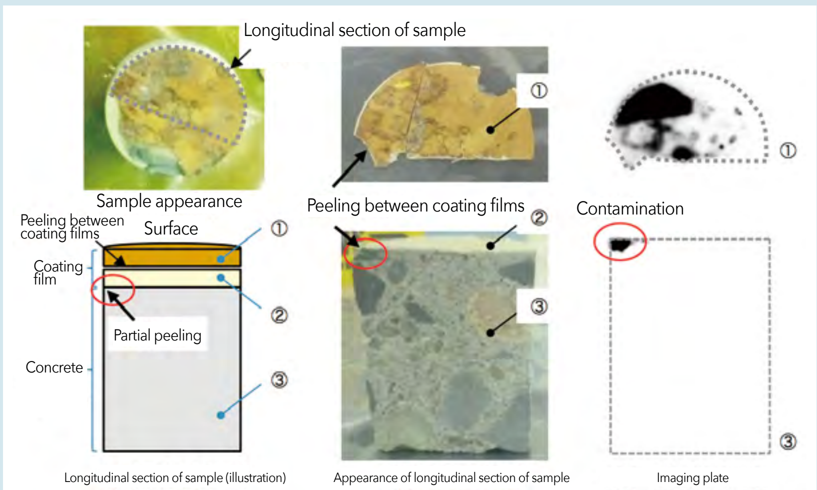


Figure 1 Imaging plate measurement results of contamination sample (Unit 2 operation floor: Floor of upper part of well)

Through measurements of sample surface and longitudinal sections, it was found that while fixed contamination was present on the surface of the epoxy coating, contamination had not permeated into the concrete. (Removed paint and contaminants found were assumed to have come from when the test sample was taken).

Main Research Results Announced/Published in FY 2014

No.	Publisher	Date	Details
1	Atomic Energy Society of Japan	July 4, 2014	Evaluation of long-term structural integrity of the fuel assemblies removed from the spent fuel pool
2	Japan Society of Maintenology	July 24, 2014	Remote decontamination technology demonstrated at the Fukushima Daiichi NPS (blast/suction decontamination using MEISTeR, core sampling)
3	Atomic Energy Society of Japan	August 22, 2014	Development of technology for criticality control in fuel debris retrieval
4	Atomic Energy Society of Japan	August 22, 2014	Evaluation of long-term structural integrity of fuel assemblies removed from the spent fuel pool
5	Atomic Energy Society of Japan	August 22, 2014	Development of technology for remotely operated decontamination inside reactor buildings (Results of dosage survey taken at southern end of Unit 1 first floor and operation floor of Unit 2)
6	Atomic Energy Society of Japan	August 22, 2014	Development of technology for remotely operated decontamination inside reactor buildings (High pressure water jet)
7	Atomic Energy Society of Japan	August 22, 2014	Localized corrosion behavior of stainless steel in zeolite containing diluted seawater under gamma ray irradiation
8	Atomic Energy Society of Japan	September 8, 2014	Identifying conditions inside reactor through application of severe accident analysis code (MAAP5 revised version)
9	Atomic Energy Society of Japan	September 8, 2014	Identifying conditions inside reactor through application of severe accident analysis code (analysis code for casting process simulation)
10	Atomic Energy Society of Japan	September 8, 2014	Identifying conditions inside reactor through application of severe accident analysis code (SAMPSON code)
11	Atomic Energy Society of Japan	September 8, 2014	Identifying conditions inside reactor through application of severe accident analysis code (sophisticated MAAP)
12	Atomic Energy Society of Japan	September 8, 2014	Development of technology for remotely operated decontamination inside reactor buildings
13	WRFPM2014	September 14, 2014	Evaluation of long-term structural integrity of fuel assemblies removed from the spent fuel pool
14	WRFPM2014	September 14, 2014	Development of technology for criticality control in fuel debris retrieval
15	The 9th Takasaki Advanced Radiation Research Symposium	October 9, 2014	Evaluation of radiation resistance of solidified inorganic specimen simulating sludge generated from water treatment in the Fukushima Daiichi NPS
16	NuMAT2014	October 27, 2014	Identifying conditions inside reactor through application of severe accident analysis code
17	ICMST-Kobe2014	November 2, 2014	Development of technology for remotely operated decontamination inside reactor buildings
18	ICMST-Kobe2014	November 2, 2014	Evaluation of radioactivity inventory of ^{137}Cs in waste generated from water treatment system at the Fukushima Daiichi NPS
19	5th Applied Laser Technology Institute Meeting to Report Results	November 20, 2014	Development of technology for fuel debris characterization and treatment of fuel debris/Development of technology for treatment and disposal of accident-generated waste
20	Japan Society of Corrosion Engineering	November 26, 2014	Development of technology for evaluating integrity of the RPV/PCV
21	Japan Atomic Energy Agency Report Meeting	November 27, 2014	JAEA efforts towards technology for the decommissioning of the Fukushima Daiichi NPS
22	2014 Specialists' Meeting on Radioactive Waste Management	November 28, 2014	Development of technology for treatment and disposal of accident-generated waste
23	Workshop on Radiation Exposure Management at the TEPCO Fukushima Daiichi NPS	December 11, 2014	Development of technology for remotely operated decontamination inside reactor buildings
24	University of Tokyo The First Human Resource Development Seminar	December 11, 2014	Radioactivity analysis for the treatment and disposal of accident-generated waste
25	Japan Atomic Energy Agency	February 12, 2015	Sector of Fukushima Research and Development Meeting to Report Results for 2014 Results of evaluation of fuel debris characteristics, study on method of processing damaged fuel, identification of conditions inside reactors, and treatment/disposal of waste
26	IAEA International Specialists Meeting	February 19, 2015	Further investigation of the fabrication and production phases of simulated fuel debris by melting and solidification of fuel and B4C control rod material.
27	Workshop with the Idaho National Laboratory (INL)	March 2, 2015	Development of technology for collection, transfer and storage of fuel debris
28	Atomic Energy Society of Japan	March 20, 2015	Development of technology for remotely operated decontamination inside reactor buildings
29	Atomic Energy Society of Japan	March 20, 2015	Treatment of contaminated water at the Fukushima Daiichi NPS and the current state of waste products generated from the water

Main Research Results Announced/Published in FY 2014

No.	Publisher	Date	Details
30	Atomic Energy Society of Japan	March 20, 2015	Preparation and characterization of simulated MCCI products (Phases and microhardness of molten solidified samples prepared by arc melting)
31	Atomic Energy Society of Japan	March 20, 2015	Preparation and characterization of simulated MCCI products (Characterization of interface between concrete and molten core)
32	Atomic Energy Society of Japan	March 20, 2015	Study on dissolution technique for damaged fuel in the Fukushima Daiichi NPS (Dissolution test of fuel debris sample from TMI-2)
33	Atomic Energy Society of Japan	March 20, 2015	Impact evaluation of seawater on corrosion of materials used in reprocessing process
34	Atomic Energy Society of Japan	March 20, 2015	Development of technology for remotely operated decontamination inside reactor buildings (Analysis of concrete core sample from south side of the first floor in Unit 1)
35	Atomic Energy Society of Japan	March 20, 2015	Radiochemical analysis of trees collected from the Fukushima Daiichi NPS
36	Atomic Energy Society of Japan	March 20, 2015	Development of inventory evaluation methods for the radioactive wastes of the Fukushima Daiichi NPS
37	Atomic Energy Society of Japan	March 20, 2015	Study on long-term storage method of spent zeolite absorption vessel
38	Atomic Energy Society of Japan	March 20, 2015	Current status of secondary waste stemming from contaminated water treatment and efforts towards processing and disposal

Main R&D Installations/Equipment

(Over 1 million yen)

No.	Project Name	Item Name
1	Evaluation of long-term structural integrity of the fuel assemblies removed from the spent fuel pool	Ultrapure water making apparatus
2	Evaluation of long-term structural integrity of the fuel assemblies removed from the spent fuel pool	Multi water quality meter
3	Evaluation of long-term structural integrity of the fuel assemblies removed from the spent fuel pool	Automatic tritator
4	Evaluation of long-term structural integrity of the fuel assemblies removed from the spent fuel pool	Spectrophotometer
5	Study of methods to process damaged fuel removed from the spent fuel pool	Pressure reduction type immersion corrosion test equipment
6	Development of technology for investigation inside the PCV	Investigation equipment for pedestal interior
7	Development of technology for investigation inside the PCV	X-6 penetration hole boring equipment
8	Development of technology for investigation inside the PCV	Penetration cable removal equipment
9	Development of technology for investigation inside the PCV	Sediment removal device
10	Development of technology for investigation inside the PCV	Alternate shielding body
11	Development of technology for investigation inside the PCV	Pre-confirmation Equipment for pedestal interior
12	Development of technology for investigation inside the PCV	Mock-up of PCV internals
13	Development of technology for investigation inside the PCV	Shield block removal equipment
14	Development of technology for investigation inside the PCV	Shape-changing device for investigating pedestal exterior
15	Development of technology for investigation inside the PCV	Scattering prevention device for use in equipment to investigate pedestal exterior
16	Development of technology for investigation inside the PCV	Equipment for measurement of fuel debris shape
17	Development of technology for detection of fuel debris in the reactor	Small-scale muon tracking system using scattering method
18	Development of technology for detection of fuel debris in the reactor	Muon tracking system using scattering method for 1F
19	Development of technology for retrieval of fuel debris and reactor internals	Equipment for evaluating sheet to prevent spread of contamination
20	Development of technology for retrieval of fuel debris and reactor internals	Remotely operated arm (artificial muscle robot)
21	Development of technology for fuel debris characterization and treatment	Ceramography image processing equipment
22	Development of technology for fuel debris characterization and treatment	Carbon coater
23	Development of technology for fuel debris characterization and treatment	Fuel debris sonic speed measuring device
24	Development of technology for fuel debris characterization and treatment	Fuel debris pressure test equipment
25	Development of technology for fuel debris characterization and treatment	High capacity thermobalance and simultaneous heat analysis equipment
26	Development of technology for fuel debris characterization and treatment	Crystal piezoelectric four-component dynamometer
27	Development of technology for fuel debris characterization and treatment	SEM elemental analysis system
28	Development of technology for fuel debris characterization and treatment	Hydraulic auto mounting machine
29	Development of technology for fuel debris characterization and treatment	Inverted metallurgical microscope
30	Development of technology for fuel debris characterization and treatment	Vacuum arc melting furnace
31	Development of technology for analysis of debris properties	Ball mill
32	Development of technology for analysis of debris properties	Small thermostatic dryer
33	Development of technology for analysis of debris properties	Measurement control system for waste sorting measurement technology
34	Development of technology for treatment and disposal of accident-generated waste	Small-scale filter press
35	Development of technology for treatment and disposal of accident-generated waste	Full-scale filter press
36	Development of technology for treatment and disposal of accident-generated waste	Slurry conditioning tank equipment
37	Development of technology for treatment and disposal of accident-generated waste	Particle size analyzer
38	Development of technology for treatment and disposal of accident-generated waste	Atomic absorption spectrophotometer
39	Development of technology for treatment and disposal of accident-generated waste	Water purification system

Joint Research/Contract Research

No.	Project Name	Type	Title	Partner	Period
1	Development of Technology for Investigation Inside the RPV	Contract research	RPV internal investigation. Element technology feasibility study (transport technology)	Tokyo Institute of Technology	Dec. 2014 - Feb. 2015
2	Development of Technology for Investigation Inside the RPV	Contract research	RPV internal investigation. Element technology feasibility study (transport technology)	Meijo University	Dec. 2014 - Mar. 2015
3	Development of Technology for Detection of Fuel Debris in the Reactor	Contract research	Measurement of fuel debris inside Unit 1 reactor at the Fukushima Daiichi NPS using muon transmission method	High Energy Accelerator Research Organization (KEK)	Jun. 2014 - Mar. 2015
4	Development of Technology for Fuel Debris Characterization and Treatment	Contract research	Evaluation of characteristics of debris which has melted and solidified	Central Research Institute of Electric Power Industry	Nov. 2014 - Feb. 2015
5	Development of Technology for Treatment and Disposal of Accident-generated Waste	Contract research	Study related to technology for evaluating waste product inventory	Central Research Institute of Electric Power Industry	Sep. 2014 - Jan. 2015
6	Development of Technology for Treatment and Disposal of Accident-generated Waste	Contract research	Study on evaporation of remaining water through zeolite	Central Research Institute of Electric Power Industry	Oct. 2014 - Jan. 2015
7	Development of Technology for Treatment and Disposal of Accident-generated Waste	Joint Research	Study on precision technology for analysis of actinide analysis reagents using capillary electrophoresis	Saitama University	Aug. 2014 - Jan. 2015
8	Development of Technology for Treatment and Disposal of Accident-generated Waste	Joint Research	Zeolite thermal conductivity measurement and analysis	Fukushima National College of Technology	Sep. 2014 - Jan. 2015
9	Development of Technology for Remotely Operated Decontamination Inside Reactor Buildings	Contract research	Study on improvement of remote operability of manipulator	Kobe University	Dec. 2014 - Aug. 2015
10	Development of Technology for Remotely Operated Decontamination Inside Reactor Buildings	Contract research	Study on 3D recognition technology surrounding robots	University of Tsukuba	Dec. 2014 - Aug. 2015
11	Development of Technology for Remotely Operated Decontamination Inside Reactor Buildings	Contract research	Study on camera calibration method for simulated bird's eye view image generation system mounted on remotely operated robot	University of Tokyo	Nov. 2014 - Aug. 2015

IRID International Research Institute for Nuclear Decommissioning

5F, 3 Toyokaiji Building, 2-23-1 Nishi-Shimbashi,
Minato-ku, Tokyo 105-0003, Japan
Phone:+81-3-6435-8601
URL http://irid.or.jp