Technical Strategic Plan 2016 for Decommissioning of the Fukushima Daiichi Nuclear Power Station of Tokyo Electric Power Company Holdings, Inc.

July 13, 2016

Nuclear Damage Compensation and
Decommissioning Facilitation Corporation

1. Introduction

It has been five years since the accident at the Fukushima Daiichi Nuclear Power Station of Tokyo Electric Power Company Holdings, Inc. (Fukushima Daiichi NPS). At the time immediately after the accident, the fragments of the building materials and rubble in various sizes shattered by the explosion were scattered around the building on the site. The rubble has been moved and on-site radiation level has decreased, therefore allowing preparation work for decommissioning to be carried out smoothly.

R&D and projects have been developed based on the "Mid-and-Long-Term Roadmap towards the Decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Station Units 1-4" (the Roadmap) released by the Japanese Government in December 2012. The plant situation of each unit has become clearer in some ways. As a result, the technical difficulties and challenges in the decontamination of the Reactor Building (R/B), water sealing and repairing of the Primary Containment Vessels (PCV) are now better understood.

Thus, progress will be made in the internal PCV condition analysis resulting in the identification of further technical issues, field work and R&D, acquisition of data on the applicability of the developed equipment, and investigation of related technologies in Japan and abroad; these processes and techniques will be adapted as the social climate and expectations change. Under these circumstances, there is a growing necessity to make technical judgments based on the latest information by incorporating the concept of the PDCA cycle[1]. As a result, it has now reached the stage where strategic and realistic considerations and decisions from firm technical bases need to be carried out. In accordance with the latest information for the conditions, the successful retrieval of fuel debris[2] will be carried out despite the various technical challenges.

The overall approach to the decommissioning of the Fukushima Daiichi NPS was started based on the Roadmap. Since the establishment of the Roadmap, measures have been taken for the most urgent issues, such as the contaminated water management; however, the development of a mid- and long-term decommissioning strategy is also essential "to reduce the risks posed by radioactive materials over a long period of time" in addition to the short-term measures.

For this reason, Nuclear Damage Compensation and Decommissioning Facilitation Corporation (NDF) was established on August 18, 2014 by reorganizing the former Nuclear Damage Compensation Facilitation Corporation, as an organization responsible for technical studies to deliver decommissioning properly and steadily from mid- and long-term perspective.

The NDF launched the "Technical Strategic Plan for Decommissioning of the Fukushima Daiichi Nuclear Power Station of Tokyo Electric Power Company (the Strategic Plan)" as part of its statutory obligations "to provide advice, guidance and recommendations for ensuring an appropriate and steady conduct of decommissioning of the Fukushima Daiichi NPS" and of "R&D for technologies required for

[1] PDCA cycle (plan-do-check-act cycle) is a method that facilitates the manufacturing control for the operations and control work such as quality control. Operations are continuously improved by repeating the four steps: Plan→ Do→ Check (evaluation) → Act (improvement).

[2] Nuclear fuels molten and mixed with parts of reactor internals due to loss of reactor coolant and resulted in a re-solidified state.

decommissioning" based on the Nuclear Damage Compensation Facilitation Corporation Act. The Strategic Plan was formulated as a mid- and long-term strategy according to actual conditions of Fukushima Daiichi NPS.

The NDF released the Strategic Plan 2015 on April 30 2015 after a series of discussions with the related organizations including the Japanese Government and Tokyo Electric Power Company Holdings, Inc. (TEPCO), and R&D institutions including International Research Institute for Nuclear Decommissioning (IRID) and Japan Atomic Energy Agency (JAEA) regarding the situation at the Fukushima Daiichi NPS and the status of the R&D on the decommissioning and its concerns.

The Strategic Plan 2016 was formulated based the status of the progress made in the site conditions and technical development made over the last year since the Strategic Plan 2015 was released.

The Fukushima Daiichi NPS is appointed as Specified Nuclear Facility on November 2013 based on the Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors. TEPCO takes responsibility as an operator to implement of the decommissioning of the Fukushima Daiichi NPS. They are licensed to implement decommissioning by the submission of “Implementation Plan of the Measures for the Specified Reactor Facilities at Fukushima Daiichi Nuclear Power Station" (implementation plan) based on the "The matters for which measures should be taken" (Matters to be addressed) issued by Nuclear Regulation Authority (NRA).

The Japanese Government establishes the Roadmap and manages the progress of the decommissioning of the Fukushima Daiichi NPS and contaminated water issue based on the Roadmap to find the fundamental solutions.

The research institutions including IRID and JAEA that performs R&D based on the Roadmap, efficiently and effectively carry out the R&D required for the decommissioning gathering the knowledge and experiences from all over the world.

The NDF examines the key issues presented by the Japanese Government and reports them as the Strategic Plan. For TEPCO, the NDF provides technical advice and guidance in order to realize steady progress in decommissioning. It is also expected to play a central role in technical areas, such as establishing a close alliance with R&D institutions such as IRID and JAEA, and sharing information for the progress and challenges for the successful advancement of R&D projects.

The activities to make the basic technology research to be applied to the actual site are being enhanced through the Decommissioning R&D Partnership Council established for the purpose of contributing to the decommissioning of the Fukushima Daiichi NPS in July 2015, and a link with Collaborative Laboratories for Advanced Decommissioning Science (CLADS) of JAEA has been enhanced.

Furthermore, technical cooperation agreement was concluded with Chubu Electric Power Co., Inc. in April 2015 and exchange of information for the decommissioning of Unit 1 and 2 of Hamaoka NPS which is the same type of the reactor at Fukushima Daiichi NPS and inspection of power station have been carried out.

Furthermore, cooperative relations have been established with Nuclear Decommissioning Authority in the U.K. (U.K. NDA), Commissariat a l'energie atomique et aux energies alternatives in France (France CEA) and U.S. Department of Energy (U.S. DOE) and contract researches and information exchange have been performed on the decommissioning strategy, analytical techniques, and risk assessment method.

Figure 1-1 shows the roles of the organizations involved in the decommissioning project of the Fukushima Daiichi NPS and positioning of the NDF. Also, the structure of decommissioning project by the Japanese Government is shown in Appendix 1.

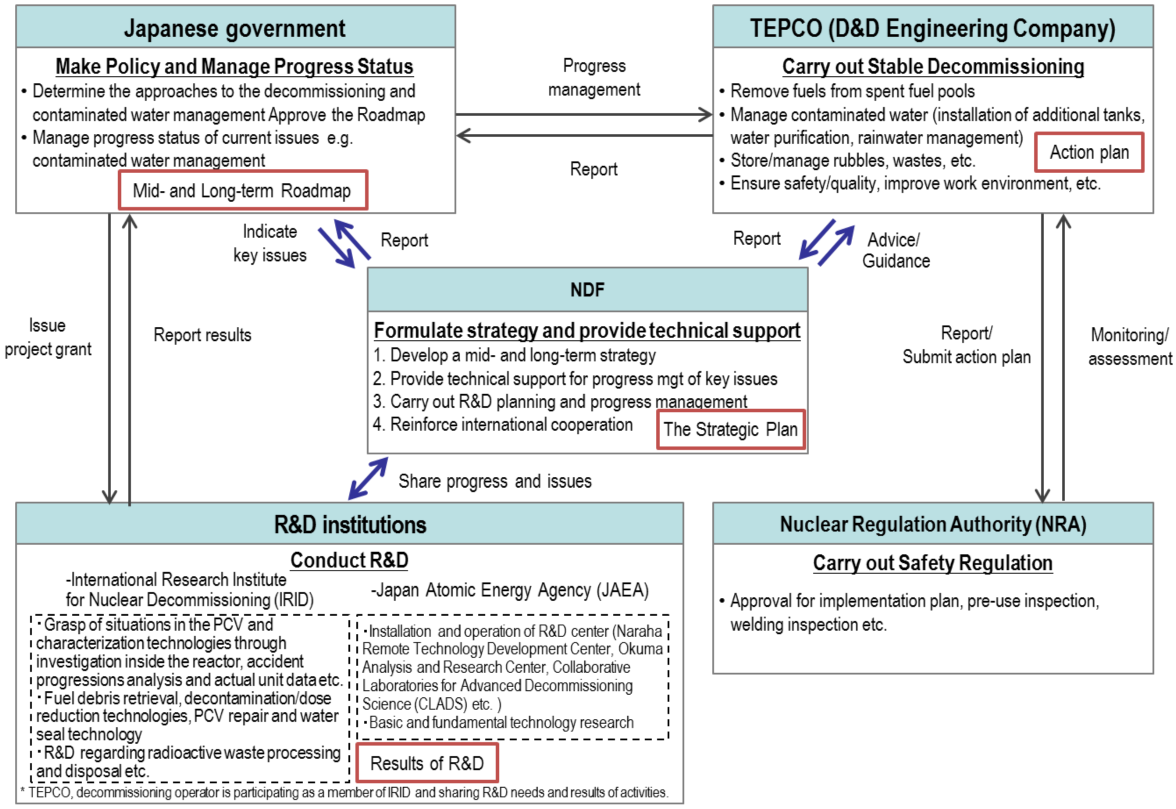


Figure1-1 Roles of the organizations involved in the decommissioning project

2. The Strategic Plan

2.1 Progress toward the decommissioning of the Fukushima Daiichi NPS

Since the release of the Strategic Plan 2015 in April 2015, there has been progress at the Fukushima Daiichi NPS as follows:

(1) Management of contaminated water

Measures based on the three fundamental policies (removing contaminated sources; isolating contaminated sources from the water; and preventing leakage of contaminated water) are being taken on the contaminated water generated from mixture of the groundwater flowing into buildings and cooling water for the fuel debris.

With regard to the measure of removing contaminated sources, progress was made in purifying the contaminated water such as by using multi-nuclide removal systems. The treatment of highly contaminated water containing strontium (reverse osmosis salt water) was completed on May 27, 2015. Efforts are still being made to the treated water requiring further purification and to treat the contaminated water that is newly generated. Removal of highly contaminated water in the trenches has also been finished.

In the measure of isolating contaminated sources from the water, water is being pumped from wells near the buildings (subdrain system) to reduce the amount of groundwater flowing into the R/B and T/B, in addition to pumping out the groundwater using groundwater bypassing systems. The flow rate into buildings was approx. 400 m^3/day during the peak time; however, it is now reduced to approx. 150–200 m^3/day by operating the subdrain system. Installation of freezing pipes has been completed for the land-side impermeable walls to block groundwater, and now the mountain-side area has been gradually frozen in conjunction with the sea-side area. In addition, measures are being taken including the construction of waterproof pavement to prevent rainwater from being permeated into the soil (facing).

With regard to preventing leakage of contaminated water, the sea-side impermeable walls that are designed to block groundwater flowing out from Units 1–4 sites to the port in order to prevent the spread of contamination in the ocean were closed on October 26, 2015. The groundwater flowing into the sea-side is blocked by the sea-side impermeable walls and it is pumped out of wells (the groundwater drain) set on the revetment. Although additional tanks (including replacements with welded tanks) for storing water purified by multi-nuclide removal systems have been placed, ensuring water storage capacity, there is a concern about the impact on the future use of site.

(2) The removal of fuels from the SFP

Removal of fuels from the Spent Fuel pool (SFP) of Unit 4 was completed on December 22, 2014. At Unit 1, since rubble on the operating floor are impeding the removal of fuel, measures against dust scattering caused by the removal of rubble is being prepared following the completion of removal of the roof panels of the building cover on October 5, 2015. Fuel removal is scheduled to start in FY 2020. At Unit 2, review is underway with the idea of demolishing the entire surface of the top of R/B operating floor before starting to remove fuel debris. At present, studies are being carried out to ensure the work area such as for installing large heavy equipment. At Unit 3, decontamination of the operating floor and application of shielding are being carried out to start removing fuels in the SFP within FY2017. On August 2, 2015, the removal of the largest rubble (fuel handling machine) was finished.

(3) PCV internal survey:

The locations of fuel debris inside the reactor of Unit 1 were examined using cosmic ray muons during the period of February 12 to May 19, 2015. It indicated that there was no lump exceeding 1m, located around the reactor core. From April 10 to 20, 2015, there was preliminary surveillance for exploring the location and shape of molten fuels that had fallen into the PCV. Robots were inserted into the PCV for the first time, and this allowed us to obtain an image of the inside as well as information for the dose and temperature. At Unit 2, measurements of inside the reactor with muons started on March 22, 2016. There was a plan to insert robot from the PCV penetration (X-6), and conduct investigations by using the replacement rail for the Control Rod Drive (CRD) mechanism to access inside the pedestal. However, measures to reduce radiation dose is currently being studied, since it took time to remove some of the shielding blocks installed in front of X-6 and radiation dose is high in the area (in the periphery of X-6) where equipment is planned to be installed.

At Unit 3, investigation devices were inserted from the PCV penetration (X-53) during the period of October 20 to 22, 2015. This provided with an image of the inside the reactor and information for the radiation dose and temperature.

(4) Waste management

The storage amount of solid wastes has increased due to the secondary waste generated from water treatment as well as the rubble removed, associated with the progress in treating contaminated water. TEPCO is enhancing the structure of the waste management department, and working to control the generation of wastes. Also, the waste storage plan was announced based on the projection of the amount of solid wastes that will be generated for the next decade.

(5) Work environment

Decontamination work is being carried out as a dose reduction measure for improving the work environment, thereby increasing the area which does not require full-face masks to 90% of the entire site. Also the on-site radiation dose except for the peripheries of Units 1-4 was reduced to less than 5μSv/h and dose rate at the site boundary of less than 1 mSv/year (assessment value), which is the objective of FY2015 was achieved. Opening the large-scale rest center for workers on May 31, 2015, convenience for workers has been improved. Meanwhile, as for decontamination inside the R/B, the dose on the first floor was reached to approx. 3 to 5 mSv/hour. There are, however, still highly contaminated areas with radiation dose exceeding 10mSv/hour. Little progress has been made in decontamination for the investigation areas including the piping of Atmospheric Control System (AC) and peripheries of Drywell Humidity Control System (DHC) of Unit 1, revealing its difficulty. Decontamination on the top floor is scheduled in future.

(6) R&D activities

The Decommissioning R&D Partnership Council was established by the Team for Countermeasures for Decommissioning and Contaminated Water Treatment in the NDF, enhancing research and development through the promotion of collaboration with the relevant institutions. The Collaborative Laboratories for

Advanced Decommissioning Science (CLADS) was established by JAEA as a global research and development organization. JAEA also began operation of the Naraha Remote Technology Development Center where the development and the verification test of the remote operation equipment (such as robots) will be carried out.

2.2 Positioning and purpose of the Strategic Plan

(1) Approach required for the decommissioning of the Fukushima Daiichi NPS

Decommissioning of the Fukushima Daiichi NPS is different from that of a normal nuclear power plant and is an unprecedented project to date in Japan as well as overseas. There are four plants that experienced core damages and/or hydrogen explosions. Even comparing with the U.S. TMI-2 which experienced a similar accident in the light water reactor, the conditions of the Fukushima Daiichi NPS are far severe in terms of the extent such as of core damage, number of units involved and the environments. The studies from different perspectives are essential because the plant conditions are still largely unknown (especially the internal condition of the PCV) and with so many uncertain factors. Moreover, some of these risks are in a trade-off relationship.

Such decommissioning and waste management of the damaged reactors that involve a lot of uncertainties require short term approaches where major works, such as contaminated water treatment, are required to be carried out immediately and long term approaches of several to several tens of years in span with a view to a future concept. That is, if short-term approaches are not adequately addressed, it may affect the mid-and-long- term and future approaches. Predetermination on the future approaches may restrict the conditions or actual works to be implemented over mid- and- long- term. Therefore, causes and effects may be complicatedly affected each other. An optimal approach must be formulated on a time axis because, for example, treatment of retrieved fuel debris and radioactive waste generated from handling and retrieval of fuel debris must be considered in line with the studies on the fuel debris retrieval plan while considering the status of the measures to the contaminated water since it affects the storage areas.

Removing the highly contaminated water in the sea water piping trenches on the seaside, the risk of environmental pollution caused by the leakage of stagnant water was significantly reduced comparing with that of before. Thus, NRA established the Committees of Radioactive Waste Management for Specified Nuclear Facility as a structure to conduct study on the stable waste management from the stage before the implementation plan is materialized assuming the long-term decommissioning work, dissolving Liquid radioactive Discharge Working Group developmentally. In addition, it is required to establish the approaches incorporating not only ensuring safety in the decommissioning but also social risks such as a delay in the decommissioning project and trigger of reputational damage.

(2) Relation between the Roadmap and the Strategic Plan

The decommissioning of the Fukushima Daiichi NPS has been conducted in accordance with the Roadmap, a document that encompasses the policies established by the Japanese Government. The first document, reported such as to the Japanese Government and TEPCO to illustrate the mid- and- long- term plan of the Fukushima Daiichi NPS was a report titled "Results of Deliberation to Formulate a Mid- and Long-Term Strategy for Cleaning Up the Fukushima Daiichi Nuclear Power Plant" (dated December 7, 2011) produced by Advisory Committee for Formulating Mid- and Long-term Strategies to Clean up the Fukushima Daiichi NPP of TEPCO (Advisory Committee) established in Japan atomic Energy Commission. Subsequently, the first edition of the Roadmap was issued by the Japanese Government and TEPCO's Mid-to-Long Term Countermeasure Meeting established under the Nuclear Emergency Response Headquarters on December 21, 2011 and there have been three revisions since then. (Refer to Appendix 2)

The Roadmap describes important elements including objectives, policies and plans of the decommissioning project. In response to this, the NDF provides TEPCO with instructions/advice while formulating the approach toward the objectives, concept of the decision-making and order of priorities. As described in Chapter 1, the Strategic Plan was formulated to study the concrete policies and requirements as a total plan for the approaches, such as field works and research to implement the strategy while sharing the progress status and issues with TEPCO that carries out the decommissioning and the manufactures and research institutions that performs R&D.

(3) The purpose of the Strategic Plan

The purpose of the Strategic Plan is to contribute to steady implementation and revision of the Japanese Governments' Roadmap in order to facilitate appropriate and steady decommissioning of the Fukushima Daiichi NPS, that is, to provide a firm technical base to the Roadmap.

From now on, R&D for highly technical issues such as fuel debris retrieval, technical studies on the site constructions and site operations will be carried out in full scale. Therefore site conditions and R&D progress must be fully understood and a practical strategic plan with firm technical basis must be specified so as to share the procedures and concept of the decision-making for the technologies (technical strategy) among the relevant parties and workers.

In order to formulate the Strategic Plan, the Decommissioning Strategy Board consists of a group of experts to perform reviews from various technical areas as well as the Expert Committee to hear opinions from experts and representatives of related institutions on specific subjects were established. Overseas experts, appointed as the International Special Advisors, are invited to the Decommissioning Strategy Board and provided experiences and knowledge through various types of technical meetings.

(4) Perspective and scope

Although the relationship with the local communities and society and the impacts on funds are the factors that must be examined, the study will be carried out from the technical perspective in accordance with the roles of the NDF to formulate a strategy and provide technical support as described in Chapter 1. The Strategic Plan will not be limited to plans for the field work but cover an overall decommissioning plan including necessary R&D and technical studies on the site constructions as well.

The Strategic Plan was formulated to contribute to the revision and implementation of the current Roadmap and provide it with firm technical basis. Therefore, the issues important from mid- and long-term perspective, which are fuel debris retrieval and radioactive waste, are studied in this plan.

In addition to the actions carried out in the Fukushima Daiichi NPS, the scope of studies includes technical R&D required for fuel debris retrieval above and waste management as well as the R&D centers developed by JAEA near the Fukushima Daiichi NPS (Naraha Remote Technology Development Center and radioactive material analysis and research center). The demonstration and training using the facilities of Units 5 and 6 are also discussed as part of the study.

(5) Continuous review based on the progress

The Strategic Plan is required to develop future project management though visualizing and sharing the concrete actions to be taken with related organizations. Furthermore, the Strategic Plan will be continuously evaluated and reviewed in light of the project assessment using the PDCA cycle and based on the changes in the site situation and the results obtained by the research institutes.

Revised version is released as a complete version in order to share the latest site situation and R&D results instead of providing only the information revised this time.

(6) Overview of the Strategic Plan 2015

The basic concept of the Strategic Plan released in April 2015 was to continuously and promptly reduce the risks associated with the radioactive materials in the Fukushima Daiichi NPS, and risk reduction strategy was formulated for the risks represented by the significant effect (Hazard Potential) and the likelihood of loss of containment function due to radioactive materials (risk sources) such as fuels, contaminated water and waste.

The major risk sources are categorized into three levels depending on the order of priority and the measures are already being taken for those risks requiring immediate actions. Therefore this Strategic Plan focuses on the areas of study, the fuel debris retrieval which requires thorough preparations and has a number of challenging issues, and the waste management that requires to be addressed on a long-term basis.

The technical studies on the fuel debris retrieval and the waste management are set out based on the following : Five Guiding Principles to risk reduction; 1)Safe- Reduction of risks posed by radioactive materials and work safety, 2) Proven- Highly reliable and flexible technologies, 3) Efficient- Effective utilization of resources (e.g. human, physical, financial and space), 4) Timely- Awareness of time axis, 5) Field-oriented- Thorough application of Three Actuals (the actual place, the actual parts and the actual situation).

Since the Strategic Plan deals with a wide range of subjects, "logic tree" style is used for explanation of logics throughout the document in order to ensure that every detail is covered comprehensively and to serve as an aid to understand the logical development.

This strategic plan describes possible methods for the fuel debris retrieval method. After the selection of the methods to be focused on, the current status and the future actions for the technical requirements for the Submersion and Partial submersion methods are discussed.

Safe and steady storage of radioactive solid waste and study of the processing method and disposal concept from the mid- to long-term perspective are important for waste management. The internationally established safety principles on radioactive waste disposal in general are to be summarized as well as the approaches to processing which is to be considered in relation to the disposal method in preparation for formulation of more specific disposal management of radioactive solid waste.

For the facilitation of R&D, these R&D projects are reviewed and managed integrally. Overall optimization is to be carried out by clear division of roles based on the characteristics of each organization and expected results and close cooperation among the concerned organizations and the proposals were made on the plan for the next development project.

(7) Positioning of the Strategic Plan 2016

The Roadmap revised in June 2015 provided that "Determination of fuel debris retrieval policies for each unit" in summer of FY 2017 as an immediate milestone for fuel debris retrieval and "Basic concept for radioactive solid waste processing and disposal" in FY 2017. The Strategic Plan 2016 develops the concept and method according to the concept and approaches of the Strategic Plan 2015 in order to serve as an aid for smooth and steady implementation of the Roadmap.

This will provide the plans which should be addressed including the directions of studies to make a "Decision on the approaches " based on the realty with a view to the technological challenges that became to be cleared by the status of the progress made in the field work and inspections.

2.3 Basic concept of the Strategic Plan

2.3.1. Fundamental policy

As the Specified Nuclear Facility, the Fukushima Daiichi NPS has been taking necessary safety measures as obligated in the "Matters to be addressed" by the NRA and a certain level of stable condition has been maintained.

However, since the Fukushima Daiichi NPS is in the state different from the normal nuclear power plants in terms of degree of building damage, presence of fuel debris and spent fuels, contaminated water containing radioactive materials and various types of radioactive waste, the risks associated with radioactive materials may arise in the course of the future decommissioning work. Therefore it should be recognized that the decommissioning of the Fukushima Daiichi NPS has higher risks posed by radioactive materials than that of normal nuclear power plants.

If no actions are taken, the risks from the radioactive materials continue to exist. Even though the risks may gradually be reduced by radioactive decay, there may still be increase of risks resulting from degradation of the facilities over mid- and long-term. Therefore it cannot be necessarily stated that the risks simply decrease over time.

For this reason, "to continuously and promptly reduce the risks associated with the radioactive materials generated by the accidents" is the fundamental policy of decommissioning of the Fukushima Daiichi NPS. Therefore, the Strategic Plan can be called as "the design of risk reduction strategy" on a mid- and long-term basis. The Five Guiding Principles that taken into account of the Strategic Plan are described below.

2.3.2. Five Guiding Principles

This section presents the Five Guiding Principles of risk reduction in decommissioning of the Fukushima Daiichi NPS.

Principle 1: Safe- Reduction of risks posed by radioactive materials and work safety

Principle 2: Proven- Highly reliable and flexible technologies

Principle 3: Efficient- Effective utilization of resources (e.g. human, physical, financial and space)

Principle 4: Timely- Awareness of time axis

Principle 5: Field-oriented- Thorough application of Three Actuals (actual field, actual things and actual situation)

(1) Principle 1: Safe- Reduction of the risks caused by radioactive materials* and work safety

*Environmental impacts and workers' exposure

Needless to say, safety is the first priority. The safety fundamental set out obligating "protection of humans and environments against risks from radioactive materials" by International Atomic Energy Agency (IAEA).

However since the Fukushima Daiichi NPS is a damaged reactor that does not meet the safety standards required for normal operating reactors, the approaches to safety and decommissioning processes are not necessarily consistent with those for normal operating reactors. Hence, it is expected to facilitate decommissioning while taking actions corresponding to the situations.

Accordingly, recognizing the high level of risks posed by the damaged reactors, the highest priority should be given to "immediate risk reduction to achieve a safe and stable condition." While it is important to meet the basic principles of the new regulatory requirements such as defense in depth, which were reviewed as a result of the lessons learned from the accidents at the Fukushima Daiichi NPS, it is also important to make effort to effectively ensure safety and reduce risks with the awareness of total risk reduction in light of the time axis. It is important to surmise the specific safety regulations in the decommissioning of the damaged reactor and discuss with the NRA from early stage.

With regards to safety of workers, it is necessary to pay sufficient attention to their work to prevent accidents and injuries since the site is not easily accessible and the work space is also limited. In addition, field work has to be done under the highly radioactive environment therefore management of work hours, setting up of radiation shielding, and use of protective equipment must be ensured in an effort to reduce their exposure.

(2) Principle 2: Proven- Highly reliable and flexible technologies

The decommissioning of the Fukushima Daiichi NPS is an unprecedented project involving tremendous technical difficulties and a large number of elements of R&D.

To carry out the measures in a relatively short period of time, further developmental work should be minimized in order to make steady progress by minimizing the risks of failure in developments.

For that purpose, feasible technologies and knowledge with high Technology Readiness Level (TRL) available in Japan and abroad should be adopted and applied. They need to be verified and demonstrated in advance to confirm their performance under severe site conditions while being improved to be met with the site conditions at the Fukushima Daiichi NPS (e.g. by systematization).

Considering a high degree of uncertainty in the site conditions, robust technologies should be selected to enable flexible application to unexpected situations or changes of situation. The works should also be carried out step by step and the course of actions should be adjusted as necessary. The alternative plans need to be in place in case that the selected technologies cannot be applied at the site.

On the other hand, development of entirely new technologies may become critical over the course of decommissioning work. For mid- and long-term issues for such technical development, needs, objectives, and roles of relevant organizations (e.g. university, public research institution and private organization) are required to be identified including basic and generic technology researches to carry out the R&D. In specific, the radioactive environment caused by the difficulties in the decontamination is unlikely to turn for the better; therefore the use of remote-controlled technology is highly anticipated.

For example, target of operations under severe radiation environment may require a combination of (a) remote manipulation technologies, (b) remote transfer control technologies, (c) decontamination or shielding using newly developed technologies, (d) direct operation by humans and (e) related basic technology research. If reliability/assuredness in (a)-(c) is comparatively low, the combination with human operation in (d) is to be considered as technical strategy.

The robotic technologies to be applied for the decommissioning work are as follows:

(a) Remote-controlled manipulation technology

Technology to realize operation by transferring the end effectors such as gripper, decontamination head and sensor heads to the work area by arm mechanism controlled by the operator in a safe place.

(b) Remote transfer control technologies

Technology to transfer the mobile base (platform) equipped with remote-controlled manipulation system and sensing system to the work site, which is controlled by the operators who are in a safe place.

(3) Principle 3: Efficient- Effective utilization of resources (e.g. human, physical, financial and space)

Decommissioning of the Fukushima Daiichi NPS involves implementation of a large amount of complicated works and developments over a long period of time. Therefore, shortage in resources, such as human, physical, financial and space may become the restraints to the project. Reasonable and effective use of these resources will be a key factor in successful decommissioning.

As for human resources, it is necessary to plan and manage the total exposure of the construction workers during the period of construction in order to secure manpower over a long period of time for the works under highly radioactive environment. In addition, because the project requires various R&D and technical studies for site construction, impracticable and unnecessary work should be avoided and efficient operation must be pursued. Securing necessary human resources for successful decommissioning such as researchers, engineers and workers in conjunction with human resource development and technology succession are important.

As for physical resources, any facility and equipment brought into the Fukushima Daiichi NPS are highly likely to be treated as radioactive waste. A rational way should be developed in order to reduce the amount of waste by effective utilization, and not to carry in unnecessary goods but fully use them once they are carried in with keeping "3R rule (reduce, reuse and recycle)" in mind.

As for financial resources, since a large amount of work and developments are required over a long period of time to successfully implement the project, a total cost reduction including the cost effectiveness of the works, the effect of investment against technical development and facilities in conjunction with effective use of manpower must be considered.

Although the area of the Fukushima Daiichi NPS is a relatively large compared to the other domestic nuclear power stations, it is not enough to secure a vast area of land for storing contaminated water tanks and installing the waste storage facilities. Considering that such areas may further increase in the future and constrain the working areas, the available land must be used in an effective way along with the preparation and securing of the transport route such as for equipment.

For efficient uses of the resources (human, physical, financial and space), it is important to optimize individual work and each development activity, but to avoid partial optimization, it is also important to determine the order of priority to achieve overall optimum results in the long run taking into account the consequences on the future procedures.

(4) Principle 4: Timely- Awareness of time axis

To take unnecessarily long time for decommissioning of the Fukushima Daiichi NPS means continuation of high-risk situation associated with radioactive materials. "Timeliness" and “Proven” may be in a trade-off relationship, but putting off the judgment and leaving the high-risk situation unattended is not a reasonable way to deal with the risks. Works need to be carefully carried out in parallel with the studies, and optimal judgments must be made at necessary timing.

In order to be aware of "timeliness," it is important to set a certain target time for each of the "actions to be taken as soon as practicable," "actions that require steady implementation" and "actions to be carried out over a long time." Moreover, the process of the fuel debris retrieval can be divided into three phases; "beginning phase," "intermediate phase" and "completion phase," and it is also necessary to set a step by step intermediate target for "beginning phase" and the "intermediate phase" in addition to the objective of overall achievement. The "beginning phase" is a phase when the preparation for reliable method is complete and retrieval work starts, making this phase a significant point both technologically and socially. Even in the "intermediate phase," it is extremely important to have actual sense of accomplishments and to demonstrate steady progress.

In addition, multilayered preventive measures against project risks are also important to avoid time loss and additional rework. In planning preventive measures, judgments on what risks should be addressed, to what degree preventive measures should be taken, and to what degree the measures should be multilayered will be important as well. Prior clarification on details and level of safety assessment is also required to avoid time loss and additional rework.

Since "timeliness" is not the focal point of long-term issues such as waste management and decommissioning, new regulatory systems and standards may need to be established for the treatment of the damaged nuclear power station and the generated waste which has never been handled in the past. This process may take a considerable time, therefore sufficient lead time should be taken into account when planning the long-term measures.

(5) Principle 5: Field-oriented- Thorough application of Three Actuals (actual field, actual things and actual situation)

Decommissioning of the Fukushima Daiichi NPS is the risk reduction activities associated with the radioactive materials at the site; therefore, it is important to carry out the tasks in accordance with the "Three Actuals" concept thoroughly and in a field-oriented manner.

The Three Actuals means to understand the precise needs based on the actual site conditions, actual structures, systems and components, and what is actually happening at the site, and to choose technologies focusing on the applicability at the site. Special attention needs to be paid and common understanding should be developed for the risks posed by gaps in understanding between engineers and field workers who apply the developed technologies to the site, as well as between design/project management personnel and field workers.

The site applicability is to assess whether the technologies under feasibility study (FS) is actually applicable to the site conditions and the environment of the Fukushima Daiichi NPS.

Assessment of site applicability should be made based on the following points.

- Environmental resistance (e.g. radiation, temperature and humidity and light intensity).
- Accessibility and transportability (e.g. narrow routes, obstacles such as rubble, lifting device and dose rate).
- Work space (e.g. inside the buildings, and yard)
- Infrastructure development (e.g. electricity, air, communication and water)
- Necessity of liquid and radioactive solid waste treatment.
- Maintainability and capability to respond to troubles.
- Onsite operability

Understanding of site conditions may also provide information to enhance the safety of existing light water reactors. It is originally not within the scope of the decommissioning of the Fukushima Daiichi NPS, but it is expected to keep that in mind throughout the project.

On the other hand, regardless of the Three Actuals or the safety enhancement for the light water reactors, there are great difficulties and exposure accompanied through the investigation of the site conditions under highly radioactive environment of the Fukushima Daiichi NPS. Spending time to execute a thorough investigation is a trade off with safety from the total risk reduction point of view; therefore, it may be necessary to make a plan based on a certain level of assumptions. In such case, multilayered measures should be prepared to deal with unexpected situations.

For the decommissioning of the Fukushima Daiichi NPS, it is important to manage the project by comprehensive assessment of various risks taking into account of the balance of all risks that are in the trade-off relationship. Therefore, involvement of participation of various relevant parties according to the risk information that is, risk-informed decision making should be applied in assessing the risks. .

The decommissioning project depends heavily on safety regulation. The regulatory authorities also consider the use of the safety risk information, therefore the risk-informed decision making should also be applied to meet the regulatory requirements. In addition, it is essential to have consultation with the regulatory authorities from the R&D stage concerning how to address safety issues.

It is also important to explain to the society that utmost effort is being made although various risks and restrictions exist, that is, so-called risk-informed communication is necessary.

While individual areas of work are studied in accordance with the Five Guiding Principles, it is extremely important to be always aware of the interrelationship of all the areas of work and their position within the entire project from the total optimization viewpoint.

2.4 Approach to the international cooperation

(1) Gathering and using wisdom and experience

Decommissioning associated with the accident at the Fukushima Daiichi NPS was one that the world has never seen before in terms of scale and so on. In order to smoothly and promptly carry out such extremely complicated and difficult project, technologies that is far beyond what has been accumulated in the construction, operation, maintenance and decommissioning of light water reactors in Japan are required. On the other hand, there are many cases where contaminated facilities and facilities that have seen accidents overseas have been decommissioned. Learning from these similar cases will be helpful to accelerate the decommissioning project and ensure safety at the Fukushima Daiichi NPS. Thus, it is very important to actively cultivate relationships with the relevant overseas organizations that have this knowledge and experience. Their knowledge and experiences are valuable not just because of their technologies but also because of their familiarity with actions and countermeasures in response to unexpected and extraordinary situations. A special consideration and measures are also required in order to smoothly obtain such overseas decommissioning technologies and the knowledge and experiences regarding projects, even in the Japanese systems and frameworks that have been supporting our excellent light water reactor technologies. That is, creating an optimal environment to facilitate the introduction of excellent experiences and technologies from overseas should also be encouraged. Thus it requires the operators to recognize "shift to the decommissioning culture"[3] recommended by IAEA

To this end, the NDF has appointed International Special Advisors from the U.S., the U.K., and France as experts to provide us with advice. Also, in February 2015, the NDF entered into Memorandum of Understanding on the exchange of information with the Nuclear Decommissioning Authority (NDA) in the U.K. and the Commissariat à l'énergie atomique et aux énergies alternatives (CEA) in France. These are institutions related to decommissioning administration and research and development overseas, and we are working on collaborating with them. The NDF has been also participating in the framework of Decommissioning and Environmental Management Working Group (DEMWG) under the U.S.–Japan Bilateral Commission on Civil Nuclear Cooperation between the U.S. and Japanese Governments since 2015, having discussions such as with the United States Department of Energy (DOE) and the U.S. national laboratories. Further, the NDF is taking part in the activities of international organizations including the IAEA and the Organization for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA).

Making an effort to gather and use the wisdom and experience from Japan and abroad, including adequately utilizing knowledge and experiences of decommissioning projects in other countries, are

[3] IAEA, Safe and effective nuclear power plant life cycle management towards decommissioning, IAEA-TECDOC-1305, August 2002

important as ever to conduct the decommissioning project at the Fukushima Daiichi NPS efficiently and effectively.

(2) Active information dissemination

Taking into account of Japan's responsibility to international society, as a country where the Fukushima Daiichi NPS accident occurred, it is important to carry out the decommissioning project in a manner open to the international society. Activities through bilateral and multilateral frameworks, such as disseminating information proactively on the knowledge and experiences obtained from decommissioning and contaminated water measures, receiving advice and assessments will be important.

At the side event of the IAEA general conference in September 2015, the government organizations of Japan, the U.S., the U.K., and France respectively disclosed their decommissioning efforts to the world and discussed their decommissioning initiatives. In April 2016, five years after the Great East Japan Earthquake and the Accident at Fukushima Daiichi NPS, the 1st International Forum on the Decommissioning of the Fukushima Daiichi NPS was held in Iwaki City of Fukushima Prefecture. It was co-hosted by the Agency for Natural Resources and Energy in Japan and the NDF. The participants included the relevant institutions and experts in Japan and abroad, local residents and students (641 attendees from 15 countries). It is important to actively continue these activities.

In handling of fuel debris and spent fuel, it is necessary to note that a great deal of consideration is internationally requested from the perspective of ensuring physical protection and safeguards.

(3) Close cooperation among the domestic organizations

The relevant Japanese organizations are working on international collaboration respectively. TEPCO entered into the Agreement on Information Exchange with Sellafield Ltd. in the U.K., and the CEA in France. JAEA has partnerships with organizations, including the NDA in the U.K., the CEA and the Agence Nationale pour la Gestion des Dechets Radioactifs (ANDRA) in France, and the Nationale Genossenschaft für die Lagerung radioaktiver Abfälle (NAGRA) in Switzerland.

The relevant Japanese institutions are communicating on a regular basis. In leading international efforts, it is important for the Japanese Government, the NDF, TEPCO and relevant research institutions to work together closely to carry out these cooperation with the international society.

2.5 Overall structure of the Strategic Plan

The strategic plan consists of seven Chapters.

Chapter 1 provides the role of the NDF and the relationship with the relevant institutions involved with the decommissioning project of Fukushima Daiichi NPS.

Chapter 2 provides the progress status of the Fukushima Daiichi NPS since the announcement of the Strategic Plan 2015. Also, as the NDF's outline of the Strategic Plan from a mid-to long term perspective, concerning the purposes, positioning and fundamental policies of the Plan, this Chapter describes that the project will be implemented in line with the "Five Guiding Principles" and "how to develop the international cooperation."

Chapter 3 provides the strategy for reducing radioactive material risks at Fukushima Daiichi NPS and the project risk management for facilitating the steady decommissioning, and its relationship with society.

As the basis to achieve the fundamental policies of the Strategic Plan, it is essential to identify the risks of the entire Fukushima Daiichi NPS, and develop the risk reduction strategy, prioritizing risk removal and reduction. In the Strategic Plan 2015, an overall picture based on the short-term perspective and the mid-and-long-term perspective was provided, as there were various characteristics of risk sources and there were priorities in the responses. Risk reduction efforts are emphasized in the Mid-and-long-Term Roadmap as well.

The Strategic Plan 2016 provides the concept of an updated risk reduction strategy, including changes made in the risk status over the past one year; specifically, it includes a reduction in the risks due to various measures, additional risk sources to be reviewed, and the improvements of risk analysis methods. Also, it describes the concept of project risk management and importance of communication, focusing on the fact that the decommissioning work can lead to social risks such as trigger to delay in the decommissioning project and reputational damage, besides the radiation risks of radioactive materials.

Chapter 4 provides the strategy to analyze the internal PCV condition and its action status; the action status of the technical issues important in the evaluation of feasibility of three methods to be focused on (The Submersion-Top access method, Partial submersion-Top access method and Partial submersion-Side access method); and the studies on the approaches to the retrieval method for each unit.

Assuming a high-dose radiation environment, the descriptions on the internal PCV condition analysis should include the inspection strategy for comprehensively analysis/evaluation so as to obtain more reliable results by utilizing the evaluation results based on the severe accident progression analysis and plant parameter in addition to the investigation of conditions inside the actual reactor with high priority considering necessary timing and significance.

Also, it describes the action status of the technical issues (e.g. establishment of criticality control and containment of radioactive materials function) important to ensure safety in the fuel debris retrieval method and technical issues important to realize the fuel debris retrieval method. Integrating these approaches to the studies are described on “Determination of fuel debris retrieval policies for each unit” scheduled to be announced in 2017 summer.

Chapter 5 provides the strategies of waste management. It describes the issues based on the current status and the concept of safety assurance for radioactive wastes in line with the target process of developing the treatment and disposal fundamental policy in FY2017.

Concerning the fundamental concept of waste management safety assurance, the fundamental approach based on “the concept of international radioactive waste management” should be regarded as important. However, the effort considering the characteristics of wastes generated by the accident at the Fukushima Daiichi NPS (the wastes which may be categorized to radioactive waste such as by reuse and radioactive solid waste which has been stored since before the accident and is hereinafter referred to as “radioactive solid waste") will also be important. The Strategic Plan 2016 describes the concept to control the radioactive waste in a safe manner from the perspective of “stable storage of radioactive solid waste will be important for the time being." In addition, it describes the evaluation of the current efforts and clarifies the issues on the solid wastes. For example, since the amount of solid waste to be controlled increases as decommissioning work progresses, stable storage will be important from the perspective of risk reduction.

Further, the importance of waste reduction with clear priority is shown along with the concept of the overseas field-proven waste management.

With regard to the evaluation of the properties on the solid waste, it indicates the importance of identifying the properties of radioactive solid waste through analysis. It also indicates the importance of having an analysis plan focusing on the period until about FY2021, when the safety for the treatment and disposal of solid waste will have been confirmed. The importance of expanding data on the evaluation of technologies of processing of solid wastes is described. Also, the importance of studying new disposal concept according to the characteristics of solid wastes by utilizing the knowledge and experiences in Japan and abroad is described.

Chapter 6 provides the direction to head in and management of research development efforts, taking into account the status of the current activities. This is based on the research development policy of accelerating work on challenges to the decommissioning of Fukushima Daiichi NPS, which has never been experienced before with all efforts of the people gathering strength in Japan.

“The Decommissioning R&D Partnership Council” was established in the NDF to strengthen its research and development structure. This is based on the philosophy that the research and development required for decommissioning should be implemented through the concerted efforts of Japan. The Chapter also provides the activities of developing human resources to ensure long-term decommissioning, bridging technical seeds and needs, and sharing the details of research and development activities among the relevant institutions of the JAEA, other research institutions, universities, and technical colleges through mutual understanding.

By reflecting the progress in the study of decommissioning strategies, the new plans will be presented for the basic/infrastructural research & development and the research & development of decommissioning technologies using the government budget.

Chapter 7 provides future action to carry out the Strategic Plan as a summary.

3. Risk reduction strategy

For the purpose of "continuously and promptly reducing the risks associated with the radioactive materials that resulted from the accident" that is the fundamental policy for the decommissioning of the Fukushima Daiichi NPS, "the designing of a risk reduction strategy" is performed in this chapter. To do so, the measures for risk reduction shall be decided by identifying various radioactive materials, performing analysis and evaluation based on their characteristics, and then deciding priorities.

Various challenges are found in implementing the designed risk reduction strategy steadily. One of them is "the project risks" that can significantly affect the progress of the decommissioning project, including the operational risk associated with the removal of fuel debris; it is important to identify such risks and properly manage them for fulfilling the above-mentioned fundamental policy. Further, it is also important to proceed with decommissioning in cooperation with the community, through gaining the understanding by the stakeholders, including the local residents.

In developing the risk reduction strategy and steadily proceeding with it, it is important to make decisions, regarding risks as one of the important pieces of information, while taking into account other various factors. In addition, communication with the local residents is important to perform decommissioning in cooperation with the community.

3.1 Method of reviewing the risks posed by radioactive materials

Here, preparations are made to design the strategy for reducing the risk of radioactive materials, described in Section 3.2, in reference to generic risk management[4]. The concept of risk management is also effective in ensuring that decommissioning is steadily carried out as stated in Section 3.3, where reviews are made by referring back to the generic definitions in Table 3-1.

3.1.1. Terms and Definitions

As the expressions associated with risks are used in various ways, it is effective to define terms so as to improve people's understanding of the concept of risks. Table 3-1 shows the terms and their definitions that are generally used in relation to risks, as well as their usage in the strategy for reducing risks posed by radioactive materials.

The major purpose here is to protect people and the environment from the impacts of radioactive materials. Impacts of radioactive materials include as follows:

- ➢ An impact on the environment
  - ✓ Public exposure (external exposure and internal exposure)
  - ✓ Environmental pollution and wide-area dispersion
- ➢ Workers' exposure (external exposure and internal exposure)

Our purpose here is to suppress the public exposure, as a representative impact on the environment. Exposure of workers is an important point to consider when reviewing the countermeasures against risks.

[4] JIS Q 31000:2010 (ISO 31000:2009) "Risk Management - Principles and Guidelines"

Uncertainties refer to an insufficiency of information, understanding, or knowledge associated with the events, consequences, and likelihood of occurrence. For example, it is difficult to forecast the timing and the place of occurrence of natural disasters, as well as their size; once it occurs, impacts on people and the environment will be unavoidable. In addition, information concerning the risk sources themselves, such as the distribution and properties of fuel debris, is also insufficient, representing another significant uncertainty.

Consequences and likelihood of occurrence are qualitatively represented in some cases; however, the strategic plan aims to be quantitative to the extent possible. The metrics, however, are represented as relative values, and are not limited to physical ones such as effective dose. In some cases, subjective expressions may be used, in spite of aiming to be objective as much as practicable.

The level of risks is generally expressed as a combination of a consequence and the likelihood of occurrence, not always as the product of them. In the strategic plan, the level of risks is basically expressed as a product; in some cases, however, either a consequence or the likelihood of occurrence is used in various kinds of assessment.

The risk criteria are discussed in the sub-section of risk evaluation below.

Table 3-1 Terms and Definitions

Terms	General definition	Usage in a strategy for reducing risks posed by radioactive materials
Risk	Impact of uncertainties on an objective	Impact on people and the environment caused by radioactive materials Also used when a risk source, an event, a consequence, or the level of risk is represented as a whole
Risk source	A factor that has the potential to produce a risk by itself or in a combination with other factors	Radioactive materials
Event	Occurrence of or change in a certain series of surrounding conditions	Occurrence of a natural disaster or a failure, and the resultant change in the condition of a risk source or in the containment function
Consequence	Outcome of an event impacting on the specified objective (qualitative or quantitative)	Public exposure caused by the release of radioactive materials (or a metric representing this)
Likelihood of occurrence	Chance that something may happen (objective or subjective, qualitative or quantitative)	Chance that public exposure, caused by the release of radioactive materials, may happen (or a metric representing this)
Level of risk	The magnitude of a risk expressed as a combination of a consequence and the likelihood of occurrence	Product of a consequence and the likelihood of occurrence
Risk criteria	Reference for evaluating the significance of the relevant risk	Comparison of the level of risk between various risk sources or between before and after an action is taken against the relevant risk

3.1.2. Risk management process

Figure 3-1 shows the process of generic risk management. Designing of a strategy for reducing risks posed by radioactive materials, discussed in Section 3.2, is also performed according to this process.

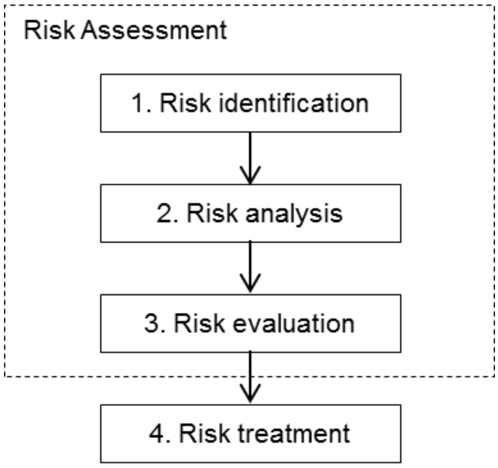


Figure 3-1 Process of Risk Management

(1) Risk identification

Risk identification is a process of detecting, recognizing a risk and clarifying its characteristics, including the identification of risk sources, events, and consequences. In this phase, it is important to extensively list up a variety of risk sources existing at the Fukushima Daiichi NPS, and to understand their characteristics.

(2) Risk analysis

Risk analysis is a process of understanding the characteristics of risks, and determining the level of the risk. To do so, a certain consequence and the likelihood of occurrence must be determined. In the case of the Fukushima Daiichi NPS, some ingenuity must be used to take into account the uncertainties associated with risk sources such as fuel debris.

(3) Risk evaluation

Risk evaluation is a process of comparing the level of risks with the risk criteria, in order to determine if the relevant risk is acceptable or tolerable. In the decommissioning of the Fukushima Daiichi NPS, this type of risk evaluation will be required sooner or later; under the present condition, however, a strategy must be developed first, e.g., to determine the priority of risk sources. Therefore, in the strategic plan, a relative comparison will be made on the level of risks for various risk sources, not a comparison with the risk criteria.

(4) Risk treatment

Risk treatment is the process of reducing the level of risks. As shown in Figure 3-2, the ways to do this include removing risk sources, reducing the likelihood of occurrence, mitigating consequences, and so on. In treating risks, through a single means or a combination of them, the level of risks is reduced accordingly for the risk sources positioned at the upper right in the figure.

In this regard, a variety of possible options shall be reviewed and the best one must be selected. To make comparisons between these options, the Five Guiding Principles shall be referred to, and attention shall be paid to the risks that might happen during treatment. It is also important for making the best selection to determine how much risks can be reduced by treating them.

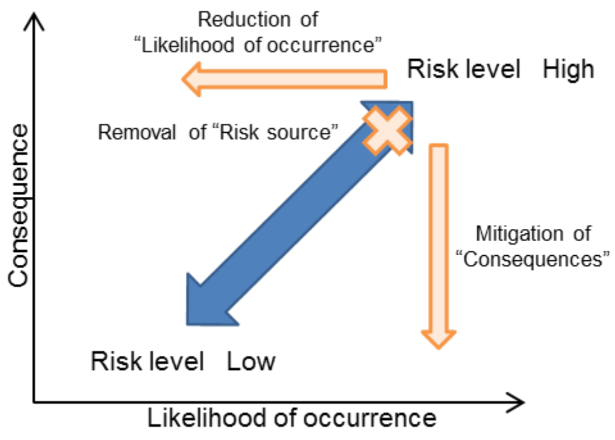


Figure 3-2 Level of Risks and Risk Treatment

Processes (1) to (3) are collectively called risk assessment. Prioritization of risk sources is resultantly performed in this process, and this is the first step in the risk reduction strategy. The methodology suitable for analyzing diverse risk sources is discussed in Section 3.1.3, and then the risks are identified in Section 3.2.1, risk analysis is performed in Section 3.2.2, and the priority for treatment is established based on the level of risks in Section 3.2.3.

The second step in the risk reduction strategy is the process of risk treatment. In Section 3.2.4, according to the established priority, a method of reducing the level of risks for individual risk sources is determined.

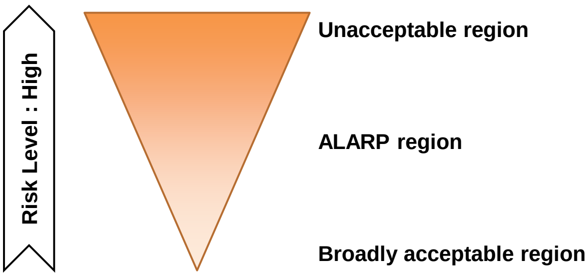
Column: ALARP and ALARA

In determining the risk criteria, the concept of ALARP (As Low As Reasonably Practicable), which has been proposed by the HSE (Health and Safety Executive) and introduced into the NDA's strategy, serves as a reference. In ALARP, risks are classified into the following three groups.

- Unacceptable region
  - Risks are significant, and cannot be justified except in extraordinary circumstances.
- ALARP region (or tolerable region)
  - Risks are tolerable only when risk reduction is impracticable, or the cost incurred with risk reduction is disproportionate to the effect acquired by risk reduction.
  - As the lower a risk becomes, the higher the cost that is commensurate with the effect of risk reduction becomes, the risk should be reduced to a level as low as reasonably practicable.
- Broadly acceptable region
  - The risk is sufficiently low; it should be continuously ensured that the risk stays at this level.

To establish a reasonable level of risk in the ALARP region, you are recommended, in addition to performing cost-benefit analysis, to refer to the past good practices, and to aim to have the best solution by extensively reviewing the alternative risk reduction options. In any case, it is essential to involve the stakeholders.

The concept of ALARP is similar to that of ALARA (As Low As Reasonably Achievable), put forward by the International Commission on Radiological Protection (ICRP). In ALARA, in order to optimize protection against radiation, it is necessary to limit the exposure dose to "as low as is reasonably achievable, taking into account the social and economical factors." The concept of ALARA is specifically to reasonably determine how much the exposure dose should be reduced, after achieving the dose limit established as the lower limit in the unacceptable region.



Reference: "The Tolerability of Risk from Nuclear Power Stations," HSE (1992)

3.1.3. Methodology of risk analysis

To perform a risk assessment targeting various risks sources, it is essential to use a methodology of analysis that can get an overview of diverse characteristics, instead of the details. This section refers to the SED score (Safety and Environmental Detriment score)[5] developed by the NDA and partially modified to be applied easily to the Fukushima Daiichi NPS. Outline of SED score is shown in Appendix 3.1.

3.1.3.1 Risk metric

According to the SED score, following "Risk metric" is used as an index that represents the risk level.

$$Risk\ metric = RHP \times (modified\ FD \times modified\ WUD)^4$$

The first term "RHP" represents Radiological Hazard Potential (RHP) of SED score. It is defined as a total amount of radioactive materials which is contained in the risk source, taking into account the properties such as gas, liquid or solid from the perspective of likelihood of leakage or migration and the available time to recover when the safety function is lost. The details are described in 3.1.3.2.

The "modified FD" in the second term is a modified Facility Descriptor (FD) of SED score. It is a factor that determines the grade of the risk source by the combinations of the elements, such as the integrity of the facilities and containment functions. The "modified WUD" is a modified Waste Uncertainty Descriptor (WUD) of SED score and is a factor that determines the grade of the risk source by the combinations of such elements as the changes in the conditions of risk source and states of packaging/monitoring. Each factor is divided into ten categories and scores are set for each category.

The first term is "Hazard Potential," which is a risk source-specific nature. It is equivalent to the "consequence." The second term is "Safety Management" that expresses the controlled state of the risk sources and two factors above are respectively related to current and future "likelihood of occurrence." Hazard Potential and Safety Management are important indices to determine not only the prioritization of the risk source but also the precautionary measures for risk treatment. For example, Safety Management enables more diverse measures than Hazard Potential. Also, fourth power of the second term is set so that the magnitude of the first and second terms will be comparable each other.

3.1.3.2 Hazard Potential

The term "RHP" represents Hazard Potential as follows:

$$RHP = \frac{Inventory \times Form\ Factor}{Control\ Factor}$$

(1) Inventory

Inventory is represented as the product of radioactivity of risk sources and specific toxic potential (STP)[6] and this corresponds to effective dose. STP is the amount of water needed to dilute the radioactive materials of 1TBq so that the ingestion of such diluted water throughout the year would not exceed the radiation dose of 1mSv. This corresponds to the dose coefficient.

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[5] NDA Prioritization – Calculation of Safety and Environmental Detriment score, EPGR02 Rev.6, April 2011.

[6] Instruction for the calculation of the Radiological Hazard Potential, EGPR02-WI01 Rev.3, March 2010.

Using Inventory to the calculation of RHP corresponds to the full release of radioactive materials contained in a risk source. Only a small fraction of radioactive materials is released in a real event, and it is very unlikely that the full amount of them will be released. Thus, using RHP as the “consequence” is a setting on the safety side. From the perspective of prioritization of risk source, RHP using Inventory is an impartial index that does not depend on the event.

(2) Form Factor

Risk sources have different properties: solids such as fuel debris and pellets in fuel rods, liquids such as contaminated water, and gases such as noble gases in fuels. There are also other various properties of risk sources, such as sludge, complex radioactive compounds on the surfaces of equipment and buildings (surface deposits), materials having potential chemical reactivity, mixtures of chemical and radioactive materials (mixed waste), and environmental pollutants in the site (rocks which adsorbed clay particulates and radioactive materials).

If the risk source is currently contained, the likelihood of its leakage depends on its property in the event of a loss of the containment function. And if the risk source is already released into the environment, the likelihood of migration or dispersion depends on its property. The factor that quantifies these impacts is the Form Factor (FF) shown in Table 3.2.

RHP is a more realistic metric than Inventory, which indicates a full release, as the difference between the likelihood of release is taken into account for each property by FF.

(3) Control Factor

Control Factor (CF) of risk sources is the factor that indicates the time allowable until recovery in the event of a loss of safety function, such as cooling and nitrogen injection, designed for maintaining the current stable condition.

CF is shown in Table 3.3. The maximum value of 100,000 is prepared for the allowable time of 10 years, meaning that a stable condition can be maintained for a sufficiently long period without special equipment.

Table 3.2 Form Factor (FF)

Properties	Score
Gas, liquid	1
Sludge, powder	0.1
Discrete solid	0.00001
Large monolithic solid, activated component	0.000001

Table 3.3 Control Factor (CF)

Category	(In hours)	Score
Hour	1 h	1
Day	24 h	10
Week	168 h	100
Month	730 h	1,000
Year	8,760 h	10,000
Decade	87,600 h	100,000

3.1.3.3 Safety Management

Modified FD and modified WUD that comprises Safety Management are those which partially modified to be applied flexibly for various risk sources at the Fukushima Daiichi NPS.

Modified FD is a factor that determines the level of the risk sources by comparing them from the perspective of containment functions, structural integrity and safety measures as the elements that characterize the facility containing the risk sources. These describe the current state of containment: its multi-layered containment functions, the maintenance of its structural integrity until the time of its retrieval, and the multiplicity of safety equipment.

Modified WUD is a factor that determines the level of the risk sources by comparing them from the perspective of degradation, corrosivity, packaging and monitoring as the elements that characterize the risk source. It shows whether or not the impact is caused to the controlled state of risk source and the future retrieval work if no action is taken for a long period of time. Furthermore, impact caused by its presence to the retrieval of other risk source and needs of the early treatment were added to the elements for characterization from the perspective of prioritization of the risks sources, although these are not considered in the concept of SED score. When the properties and locations of risk source are uncertain, preparations such as inspection should be started instead of rushing to retrieve.

In SED score, combinations of the elements characterizing the risk sources are narrowed down to 10 categories as shown in Table A3.1-3 and Table A3.1-4. However, such fixed categories have limitations in expressing the risk sources corresponding to the various combinations of the elements characterizing the risk sources. In modified FD and modified WUD, any combinations of the elements characterizing the risk sources are allowed so as to respond flexibly to the various types of risk sources at the Fukushima Daiichi NPS.

For both modified FD and modified WUD, a relative comparison will be made for risk sources based on the above-mentioned concept. Then, risk sources will be classified into 10, and the scores specified in Figure 3-3 will be applied. This score is the same as that shown in Table A3.1-3 and Table A3.1-4. The scores on the graph have been set, so that Categories 4 - 10 into which many risk sources are classified appear as a straight line on a logarithmic scale. As Hazard Potential is changed to logarithmic scale in proportion to radioactivity of risk sources, score setting of Figure 3-3 is important to handle Hazard Potential and Safety Management equally.

Since quantification of the descriptive information such as the characteristics of risk sources and the condition of facilities is not easy, the method that sets scores by ranking as described above is one of the effective methods. The risk analysis method will be improved as to be more consistent with the on-site situation, so that the results will be reflected in the decommissioning work.

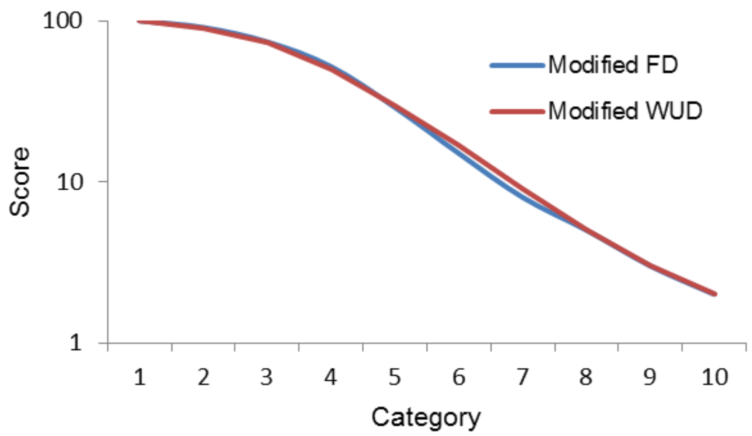


Figure 3-3 Scores of modified FD and modified WUD

3.2 Strategy for reducing risks posed by radioactive materials

3.2.1. Risk identification

Actinide nuclides such as uranium and plutonium (hereinafter referred to as “heavy nuclides”), fission products (hereinafter referred to as “FPs”) such as cesium which are easily released into the environment, and activation products accumulated in the reactors over a long period of operation (e.g. Co-60 and Fe-55) are typical radioactive materials for which the external impacts should be considered, and they contribute to risk sources.

Among major risk sources present at the Fukushima Daiichi NPS, the following are attributable to the fuels and contain radioactive materials of heavy nuclides and FPs:

- Fuel debris in the PCV (Units 1-3);
- Fuel assemblies stored in the SFP of each unit (Units 1-3);
- Fuel assemblies stored in the common pool (Fuels in the common pool);
- Fuel assemblies stored in the dry casks (Fuels in the dry cask).

The following contaminated water and waste contain FPs as radioactive materials:

- Heavily contaminated water accumulated in buildings (Contaminated water in buildings);
- Highly concentrated liquid waste stored in tanks (Concentrated liquid waste);
- Secondary waste generated from the cesium and the second cesium adsorption systems (Waste adsorption column);
- Secondary waste in the sludge storage of decontamination equipment (Waste sludge);
- Secondary waste generated from the advanced liquid processing system (ALPS), the added ALPS and high-performance ALPS (the slurry contained in a high integrity container (HIC); hereinafter referred to as “HIC slurry”);
- Rubble, felled trees and radioactive solid wastes generated by operations (including waste generated during normal operations before the accident which mainly consists of corrosion products such as Co and Mn. Hereinafter, the radioactive solid wastes contained in a storehouse is called “solid waste in storehouse,” and the radioactive solid wastes stored outdoor is called “tentatively stored solid waste.”)

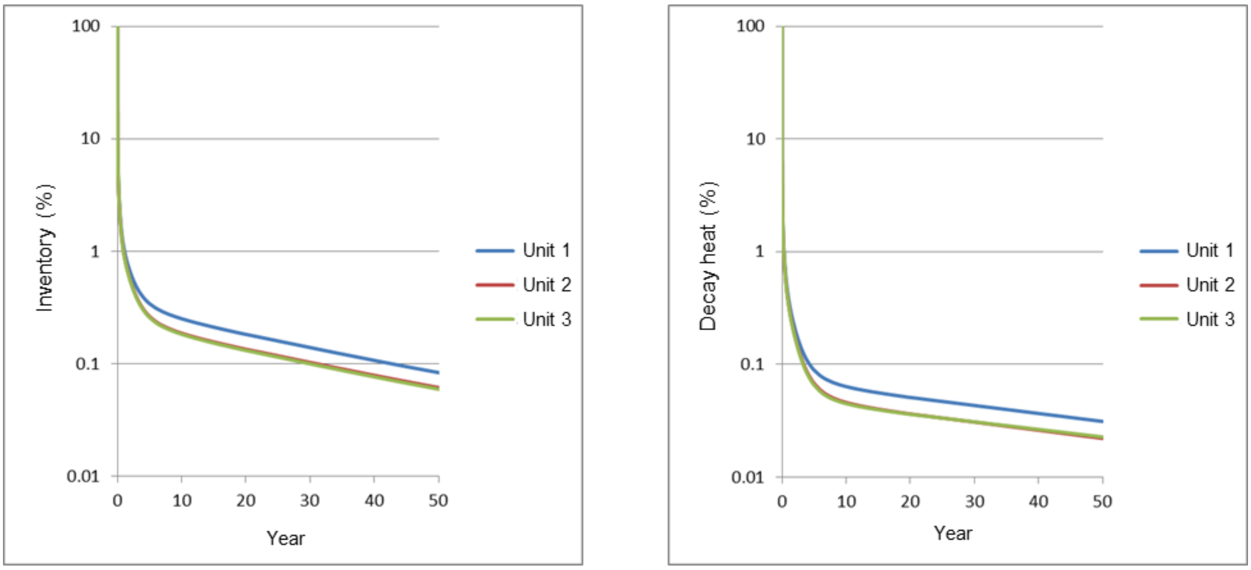
The following structures and buildings contain radioactive materials such as FPs and activation products. (These are hereinafter collectively called “PCV internal structures.”)

- Equipment that is located in the Reactor Pressure Vessel (RPV) and PCV, containing activation products, as well as the equipment contaminated by dispersed FPs (such as steam driers, steam separators, core shrouds, upper grid plates/core plates, piping, valves).
- Some equipment, piping and buildings which are contaminated by dispersed FPs.

In the strategic plan, all of them are considered as risk sources. With regards to scattered rubble, felled trees, stagnant water, and exhaust stacks, TEPCO is now working on risk reduction through a comprehensive investigation.

In addition, 1,535 fuel assemblies that were stored in the pool of Unit 4 at the time of the accident were completely removed in 2014. The transported spent fuels are stored in the common pool. Among the contaminated water, the concentrated saltwater stored in tanks and heavily contaminated water stagnating in the seawater piping trenches of Units 2 to 4 were completely removed in 2015.

The characteristics such as radioactivity, properties, and containment conditions of each risk source are summarized through the risk analysis in Section 3.2.2. Here, the properties of fuel debris will be briefly discussed. It should be taken into consideration that five years have already passed since the accident and the radioactivity and decay heat have decreased over the years. And, it should also be considered that the radioactive materials will further decay over the course of the assumed decommissioning work in the future. Core radioactivity and decay heat are shown in Figure 3-4 for each unit. All values are relative to the values at the time of the accident and the release of radioactive materials into the environment is not taken into account. The current level of radioactivity has decreased to less than 1%, and decay heat to less than 0.1%, compared to those at the time of the accident.



Reference: JAEA-Data/Code 2012-018

Figure 3-4 Evaluation of core radioactivity (left) and decay heat (right)

Risk identification includes the identification of events and consequences. As shown in Table 3-1, events represent the occurrence of natural disasters and failures, the condition of risk sources caused by them, and changes in the containment function, while consequences represent public exposure caused by the release of radioactive materials. Initiating events as the start of a series of events are as follows:

- Internal events: loss of power supply, internal fire, internal flooding, hydrogen explosion, malfunction, erroneous operation (human error), internal missile, sabotage, etc.
- External events: earthquake, tsunami, volcanic activity, tornado, external fire, typhoon, heavy rain, flooding, missile, illegal intrusion, etc.

The method of risk analysis discussed in Section 3.1.3 does not directly take into account the above-mentioned events; instead, individual factors of Safety Management will be analyzed by assuming these events. For the consequences, Hazard Potential will be used.

3.2.2. Risk analysis

3.2.2.1 Selection of nuclides for evaluation

The nuclides to be considered during the period of several decades until the completion of decommissioning will be selected. Appendix 3.2 explains the review conducted for the core and the fuels in the pool at Unit 2, focusing on the effective dose that represents the impact on people.

The selected nuclides and their characteristics are provided in Table 3-4. The seven nuclides of Pu-238, Pu-239, Pu-240, Pu-241, Am-241 and Cm-244 are selected for heavy nuclides, and the three nuclides of Sr-90, Cs-134 and Cs-137 are selected for FPs.

Table 3-4 Major radionuclides and their characteristics

Nuclides	Half-life		STP(m^3/TBq)	Feature
Pu-238	87.7	years	66,000,000,000	―
Pu-239	2.41×10^4	years	72,000,000,000	―
Pu-240	6.54×10^3	years	72,000,000,000	―
Pu-241	14.4	years	1,380,000,000	―
Am-241	4.32×10^2	years	57,600,000,000	Generated from decay of Pu-241
Cm-244	18.1	years	34,200,000,000	―
Sr-90	29.1	years	96,000,000	Medium volatile
Cs-134	2.06	years	12,000,000	High volatile
Cs-137	30.0	years	23,400,000	High volatile

Reference: Half-life from ICRP Publication 72, STP from EGPR02-WI01

3.2.2.2 Hazard Potential

Inventory, FF, and CF are determined for each risk source, and RHP is calculated as Hazard Potential. The setting values and their basis are shown in Appendix 3.3.1.

Radioactivity required for inventory of the following was estimated from public data: fuel debris, fuels in SFPs, contaminated water, secondary waste generated from water treatment, and radioactive solid waste. Radioactivity of fuels in common pool and fuels in dry casks was estimated from that of fuels in SFPs. Activity and amount of contaminants estimated from public data on the normal reactor with incorporating the partial FPs with high volatility released at the time of accident that attached on the surface are considered for the PCV internal structures. Range of estimation and variation among the data are considered as uncertainties.

With regard to FF, fuel debris is categorized as large monolithic solid, spent fuel as discrete solid, contaminated water as liquid, secondary waste generated from the water treatment as liquid or sludge; and radioactive solid waste as powder. PCV internal structures are categorized as activated components and surface contaminants. As for spent fuels, small quantity of highly volatile FPs released during the operation is located in cladding in powder form. A wide range of uncertainty is set for fuel debris and the surface contaminants such as PCV internal structures. Uncertainties are not set in the contaminated water and spent fuels and intermediate uncertainties are set in the rest of those.

To determine CFs for fuel debris, fuels in SFPs and fuels in common pool, the time margin for cooling shutdown were estimated. Cooling is unnecessary for fuels in dry casks, contaminated water and radioactive solid wastes. Possibility that the FPs on the surface of the PCV internal structures are released by the temperature rise by cooling shutdown is considered. Although secondary waste generated from the water treatment does not require to be cooled, continuous surveillance of impact of hydrogen generation is considered in HIC. Time margin is estimated for shutdown of agitation to prevent fixation for waste sludge. Although one order of magnitude is set for uncertainties, uncertainties are not set for the risk sources that do not require to be cooled.

3.2.2.3 Safety Management

The characteristics of each risk source required to evaluate modified FD and modified WUD that comprise Safety Management are described below. The detailed features and the values for modified FD and modified WUD are described in Appendix 3.3.2.

No significant damage on the PCV was observed. Safety equipment is multiplexed and important parameters are being monitored. Spent fuel pool of each Unit is sub-criticality system and cooling system is multiplexed. However, rubble and heavy weight objects fell, building ceilings was lost and seawater injection was experienced in some Units. No impact of the accident is observed on the common pool and dry casks.

Contaminated water in the buildings is kept contained by balancing with the water level of groundwater. Concentrated liquid waste contains highly concentrated radioactive materials and salt content, and stored in the welded type tanks and placed in dikes. The waste adsorption columns are shielded containers filled with Cs absorbed zeolite, which are fixed on the box culvert or storage rack. Waste sludge is stored in the agglomeration pit, and leakage monitoring and hydrogen discharge are being carried out. HIC slurry was housed in polyethylene container and collected in the stainless steel reinforcement unit, and is now being stored in the box culvert.

The solid waste in storehouse is the highly radioactive rubble that are collected in the containers and stored in the radioactive solid wastes storage building. Tentatively stored solid waste is the waste at various levels of concentration of radioactive materials which is stored in various forms and monitored.

3.2.3. Risk evaluation

3.2.3.1 Prioritization of risk sources

Concerning the major risk sources at the Fukushima Daiichi NPS, the example of risk analysis performed based on the information as of March 2016 are shown in Figure 3-5. In this figure, the impacts of uncertainties involved in the factors on Hazard Potential and Safety Management are expressed as the range of spread.

To continuously and promptly reduce the risks posed by radioactive materials, risk sources should be categorized by the level of risks, as follows:

[Category I] Risk source to be addressed as soon as practicable

- Fuels in the SFPs
- Contaminated water in the buildings

[Category II] Risk source to be addressed safely, effectively and carefully with thorough preparations and technologies to realize a more stable condition

- Fuel debris

[Category III] Risk source that requires actions to be taken for a more stable condition

- Concentrated liquid waste
- Waste sludge
- HIC slurry
- Part of tentatively stored solid waste
- PCV internal structures

Risks in Category I have high scores both in Hazard Potential and Safety Management, as well as the highest level of risks as the product of them, because of lots of radioactive materials, high mobility, and insufficient containment function and management, etc.

Risks in Category II are similar to those in Category I; however, their level of risks is rather lower.

Category III includes the risk sources with a relatively low level of risks; however, planned actions should be taken to realize a more stable condition, due to the following reasons.

- Although concentrated liquid waste will not be increased, highly concentrated liquid waste is being stored for a long period of time.
- Agglomeration pits that stores waste sludge were not designed for long-term storage. Solid wastes stored outside are not intended to be stored permanently.
- As for the PCV internal structures, activated materials are fixed inside, whereas some fission products that are attached to the surface are not stabilized.
- HIC was designed so as to store the wastes for a long period of time after the accident; however, since distillation of water occurred, drainage has been conducted while limiting the storage quantity and monitoring impacts caused by hydrogen generation.

The risk sources other than those mentioned above are sufficiently in a stable and safe condition. The common pool, dry casks, and solid waste in the storehouse have been safely designed and used, and no impacts caused by the accident are observed on them. The waste adsorption column was designed for long-term storage after the accident. For these items, sufficiently low levels of risks can be maintained by ensuring continuous management in the future.

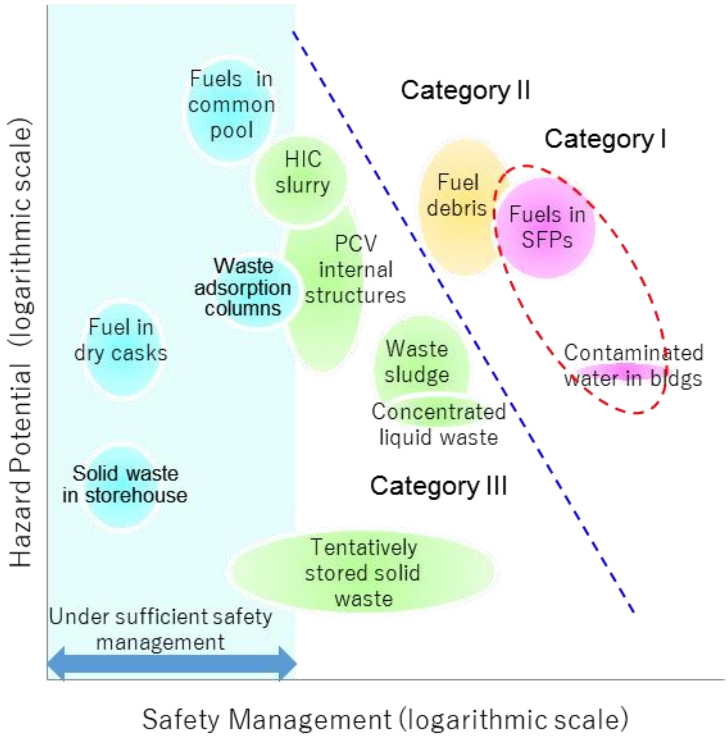


Figure 3-5 Example of Risk Analysis at Fukushima Daiichi NPS

In the above discussion, to understand and compare the characteristics of various risk sources, fuel debris and fuels in the SFP were summed up for Units 1-3. And the risk sources including dry casks and wastes that can be further broken down were also summed. Thus, while Safety Management represents each risk source, Hazard Potential assumes the release of radioactive materials all together, for example, from all wastes. Although such analysis is appropriate when targeting the common causes, each unit or each waste is required to be analysed for independent causes.

3.2.3.2 Changes in the level of risks over one year

The changes in the inventory by the treatment of contaminated water brought about noticeable effects in risk reduction among major approaches taken in the past one year are shown in Figure 3-6. The baseline value of the inventory of contaminated water has been set to 1 (as of March 2015), and the inventory value as of March 2016 is shown as a relative value. As indicated by the arrows in the figure, each inventory has been significantly reduced by the removal of contaminated water from the trenches of Unit 2-Unit 4 and completion of concentrated salt water treatment Radioactive materials contained in contaminated water did not disappear but were stored in the waste adsorption column or HIC slurry, which were the secondary waste generated from the water treatment. Since Safety Management of these is very small compared to the contaminated water, level of risks is significantly reduced.

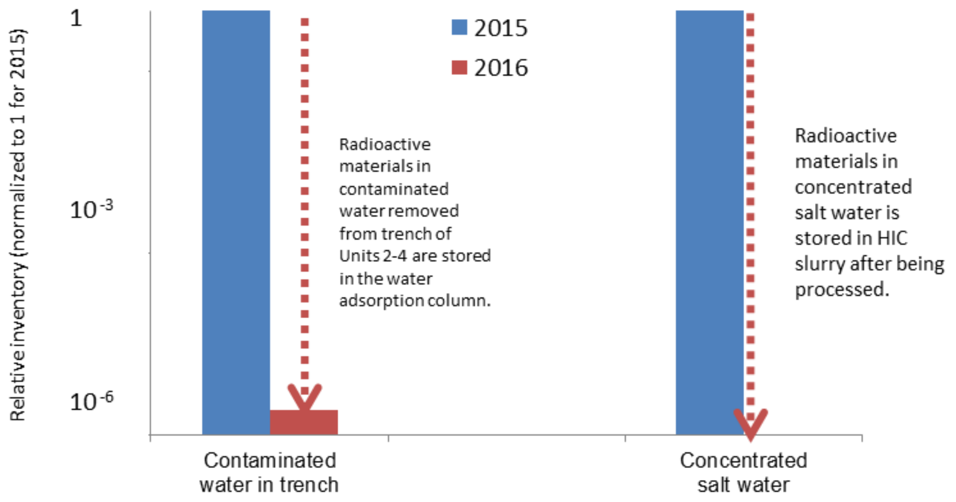


Figure 3-6 Decrease in Level of Risks through Treatment of Contaminated Water

Also, approximately 800,000 m^3 of treated water whose concentrations of Cs and Sr were significantly reduced by the contaminated water treatment is stored in the tank as of March 2016. Treated water contains tritium, whose Specific Toxic Potential (STP) as radioactive material is extremely low. Therefore, the technical studies for the various types of options were carried out in the Tritiated water task force established under the Committee on Countermeasures for Contaminated Water Treatment. Comprehensive study will be required including the issues regarding the reputational damage from the social perspective.

3.2.4. Risk treatment

3.2.4.1 Basic strategy for risk reduction

The image of the treatment of risk sources, as categorized in Section 3.2.3, and the changes in the level of risks caused by taking appropriate actions are shown in Figure 3-7.

First of all, actions for risk reduction should be taken for the risk sources in Category I. These actions have definite plans for risk reduction; while they will be challenging, there are no mid-and-long-term R&D issues, and thus, risk reduction should be conducted as soon as practicable. Specific actions are already in progress; in order to transport the fuels in the SFPs to the common pool, removal of rubble is carefully being conducted on the operation floor and in the pool. As the common pool is almost full at the moment, some of the fuels now in the common pool must be contained in dry casks. There is a plan to reduce stored amount of contaminated water in the buildings by lowering the water level and suppressing the inflow of groundwater by means of land-side impermeable walls.

For the risk sources in Category I, the NDF provides technical support for various issues found in implementing the reduction strategies; however, this is not covered by the Strategic Plan.

Secondly, the target is the fuel debris, which contains lots of radioactive materials and has many uncertainties concerning its locations and properties although it is kept in a certain stable condition. After fuel debris is removed it is contained for storage in containers which are designed to ensure sufficient safety in terms of criticality, shielding, heat removal, etc. Various issues are to be studied in parallel with the risk reduction of Category I and thorough preparations should be completed to take actions safely, effectively and carefully and then to realize a more stable condition. It is important to reduce the risks associated with the work and to achieve a more stable condition of the fuel debris by carrying out investigations and samplings.

Following these, the risk sources in Category III with a relatively low level of risks are aimed to take actions on a long-term basis. Note that the risks are reduced not only for the existing waste but also for the risk sources to be newly generated from the actions taken in Categories I and II.

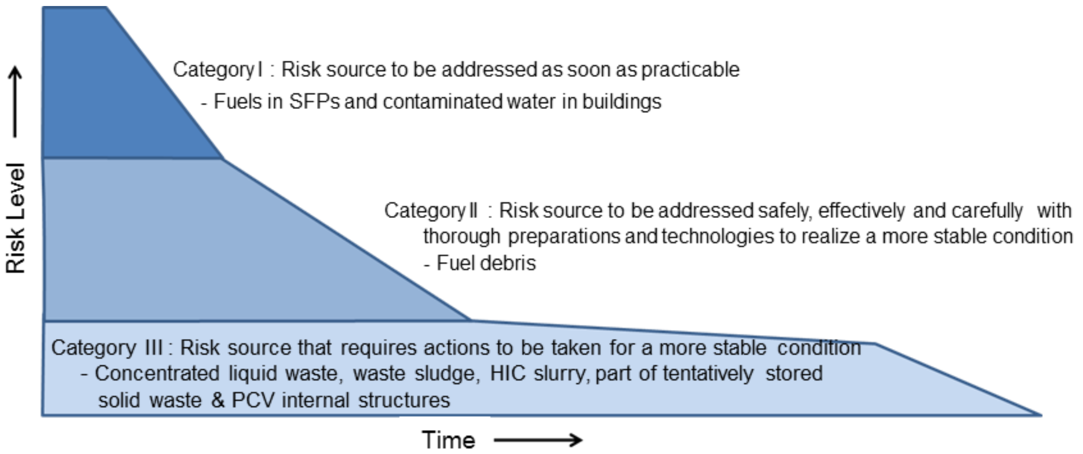


Figure 3-7 Risk Reduction Strategy

3.2.4.2 Issues in risk treatment

While decreasing the level of risks for existing risk sources by appropriate risk treatment, there is a possibility that the level of risks could temporarily increase during the work. Among the risk sources of Categories I and II, removal of the fuels in the SFP and fuel debris will require caution to ensure workers are not exposed, in addition to being careful about impacts on people and the environment caused by the radioactive materials. The outline of risk levels for major risks estimated if no measures are taken is shown in Table 3-5. In the actual operation, risk levels should be lowered by taking thorough measures. This table also shows dust dispersion during removal of the highly radioactive rubble, which is to be carried out before fuel debris removal as a risk related to the removal work.

Also various issues will be raised in the fuel debris retrieval work, including the increase in exposure of workers during the removal work, and securing of storage locations for the increasing amount of removed PCV internal structures to which fuel debris and cesium are attached. For the removal of fuel debris, these issues must be properly considered.

Table 3-5 Major risks during operation and the outline of risk level

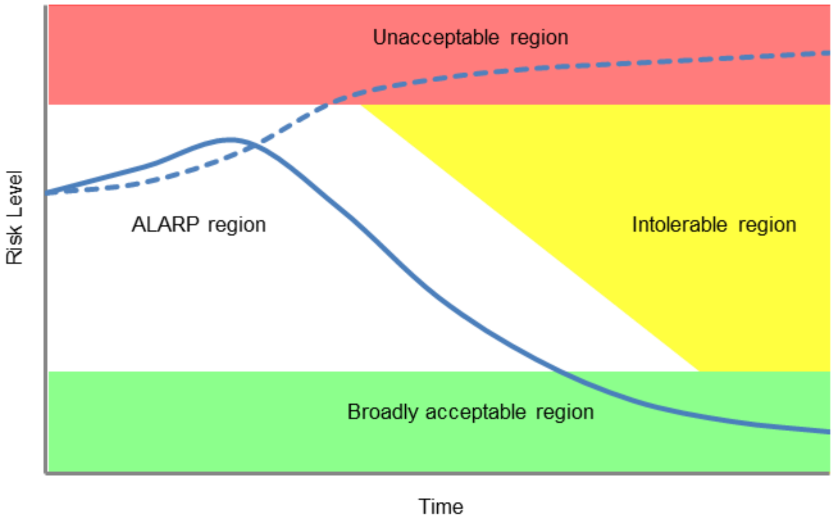
Risk source	Workers' exposure	Dust dispersion	Damage due to fall	Re-criticality	Dust dispersion during the removal of rubble	(Estimate of duration)
Fuels in the SFP	Low	None	Medium	Low	High	Up to 1 year
Fuel debris	High	High	Medium	Medium	Low	Up to 10 years

Column: Risk as a function of time

During risk treatment, there is a possibility that the level of risk may increase due to the change in the condition of facilities and risk sources as well as the work carried out. On the other hand, even when no actions are taken, risks may be emerging or increasing.

The figure shows a conceptual image derived from the references below. Even if a level of risk is present in the ALARP region, it cannot remain there permanently and eventually it gets intolerable (the yellow region). In addition, as time passes, the level of risk could increase due to deterioration of the facilities and the risk source (the dotted line).

On the other hand, by taking appropriate actions against risks, it is possible to prevent the risk sources from entering the unacceptable region. This can be done via careful preparations and thorough management, in spite of a possible temporary increase in the level of risk. Thus, it should be targeted to reduce the level of risk without entering the unacceptable or intolerable regions (the solid line).



Reference: V. Roberts, G. Jonsson and P. Hallington, “Collaborative Working Is Driving Progress in Hazard and Risk Reduction Delivery at Sellafield” 16387, WM2016 Conference, March 6-10, 2016.
M. Weightman, “The Regulation of Decommissioning and Associated Waste Management” The 1st International Forum on the Fukushima Decommissioning (April 2016).

3.3 Steady promotion of decommissioning project

To make steady progress in the designed risk reduction strategy, and to accomplish the fundamental policy of continuously and promptly reducing the risks posed by radioactive materials, risks influencing the progress of decommissioning should be identified. Analyzing their severity, measures against significant risks should be prepared. Namely, it is important to review comprehensively the project risks, such as failure of technological development, insufficient personnel and space, cost increase, and reworks caused by uncertain safety considerations.

Making progress in decommissioning the Fukushima Daiichi NPS is deeply connected with the return of the evacuees to their homes. Even minor troubles or environmental impacts may affect the residents of the surrounding areas seriously because of reputational damage. Therefore, it is essential to make a clear explanation to society about the prospect of decommissioning work and to share various risks with the local residents.

Any delay in the decommissioning would cause continuous social risks such as reputational damage, and the response to social risks might, in turn, delay the decommissioning. Therefore, these two cannot be separated.

In Sections 3.1 and 3.2, direct impacts of radioactive materials are considered as risk. In this section, however, by returning to generic definitions in Table 3-1, a wide range of risks affecting the steady progress in the decommissioning are to be considered.

3.3.1. Project risk management

(1) The Strategic Plan and project risk management

The logic tree for accomplishing the risk reduction in fuel debris and waste is shown Figures 3-8 and 3-9, respectively. Most of the strategic plan accounts for technical reviews to establish the technological requirements, which are discussed in detail in Section 4 and Section 5, respectively.

Since the technological requirements for successful risk treatment of fuel debris are challenging, multifaceted reviews are now being made. Among others, understanding the in-core condition, response to the safety requirements and securing accessibility through decontamination are essential prerequisites for success; however, under the current condition where it is difficult to access the PCV because of the high radiation dose, the possibility of not attaining them should also be taken into account. Risk management is also important in selecting a fuel debris retrieval method, and some alternatives should be reviewed, not just a single method. In a series of decommissioning tasks, it is also important to secure storage locations for equipment and wastes as well as working areas. As an example, the increase in the tanks for contaminated water including treated water is pressing the on-site area.

To accomplish the technological requirements for successful risk reduction of waste, reviews are being made from a long-term viewpoint. In particular, understanding the amount of generated waste and its properties is important to develop a risk reduction strategy for waste; thus, it is essential to grasp the in-core condition for a successful strategy.

In addition, a delay in promotion of R&D to support the actions against these risks, as well as fostering and securing of human resources is essential as a project risk. To succeed in R&D, the Decommissioning R&D Partnership Council has been established to manage in an integrated fashion from fundamental to field application studies conducted by IRID, JAEA, universities and research laboratories. JAEA has large-scale testing facilities where a group of researchers in Japan and abroad are concentrated to work jointly on the R&D. For fostering and securing of human resources, Center of Excellence has been organized such as on universities. These are discussed in detail in Chapter 6.

(2) Method of project risk management

As discussed above, project risk management is to identify the risks associated with the progress in the decommissioning project and to take actions against significant risks in advance, and this is nothing but deploying specific efforts through the Strategic Plan. Examples of the technique to manage project risk in more systematic manner are described in the following. Project risk management needs to be implemented in reference to such examples. Although we tend to focus on problem solving, it is also important to prepare the countermeasures in case the desired result is not achieved.

As a generic risk management method, FMEA (Failure Mode and Effects Analysis) is well known. In this method, potential failures are listed for each process or for each function, three indices of the consequence and its magnitude, the cause of a failure and the likelihood of its occurrence, and the method of controlling or detecting it and its feasibility are assessed, then the severity of the risk is determined by obtaining the product of them. Then, countermeasures are developed for the risks of large severity, and effective actions will be taken by forecasting how these three indices would be reduced.

DRiMa[7] (Decommissioning Risk Management) of IAEA has been organized to develop a risk management method for the safe and steady promotion of decommissioning. It uses a similar method to that of FMEA; however, risk is regarded as impacts on the progress of decommissioning (both a threat and an opportunity) caused by uncertainties. Certain assumptions will be made for uncertainties and the possibility that such assumption fails to become reality will be the risk. If the impact when assumption is not realized will impede the progress of decommissioning, measures should be prepared in advance so as to minimize the impact. Meanwhile, if it will be the opportunity to advance the decommissioning, assumption is to be challenged so. Such action is recognized as risk management. A list of risk factors associated with the decommissioning is developed in reference to past experience, and it will be useful in understanding where a risk would emerge.

These risk factors are also arranged in the Value Framework[8] of NDA, which serves as a reference when reviewing the project risks. In the Value Framework, the factors are arranged in a 3-tier hierarchical structure, and the top tier consists of the following factors. The middle tiers and lower tiers consist of 24 and 54 factors, respectively. As shown below, the risk reduction of radioactive materials, which is analyzed using the SED score, is only one of the factors; thus, various factors should be considered in reviewing the entire decommissioning project.

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[7] Risk Management on Strategic and Operational Level during Decommissioning – First Outcomes of the DRiMa Project at IAEA – 14467, WM2014 Conference, March, 2014

[8] The NDA Value Framework, January, 2016

- Health and safety during the decommissioning work
- Security of radioactive materials, etc.
- Protection of the environment from radioactive materials and chemical substances
- Reduction of risks (or hazards)
- Impacts on the social economy such as the local employment and infrastructure
- Financial affairs such as costs and return on investment
- Enabling missions other than the progress in decommissioning, such as setting precedents and building capability

(3) Basic concept for ensuring the safety

The regulatory requirements to the Fukushima Daiichi NPS as a specified nuclear facility are established as the matters for which actions are to be taken. To carry out risk reduction such as fuel debris retrieval, designing of installations and the working plan shall be developed by ensuring safety in compliance with the matters for which actions are to be taken. However, no precedents are expected in many cases, and it is inevitable that various assumptions will be made on what are the safe conditions and how their achievement could be confirmed. As a result, in case there are inappropriate assumptions, designing or scheduling might be prolonged, or rework such as reconsideration of the assumptions might be required.

To avoid such a situation, it is helpful to develop a basic concept for ensuring safety in compliance with the requirements of the matters for which actions are to be taken, and share it with the stakeholders in advance. In doing so, the fundamental policy of continuously and promptly reducing the risk of radioactive materials should be reconfirmed, and the following differences should be understood between the Fukushima Daiichi NPS and a power reactor.

- The Fukushima Daiichi NPS will not be restarted. Thus, it does not necessarily advanced technologies or installations that are essential to maintain constant power operation in a power reactor.
- Noble gases and volatile FPs were released at the time of the accident. Also, radioactivity and decay heat are reduced over time since the accident, and hence sufficient time is available for the recovery in case of failure in filling nitrogen or in cooling.

3.3.2. Relation with the society

(1) Sharing risk awareness with the local communities

In steadily proceeding with the decommissioning, it has been pointed out that communication with local residents is greatly important, not only by domestic experts but also by experts of foreign countries and international organizations with experience in decommissioning. As a first step, accurate and timely information should be sent out; the fact that great efforts made by the workers are contributing to the progress in decommissioning should be conveyed, as well, of course, as the troubles that actually happened.

In addition, it is expected to explain the risk status, as well as its control method, at every milestones of the decommissioning process. Through such risk communication, it is also expected to establish common understandings on the target level of risks, to be achieved within the concept such as of ALAP. Such target risk level is expected to meet with the safety objectives in line with the regulatory requirements and the international standards and besides, achieving the target will provide the local residents with a great reassurance.

Particularly, fuel debris containing a large amount of radioactive materials is now kept under a certain level of containment. If safe and reliable removal methods become available, risks can be reduced without causing serious troubles. However, if the retrieval work is performed too hastily without thorough preparations, it would remain susceptible to unexpected troubles until the retrieval work completes. Namely, the risk reduction strategy involves a trade-off between promptness and carefulness. Thus, it is necessary to separate the risks that should be removed as soon as practicable from those requiring careful actions; this awareness must be shared with the local residents.

Such communication will not be accomplished just by sharing the information between the sender and the receiver. It is important to accept the understanding by the receiver, to respect the opinions gained, to make mutual efforts so as to reduce the gap between the sender and the receiver, and eventually to cooperate toward decision making.

(2) Reputational damage

In considering the impacts on people and the environment caused by radioactive materials, direct impacts such as exposure of the public and the workers were discussed in Sections 3.1 and 3.2. Apart from this, there are also indirect impacts, and a typical example is reputational damage.

A representative reputational damage is "an economical damage caused through suspension of spending and sightseeing by the people who regard the inherently safe food, products or land as dangerous when a particular event, accident, environmental contamination or disaster is sensationally reported." In the case of the Fukushima Daiichi NPS, reputational damage may be caused not only by the emergence of a risk through leak of radioactive materials but also by the only presence of the risk. In addition, magnitude of its consequence is taken into account, but the likelihood of occurrence tends to be unperceived.

At the Fukushima Daiichi NPS, reputational damage and actual economic damage are present in the current situation. In overcoming the damage and taking some actions, additional means to prevent further occurrence of reputational damage may be required, or the understanding for taking such actions itself may not be gained. As a result, there is a possibility of a delay in improving the current situation, thereby causing prolonged reputational damage.

Similarly, in taking actions against the existing risk sources, there is a possibility that a delay in risk treatment or increase in the worker dose and the cost may occur, as a result of fearing a possible occurrence of reputational damage. This may lead to lowering the evaluation by society with regards to efforts in the decommissioning, which, in turn, leads to a vicious circle of further delay in taking actions against risks.

To prevent further occurrence of such reputational damage, it is most important to properly manage the radioactive materials so as not to lead to leakage, and to promptly reduce the existing risks. Further, it is important to continuously provide accurate information to, in addition to the local residents, the people in the press having a great influence on society, and the market participants and distributors, as well as the consumers in Japan and also abroad.

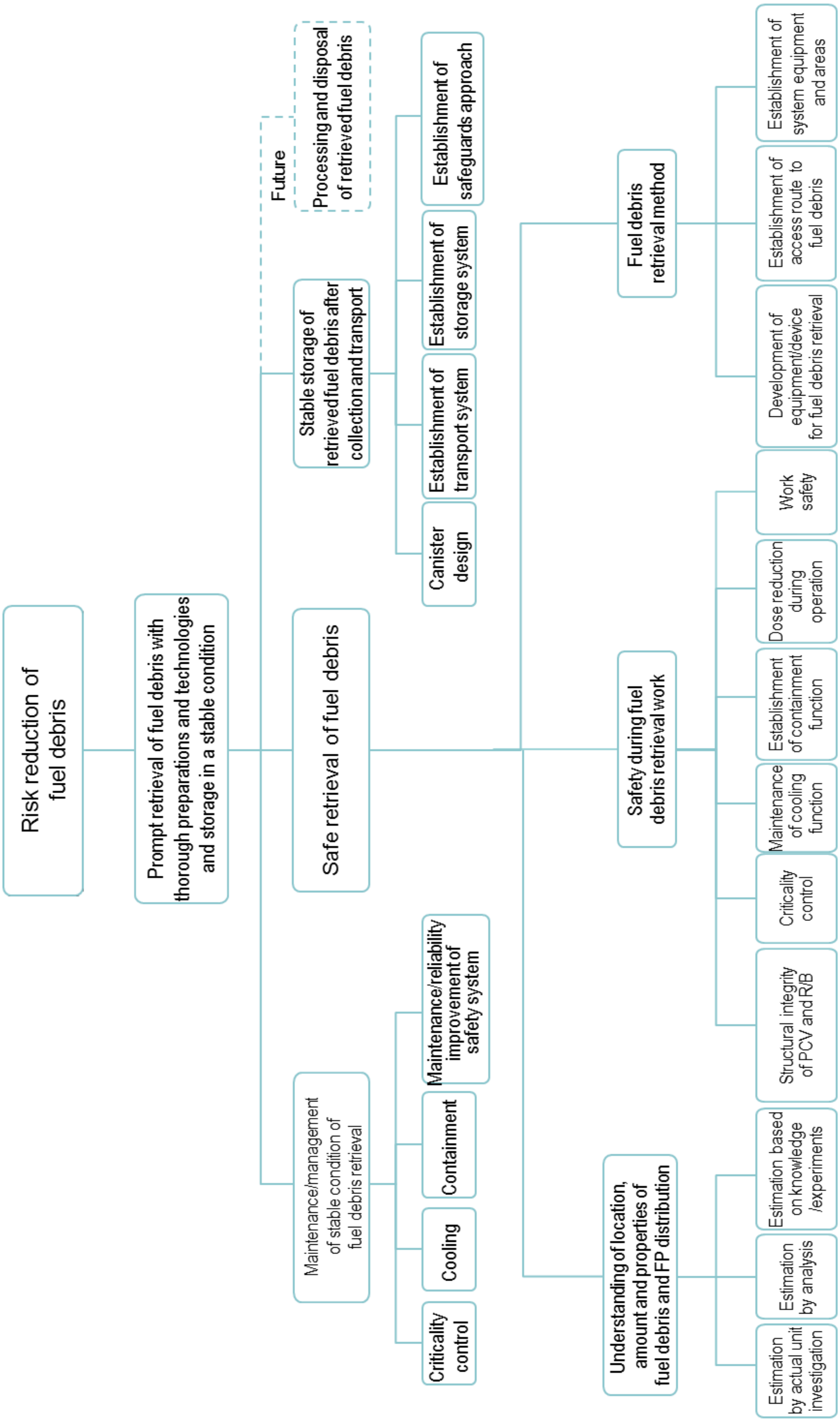


Figure 3-10 Logic Tree for Risk Reduction of Fuel Debris

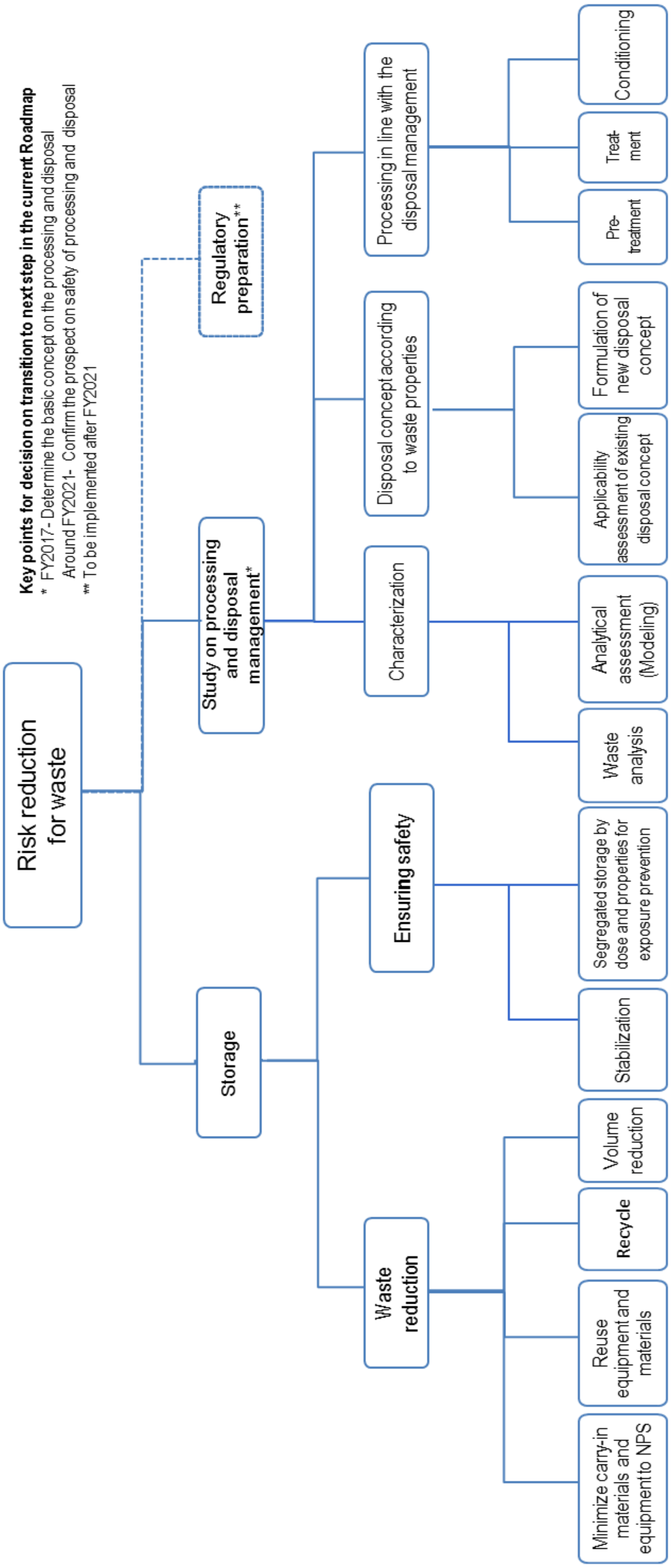


Figure 3-11 Logic Tree for Risk Reduction of Waste

4. Strategic plan for fuel debris retrieval

4.1 Study plan for fuel debris retrieval (risk reduction)

Fuel debris is characterized by "being in a state that contains nuclear fuel materials without being contained in the cladding, and combined with other materials"; therefore they have factors of risks related to criticality, decay heat, containment, risk of high radiation, hydrogen generation, and degradation of the integrity of supporting structures. Followings are difficulties in managing such risks.

-"Uncertainties" of the reactor condition,

-"Instability" of molten fuels and damaged facilities caused by the accident and

-"Insufficient management" due to severe access condition under a high radiation environment.

The amount of radioactivity (Bq) of fuel debris has been greatly decreased to about one several-hundredth of the amount immediately after the accident. Also plant parameters for criticality, cooling and containment are stabilized by the control performed according to the implementation plan issued by TEPCO. (For details, refer to 4.2.2 Stable state maintenance/management).

However, to continuously and promptly reduce the risks associated with fuel debris, which is the fundamental policy, the strategy to reduce risks from two points of view from mid-and long-term will be required.

Mid-term risk is a risk of a deviation from "a certain stable state" which is currently maintained for the fuel debris. Examples of such deviations include re-criticality, cooling related issues, degradation of reactor internals and leakage of radioactive materials. These deviations are unlikely to take place as long as the current stable state is maintained. However, the understanding of the conditions and planning of the measures against risk sources are desired to be established in the early stage since the methods which can directly control the conditions inside the reactor have yet to be established. More stable state of the reactor could be achieved by the appropriate measures, such as collection of highly unstable fuel debris (the debris with the locational instability and in physically and chemically unstable state) and checking the conditions of the fuel debris and reactor internals.

Long-term risk is a risk of environmental contamination caused by the leakage of highly toxic nuclear fuel materials due to the deterioration of the buildings. To secure the ultralong-term safety by isolating spent fuels from human environment, in Japan, high level waste is isolated and stabilized after being reprocessed (geological disposal) as a fundamental policy. To leave fuel debris corresponding to the spent fuels of about 270t in Units 1-3 does not comply with this fundamental policy. This is because the durability of damaged R/B is limited and its containment function cannot be assured for a long period of time. The fuel debris are, therefore, to be collected within a time frame (about several decades) where the containment can be maintained by the R/B and be brought into a stable state under sufficient control. To make the risk level eventually the same as the risk of back-end project is a fundamental policy.

Having considered the above discussion, and seen the consequences of the initial measures to reinforce the R/B taken at the accident of Chernobyl nuclear plant Unit 4, leaving the nuclear materials for a long time without establishing the proper collection policy - just thinking about short-term containment without long-term safety considerations - must be labelled as an irresponsible postponement of the issue to the next generations, as long-term safety can never be envisaged in such a situation.
In the decommissioning of the 1F NPPs, therefore, the measures taken at the Chernobyl nuclear plant Unit 4 can never be an option, and, fuel debris retrieval as explained below has an absolute priority.

As a mid-term risk reduction strategy for the fuel debris, the following items are to be studied based on the current stable state to solve the difficulties in the risk management described above, and "the stably controlled condition stably controlled condition based on the more accurate information" is aimed to be achieved.

(1) Understanding of the conditions and properties of fuel debris (decrease in uncertainties)

- Current stable condition can be ensured as uncertainty decreases through the understanding of the conditions and properties of fuel debris. Also information obtained will be important inputs to the studies on the safe and proven method for fuel debris retrieval.

(2) Improving the reactor condition during the fuel debris retrieval (resolution of instability)

(3) Maintaining the fuel debris in a stable storing state (improvement of management level)

Both mid-term and long-term risk reduction are important for the fuel debris retrieval. For the former, an early stage execution and effectiveness of stabilization inside the reactor are desired, and for the latter, high collection rate for the fuel debris is expected t even if it takes some time. For this reason, in the initial operation of fuel debris retrieval, mid-term risk reduction is required to be focused on while selecting the method which can collect fuel debris efficiently. If a certain amount of fuel debris is retrieved by this method, reducing mid-term risk and safety of the R/B is maintained by a passive method*, "a low risk level widely accepted by the society" is said to have been achieved. On that basis, the risk elimination (removal and isolation of nuclear fuel materials) is to be aimed from a longer term perspective through the subsequent works such as further fuel debris retrieval and facility disassembling. Consequently, the fuel debris retrieval to reduce mid-term risk is required to be aimed for the time being.

*A state where cooling of the fuel debris and prevention of re-criticality, leakage such of radionuclide and hydrogen explosion are ensured by a passive method.

In the meantime, in the studies of fuel debris retrieval strategy, note that the retrieval work itself will not be justified if the level of the risk caused by the fuel debris retrieval work (leakage of radioactive materials and workers exposure due to the failures during the operation) is higher than the permissible level. Also, the human resources and time assigned for decommissioning are not unlimited; therefore it is important to reduce risk by searching a realistic technical strategy for fuel debris retrieval while ensuring safety. The approaches based on the concept (Refer to Colum on P.3-4) of ALARP (As Low As Reasonably Practicable) will be required. The basic approach needs to be recognized as a "risk-aversion-oriented approach instead of focusing on the outputs" described in the Roadmap (Revised on June 2015). The process of fuel debris retrieval is to be set flexibly while assessing the risks.

That is, technical strategy of fuel debris retrieval is to explore the optimal point between "resolution of mid- and long-term risk of damaged reactors" and the "risk involved in the retrieval work," that are in a trade-off relationship while balancing the issues such as regarding technical specifications, time period, work safety, and actual work site conditions. This Chapter describes the concept to select a method with high accuracy and its technical background.

The Roadmap declares that "Start of fuel debris retrieval at the first implementing unit" is to start by December 2021, and, as major milestones for that, “Determination of fuel debris retrieval methods for the first implementing unit” is to be made in the first half of FY2018 and " Determination of fuel debris retrieval policies for each unit" in summer of FY 2017.

The detailed approach to the studies on the fuel debris retrieval method and sharing of the roles among the related organizations involved in the fuel debris retrieval work are described in the following chapter.

Column: What the fuel debris is

According to the definition introduced by IAEA, fuel debris is "fuels melted and resolidified with fuel assemblies, control rod and internal structures." As a similar term, there is "corium." This term is defined as "the once molten mixture of components of a nuclear reactor. It can consist of nuclear fuel, fission products, control rods, structural materials from the affected parts of the reactor, products of their chemical reaction with air, water and steam, and, if the reactor vessel has been breached, concrete from the structure of the reactor space" according to the definition by IAEA. Although the debris caused by the accident at the Fukushima Daiichi NPS are rather corium, they will be called “fuel debris” even from now in Japan since it is composed mostly of fuels and the term is entrenched.

Samples of fuel debris caused by the accident at the nuclear power plant in the world are as follows.

Photograph A
Fuel debris of TMI-2

Photograph B
Fuel debris of Chernobyl

Photograph C
Fuel debris of Fuel debris of Windscale

As it can be recognized by those photos, the appearances of the fuel debris are significantly different depending on the reactor types and accident progression. The retrieval measures should be therefore taken corresponding to the situation of each case. In fact, out of the plant above, the plant where fuel debris was retrieved was only TMI-2 and the fuel debris still remains in Chernobyl and Windscale.

According to the definitions above, since fuel debris is molten and solidified fuels, fuel assemblies that remain unmelted and FPs that vaporized, leached and scattered when being melted, and actinide are not considered fuel debris. However, if there is fuel assembly which was not melted, it needs to be retrieved. Also, the reactor internals that melted and solidified by the impact of the accident will be not fuel debris if fuels are not mixed with them.

Photographs A-C are reprinted from the following under the permission of IAEA obtained by the NDF.
Photograph A and B: IAEA Experience and Lessons Learned Worldwide in the Cleanup and Decommissioning of Nuclear Facilities in the Aftermath of Accidents, IAEA Nuclear Energy Series No.NW-T-2.7, IAEA, Vienna (2014)
Photograph C: Managing the Unexpected in Decommissioning, IAEA Nuclear Energy Series No.NW-T-2.8, IAEA, Vienna (2016)

4.1.1 Approaches to the study on fuel debris retrieval

According to the basic concept of fuel debris retrieval (risk reduction), fuel debris retrieval work is a series of the process that includes not only the retrieval but also the collection, transport and stable storage of fuel debris. To select a method to realize those processes, following items are required to be studied. The flow of the studies is shown in Figure 4.1-1 Logic tree.

Note; Definition of "Fuel debris retrieval method" and "Fuel debris retrieval methodology" in the Staretgic Plan are described below.
Fuel debris retrieval method: an individual process to retrieve fuel debris. At this point, it refers to "Submersion-Top access method," "Partial submersion-Top access method" and "Partial submersion-Side access method" which are focused on are aimed to be achieved. (For details, refer to 4.3.1).
Fuel debris retrieval methodology: it considers the process from the commencement to the completion of fuel debris retrieval work for each Unit, and refers to the contents including the method to be applied and its order of the application, if multiple methods are applied.

(1) Understanding and estimation of the situation inside the PCV (For details, refer to 4.2)

The studies on fuel debris retrieval methods and development of fuel debris retrieval equipment and devices require the information regarding the amount and properties of fuel debris for each location, such as the core region, bottom of the RPV, CDR housing, and the inside and outside of the RPV pedestal. In addition, the studies on the establishment of an access route to the fuel debris and of the system equipment and working areas require information regarding the FP distribution and damaged condition of reactor internals that affect the radiation evaluation. Also, a certain period of time will be necessary before the commencement of the fuel debris retrieval work. By then, in order to ensure safety, it is important to identify the state of FPs such as of the plant, fuel debris and cesium, and maintain and control them in a stable manner.

That is, the following information regarding the conditions inside the reactor including fuel debris which are in the scope of the study is required as an input to the study of fuel debris retrieval (risk reduction).

a. Locations, amount and properties of fuel debris
b. FP distribution inside the reactor (dose rate)
c. Damaged condition of reactor internals

These pieces of information are desired to be obtained as actual data from the actual plant. However, observation of the distribution conditions of fuel debris and FPs in the PCV involves high degree of difficulties even if the current radioactive environment in the PCV of each Unit is lower than that after the accident, since the radiation dose is still high as described in Appendix.4.1.

Therefore, in the preliminary understanding and estimation of internal PCV condition, its needs are to be clarified and comprehensive analysis/evaluation are aimed to be carried out based on the all available information using not only visual inspection with remote devices but also the findings and experiments in the past and following analyses.

a. Estimation based on plant investigation

i) PCV internal survey

ii) RPV internal survey

iii) Muon detection

iv) Investigation of the leak locations of the PCV

v) Sampling of fuel debris

b. Estimation by analysis

i) Improvement of severe accident progression analysis code (MAAP，SAMPSON)

c. Estimation based on knowledge and experiments

i) Plant parameter analysis

ii) Heat balance method

iii) Results of test and research of severe accidents in the past

iv) Properties test using simulated fuel debris

(2) Feasibility study (FS) of fuel debris retrieval method (For details, refer to 4.3)

To realize the fuel debris retrieval, required technical requirements are identified as follows. The approach will be taken so as to study, design and develop the fuel debris retrieval method that satisfies these nine requirements and evaluate its achievement and prospect.

i) Securing the structural integrity of the PCV and the R/B

ii) Criticality control

iii) Maintaining the cooling function

iv) Securing containment function

v) Reduction of workers' exposure during operation

vi) Ensuring work safety

vii) Establishment of an access route to the fuel debris

viii) Development of fuel debris retrieval equipment and devices

ix) Developing system equipment and working areas

Also, i) - iv), out of these technical requirements are those related to the ensuring safety during the fuel debris retrieval work and vii)-ix) are to the fuel debris retrieval method. Six technical requirements on ensure safety need always be satisfied in each step of fuel debris retrieval work including the preparation work. Most of them are required to be designed to satisfy the required specifications as a system structure. The feasibility of installation area for system equipment is also required to be evaluated. That is, those are required to be considered in conjunction with the technical requirements on the fuel debris retrieval method.

The Fukushima Daiichi NPS has been designated as specified nuclear facilities and there are no specific standards for the technical requirements on ensuring safety. For this reason, the studies on the concepts and standards on ensuring safety are also required to be performed based on the on-site situation and communication with regulatory body.

As described above, there are many technical requirements to be satisfied for the fuel debris retrieval method. To evaluate those feasibilities, the methods subject to the studies and developments are required to be examined in detail.

For this reason, in the original Roadmap, investigations and developments were started based on the Full submersion-Top access method which is used during the periodical inspection of standard power station, and is also applied for TMI-2. This method is performed by raising the water level to the upper part of the reactor well to cool down, provide shielding and prevent the scattering of radioactive dust.

The situation of the Fukushima Daiichi NPS is, however, different from TMI-2 and fuel debris may have reached to the bottom of the PCV, which is the outside of the RPV pedestal, instead of staying in the RPV. Also, the PCV has been damaged and the water injected for cooling the fuel debris was leaked from the PCV and it is staying inside the building. To realize the submersion-top access method under such conditions, there are major issues which are to travel a significantly long distance to the fuel debris located at the bottom of the PCV, specifically outside of the RPV pedestal, and to raise PCV water levels by repairing the PCV leak locations.

Therefore, assuming the difficulties in developing the Full Submersion method -Top access method, the development of the methods with different access directions and different PCV water levels are discussed in the Strategic Plan 2015. Out of the method combining the possible access directions and PCV water levels, three methods which have less difficulty in the element of the development were selected as the method to be focused on. The conceptual design and FS will be performed for these three methods and their feasibility will be evaluated. (For details, refer to 4.3.1)

- Submersion-Top access method (including Full Submersion method)
- Partial submersion-Top access method
- Partial submersion-Side access method

Although the technical requirements above are required to eventually be satisfied, it is efficient to focus on the following requirements particularly important for each method in evaluating the feasibility of fuel debris retrieval method.

- Submersion method:
  - PCV repair and establishment of the water level control system
  - Ensuring the structural integrity of the PCV and the R/B considering its load and aged deterioration when submerged
  - Maintaining sub-criticality when the water level rises
- Partial submersion method:
  - Shielding for high radiation from fuel debris
  - Prevention of the impact on workers and environment caused by the dust scattering to the outside of the building
  - Confirmation of radiation resistance such as of fuel debris retrieval equipment.
- Top access method
  - Establishment of access route to the R/B operating floor

- ➢ Securing the access route for the periodical inspection (e.g. removal of reactor well shield plug, PCV upper head, insulator for the RPV upper head, RPV upper head, steam dryer, steam separator)

- ● Side access method
  - ➢ Dose reduction for the CRD hatch on the 1st floor of the R/B and equipment hatch.
  - ➢ Removal of objects on the access route, such as Primary Loop Recirculation (hereinafter referred to as PLR) pumps, valves, piping and supports outside the RPV pedestal and CRD handling machine, operating floor inside the RPV pedestal to establish access route to the Drywell of the PCV (hereinafter referred to as "D/W").

(3) Studies on handling to stabilize conditions of retrieved fuel debris (For details, refer to 4.4)

The retrieved fuel debris is expected to be stored in the storage canisters and transported to the outside of the R/B to be stored within the site. The study, design and development are required to be performed so that a series of operation is implemented as a system. Also, the nuclear fuel materials to be transported needs to be studied in consideration with the following systems since they are subject to the safeguards.

a Storage canister design

b Development of the transport system

c Development of the storage system

d Establishment of safeguards measures

The fuel debris retrieval method should be studied, designed and developed so as to satisfy the technical requirements related to the fuel debris retrieval. Understanding and estimation of the situation inside the PCV is to be carried out through the close collaborations and arrangement since its results will be the input of this study. Although technical requirements on the collection, transport and storage of the retrieved fuel debris are required to be studied for any type of fuel debris retrieval method, coordination will be required since the requirements may be affected by the type of the method. The study on the storage area is equally important to determine the retrieval method, since retrieval cannot be achieved if there is no storage space.

(4) Studies on the approaches to the retrieval method for each Unit (For details, refer to 4.5)

The approaches to the retrieval method for each Unit are studied based on the (1)-(3) above. Note that there is a difference in the applicability of the methods depending on the locations of fuel debris inside the PCV. That is, the applicability of each method will be able to be evaluated by the results of the understanding and estimation of the situation inside the PCV in each Unit.

According to the current estimation of the situation inside the PCV (interim evaluation), fuel debris of all Units was estimated to be scattered at the bottom of the RPV, bottom of the PCV (inside the RPV pedestal), and bottom of the PCV (outside the RPV). All the fuel debris scattered in each Unit are not necessarily retrieved by one method and the policy may be made by combining multiple methods. For example, the location where first fuel debris is retrieved is determined first. Then, the method to be used in other locations are examined and studied together with the preceding retrieval work and the work in the next stage will be continued based on the improved retrieval method. Specific designs/studies and technical developments needs to be accelerated toward the "Determination of fuel debris retrieval methods for the first implementing unit," planned to be made in summer of FY 2017.

The whole image towards fuel debris retrieval is shown in Figure 4.1-2. A part of the FS, conceptual design and element test are currently being carried out for "Determination of fuel debris retrieval policies."

Also, in the studies on the policies, the fuel debris retrieval method and application technology is evaluated and determined based on the perspective of Five Guiding Principles as shown in the Strategic Plan 2015.

"Safe" should be evaluated with the highest priority. A target level foreseen from the perspective of safety regulations or independent safety ensuring should be set up for the safety; it is first required to carry out a conceptual design to meet the target. Seeking the realistic constraints for the resources (e.g. human, physical, financial and space) required for realization of this conceptual design from two perspectives (i.e. "efficient" and "field-oriented"), only the concepts judged feasible by the coordination with the field work are selected. Since the notion of "timely" should basically be used for priority setting in the technology where the conditions of "safe," "efficient" and "field-oriented" are satisfied, it is not desirable that safety, efficiency and site conditions are disregarded as a result of "timely" being preferred. With regard to "proven," the technologies with high technical maturity and high applicability are basically adopted. Therefore, the maturity of application technologies should be thoroughly evaluated by comparing and checking with various requirements to overcome, which are needed to put the methods into practice.

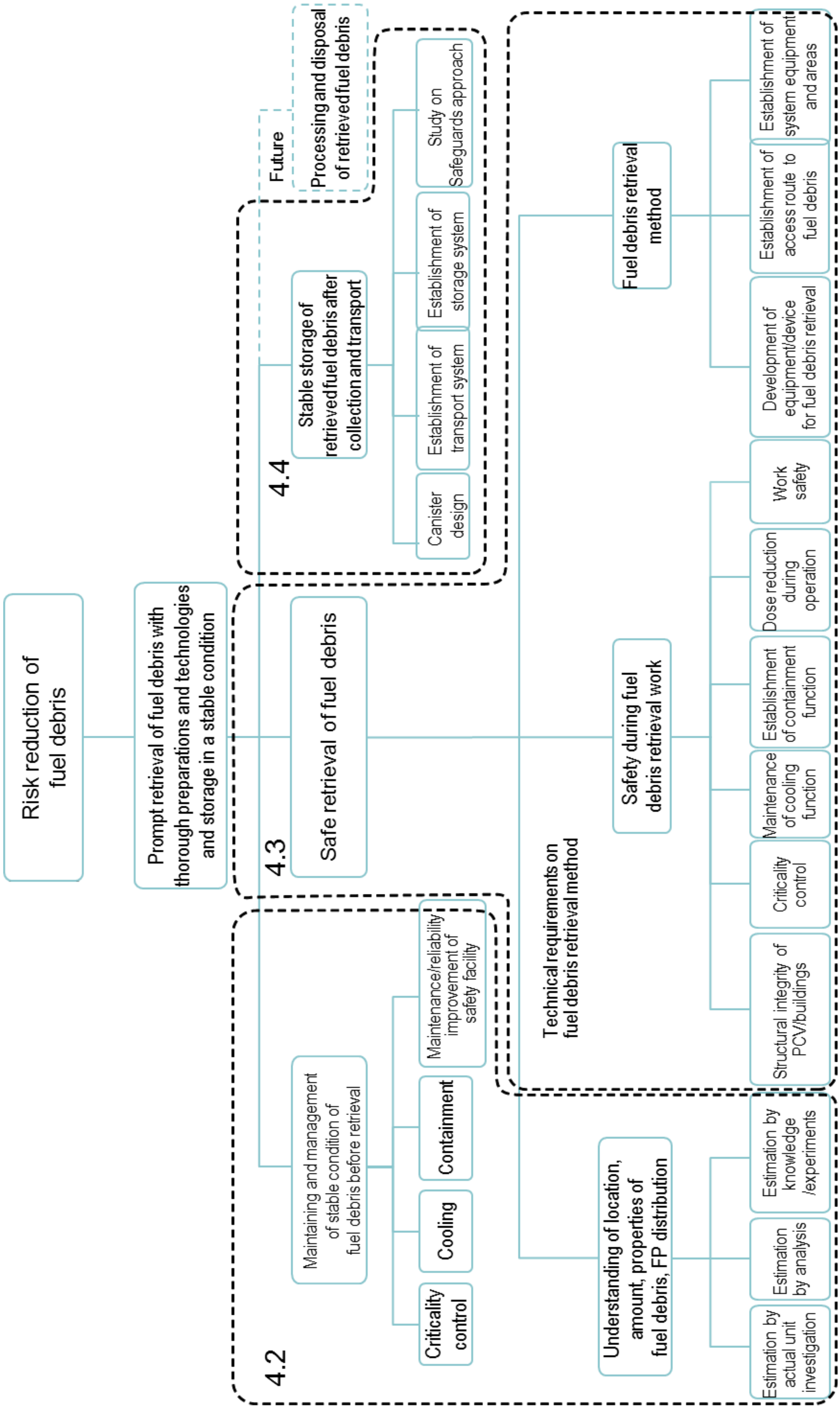


Figure 4.1-1 Logic tree on risk reduction for fuel debris

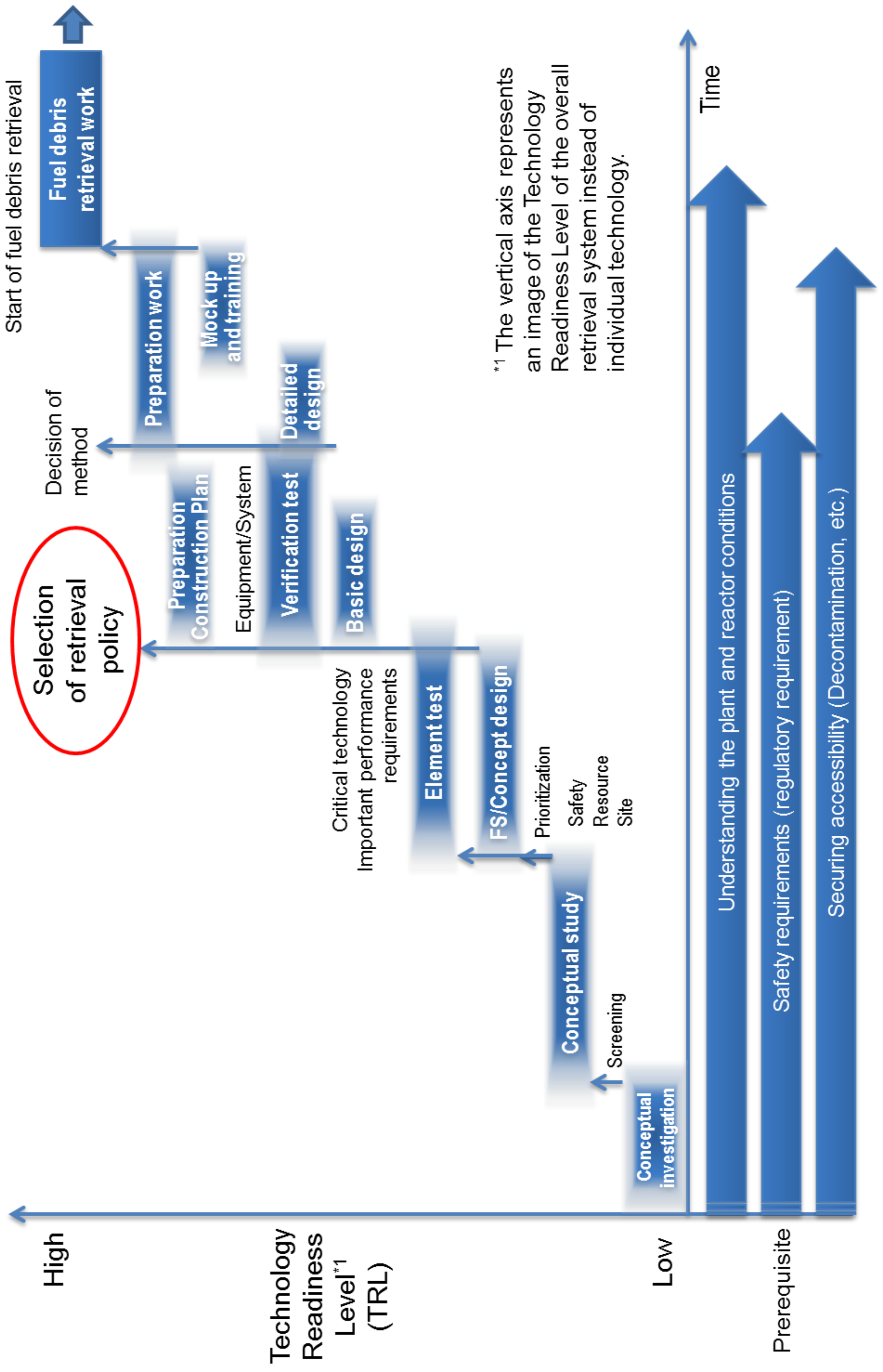


Figure 4.1-2 Processes toward fuel debris retrieval

4.1.2 Role-sharing among related organizations

Since the fuel debris retrieval work involves highly-difficult technical challenges which have never been experienced, not only collaboration with related industries and research institutes, but also acquisition of the knowledge and technologies from the broad range of fields in Japan and abroad including the field other than nuclear engineering. Also it is required to implement necessary R&D and apply the developed technologies to fuel debris retrieval work in the actual field.

The role sharing toward decommissioning of the Fukushima Daiichi NPS is described in Figure 1-1 of Chapter 1. Here, the fundamental views of role sharing of each organization that participates in fuel debris retrieval are briefly described below:

(1) TEPCO

- Investigation of plant conditions, basic design, procurement, detailed design, fabrication, construction work plan, training, technical studies on the on-site construction work and field work.
- Development and delivery of implementation plan (discussion with the NRA)
- Presentation of needs for the highly technical R&D projects to be carried out as a subsidized program of the Japanese Government, and management of field demonstration tests for review and evaluation for practical application.
- Collaboration and ensuring conformity among the technical studies, field works and R&D projects.
- R&D carried out by TEPCO itself

(2) The Government of Japan

- Policy (e.g. Basic policies for the Roadmap) decision and progress management of decommissioning.
- Budgetary measures for research and technical development that involves high degree of technical difficulties

(3) NDF

- Development of a strategic plan for the retrieval method
- Provision of support and progress management of technical studies
- Operation of advisory committee for fuel debris retrieval project
- Planning, coordination and management of R&D

(4) Research institutes (e.g. IRID)

- Development of R&D project plan and its implementation
  (e.g. development of equipment/device, development of the evaluation method and collection of data/information required)
- R&D progress management, collaboration and ensuring conformity among R&D projects

4.2 PCV inspection strategy and latest information

This chapter describes the positioning and the basic concept of the internal PCV conditions analysis, collection and evaluation methods of required information, current analysis results, and future inspection strategy.

4.2.1 Basic concept of internal PCV condition analysis

To understand the internal PCV conditions including plant conditions and fuel debris is extremely important to carry out the studies on the fuel debris retrieval method. However, considering the severe environmental conditions due to high radiation, to conduct inspection for all kinds of required information using with remote devices will be difficult from technical and temporal perspective.

For this reason, in order to obtain highly accurate results, making maximal use of not only plant investigation but also the results of the severe accident progression analysis and the evaluation based on plant parameter, required information should be analyzed and evaluated comprehensively based on the priorities set by the required timing, accuracy level, and significance.

Based on the above, the internal PCV conditions analysis is to be performed as follows.

(1) Considering the necessity of information, the collection, analysis, evaluation of information for internal PCV condition analysis are carried out according to the priorities given.

(2) Utilizing the information obtained effectively, the most probable results are to be achieved in the comprehensive analysis and evaluation of internal PCV conditions.

(3) Considering the balance of "safety measures, retrieval equipment and equipment design/cost" in the "workload, time and cost" for obtaining information, valuable information is to be obtained to the extent possible with a view to the time and cost allowed.

If it is difficult to obtain information in advance, assessment is to be performed based on the maximum likelihood method. A conservative process plan is to be established including contingency plan, and conditions of actual unit are confirmed while carrying out the work. The approach where process plan is narrowed down, materialized, and reviewed following the improvement of the accuracy of internal PCV condition analysis is also to be studied. This concept is based on an experience of TMI-2 which is similar accident plant. Figure 4.2-1 shows the image of the procedures.

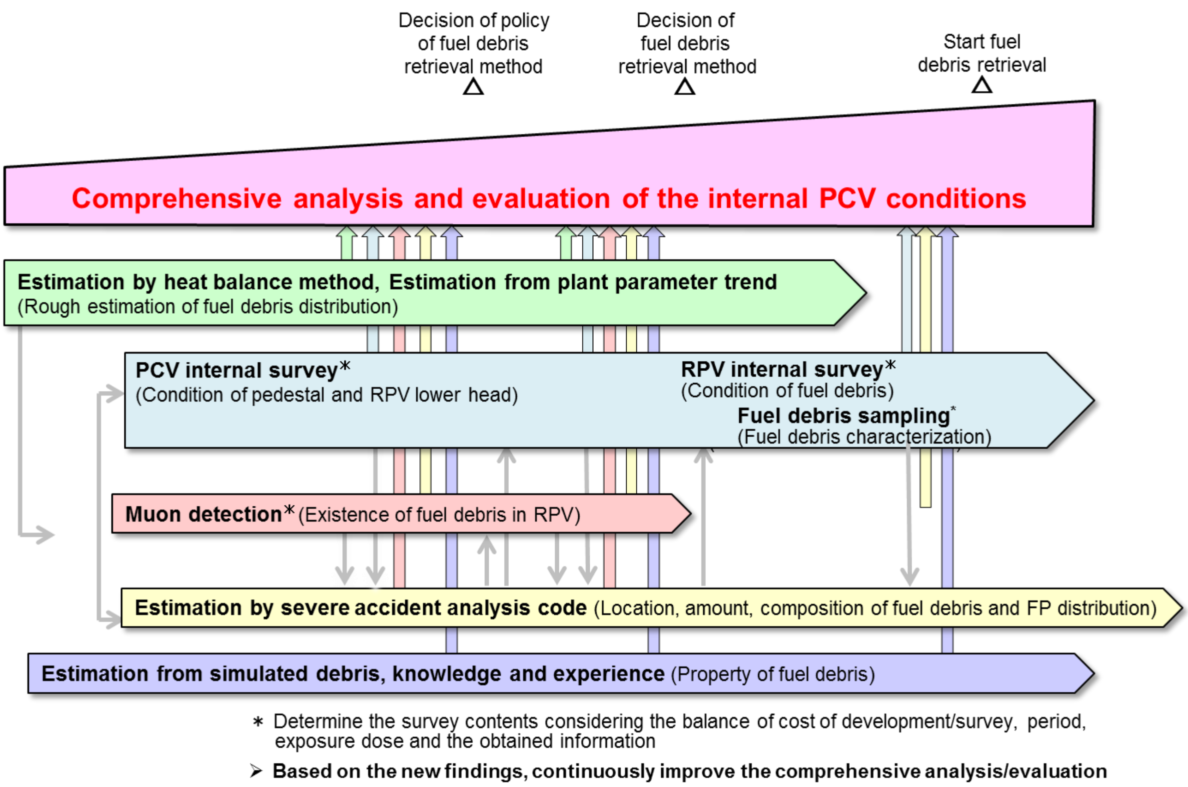


Figure 4.2-1 Strategy for internal PCV condition analysis

(Comprehensive analysis and evaluation)

Information required to study the fuel debris retrieval is considered to be used mainly for the purposes as follows.

Table 4.2-1 Information required for fuel debris retrieval studies

Purpose	Required information	Required timing
(1) Decision on approaches to the retrieval method	Fuel debris distribution	In summer of FY 2017
(2) Advancement of ensuring safety	Fuel debris distribution and properties	In summer of FY 2017
(3) Optimization of retrieval equipment and equipment design	Improvement of accuracy of information above, and FP distribution.	After FY2018 on a timely basis
(4) Further optimization for retrieval method and improvement of applicability	Detailed internal PCV conditions and fuel debris properties by sampling	To be continued including the period after the commencement of fuel debris retrieval

The purposes of the information above are as follows:

(1) To study the access direction and flow lines of fuel debris retrieval and concept of the systems.

(2) To ensure safety, such as the evaluation of re-criticality and cooling conditions. Allowance of the method can be reviewed reasonably according to the information.

(3) To rationalize the application to the actual plant depending on the quantity of information.

(4) To perform retrieval work according to on-site situation before and after the commencement of the retrieval work.

Note that the required accuracy and quantity of information are varied depending on the degree of the progress of the process even if these pieces of information are the same.
At this point, the analysis and evaluation of the information required to determine the approaches to the fuel debris retrieval method are being carried out as the first priority. The information obtained will be utilized not only for (1) but also for (2), (3) and (4). Therefore, when collecting information for (1), information for (2), (3), and (4) will also be obtained if it is reasonable. Table 4.2-2 shows major collection method and degree of importance of the information required for the decision on the approaches and method for fuel debris retrieval.

Also, in order to conduct study required to understand the amount, locations and properties of fuel debris and FP distribution, comprehensive analyses and evaluations are conducted by the estimation of three items which are (1) Investigation of conditions inside the actual reactor, (2) Estimation by severe accident analysis codes and (3) Estimation based on knowledge and experiments according to the logic tree shown in Figure 4.2-2. The information collection, analyses and evaluation are required to be performed utilizing its features since they have different features. The features for each item are shown in the Table 4.2-3.

Table 4.2-2 Information required for the decision on approaches to the retrieval method and retrieval method-Important issues and degree of importance of investigation method in the studies (Summary)

Important items in the studies	Survey method (data/information collection)						Importance level of decision of the policy	Importance level of decision of the method
	Severe accident analysis	Heat balance method /plant parameter	PCV internal survey	RPV internal survey	Muon detection	Fabrication of simulated debris /past findings		
1. Distribution of the fuel debris								
1) Fuel debris remained in the core region (including stub-shaped fuels)" *Important for the evaluation of possibility of criticality during Submersion.	◇	◇		◇	◇		◎	◎
2) Level of fuel debris remained inside the bottom of the RPV	◇	◇		◇	◇		◎	◎
3) Degree of adhesion to the CRD housing	◇	◇	◇				△	○
4) Transferring level of the fuel debris to the PCV	◇	◇	◇				◎	◎
5) MCCI behavior of fuel debris transferred to the PCV	◇						○	○
6) Possibility of expansion of fuel debris to the outside of the pedestal (including shell attack)"	◇		◇				◎	◎
2. Securing PCV/building structural integrity								
1) Evaluation of degradation due to being exposed to the high-temperature environment	◇	◇					◎	◎
2) Damage of the pedestal due to fuel debris	◇		◇				○	◎
3) Damage of the reactor internals	◇			◇			△	○
3. Criticality control								
1) Possibility of remaining stub-shaped fuels	◇			◇	◇		◎	◎
2) Various fuel debris state for criticality evaluation	◇			◇		◇	○	○
4. Maintenance of cooling function								
1) Evaluation of generation of heat considering shape and condition of fuel debris	◇	◇	◇		◇	◇	◎	◎
5. Provision of containment function								
1) Evaluation of the leakage amount of radioactive materials from the PCV gas phase	◇			◇		◇	◎	◎
2) Contaminated water leakage from PCV liquid phase	◇		◇				◎	◎
6. Reduction of exposure dose to workers during operation								
1) Study on decontamination method	◇			◇		◇	○	○
2) Study on shielding capacity during operation	◇			◇		◇	○	○
7. Development of fuel debris retrieval equipment and devices								
1) Study on fuel debris cutting /sampling method						◇	△	○
2) Study on specifications of equipment/device	◇	◇	◇	◇	◇	◇	△	○

(Note: internal information obtained by RPV internal survey is expected to have high reliability but high degree of difficulty is anticipated in the technical development and it may be difficult to be obtained at the time of determining the method for initial unit.
In that case, it is required to study the investigation results obtained by other means. It will contribute to the optimization after the manufacturing of the actual unit equipment.
Symbol◇: Information research method, ◎: High degree of importance, ○: Moderate degree of importance,
△: Low degree of importance Further detailed requirements for the survey with high degree of importance will be arranged with related institutions.

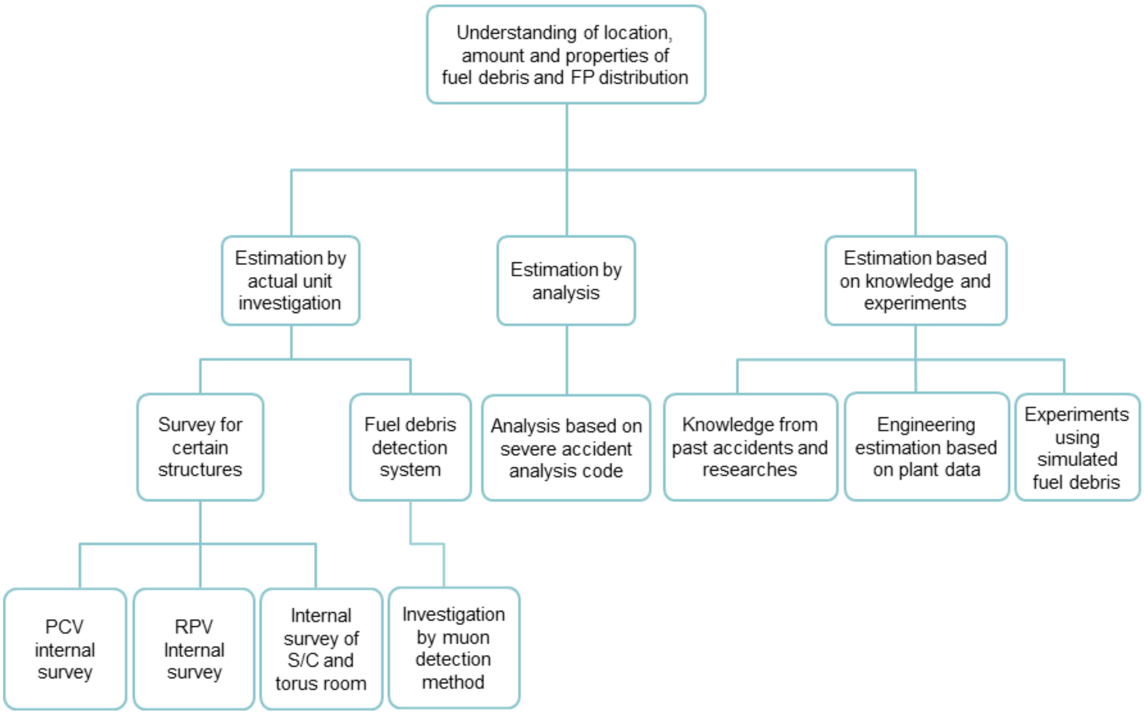


Figure 4.2-2 Logic tree for understanding the situations of fuel debris and FP

Table 4.2-3 Features of method of collecting information for internal PCV condition analysis

	Method (expected information)	Features
Investigation of conditions inside the actual reactor	・Visualization and measurement by PCV internal survey (i.e. Verify fuel debris distribution, damage state of the bottom of the RPV and shell attacks)	・High reliability of information obtained for actual measurement. ・Not suitable for comprehensive understanding due to local data.
	・Visualization and measurement by RPV internal survey (verify fuel debris distribution)	
	・Muon detection (Confirm fuel debris inside the RPV)	・Capable for understanding the fuel debris distribution inside the RPV. ・Small fuel debris cannot be evaluated due to low resolution.
Estimation by severe accident analysis code	・Reduction of uncertainties in the analysis using severe accident progression analysis code and estimation of internal PCV condition by the sensitivity analysis and inverse calculation. (Reduce uncertainties in the locations, amount and properties of fuel debris and FP distribution, and amount and locations of fuel debris)	・Capable for the estimation of plant parameters which were not measured at the time of the accident and comprehensive understanding of fuel debris distributions. ・Difference may be caused depending, such as on the conditions and models because of the uncertainties.
Estimation by knowledge and experiments	・Fabrication of simulated debris and findings from the past (Mechanical/chemical properties of fuel debris)	・High reliability of information obtained for actual measurement. ・Depends on the conditions of simulated debris.
	・Consideration from the heat source (fuel debris) evaluation by heat balance method and plant parameter (Verify fuel debris distribution trend)	・Low dependency for the models based on the measured value. Capable for comprehensive understanding. ・Low quantification performance. Only the presence of the fuel debris can be identified.

4.2.2 Maintaining management of stable condition

The current conditions of Units 1-3 can be estimated to maintain the stable cold shutdown state based on the plant data of the inside of the PCV, such as the temperature, hydrogen concentration and pressure recorded since the accident. Appendix 4.2 shows its details.

Moreover, technical developments required for practical application of the fuel debris retrieval methods, such as the development of equipment and facilities for remote decontamination, investigation, and work are being advanced. It is, therefore, important to maintain, control and monitor the condition of plant, fuel debris and FP such as Cs in ensuring safety until fuel debris retrieval starts.

Although at this moment, it is difficult to observe the condition of fuel debris directly due to high radiation, the information is required to be understood from the perspective of maintaining and management of safety state. The status of the information obtained to date is as follows:

(1) Criticality control

a. Xe-135 concentration, which is a short half-life FP, is continuously monitored by the gas radiation line monitor installed in the PCV gas control system of each Unit. While the criticality criterion has been set to 1Bq/cm3, no sign of criticality has been shown. In addition, evaluations have been carried out by using various conditions, such as compositions and shapes of fuel debris, deposition shape, composition and mixed amount of structural materials. The results indicated that the possibility of reaching criticality was low.

b. A boric acid water injection system has been installed in order to make it subcritical or prevent criticality for the case when the fuel debris reaches re-criticality or there is a possibility of re-criticality. Two tanks of boric acid water has been installed (one for spare), and alkalescent sodium pentaborate solution which has little effect on structures will be injected through the reactor coolant injection system. This system can achieve the boron concentration of 510ppm, which is equivalent to the reactivity of more than 5%.k. In addition, when boric acid is exhausted up, the sea water having the reactivity reduction effect of approx. 3%.k will be injected. The time required from the occurrence of criticality to the completion of injection takes usually 6 hours and 22 hours at the longest even assuming the equipment is damaged.

c. The assessment of the impact during re-criticality was performed conservatively to evaluate the radiation dose at the boundary of the site assuming that the critical state with the output level, equivalent to 100 times the criticality criterion for Xe-135 concentration continues for one day. It indicated that the evaluated exposure dose at the site boundary was 2.4 x 10-2 mSv and it does not have significant impact.

(2) Cooling

a. TEPCO is carrying out the maintenance and management of the equipment for cooling fuel debris as well as continuous monitoring of the parameters including the temperatures of the reactors. The buffer tanks which are the water source of the circulating injection cooling facility that cools down the fuel debris was replaced with condensate storage tank (hereinafter referred to as "CST") on July 2013 and the operation of the CST reactor coolant injection system has been started. Thereby, improvement of seismic resistance and increase in capacity are being achieved for the tank, reducing the risk of losing water injection function due to reduction of a reactor water discharge line length. In addition, the desalination system (RO), which is one of the circulating injection cooling facilities, was installed and the construction for smaller circulation loop in the T/B of Unit 4 is being conducted. This construction will reduce the circulation loop (outdoor transfer pipes) from approx. 3 km to approx. 0.8km (approx. 2.1km if transfer line for stagnant water included).

b. As described above, the parameters including the temperatures of the reactors indicated stable value which was lower than that immediately after the accident. Also the temperatures in the RPV and PCV have been declined due to the continuous cooling and decrease in decay heat, and it can be estimated to be in a stable cold shutdown state.

c. In the implementation plan, the reactor core re-damage frequency is approx. 5.9 x 10-5 / year according to the risk assessment of the reactor coolant injection system by probabilistic risk assessment. It can be confirmed that the risk is reduced compared to the reactor core re-damage frequency of approx. 2.2 x 10-4 / year evaluated in the "Report (Part 1) on facility operation plan (Revision 2) (December 2011). In addition, according to the evaluation of the reactor coolant injection system in the event of abnormality, even in the case where the amount of radioactive materials of three plants are assumed to be released due to an event equivalent to a severe accident greatly exceeding an expected level (water injection shutdown for 12 hours), the effective doses are approx. 6.3 x 10-5 mSv / year at the site boundary, approx. 1.1 x 10-5 mSv / year at a 5 km point from the Specified Nuclear Facility, and approx. 3.6 x 10-6 mSv / year at a 10 km point from the Specified Nuclear Facility and therefore, it is considered that there will not be a significant risk of exposure to the general public in the vicinity.

(3) Containment

a. Prevention of leakage of radioactive materials from the PCV gas phase part

The radioactive materials released to the environment are reduced by extracting and filtering the gas in the PCV of Units 1-3 using the PCV gas control systems and by monitoring the concentration and amount of radioactive materials using radiation monitoring and control equipment. Also, since the pressures in the PCV of Units 1-3 is slightly positive, no serious damage is considered to have been caused in the gas phase of the PCV.

b. Prevention of a leakage of the contaminated water (liquid phase) from the R/B

The contaminated water leaking from the PCV of each Unit is accumulated such as in the R/B. The water level gauges are installed to monitor the conditions of contaminated water accumulated in the building and other facilities so that the contaminated water does not leak and the level of the accumulated water is controlled so as to be kept lower than the underground water level. In addition, the underground water level is being checked by the water level gauges installed in appropriate sub-drains in the vicinity of the building.

c. Hydrogen explosion prevention

i) The nitrogen filling to RPV/PCV is continued. The amount of injecting nitrogen gas is controlled when injecting nitrogen so that the hydrogen concentration in the PCV of each Unit does not exceed the burning limit concentration (4%) and the hydrogen concentration are monitored.

ii) In addition to these efforts, for Unit 1 where intermittent increase in hydrogen concentration in the PCV was observed, hydrogen released from the water remains in the suppression chamber (hereinafter referred to as "S/C") to the upper part of the S/C was replaced with nitrogen, and the stable condition has been achieved. Because a small amount of hydrogen continues to be released from the accumulated water in the S/C, nitrogen gas injection is carried out in order to maintain the stable condition, reducing the risks due to hydrogen. For Unit 2 where increase in the hydrogen concentration in the PCV due to by pressure fluctuation was observed, nitrogen gas injection to the S/C was carried out and replacement by nitrogen gas was completed. The changes made in the parameters are being checked. For Unit 3, because the increase in hydrogen concentration has not been observed and it is considered that the condition of the closed space in the S/C is stable, the change of the parameters is being checked.

iii) Hydrogen concentration in the PCV indicates that a certain value as previously described and its concentration is being controlled sufficient low to the inflammability limiting concentration (4%).

(4) Maintaining and management of stable condition

With respect to the fuel debris, the circulation cooling system has been installed, the measures are also taken to improve reliability, such as installation of multiple of equipment and the stable cooling condition is being maintained. It is important to continuously maintain and manage the stable condition from the safety perspective.

It should be noted that maintaining the stable condition until the commencement of fuel debris retrieval affects ensuring safety during the fuel debris retrieval and has continuity.

Therefore, maintaining and improving the functions to control and manage the radioactive material release, to cool a reactor, to prevent criticality and to prevent hydrogen explosion, the cooling of fuel debris and the concentration and amount of radioactive materials are being monitored. In addition, it is important for the monitoring to create databases of plant data so as to effectively manage the plant information, such as temperatures of RPV/PCV and injection flow rate of cooling water.

4.2.3 Current survey status

The immediate observation of the distributions of the fuel debris in the PCV and FPs is difficult to be realized due to high radiation environment inside the PVC of each Unit. For this reason, distribution and properties of the fuel debris and FP distribution are obtained through the comprehensive analysis and evaluation based on the all pieces of information available. The information collection and evaluation method for comprehensive analysis and evaluation and the status of comprehensive analysis and evaluation conducted to date are described below.

4.2.3.1 Current status and evaluation of investigation of conditions inside the actual reactor

As investigation of conditions inside the actual reactor, inspections for specific locations, such as the inside of the PCV, RPV, S/C and torus room, and measurements of the fuel debris distribution using muon detection system are to be conducted. Muon detection technology is a measurement technology utilizing the characteristics of cosmic ray muon that scatters exclusively in high density materials and travels in a straight line if there is none. It is categorized to transmission method, scattering method and transmission method using nuclear emulsion plate depending on the detection method. Also, PCV internal survey can be categorized to the observation using a fiber scope camera, sampling and analysis of stagnant water, CCD camera, dosimeter and unmanned robot equipped with a thermometer.

The following is the status of the visual inspection using remote devices.

4.2.3.1.1 PCV/RPV internal survey

The internal PCV/RPV survey is an effective method to study the fuel debris retrieval policy by getting the information on the current status of the plant including the photographs, damage state of the equipment, radiation dose and temperature. This section describes the status of the study on the PCV internal survey for Units 1-3 performed in FY 2015 and study status of RPV internal survey. Also, the status of the PCV internal survey performed to date is shown in Appendix 4.3.

(1) Unit 1 PCV internal survey

a. Purpose: Collection of information of the “Grating on the 1st floor inside the PCV.”

b. Method: Inspection device was inserted from the PCV penetration (X-100B penetration) and inspection of the outside of the pedestal (B1 inspection) was conducted using shape-changing robot on April 2015.

c. Information obtained:

i) No large scale damage on the existing facilities (e.g. PLR pump, wall inside the PCV, HVH) was observed. (No fuel debris was found.)

ii) Dose rate was approx. 10 Sv/h.

iii) PLR piping shielding units were confirmed fallen.

iv) The access route to the bottom of the D/W was confirmed but the deposits are scattered over a wide range.

d. Considerations: It can be estimated that the temperature at the periphery of the grating on the 1st floor might exceeded 328 deg. C, which is the melting point of lead since PLR piping shielding units (lead wool mattress) have fallen.

e. Issues: Back and forth motion should be used while checking the crawler portions. During the counterclockwise inspections, inspection crawler robot was stuck in the gaps between grating bars in the area between PLR pump and air-conditioning unit. Also, when installing thermometer after the B1 inspection, low visibility was occurred due to the sediments stirred up in the stagnant water. Therefore, the internal survey of the outside of the pedestal in the PCV (B2 inspection) was postponed to FY2016.

(2) Unit 2 PCV internal survey

a. Purpose: To verify fallen objects on the platform, damage states, and access route to the periphery of the bottom of the PCV using the internal survey robot.

b. Method: Inspection using internal survey robot inserted from X-6 penetration

c. Information obtained:

i) Although internal survey for the pedestal inside the PCV (A2 inspection) was planned, eluted materials were confirmed near the CRD hatch (X-6 penetration) and peripheral dose rate exceeded the assumption significantly.

d. Issues: Timing of the inspection was postponed to FY2016 since the measures are required to reduce the radiation dose around the X-6. The future scope of the PCV repair including some peripheral areas will be required since low temperature history for the X-6 penetration during the progress of the event is assumed, instead of leaching from X-6. Also, multiple methods have been applied to provide decontamination to the areas close to the X-6 penetration but it takes time more than expected.

(3) Unit 3 PCV internal survey

a. Purpose: Information collection to contribute to the verification of cooling state in the PCV and studies on the future investigation method.

b. Method: The inspection device (camera, thermometer and dosimeter) was inserted via PCV penetration (X-53 penetration) in October 2015. The dose rate measurement, PCV internal survey using CCD camera, and stagnant water sampling were performed.

c. Information obtained:

i) Sediments were observed on the CRD rail and gratings on the 1st floor. (transparency under the water inside the PCV was fine)

ii) Water level inside the PCV was OP: approx. 11,800mm. Almost consistent with the estimated value.

iii) The maximum radiation dose detected in the gas phase inside the PCV was approximately 1Sv/h.

d. Consideration: The radiation dose inside PCV is the lowest among Units 1-3. This is considered to be because of shielding due to high stagnant water level.

e. Issues: Water level coordination or waterproof equipment will be required for PCV internal survey since stagnant water level is high.

(4) RPV internal survey

It is very effective to confirm the conditions of the fuel debris inside the RPV, structures and environment directly before the commencement of the retrieval work in order to carry out the retrieval work in a reasonable manner. As a result of the studies on the RPV internal survey method, the investigation accessing from the piping connected to the nozzle located on lateral side to the inside of the RPV, it was found difficult to reach the appropriate locations and conduct inspections from technical perspective. Accordingly, internal survey method which establishes holes on the shield plug and upper part of the PCV from the operating floor to access the inside the RPV was selected as a development object. The development to confirm the feasibility is currently underway. With regards to creating an opening in the upper part of the PCV as major technical issues, the possibility of sealing technology to control radioactive materials released from the inside of the PCV is confirmed through the element test. Also, the element tests were conducted for the technology to confirm feasibility of creating an access hole through the complicated internal structures from reactor core region from the operating floor level. In addition, the concepts of the system required for the RPV internal survey on the site was studied and it indicated the necessity of considerable preparation for the inspection-related systems including the measures to control the release of radioactive materials.

The requirements on the entire system during the boring and inspection are to be summarized as future issues. The detailed studies are to be conducted for technologies for the internal survey to be applied as well as the verification of the possibility. On the basis of the site conditions, the detailed studies are conducted for the improvement of reasonable plan for the RPV internal survey and timing of implementation for each Unit. Also, since the scales of the system may become larger, survey items and its degree of importance in the survey needs are to be studied combined with the technical FS. The evaluation will need to be performed from the perspective of cost effectiveness in consideration of the risk involved in the survey. It is important to conduct these evaluations systematically while making decision in appropriate timing.

Also, before the commencement of the development, it is desirable to perform the researches on the details of other technical development and technical information in Japan and abroad with incorporating the reasonable method flexibly.

4.2.3.1.2 Muon detection

Measurement of the fuel debris distribution using muon detection technology is as follows:

(1) Unit 1

The fuel debris distribution measurements were performed by muon detection of transmission method twice from February to May and from May to September, 2015. The results of these measurements indicated that in the original reactor core region indicated that there was neither water nor fuels larger than 1 m, which can be identified by muon detection using transmission method. Detailed measured results are shown in Appendix 4.4.

(2) Units 2 and 3

The distribution measurement of the fuel debris was performed by muon detection of transmission method using nuclear emulsion plate for Unit 2 and it implied there are no high density materials (fuels) in the reactor core regions. The measured results are shown in Appendix 4.4.

Also, the measurement for the core region and RPV lower plenum by the transmission method has been started from May 2016. The evaluation will be performed after the data measurement for more than three months.

The measurement plan is also required for Unit 3.

4.2.3.2 Estimation by severe accident analysis codes

The amount, locations and composition of fuel debris and FP distribution are to be estimated using severe accident analysis codes. Developing a model specific to each analysis code is to be based on the data obtained from the zircaloy oxidation tests and melting tests of uranium oxide in the past. The result of analysis is to be obtained according to the progress of the accident and scenario, such as the amount of injected water and opening and closing of SR valve.

The severe accident progression analysis is highly depending on the computational model employed and estimation scenario and calculated results contain the uncertainties. However, quantitative information, such as the amount and composition of fuel debris and FP distribution at several locations inside the reactors will be able to be obtained and it will be effective method to understand a whole situation of the severe accident. Also, estimating the temperature history inside the reactors during the progression of the severe accident through the severe accident progression analysis, the states of the major internal structures and equipment is to be presumed by the calculation results.

Furthermore, OECD/NEA BSAF (Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station) project has been carried out as an international joint research. In this project, the estimation of internal PCV conditions is being performed through the severe accident progression analysis by 13 institutions from Japan and abroad.

The estimation by analysis situation performed to date is described below.

(1) Analysis results using severe accident analysis codes

Using the MAAP and SAMPSON codes, which are the severe accident analysis codes, the amount and locations of the fuel debris and FP distribution are estimated. Addition and improvement of physical phenomena model required to evaluate the fuel debris and FPs at the Fukushima Daiichi has been performed for both codes. These improvements were completed in FY2015. The amount and locations of fuel debris and distributions of FP were analyzed using the improved version of the MAAP and SAMPSON codes. The overview of the MAAP and SAMPSON codes and improvement for the severe accident analysis codes are shown in the Appendices 4.5 and 4.6.

Also, in FY2015, the analyses focusing on the event specific to each Unit were conducted. The analysis addresses the clarification of the mechanism of the plant behavior using sensitivity analysis, and reduction of the uncertainties in the analysis. For example, since pressure increase behavior (pressure spikes) was observed three times in Unit 2 after the depressurization of RPV, this pressure behavior was reproduced by the steam and hydrogen and water vapor generated by the reaction of the fuel debris and injected water using the MAAP code. Samples of sensitivity analysis using MAAP code is shown in Appendix 4.7.

The MCCI (Molten Core Concrete Interaction) evaluation implies that most of the fuel debris was highly likely to have fallen to the pedestal in Unit 1 due to the damage to the RPV. It is important to evaluate the erosion of the concrete and amount of MCCI. Since the shape of the pedestal of the Fukushima Daiichi NPS is complicated including sump pits, relocation and diffusion model for the concrete are added to the MCCI evaluation module of the SAMPSON code and the extent of the scattering and erosion behavior of the fuel debris in Unit 1 were evaluated. The evaluation results suggest that the fuel debris are scattered to a fairly wide range of the D/W floor of Unit 1. The evaluation results of MCCI in Unit 1 are shown in Appendix 4.8.

The results of the analysis of the amount and locations of fuel debris and FP distribution using the MAAP and SAMPSON codes are described below. Also, the estimation results of the conditions of the reactor internals and equipment based on the temperature estimated by the severe accident progression analysis are as follows:

a. Amount and locations of the fuel debris

The results of analysis of the amount and locations of the fuel debris are shown in Table 4.2-4. Since severe accident analysis codes have characteristics of the models used for each code and uncertainties in the input scenario, it is required to take into account the uncertainties contained for use the results. Issues to be noted in the comparison among the result of severe accident analysis codes and Units are show in Table 4.2-4.

Table 4.2-4 Analysis results using the severe accident analysis code (Unit: ton)

Locations	Unit 1		Unit 2		Unit 3	
	MAAP	SAMPSON	MAAP	SAMPSON	MAAP	SAMPSON
Core region	0	0	0	13	0	29
Bottom of the RPV	15	10	25	58	25	79
Inside the pedestal	109(78)	79(130)	92(37)	76(14)	103(51)	53(20)
Outside the pedestal	33(52)	52(0)	102(4)	5(0)	96(6)	0(0)
Total amount (concrete included)	287	271	260	166	281	181

Note: The weight inside and outside the pedestal is the weight of fuels/structural materials (excluding the weight of concrete). The weight of concrete is indicated in ( ).

・In Unit 1, the RPV had been damaged before water injection was started. Analysis results by both MAAP and SAMPSON codes showed that most of the debris had fallen on the pedestal.

・In Units 2 and 3, analysis using SAMPSON code indicated that the particles of debris were cooled by water injection and accumulated inside the RPV. At core region and bottom of the RPV, amount of debris estimated by the SAMPSON code are larger than that by the MAAP code.

・In SAMPSON code, the particles of debris were modeled as one lump for and it will be cooled if there is water. In the actual phenomenon, it can be estimated that cooling water is maldistributed and granular debris which does not contact with cooling water will be re-melted and dropped on the lower plenum.

・Results of analysis using the SAMPSON codes showed the fuels of a part of the outer periphery of the core were remained in stub-shape, since water stayed outside the reactor core shroud and contributed to the cooling. The fuel will be melted and dropped if there is no water.

Reference: IRID Completion report for "Improvement of recognition regarding the internal PCV condition using severe accident progression analysis and actual plant data"

b. FP distribution

The FP distribution inside the RPV, PCV and R/B were analyzed by the MAAP and SAMPSON codes. The results indicated a large difference which was depended on properties of FP nuclides using both codes. The results of analysis of the distributions of Cs and Sr, which are the representative FP nuclides, are described in Appendix A4.9-1. A large difference (uncertainties) between those codes is caused by the differences in FP evaluation models and chemical form of the FP nuclides which were considered in the evaluation model.

c. Estimation for the status of major structures and equipment

Although it is necessary to know the current conditions of equipment inside the reactor for the fuel debris retrieval, there is no measured value of the environment (temperature) that the equipment have experienced during the severe accident. Therefore, the conditions of the equipment inside the reactor were estimated based on the temperature evaluation results from the severe accident progression analysis (MAAP and SAMPSON codes). This estimation was performed by referencing not only the results of analysis but also the on-site situation. The high-temperature deformation, creep rupture and corrosion degradation were considered as degradation events induced in the structures and equipment in the scope of the evaluation. The Evaluation criteria for each degradation event are shown in Appendix 4. 10.

The evaluation results indicated that the creep deformation may be induced for the steam dryer, steam separator assemblies, upper grid plates and core support plates in all Units. For use the evaluation results, the uncertainties are required to be considered in the severe accident progression analysis (MAAP and SAMPSON codes) performed for this time. All estimations results including those of other reactor internals are shown in Appendix 4.11.

Since the evaluation results of the temperature refers to those of FY2014, it is necessary to confirm the effects from the latest analysis results in the future.

Also, considering the thermal data obtained from the analysis after the severe accident base on the corrosion rate for 40 years after the severe accident, the seismic stress evaluation was performed in the "Development of technology for RPV/PCV integrity evaluation.” The results of the analysis indicated that the induced stress at the RPV, PCV and pedestal of Unit 2 fell below the evaluation criteria.

(2) Results of analysis by OECD/NEA BSAF

In the Phase-1 of OECD/NEA BSAF project, the severe accident progression analysis of Units 1-3 for six days after the earthquake was performed by 13 institutions in Japan and abroad. The result of analysis is shown in Table 4.2-5.

The result of analysis of Units 2 and 3 was categorized into two cases, which are that the fuel debris remains in the RPV, and falls down to the PCV. The result of Unit 2 can be considered depending on the modeling of the fuel debris relocation from the reactor core to the lower plenum and the assumption amount of the water injection by the fire engine which has large uncertainties. With regard to Unit 3, the results were affected by the difference in the assumption of HPCI water injection behavior (amount of injected water in reducing the RPV pressure), that is, the maximum quantity and cycle of the steam flow that drives HPCI were different significantly among the institutions that performed the analysis.

The Phase-2 of the OECD/NEA BSAF project is being carried out following the Phase-1. In the Phase-2, setting the implementation period at three years, which is from April 2015 to March 2018, the severe accident progression analysis of three weeks after the earthquake is being performed by 22 institutions from 11 countries to advance the improvement of the analyses. Also, the findings of the severe accident research analysis are shared through the workshop regarding the state of the FP adhered to the reactor internals and features of MCCI. The progress (PRG, Program Review Group) meeting and workshop are being held about twice a year. Interim reports and final reports are planned to be established at the end of FY2016 and in May 2018 respectively.

Table 4.2-5 Evaluation results of fuel debris distribution in BSAF Phase-1 (Unit: ton)

Area (Unit and No. of Institutions)	Unit 1	Unit 2		Unit 3	
	9	6	3	4	5
Core region	0- (3)	0-14	0-32	0-21	0-36
Bottom of the RPV	0-8	0-91	0	8-81	0
PCV	105-164	0	147-240	0	140-268

(Provided by IAE)

Integrating the knowledge around the world, OECD/NEA SAREF (Safety Research Opportunities Post-Fukushima) also studies the project regarding the decommissioning and safety assessment.

4.2.3.3 Estimation based on knowledge and experiments

Estimation from the knowledge and experiments can be categorized into three cases, which are the estimation based on the past accidents and researches, engineering estimation on plant data and experiments conducted using simulated fuel debris.

The examples of core meltdown accident include the accident at TMI-2 and Chernobyl Nuclear Power Plant Unit 4. The findings to be obtained will be utilized in the estimation of the behavior inside the RPV and estimation by the MCCI. The researches performed in the past include the FP tests performed at the Phebus reactor in France and MCCI test at the U.S. Argonne National Laboratory. Those results were reflected to the models of the severe accident analysis code.

The engineering estimation based on the plant data is considered in line with the plant parameter, such as the fuel debris distribution evaluation by the heat balance method.

The simulated debris for testing was fabricated considering the severe accident phenomenon progress at the Fukushima Daiichi NPS in reference to the TMI-2. Using the simulated debris, the data on the mechanical, chemical and physical properties are collected.
In this section, the status of estimations for fuel debris distribution based on the plant data and for the fuel debris properties based on the experiment using simulated debris is summarized.

4.2.3.3.1 Engineering estimation based on the plant data

(1) Estimation by heat balance method

The ratio of the fuel debris in the RPV and PCV was estimated based on the heat balance assuming that the temperature of the cooling water injected to the RPV is elevated to temperature of the accumulated water by the heat source (fuel debris) inside the RPV/PCV. That is the balance of the heat input (heat capacity of injected water and decay heat) and heat emittance (heat emittance from the PCV wall surface to the building or to the atmosphere and temperature elevation of cooling water by the fuel debris). The estimation results using heat balance method and the overview and estimation results of heat balance method are shown in Table 4.2-6 and Appendix 4.12 respectively.

Table 4.2-6 Results of estimation of fuel debris distribution by the heat balance method

Unit	Estimation results
Unit 1	Heat source equivalent to the decay heat of approx. 45% may exist in the PCV. (Evaluated assuming no heat source exist in the RPV based on the results of the analysis using MAAP code, which is (no decay heat in the RPV))
Unit 2	30-60% heat source (fuel debris) may exist in the RPV.
Unit 3	20-70% heat source (fuel debris) may exist in the RPV. The amount of debris exist in the RPV as a heat source is, however, likely to be fewer since the temperature of RPV accumulated water does not follow the temperature of injected water.

Note: This estimation includes the uncertainties of the decay heat of the fuel debris that fell to the bottom of the PCV (according to the evaluation performed by the JAEA, it will be reduced to about 60% if all of the highly volatile nuclide are released), possibility of heat conductance from the fuel debris to the floor concrete, and uncertainties in the evaluation of heat transfer rate in the heat emittance from the PCV side to the outside. (Provided by IRID)

(2) Estimation based on the trend of plant parameter

The heat source (fuel debris) in the RPV was estimated based on the trend of the temperature around the RPV in the post-accident condition, water temperature of S/C, amount of injected water via feedwater (FDW) system and reactor core spray (CS) system. The FDW system is a system that introduces the water cooled by condenser into the RPV during the normal operation of BWR. If the integrity of the bottom of the RPV is maintained, the cooling water will accumulated inside the RPV and water level will be raised. However, since increase in the water level was not observed, it is assumed that the bottom of the RPV has been damaged and the cooling water was dropping from the damaged portion to the inside of the pedestal. This indicates that the water injection using the FDW system was not able to cool down the reactor core even it can cool down the bottom part of the RPV in the post-accident conditions. On the other hand, the CS system is a core spray system for the coolant loss accident and is installed along the walls of the core shroud immediately above the reactor core. In the CS system water injection, the cooling water is flowing through the space from the reactor core to the bottom of the RPV and the scape can be cooled down. Table4.2-7 shows the estimation results of the fuel debris locations for each Unit. Also, Appendix 4.13 shows the estimation method described above and its results.

Table 4.2-7 Results of estimation of fuel debris locations based on the trend of plant parameter

Unit	Estimation results
Unit 1	Few heat sources are highly likely to exist in the RPV. The heat source is likely to exist on the water injection paths of the FDW system. The removed heat is being transferred to the S/V in response to the water injection.
Unit 2	A certain amount of heat source is likely to exist in the RPV. The distance to the heat source from the RPV lower plenum is shorter than that of feedwater nozzle N4B. The removed heat is estimated to transferred to the S/C.
Unit 3	A certain amount of heat source is likely to exist in the RPV. It is difficult to estimate since changes are less likely to be recognized by the parameters in the PCV due to a large amount of accumulated water.

4.2.3.3.2 Estimation of fuel debris properties based simulated fuel debris experiments

The studies, such as on the retrieval, collection and storage of fuel debris will require the data on the features of the fuel debris located inside the reactor. Therefore, in addition to the knowledge obtained to date (e.g. TMI-2 accident and severe accident research), analyses and tests using simulated debris are performed and fuel debris properties are estimated based on these data.

Also, the developments are being carried out for the technologies required to analyze and measure the fuel debris which will be retrieved from the reactors.

(1) Understanding of features using simulated debris

In the characterization using simulated debris, evaluation of the features of metal debris, features of products caused by the reaction specific to the severe accident at the Fukushima Daiichi NPS and features related to the heterogeneity.

1) Characterization of metal debris

Mechanical of metal oxides properties, such as of Zr (O), which is zirconium that oxygen is dissolved in, estimated to be contained in the metallic layer of the fuel debris, has been measured.

2) Characterization of the products caused by the reaction specific to the accident at the Fukushima Daiichi NPS.

The following data is being collected: mechanical properties and the formation phase of simulated fuel debris forms a solid solution of fuel, oxides of stainless steels, FP elements, and sea salt compositions.

3) Characterization for the heterogeneity

Since the evaluation of the heterogeneous properties as a large lump was difficult to be performed in Japan, the evaluation tests for the mechanical properties are being conducted for the large scale products of MCCI at CEA, France.

Also, at the National Nuclear Center, Kazakhstan, the data of the fuel debris in a powder form, which is the molten material that solidified by water cooling and physical property such as the particle size, density, phase of the condensed solidified materials has been obtained in the characterization test for the large scale molten and solidified metal ceramics.

In addition to the results of characterization using simulated debris described above, the fuel debris properties are estimated based on the knowledge obtained to date (e.g. experience from TMI-2 accident, severe accident research) and summarized in the list. In particular, the macro properties, e.g. compressive strength and uranium content, and micro properties e.g. the mechanical properties and thermal properties such as thermal conductivity are summarized based on the literature survey and experimental results of each location of the fuel debris estimated by the severe accident progression analysis.

Also, the estimation of the properties, such as external appearance and shape of the fuel debris are being carried out temporarily for the fuel debris which has been estimated by the severe accident progression analysis based on the TMI-2 accident and tests, in the RPV/PCV. The estimation results are shown in Appendix. 4.14.

Based on the needs of the information on the fuel debris properties which are required for the studies on the fuel debris retrieval, the list of the above properties will be updated in collaboration with the comprehensive analysis and evaluation of internal PCV condition.

(2) Analysis of actual debris properties

The properties of actual debris (e.g. mechanical properties and chemical composition) are required information for safe fuel debris retrieval. Therefore, the needs of analysis from the related projects are summarized as an analysis plan so as to collect the required information on the fuel debris properties reasonably and reliably.

Also, since the fuel debris formed in the state which has never been experienced before will be handled, the analysis flow is being studied comprehensively to identify the required development items, and the analytical techniques including the dissolution method of actual fuel debris and analysis method for chemical form are currently developed. In this regard, the studies are conducted for the cask required for transportation specimen with high radioactivity. These studies are to be conducted steadily according to the processes for fuel removal from SFPs and establishment of analysis and research facilities.

As a facility for analysis of the actual debris, testing facilities where specimen with high radioactivity can be handled will be required. Although the existing facility of JAEA located in Ibaraki district can be utilized at this moment, the facilities are not considered to meet a broad range of demands. Radioactive Material Analysis and Research Facility No.2 is, therefore, planned to be established in Okuma machi.

The studies on the plan for the analysis of actual fuel debris properties are to be revised including the priorities, frequency and timing of the analysis. Also it is important to reflect the studies to the specifications and operation method of the Radioactive Material Analysis and Research Facility No.2 appropriately. Also, the analysis needs in the stages of fuel debris retrieval, stable fuel debris storage, waste conditioning processing and disposal, and in addition, analysis needs over the mid and long term, such as the safe research needs of damaged reactors are to be satisfied in the establishment of analysis plan. With regard to the facility for analysis, it is appropriate to study the use of the existing facilities of JAEA in Ibaraki district, as needed.

To analyze and evaluate the obtained data, it is also require establishing a structure that incorporates the opinions from the experts in Japan and abroad.

The storage method, canisters and transportation casks of the specimen for analysis are required to be studied separately from the collection and storage for the full-scale retrieval. Also, the technical issues related to the exportation are required to be studied as needed.

Also, in parallel, the issues related to the analytical techniques using a small quantity of sample which can be obtained by the PCV internal survey are required to be carried out.

4.2.4 Comprehensive evaluation for the internal PCV condition and future actions

The comprehensive analysis and evaluation is performed for the internal PCV condition based on the information obtained by the visual inspection with remote devices, estimation by severe accident progression analysis, estimation based on the knowledge, experiments and internal PCV conditions, such as the amount and locations of fuel debris in each Unit assuming the uncertainties are being estimated. Current issues and future actions are described based on the above results.

4.2.4.1 Summary of comprehensive analysis and evaluation of internal PCV condition

Comprehensive analysis and evaluation of fuel debris distribution are being conducted based on the information on the fuel debris distribution shown in 4.2.3.1-4.2.3.3. The evaluation results and the results of plant investigation, such as radiation dose rate inside the PCV are shown in Table 4.2-8. The information used for the comprehensive analysis and evaluation of fuel debris distribution involves uncertainties. For example, the severe accident progression analysis, it involves uncertainties since it lacks the data measured at the time of the accident and depends on the computational model and assumption scenario adopted for each accident progression analysis code. Thus, the value of the results obtained by the comprehensive analysis and evaluation (evaluated value) should have a margin since uncertainties are inherent in any information. Also, the fixed value is indicated as a probable estimate at this point, since it is used as a representative value in the various evaluations and studies relating to the future fuel debris retrieval method. Appendix 4.15 shows the results of the analysis using severe accident progression analysis code used for the comprehensive analysis and evaluation above, estimation results based on the trend obtained by heat balance method and plant parameter, and information based on the results of muon detection, PCV internal survey and BSAF Phase-1.

(1) Unit 1

According to the record of D/W pressure, some evaluations indicate that the water injection was not achieved until 11 days after the station black out. Based on this evaluation, the decay heat and heat caused by the Zr-vapor reaction were not removed and it was exposed to the most severe condition over a long period of time and the RPV was damaged significantly. This damage has been estimated to cause the most of the fuel debris to leak outside the RPV. This estimation is consistent with the results of the severe accident progression analysis, estimation from the plant parameter, muon detection and results obtained in BSAF Phase-1.

Results of estimation of fuel debris locations

- Most of the fuel debris had fallen to the lower plenum. Little fuel remained in the core region (little criticality risk due to the stub-shaped fuels)
- Most of fuel debris that had fallen to the lower plenum fell to the bottom of the PCV.
- Although it transferred to the outside the RPV pedestal, there is no possibility of a large scale shell attack.

(2) Unit 2

The RCIC started to operate after the shutdown of the reactor and continue cooling the inside the reactor over three days. In specific, since decay heat immediately after the shutdown of the reactor could be removed during the time when its temperature is high, the fuel melting state is estimated to be the lowest among Units 1-3. Since there is no data, such as on the temperature during the accident and fire engine amount of injected water, the margin for the evaluated value is large.

Results of estimation of fuel debris locations

- Most of the fuel debris had fallen to the lower plenum, and very little fuel remaining in the core region. (Small risk caused by criticality due to stub-shaped fuels)
- More than half of the fuel debris had been fallen to the lower plenum fell at the bottom of the PCV. However, since there is no data on the temperature during the accident and fire engine amount of water injected, the uncertainties are the largest among Units 1-3.
- Although they may be moved to the outside of the RPV pedestal, its extent is smaller than that of Unit 3.

(3) Unit 3

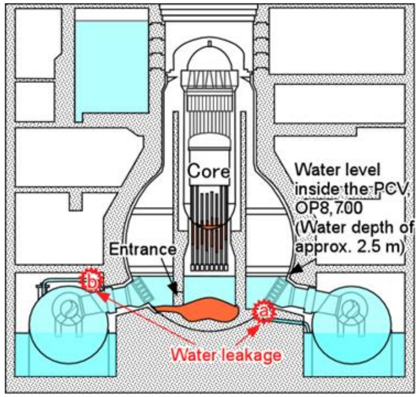
The RCIC and HPCI started to operate after the shutdown of the reactor and continued cooling about 1.5 days. As with Unit 2, although there is no data on the temperature during the accident and amount of injected water, the degree of severity of the accident is estimated to be in between Unit 1 and Unit 2 since cooling level is estimated to be in between Unit 1 and Unit 2. Since fuel debris is assumed to be located in the RPV based on the trend of plant parameter and its quantity is estimated to be equal to or fewer than that of Unit 2.

Results of estimation of fuel debris locations

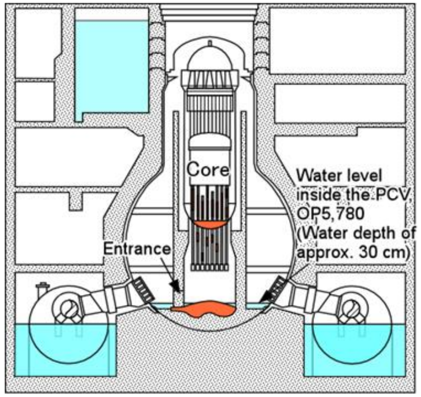
- Most of the fuel debris had fallen to the lower plenum, and very little fuel remaining in the core region. (Small risk caused by criticality due to stub-shaped fuels)
- Most of fuel debris that had fallen to the lower plenum fell to the bottom of the PCV.
- Although they may be moved to the outside of the RPV pedestal, its level is lower than that of Unit 1.
- The extent of falling down to the lower plenum by the fuel debris and transporting to the RPV pedestal are moderate among Units 1-3.

Table 4.2-8 Plant conditions of Units 1-3 (including the estimation of fuel debris distribution)

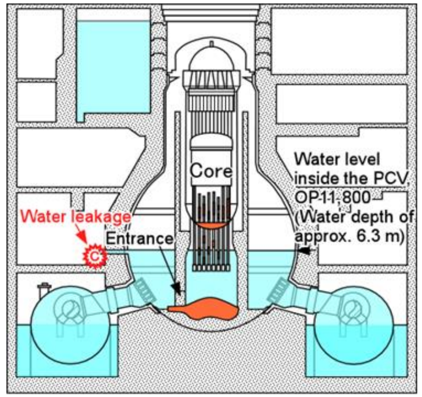
Unit 1



Unit 2



Unit 3



[Unit : ton]

		Unit 1		Unit 2		Unit 3	
	Location	Range of Estimation*1	Typical Value*2	Range of Estimation*1	Typical Value*2	Range of Estimation*1	Typical Value*2
Estimated Distribution	Core	0-3	0	0-51	0	0-31	0
	RPV Lower Head	7-20	15	25-85	42	21-79	21
	Inside RPV Pedestal	120-209	157	102-223	145	92-227	213
	Outside RPV Pedestal	70-153	107	3-142	49	0-146	130
	Total	232-357	279	189-390	237	188-394	364
Current Status	Dose Rate measured in PCV	Approx. 5-10 Sv/h (measured on Apr. 10-16, 2015, in gas phase of 0.7 m from the water level, on the grating)		Approx. 31-73 Sv/h (measured on Mar. 27, 2012, in gas phase of 3.7 to 6.7 m from the water level, around X-53 penetration)		Approx. 0.75-1 Sv/h (measured on Oct. 20, 2015, in gas phase of 0.55 m from the water level, around X-53 penetration)	
	Observed water leakage points	Water flows were detected from one sand cushion drain pipe (ⓐ) and one expansion joint cover (ⓑ) of the S/C vacuum break line.		It is expected water flow from somewhere below water level inside the torus room, since there is no trace of water leakage in gas phase.		Water flow was detected from the expansion joint (ⓒ) of the MS Line D in the MSIV room.	
	PCV internal survey	-No large scale damage on the existing facilities (e.g. PLR pump, wall inside the PCV and HVH). -Deposits are scattered over a wide range. -PLR piping shielding units fallen.		-The structures of the RPV bottom was confirmed by the photographs taken from RPV pedestal opening. Breakage of the RPV bottom is unlikely to be a large.		-PCV internal survey using inspection device inserted from PCV penetration indicated that no damage of the walls and PCV was found within the scope of the survey.	

*1: Range of results based on Severe Accident codes

*2: The most reliable value by a plurality of analysis results in the current

*3: Weight of fuel debris, indicating the weight of the fuel + melted and solidified structural material (including a concrete component).

Fuel debris distribution: Based on the document provided by IRID
Plant investigation status: Based on the document released by TEPCO

4.2.4.2 Issues on the internal PCV condition analysis and future actions

Since the estimation results of the internal PCV conditions obtained to date involves uncertainties, it is required to implement the PCV internal survey and muon detection which can obtain the actual measurement data of the inside of the reactor should be performed. Furthermore, it is important to reduce the uncertainties in the estimation results of internal PCV condition by reflecting these actual measurement data to the comprehensive analysis and evaluation of the internal PCV condition.

In addition to the results obtained from various plant data and investigation of conditions inside the actual reactor, estimating the physical phenomenon caused in the plant at the time of the accident, such as generation of the fuel debris and FPs and relocation behavior, the comprehensive analysis and evaluation of the internal PCV conditions are required to be continued. Also, clarifying the source of uncertainties of the estimation results such as the fuel debris locations and distributions, the sensitivity analysis is to be carried out using analysis code to reduce such uncertainties to improve the accuracy of the comprehensive analysis and evaluation. The major implementation items are described below. Also, the future actions for internal PCV condition analysis are shown in Figure 4.2-3.

<u>Investigation of conditions inside the actual reactor</u>

(1) The PCV internal survey (e.g. Unit 1 B2 inspection, Unit 2 A2 inspection and Unit 3 inspection using swimming robot) which is currently planned and muon detection (Unit 2) are to be reliably performed. Also, RPV internal survey is to be conducted systematically. This is very important in terms of confirming the estimation results of the fuel debris distribution of each Unit by the site data. In particular, in the evaluation of Unit 2, the confirmation should be obtained through the visual inspection with remote devices, since the changes in the amount of the fuel debris remain at the bottom of the RPV is large due to the amount of injected water at the time of accident (with large uncertainties).

(2) The comprehensive analysis and evaluation is required to be reviewed as appropriate by utilizing the latest information on the fuel debris and FP distribution which can be obtained from the PCV internal survey, RPV internal survey and muon detection.

(3) According to the PCV internal survey for Units 1 and 3, there were some deposits on the structures in the accumulated water in the PCV, and those should be considered in the future internal survey and the studies of fuel debris retrieval methods.

(4) Since decontamination work around the X-6 penetration as a preparation of internal survey for Unit 2 PCV takes significantly longer time than expected, the decontamination/radiation dose reduction work to be performed in the area closer to the PCV requires more thorough preparation and approaches.

<u>Estimation by the analysis</u>

(1) The issues are to be identified to reduce uncertainties in the analysis of accident progression scenario and state of post-accident fuel debris. As identified issues, the uncertainties in the analysis and evaluation conducted to date are to be reduced through the estimation of physical phenomena occurred in the plant at the time of accident, such as the fuel debris and generation and migration behavior of FPs,

which were identified by the findings obtained in the past cases and research, and the estimation by the sensitivity analysis and the inverse analysis that takes into the boundary conditions and the analysis model of the severe accident progression analysis code.

(2) The study on the FP distribution are required in order to perform comprehensive analysis and evaluation for the matter to be noted in the study on the fuel debris retrieval method, such as the amount and locations of FPs, and possibility of contamination of FPs, attached to the reactor internals. It should be based on the findings of the evaluation of the residual quantity inside the reactor and chemical properties, in addition to the results of on-site radiation dose measurement.

(3) Utilizing the opportunity of international joint research, such as BSAF-2 and SAREF project, it needs to collect the result of analysis and evaluation performed by organizations overseas in addition to the data and information regarding the fuel debris and FPs, and reflect them to the comprehensive analysis and evaluation.

Estimation based on knowledge and experiments

(1) The list of the fuel debris properties established in FY2015 is required to be verified and revised continuously in collaboration with the comprehensive analysis and evaluation of internal PCV condition based on the needs of information on the fuel debris retrieval.

Also, investigation of conditions inside the actual reactor and so on is required to be continued. Performing the comprehensive analysis and evaluation of internal PCV condition based on the information obtained, more probable estimation results of internal PCV conditions are required to be provided in the studies on the fuel debris retrieval method.

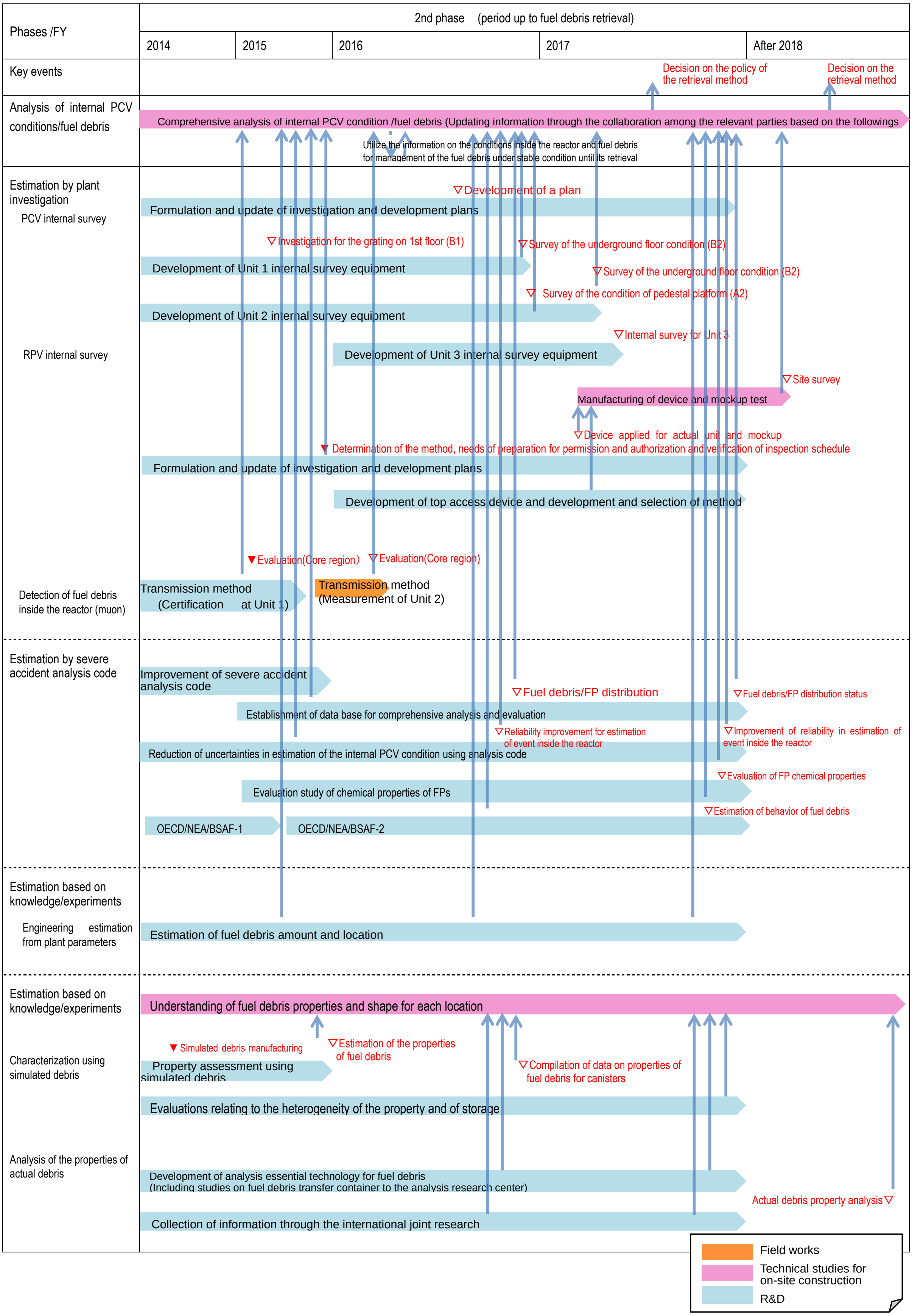


Fig.4.2-3 Future actions for the gasp of situations inside the reactor and fuel debris

4.3 FS of fuel debris retrieval method

This section describes the status of the studies on feasibility of the safe retrieval method of the fuel debris. Following the background of the selection of three methods to be currently focused on, the features of retrieval methods are described. Also action status, issues and future actions for nine technical requirements, which are the key issues required to realize the method.

4.3.1 The selection of fuel debris retrieval methods and their features

As the method to retrieve the fuel debris for the Fukushima Daiichi NPS, the method applied to the retrieval at TMI-2, which is the preceding case, has been studied since the release of original Road map. This is a method that retrieves fuel debris by filling water to the upper part of the PCV and the radiation dose is expected to be reduced by water shielding effect. On the other hand, there are many challenges involved in developing the technologies required to repair the PCV damaged by the severe accident so as to be filled with water. Hence, assuming that the challenges are involved in submerging the entire fuel debris, the studies are being conducted for methods of retrieving the fuel debris while they are partially exposed to the air and without filling the PCV with water to the top.

Unlike TMI-2 where fuel debris remained in the RPV, since the fuel debris are estimated to be scattered broadly in the PCV, it may be difficult to reach and retrieve the fuel debris from top of the RPV as it was performed in TMI-2 depending on the location of the fuel debris.

In consideration of such situations, the fuel debris retrieval methods combining the levels of water in the PCV and directions of accesses to the fuel debris are to be identified in order to realize the fuel debris retrieval corresponding to distribution of fuel debris for each Unit and differences between site conditions. Through the evaluations for the applicability based on such features, some methods to be focused on are to be selected.

(1) Study on the fuel debris retrieval method considering the PCV water level and access direction

a. PCV water level

In the studies on the retrieval method, the features of the retrieval methods vary depending on the water level of the PCV. The categories of the method according to the PCV water level during the fuel debris retrieval are defined as follows. Image of water levels are shown in Figure 4.3.1-1.

- Full submersion method: A method in which water fills to the top of the reactor well.
- The Submersion method: A method in which the water fills to a level above the highest point of the fuel debris distribution.

Note: This method is called the Submersion method assuming there is no fuel debris scattered above the reactor core region and water level is above the uppermost part of the reactor core region.

- The Partial submersion method: A method in which the water is filled to a level below the highest part of the fuel debris distribution. To retrieve fuel debris in the air water is poured on the cutting point and its periphery.

Note: This method is called the Partial submersion method assuming there is fuel debris exposed to the air if the water level is lower than the uppermost part of the reactor core region.

● Dry method: All the areas where fuel debris is scattered are exposed to the air, and neither water cooling nor water pouring is involved.

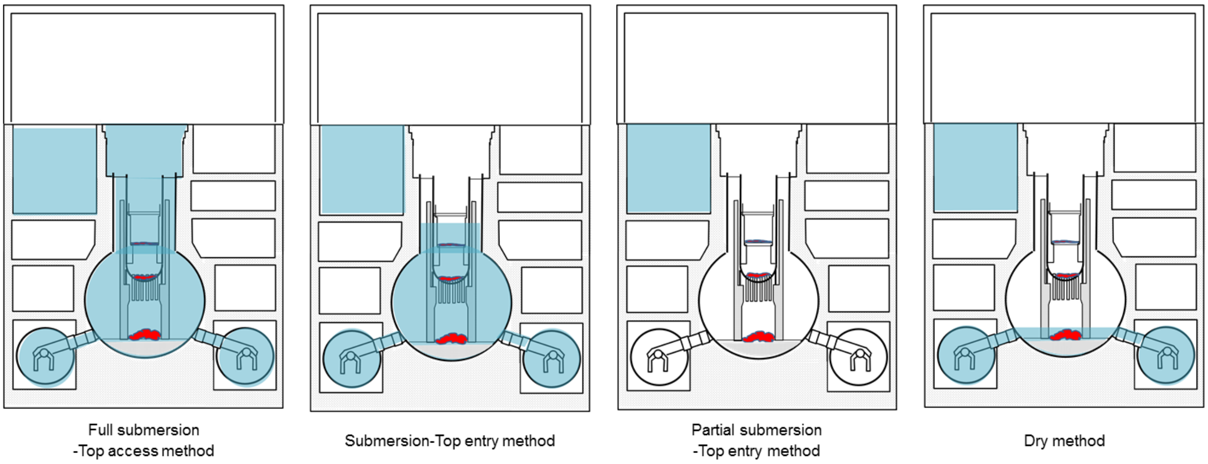


Figure 4.3.1-1 Classification of method according to the PCV water level

b. FS of access route for each direction of accesses to the fuel debris

As shown in Figure 4.3.1-2, three directions of access to the fuel debris can be considered: from the upper part of the PCV (Top access), from the side of the PCV (Side access), and from the bottom of the PCV (Bottom access). The feasibility of the access route for each direction is evaluated as follows.

i) Access from the top of the PCV (Top access)

The access route from the top of the PCV into the core has already been secured structurally for the fuel replacement work during the periodical inspection. The access into the reactor can be made by removing the well shield plug, PCV upper head, insulator for the RPV upper head, and RPV upper head. The upper grid plate immediately above the core may be made by removing the steam dryer and steam separator allowing access to the fuel debris inside the RPV. There is however a possibility that the equipment to be removed may be deformed due to high temperature during the accident and may be difficult to be removed using a normal method. In such case, it is necessary to cut them before removing. Also since the FPs such as Cs released from the fuels at the time of accident may have been absorbed by these pieces of equipment, the protective measures against extremely high radiation will be necessary.

ii) Access from the side of the PCV (Side access)

On the side of the PCV, there are equipment hatch and X-6 penetration (CRD hatch) leading into the PCV. Although the size of these openings is limited, the access route into the PCV is structurally secured. At the bottom of the D/W inside PCV, PLR pumps, valves, pipes and supports are equipped on the outside of the RPV pedestal, and CRD handling machine and operating floor (grating) are equipped inside the RPV pedestal. Since those pieces of equipment may impede the access to the fuel debris, they are required to be cut off and removed.

iii) Access from the bottom of the PCV (Bottom access)

Since there is no access route into the PCV from the bottom of the PCV available, a new access route to the bottom of D/W will be necessary.

Although it is theoretically possible to establish the underground access tunnel leading to the bottom of the D/W from the outside the R/B via underground, such route may affect the underground water management. Also, the tunnel is required to penetrate the bedrock that supports the R/B, R/B foundations, D/W bottom shell and D/W bottom foundation; therefore there is a concern of reducing the strength of R/B, D/W bottom shell and RPV pedestal foundation.

Regarding this access type, as a result of the basic study, the prospect for the technical issues on the excavation construction to establish an underground access tunnel to the basement of the R/B civil engineering work was confirmed and the results indicated that it can be technically solvable. On the other hand, there are some problems to be solved by development to perform drilling upward from the basement of the R/B to the bottom of the D/W. Furthermore, the following issues with high degree of difficulty are required to be solved to realize the fuel debris retrieval method. (Appendix 4.16: FS of the fuel debris retrieval from the bottom of the R/B)

- Unless all necessary conditions to perform Dry retrieval method are satisfied, there is contaminated water accumulated at the bottom of the D/W, which is to be controlled not to leak outside at the time of creating an opening at the bottom of the D/W in the final stage of the establishment of the access route from the bottom.
- With regard to the issues described above, the leakage of contaminated water from the bottom of the PCV can be controlled for a long period of time during the fuel debris retrieval work in the subsequent process where the access route from the bottom is used.
- The fuel debris retrieval equipment is to be installed at the bottom of the D/W through the underground access route with severe dimensional limitations while keeping water sealability. In this regard, the fuel debris retrieval equipment which can satisfy with these requirements is to be prepared.
- The fuel debris retrieval is to be performed for the all areas including the area distant from the opening to access to the bottom of the D/W.
- The retrieved fuel debris is transported to the storage facility via the access tunnel to the basement of the R/B.

Since the degree of difficulty of the issues described above is high, it is considered necessary to perform the development and verification over long term.

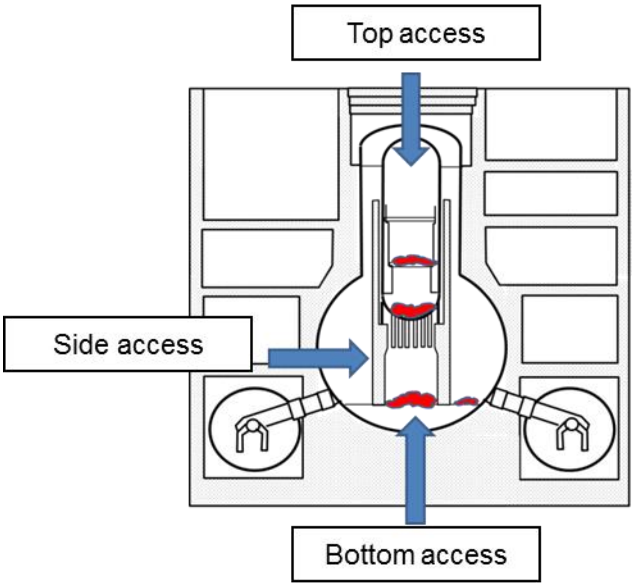


Figure 4.3.1-2 Direction of access to the fuel debris direction of access to the fuel debris

c. Assessment and narrowing of retrieval methods by considering the combination of the PCV water level and access direction

Twelve types of the combinations can be made by the directions of the access to the fuel debris and PCV water levels described in Figure 4.3.1-5.

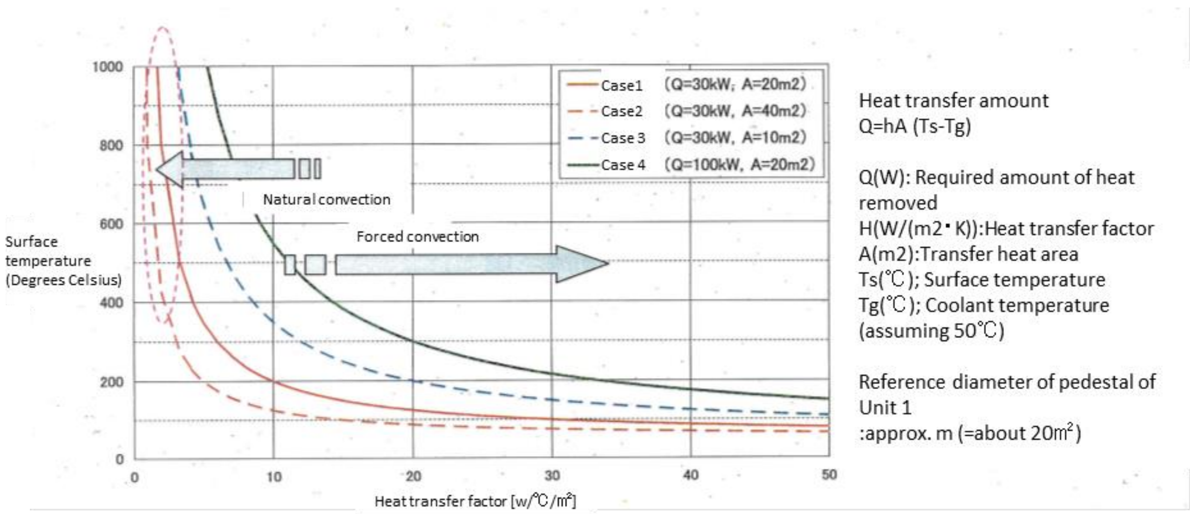
The methods to be focused on will be selected from the combinations through the studies on their applicability to the actual unit.

The fuel debris retrieval requires the development that involves extremely high degrees of difficulty. The methods which can be developed with high feasibility, which is "simple and requiring few development factors" are to be focused in the screening for the selection. This is considered to be a procedure that satisfies Five Guiding Principles.

As described in Figure 4.3.1-5, for the Full submersion method and the Submersion method, the access port will be below the PCV water level for side and bottom access. Therefore a large water tight hatch will be required to prevent water flow from the access port when bringing equipment and materials in and out, or retrieving the fuel debris. Fully remote-controlled and automatic system will also be required, and there are a number of challenges will be involved in the actual utilization of this method. Those challenges include maintenance works through the water tight hatch or dealing with construction troubles and will be caused for the Partial submersion method by side access if the access port is below the PCV water level. These options will not be focused in the studies for the practical application.

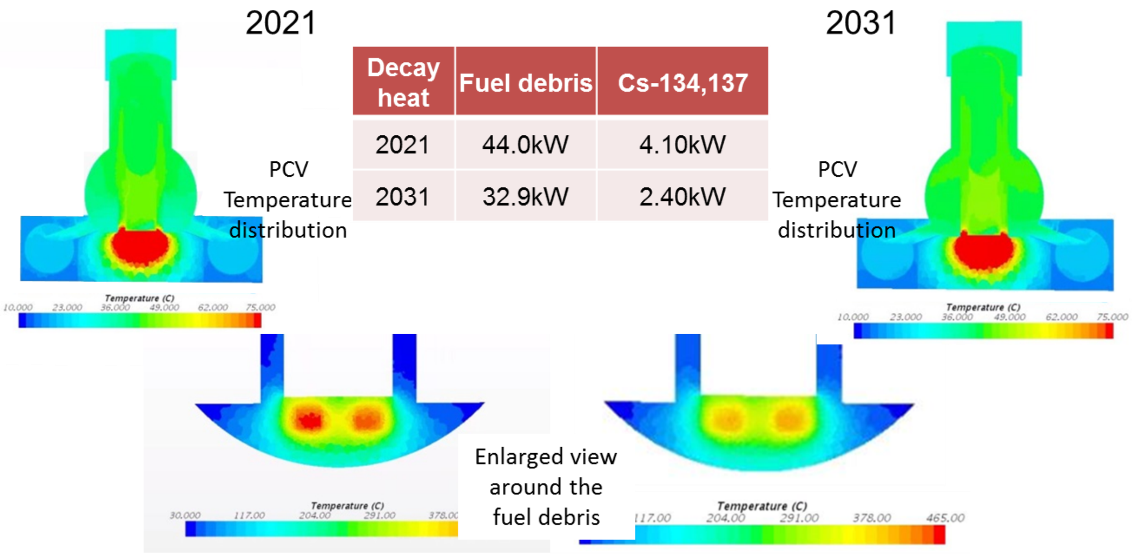
As described in (2) b., regarding bottom access retrieval method, even if a route for the bottom-access can be established, a lot of critical issues are involved in the subsequent management of contaminated water and fuel debris retrieval work (cut off of the fuel debris in a broad area and export of the fuel debris via access tunnel). Therefore, since its feasibility was evaluated as low for a short or mid-term time period to be developed and verified. This option will not be focused in the studies for the practical application.

With regard to the Dry method, surface temperature of the fuel debris at the bottom of the D/W is estimated at 400 deg. C based on the assumption as follows: all fuel debris in Unit 1 fell inside the pedestal at the bottom of the D/W and located as disk shape, amount of decay heat in the target commencement period of fuel debris retrieval (2021), and condition of air cooling by natural convection. The temperature inside the fuel debris will be higher than surface. (Figure 4.3.1-3)
A sample of the analysis results of temperature distribution of fuel debris under air cooling condition by making 3D models of Unit 1 based on the assumptions above is shown in Figure 4.3.1-4. The result of the analysis using decay heat of the fuel debris expected in 2021 and 2031 as input conditions indicated that the highest temperatures on the surface of the fuel debris are expected to reach approx. 350 deg. C and approx. 320 deg. C respectively. (Appendix 4.17 Overview of the analysis evaluation for air cooling in the reactor)
In consideration of the maintenance of concrete strength, the target temperature of the concrete should be approx. 100 deg. C or less. Using the Dry method will pose a high degree of difficulty in the air cooling of the fuel debris. It is, therefore, estimated that this condition will not be satisfied in the early stage of the retrieval work for the fuel debris located at the bottom of the D/W. If the amount of fuel debris remained inside is decreased as the retrieval work progresses or if the realistic air cooling method is established in addition to decrease in the fuel debris amount, retrieval could be conducted by Dry method.



(Provided by TEPCO)

Figure 4.3.1-3 Complete air cooling for Unit 1
Surface temperature of fuel debris and surface heat transfer rate



Decay heat	Fuel debris	Cs-134,137
2021	44.0kW	4.10kW
2031	32.9kW	2.40kW

-	Max. temperature of fuel debris [℃]	Surface temperature of fuel debris [℃]
2021年 (Assuming full submersion)	478	348
2031年 (Assuming full submersion)	415	322

(Joint analysis evaluation by Hokkaido University and the NDF)

Figure 4.3.1-4 Complete air cooling for Unit 1

Analysis results of 3D temperature distribution

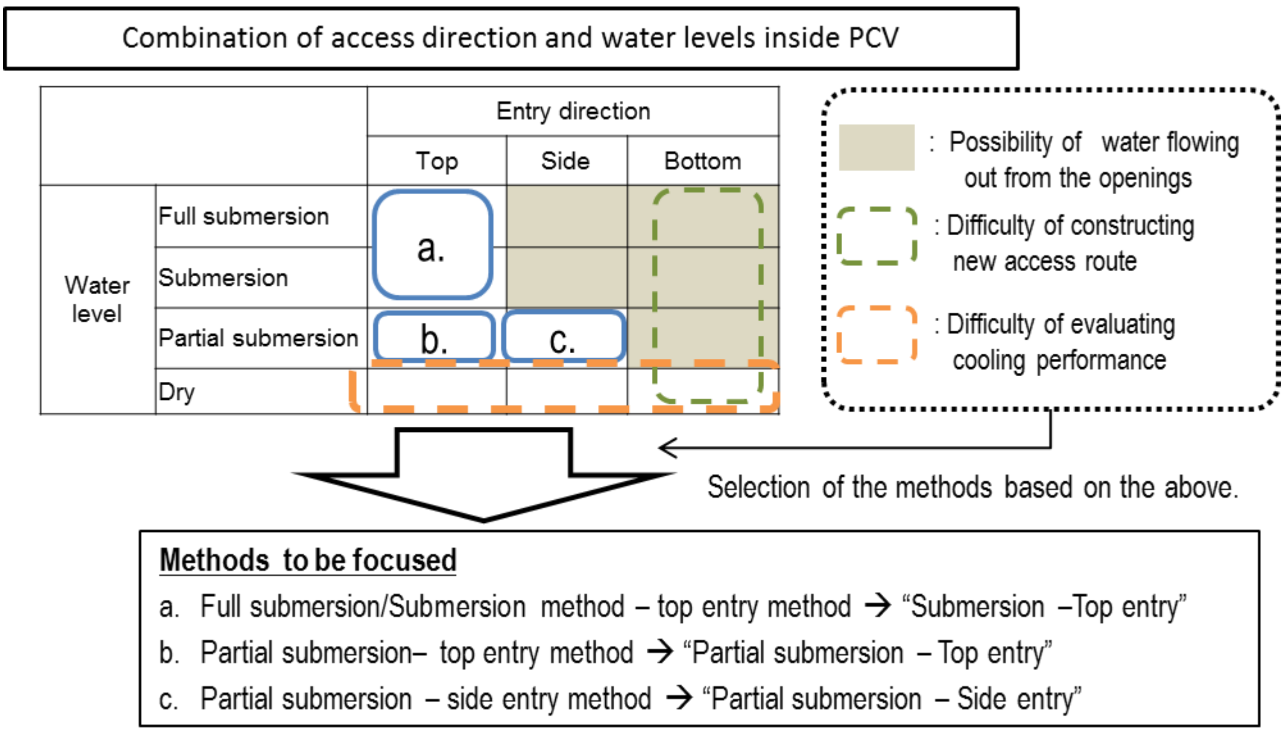
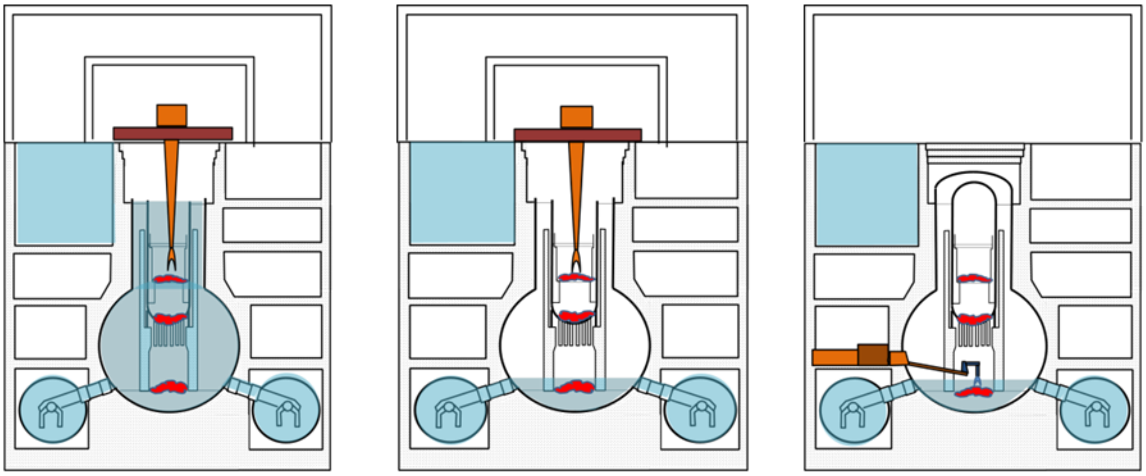


Figure 4.3.1-5 Narrowing of retrieval methods based on the combination of the PCV water level and access direction

Based on the results described above, the methods were narrowed down to those listed in the lower frame described in Figure 4.3.1-5. As fuel debris retrieval method, "Submersion-Top access method," "Partial submersion-Top access method" and "Partial submersion-Side access method" will be focused and studied. Schematic drawing for three methods of the fuel debris retrieval to be focused on is shown in Figure 4.3.1-6.



Submersion-Top access method	Partial submersion -Top access method	Partial submersion - Side access method
Assuming the in-core structures above the fuel debris are removed	Assuming that the in-core structures above the fuel debris are removed	Assuming that the equipment and other objects outside the RPV pedestal in PCV are removed

Figure 4.3.1-6 Schematic drawing of the fuel debris retrieval to be focused on

Actual selection of the fuel debris retrieval is not simple, such as selecting the one from three methods. There are variations of selection by setting or combining the water level of the PCV and access route, and conducting detailed engineering according to the plant conditions including the inside of the reactor of each Unit and situation of technical development.

In the fuel debris retrieval method, the concept of the prevention of the release of the radioactive materials from the PCV is important, especially for ensuring work safety. The method to confine the radioactive materials during the fuel debris retrieval work is being studied and the current concepts of three methods to be focused on are introduced as a reference for subsequent studies. The study plan for the Submersion-Top access method is shown in Figure 4.3.1-7. The concept for the containment is to make its pressure negative, in contrast to the outside pressure, by repairing the PCV, establishing the primary boundary, and establishing the secondary boundary by repairing or covering the R/B.

Since it would be difficult to completely prevent leakage by repairing the PCV and R/B, containment by negative pressure control and water level difference management is required to be considered. The concept of the containment boundary for the Partial submersion-Top access method and Partial submersion-Side access method is shown in Figure 4.3.1-8. In principle, it is based on the same concept which establishes the double containment boundaries to prevent release of radioactive materials by controlling negative pressure. For the reference in future studies, an example of the flow lines for the fuel debris during the retrieval work by the Partial submersion-Side access method which is currently studied is shown in Figure 4.3.1-9.

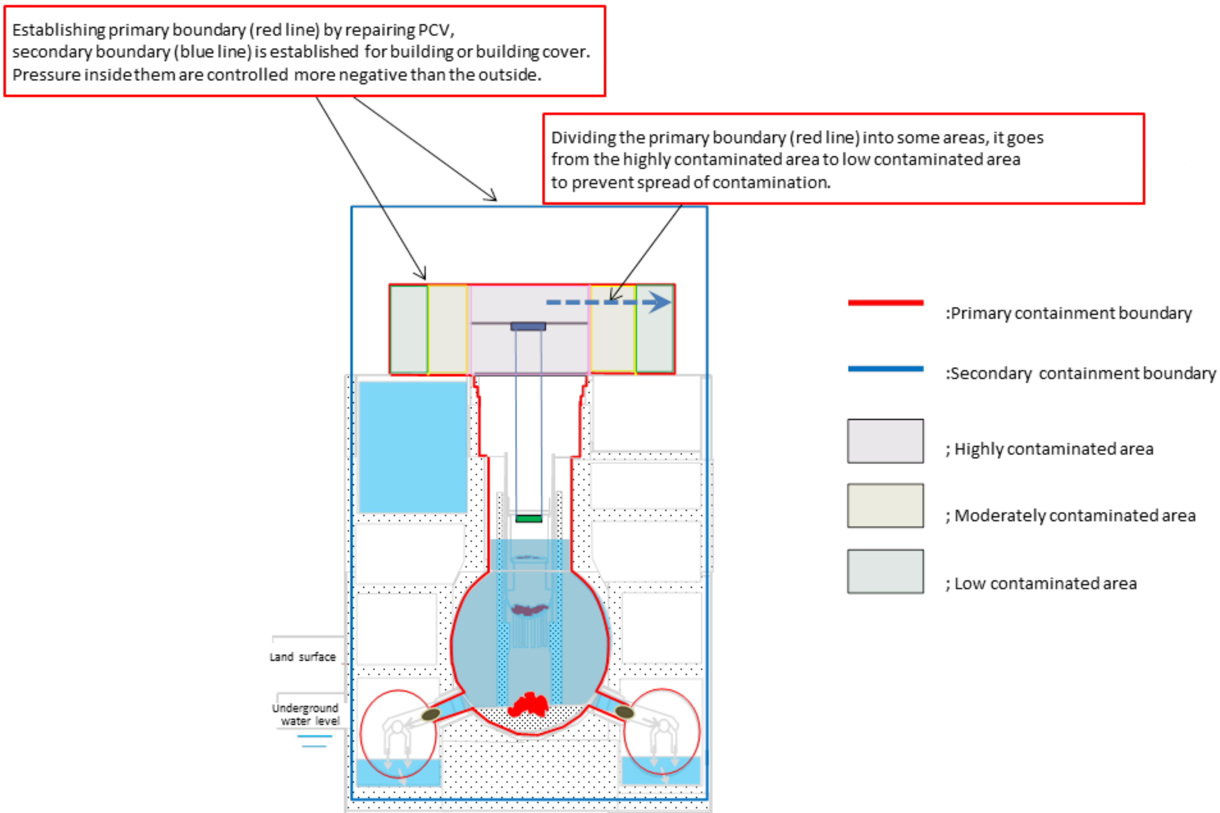
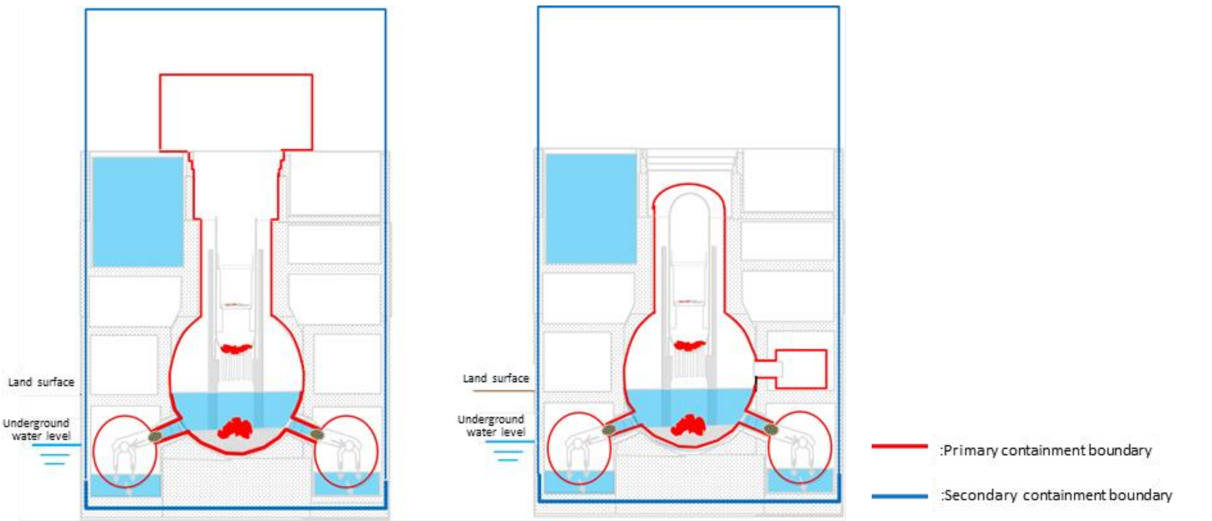


Figure 4.3.1-7 Study plan of the containment boundary for the Submersion-Top access method



Partial submersion-Top access method   Partial submersion-Side access method

Figure 4.3.1-8 Image of the containment boundary for the Partial submersion method

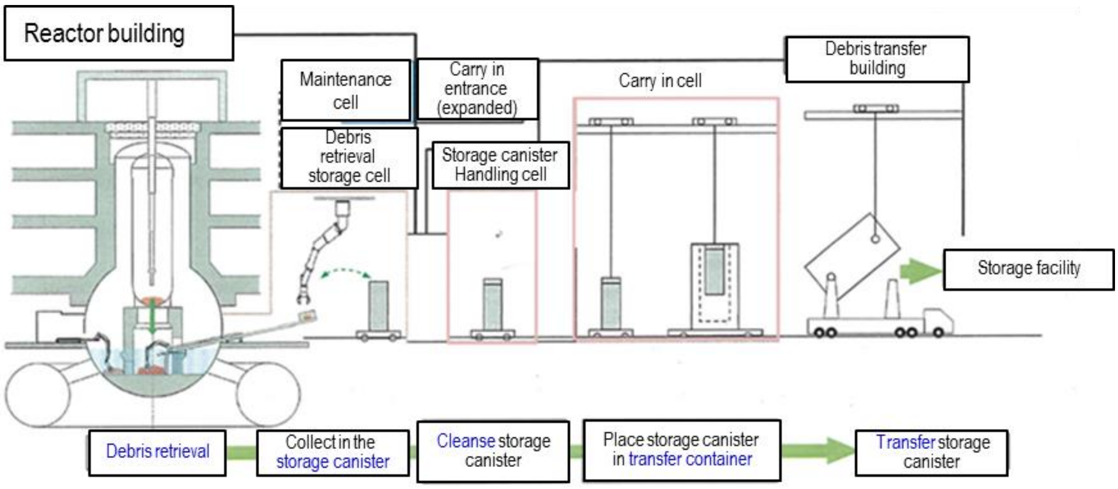


Figure 4.3.1-9 Flow line for the fuel debris in the Partial submersion-Side access method

(2) Study on the applicability of each retrieval method to the location of fuel debris

The fuel debris is estimated not only in the RPV (core region and the RPV lower plenum) but also inside and outside the RPV pedestal at the bottom of the D/W. Also, a part of the fuel debris around the RPV lower plenum is estimated to be attached to the CRD housing.

The schematic drawing of estimated fuel debris distribution is shown in Figure 4.3.1-10.

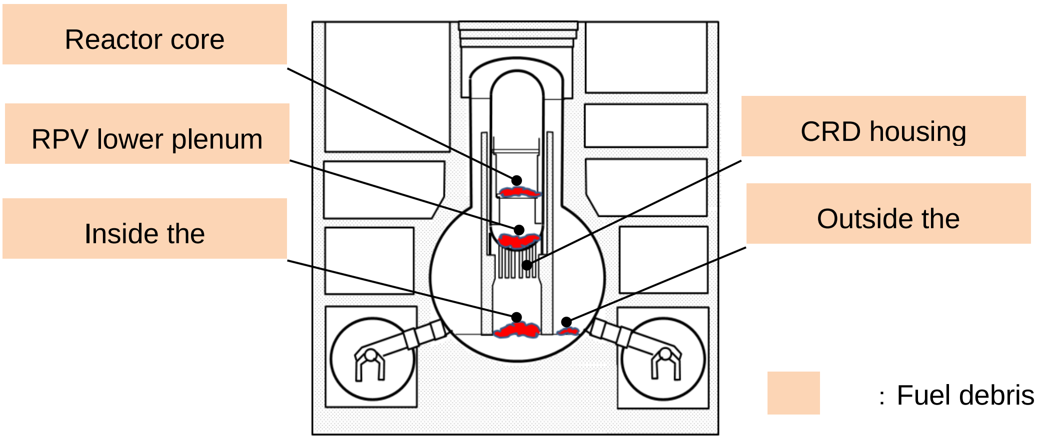


Figure 4.3.1-10 Schematic figure of estimated fuel debris distribution

The results of the evaluation for the applicability of fuel debris retrieval for these fuel debris locations for three methods from the degree of difficulty in feasibility are shown in the Table 4.3.1-1. (For details, refer to Table A4.18-1 of Appendix 4.18)

Features of applicability are as follows:

・High applicability of the Top access to the fuel debris in the RPV
・High applicability of the Side access to the outside of the RPV pedestal
・Both Top and Side access are applicable to the inside of the RPV pedestal

In either case of the possible fuel debris location (inside of the RPV (core region, RPV lower plenum and CDR housing), inside and outside of the RPV pedestal), retrieval can be done by either three options of fuel debris retrieval method. It will be necessary to retrieve fuel debris by combining these methods depending on the locations of fuel debris.

Table 4.3.1-1 Applicability of the fuel debris retrieval method according to the locations

Fuel debris retrieval Method	Fuel debris locations		
	Inside the RPV (core region, RPV lower plenum and CDR housing)	Inside the RPV pedestal	Outside the RPV pedestal
Submersion - Top access	○	○	△
Partial submersion- Top access	○	○	△
Partial submersion- Side access	△	○	○

*The fuel debris locations where triangle marks "△" are placed are the areas where the method is not essentially highly applicable to. The retrieval may be done depending on the detailed locations and amount of existing fuel debris based on the comprehensive evaluation for the reasonability of the whole retrieval process.

(3) Applicability of retrieval method in other conditions

To apply the fuel debris retrieval method on site, applicability to the various kinds of restrictions will require be studying and evaluating. Items to be considered are as follows:

・Applicability to the limitations on the PCV water level which may be required in the future.

・Applicability to the other on-site operations requiring considerations on the interface with fuel debris retrieval work

・Applicability to the accuracy of estimated internal PCV condition for each location obtained at the commencement period for the fuel debris retrieval.

The approaches to the retrieval method for each Unit is to be studied from technical perspective considering the conditions of Unit and features described in the Section (2) and (3), and are evaluated in light of the Five Guiding Principles.

4.3.2 Approach to key issues on fuel debris retrieval

This section clarifies the purpose and major requirements on each technical requirement. Also, with regards to the approaches required to satisfy the requirements above, this section describes the studies to assess the success or failure, additional issues to be studied and future action policies based on the current action status (results and plans) such as of R&D projects, and its evaluation.

In addition to the conclusions of these nine technical requirements, the perspectives to be focused on in the detailed study to be conducted for each method toward the study on the approaches to the fuel debris retrieval are described below.

Nine technical requirements as the key issues to achieve the fuel debris retrieval method are as follows:

- Technical requirements especially related to ensuring safety during the fuel debris retrieval
  (1) Securing the structural integrity of the PCV and R/B
  (2) Criticality control
  (3) Maintaining the cooling function
  (4) Ensuring containment function
  (5) Reducing workers' exposure during operation
  (6) Ensuring work safety

- Technical requirements as key issues directly relates to the fuel debris retrieval work
  (7) Establishment of access route to the fuel debris
  (8) Development of the fuel debris retrieval equipment and devices
  (9) Establishment of the system equipment and working areas

Technical requirements on ensuring safety for fuel debris retrieval are described in (1)-(6). As for "safety" in this retrieval work, it is not appropriate to apply the same safety standards as the normal nuclear power station to the Fukushima Daiichi NPS, which is a Specified Nuclear Facility. NRA indicated "The matters for which a Specified Nuclear Facility operator should take measures," for the Fukushima Daiichi NPS based on "Law on the Regulation of Nuclear Source Material, Nuclear Fuels Material and Reactors." In response to that, TEPCO has developed "Implementation Plan for the Specified Nuclear Facility" and is addressing the risk reduction work for the damaged NPS while ensuring safety.

The concepts for ensuring safety to be considered in the fuel debris retrieval work are as follows:

- The purpose of ensuring safety is to protect (1) Residents and environment, and (2) Workers from the impact caused by radioactive materials.
- The objective is to reduce risks by fuel debris retrieval from the current level of conditions, where the severe accident occurred. Although volatility fission products by the accident were released and decay heat are deteriorated in each Unit, the facilities such as R/B and PCV are still in a state of damage by the accident with high radiation environment.
- The functions to "shut down," "cool down" and "contain" a reactor, which are the bases of nuclear safety are managed by monitoring the plant parameters after the accident.
- During the fuel debris retrieval work (normal and expected abnormal operation), the temporary risk increase from current state is to be controlled as small as possible and is kept under a certain level of limit. The limit is examined by assessing the impacts on the residents and environment. The measures for ensuring safety should be recognized as a high-priority issue and be addressed in the effort from the early stages, such as prior examination and preparation.
- The risk that attributable to external events (e.g. seismic, tsunami, tornado) is to be studied. In response to the study, the concept of defense in depth assuming the plant that experienced severe accident is to be studied.

- Safety during operation may be ensured based on the safety standard by prediction or assumption at a certain point in time, due to limited information on fuel debris. Sharing "safety-related information" clarified at each operation step among the relevant organizations and reflecting them to the retrieval methods or safety management, safety requirements should be flexibly revised so that the safety is ensured more appropriately.
- Conformity and response such as to the expected safety-related requirements and inspection requirements by regulation should be considered in the development of equipment or construction methods.

It is important that the safety requirements during the fuel debris retrieval work are to be established before commencement of the works. It should be appropriately reviewed based on the facts clarified as the work progresses. In addition, response policy relating to ensuring safety and observed data will need to be indicated in the early stage while having active dialogue with nuclear regulatory authorities.

4.3.2.1 Securing the structural integrity of the PCV and R/B

(1) Purpose

The R/Bs, PCVs, RPVs, and the peripheral components, which have been damaged by the accident, shall maintain the following significant safety functions during the fuel debris retrieval work period at seismic events.

1) The R/Bs shall maintain its support functions of the important components and systems such as the PCVs, RPVs and the peripheral components.
2) The PCVs shall keep up its containment capability to prevent a massive release of radioactive materials.
3) The RPVs shall maintain the circulation path of cooling water.

The schematic sketch such as of R/B, PCV and RPV is shown in Figure 4.3.2-1.

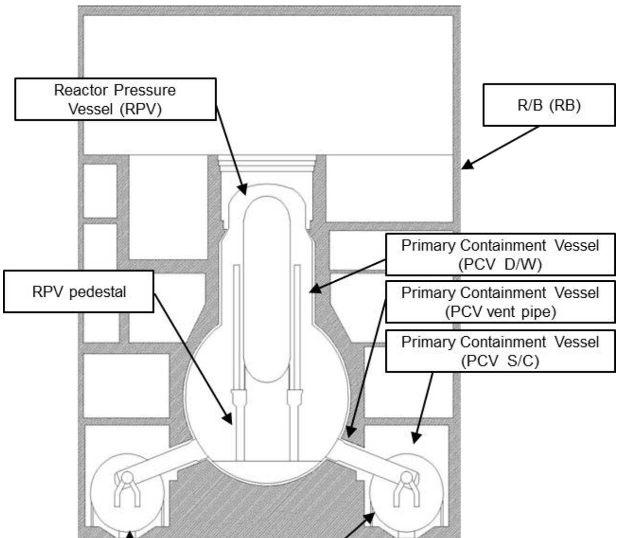


Figure4.3.2-1　Cross section of R/B and major components

(2) Major requirements

a. Evaluation of aseismic performances of the Submersion/Partial submersion methods at seismic events

The required functions described in above 1) - 3) of (1) shall be maintained during fuel debris retrieval work by the Submersion and the Partial submersion method even at seismic events.

The following conditions need to be taken into consideration in the evaluation of aseismic performances.

- Establish appropriate design seismic ground motions and evaluation criteria considering the damaged states of the systems, structures and components induced by the accident.
- Consider damages caused by the accident, degradation of materials due to exposure to high temperature, corrosion due to seawater injection as well as further degradation and corrosion that may progress to the time of the completion of fuel debris retrieval.
- Take account of the additional weights based on fuel debris retrieval work plan such as fuel debris, water inside of the PCV for cooling and submersion of fuel debris, water sealing material and fuel debris retrieval equipment including, radiation shielding and supporting structures.

b. Development of corrosion prevention measure for RPVs/PCVs and related piping systems, and the verification of its applicability

Develop the prevention measures for corrosion progression in the RPVs/PCVs and related piping systems over a long period of time during future fuel debris retrieval work in order to maintain the current status, and verify their applicability to the actual equipment.

(3) Action status and evaluations and issues

a. Evaluation of aseismic performance of the Submersion and Partial submersion methods at seismic event

Following approaches/activities have been carried out for the evaluation of aseismic performance of the Submersion and Partial submersion methods.

- Seismic safety assessment of the R/Bs taking into account its damage induced by the hydrogen explosions
- Evaluation of the load bearing capacities and stiffness of the reinforced concrete RPV pedestals which had experienced high temperature exposures during the accident
- Evaluation of aseismic performance of the Submersion and Partial submersion methods considering the wall thinning of the PCVs, RPVs and the peripheral components by corrosion, taking into account those mentioned above as well.
- Development of a simplified seismic evaluation method for of fuel debris retrieval methods to quickly understand the approximate results under various conditions related to fuel debris retrieval methods.

The findings and issues obtained to date are summarized below.

i) Seismic safety assessment of the R/Bs taking account of its damage

The following studies had been carried out by TEPCO for the seismic safety assessment of the R/Bs of the Fukushima Daiichi NPS Units 1-4.

- Evaluation of seismic responses of the R/Bs under the Great East Japan Earthquake motions (a massive earthquake of the moment magnitude, Mw, of 9.0)
- Seismic safety assessment of the R/Bs, taking into account the damages induced by the hydrogen explosions during the accident
- The investigation on degradation of the reinforced concrete seismic resistant walls of the R/Bs

The overviews for each item above are described below.

The acceleration time histories of the Great East Japan Earthquake were recorded on the base mats of R/Bs. The analytical responses of the R/Bs under the recorded acceleration time histories were evaluated to be well below the evaluation criteria, although some of the responses of the bearing walls exceeded the responses slightly under the current design basis seismic ground motion, Ss (600Gal).[9]

The seismic safety assessment results of the R/Bs taking into account the damages induced by the hydrogen explosions under the design basis seismic ground motion, Ss, indicates that the responses of the major seismic resistant walls and the SFPs were below the evaluation criteria with decent seismic margin.[10]

TEPCO has evaluated the ground motion (900Gal) based on the new regulatory requirements for light water nuclear power plants issued by NRA also.[11] However, the R/Bs and PCVs damaged by the accident have great difficulties for the repair works and reinforcement due to high radiation. Under such conditions, if the design seismic ground motion and evaluation criteria which might contain a high degree of safety margin were used for seismic safety assessment of Fukushima Daiichi NPSs, the whole risk reduction achieved by the fuel debris retrieval may be delayed due to the extension of the design and construction period of the structures and facilities for the fuel debris retrieval work which have to secure higher aseismic performance. Therefore, the design seismic ground motions and evaluation criteria shall be selected appropriately from the perspective of optimum reduction of the entire risk at Fukushima Daiichi NPS.

The periodic investigation on the degradation of the reinforced concrete seismic resistant walls has been implemented at Unit 4 R/B which seems to be most severely damaged by the hydrogen explosion, because its radiation dose is comparatively low.[12] According to the investigation results,

9 Nuclear and Industrial Safety Agency "Impacts of the Great East Japan Earthquake on the nuclear power plants (Seismic response analysis results of the buildings, structures, components and the piping systems) (Fukushima Daiichi and Daini nuclear power plants)," Dec., 2011 (Japanese)

10 TEPCO, "Seismic safety evaluation of the main office buildings of the unit 1 through 4 of the Fukushima Daiichi nuclear power plants against the design basis seismic ground motion Ss," No. 5-1, 4th Study group on monitoring and assessment of specified nuclear facilities, Feb. 21, 2013 (Japanese)

11 TEPCO, "Considerations on protection against the external events for the Fukushima Daiichi nuclear power plants," 27th Study group on monitoring and assessment of specified nuclear facilities, Oct. 3, 2014 (Japanese)

12 For example, TEPCO, "Periodical investigation results (9th) on structural integrity of the reactor building of unit 4 of the Fukushima Daiichi nuclear power plant," Jul. 31, 2014 (Japanese)

no harmful cracks that may cause corrosion of the rebar were observed on the major seismic walls and SFP walls. Also, the concrete compressive strengths of roughly $35N/mm^2$ or higher are being obtained in every investigation of the concrete strengths at seismic resistant walls, which are much higher than the design concrete strength of $22.1N/mm^2$ and no sign of degradation has been observed. Although a similar investigation is necessary for the degradation evaluation of Units 1-3 R/Bs when their radiation dose situations are improved, the degradation of them is currently supposed to be relatively low based on the investigation results of Unit 4.

ii) Evaluation of load bearing capacity and stiffness of reinforced concrete RPV pedestal influenced by high temperature exposure

The severe accident analyses have been conducted for Units 1-3 by using analysis codes such as MAAP and SAMPSON. The analytical results indicate that the molten fuels may be released from the bottom of the RPV to the PCV bottoms inside of the RPV pedestals.[13] Therefore, load bearing capacities and stiffness of RPV pedestals are considered to be decreased due to the effects of exposure to the high temperature. The many experiments using various size and types of specimens under several temperature conditions have been conducted by IRID for the evaluation of the impact of high temperature exposure followed by moistening by cooling water injection. One of typical experiments was conducted using 3 sets of 1/6 scale model of reinforced concrete pedestals which were submerged in the water after temperature experience of ambient temperature, 400 and 800 deg. C, respectively. Cyclic horizontal incremental loadings were applied up to the failures of the test models under vertical static loads corresponding to deadweights of components supported by the pedestals.[14] According to this test results, the maximum resistance force (load bearing capacity) and maximum deformation of the test model submerged in the water after the high temperature experience of 800 deg. C was reduced to approximately 70% of those of the test model of ambient temperature experience. Furthermore, the load bearing capacity of pedestal exposed to high temperature of 800 deg. C followed by moistening by cooling water injection is evaluated to be higher than the load induced by the design basis seismic ground motion Ss.

The evaluation of load bearing capacities of the RPV pedestals are required to be reviewed based on the latest findings to be obtained from the further investigation inside the PCVs and progress of severe accident analyses.

Also, the severe accident analysis results and other findings implied that the molten fuel debris have been fallen onto the PCV bottoms inside of RPV pedestals. The impacts of the erosion on the RPV pedestals by molten fuel debris may be required to be evaluated, depending on the future observations of the states of the actual spread of the fuel debris on the PCV bottoms through the investigation of the inside of the PCVs and that of the RPV pedestals.

iii) Evaluation of aseismic performance of the Submersion and Partial submersion methods based on i) and ii) above.

[13] IRID, “Improvement of recognition regarding the internal PCV condition using severe accident progression analysis and actual plant data,” Nov. 2015 (Japanese)

[14] IRID, “Development of the structural integrity evaluation methods of the RPVs and PCVs,” Nov. 2015 (Japanese)

In the R&D program by IRID, the FS for seismic safety of RPVs, PCVs and other related major components against seismic input of Ss (600gal) was carried out for the time of the fuel debris retrieval work for both the cases of the Full submersion method (water filled up to the top of the PCV) and Partial submersion method (current water level), in which, additional weights on the operation floor such as container and fuel debris retrieval equipment including radiation shielding and supporting structure as well as water inside of the PCV for cooling and submersion of fuel debris, water sealing material were considered. Especially, the detailed evaluation of S/C supports by elastoplastic FEM analysis has been carried out from FY2015. In parallel, in case of low seismic margin of the SC supports, technical development has been carried out for establishing the alternative back up method such as to place high-flow anti-washout underwater mortar on the torus room floor as an option for the reinforcement of the S/C supports. This reinforcing method raise the floor level of the torus room, consequently, it requires to control appropriately the water level in the torus room lower than that of groundwater.

In FY2016, the assessment of impact on the important safety functions of the RPV/PCV described in (1) under a severe earthquake event need to be performed. If there is any possibility of failure of the SC supports, for example, its consequent impact (ripple effect) is to be evaluated. Its countermeasures (preventive measures and mitigation measures) are also need to be studied. The prospect of their effectiveness is to be verified through the detailed analyses and experiments, as needed.

iv) Development of simplified seismic evaluation method for aseismic performances of fuel retrieval methods under various options

Development of the simplified seismic evaluation method was completed in FY2015. This method is effective to quickly evaluate the rough estimate of an impact of water level of inside of the PCV, and of additional weights on the operation floor such as container and fuel retrieval equipment including radiation shielding and supporting structure.

b. Development of corrosion control measures for RPVs/PCVs and piping systems and their verification of applicability to the actual units

The one of the concerns at the Fukushima Daiichi NPS Units 1-3 is the progress of corrosion of the RPVs/PCVs caused by exposure to the seawater and by the fallen fuel debris with a high temperature. It is necessary to confirm the applicability of the corrosion control measure to the actual components in order to prevent the progress of corrosion in the structural materials of the RPVs/PCVs and related piping systems and to maintain the current status of them for a relatively long period of time.

The circulation water has been injected into the PCVs to cool down the fuel debris and nitrogen is also injected to prevent hydrogen explosions. Since dissolved oxygen in the cooling water is reduced in the nitrogen atmosphere, corrosion of steel material is supposed to be inhibited. Since the PCV will be opened to atmosphere during the fuel debris retrieval work period, corrosion may progress due to the inflow of ambient air and increase the dissolved oxygen in the water. The corrosion progress is concerned to have an impact on the structural strengths of components at seismic event and consequently the functions to maintain the boundaries of the PCVs and S/Cs. Although addition of rust inhibitor in the cooling water during the fuel debris retrieval work period is an alternative candidate of the corrosion control measures instead of nitrogen injection, it is necessary to consider the impact on the sodium pentaborate to be injected as a criticality prevention agent and on cooling water quality control system by small circulation loops which are currently to be studied.

As the candidates, rust inhibitors of inorganic oxide film type used in the general water treatment system as well as those of precipitated film type were selected. The candidates of rust inhibitor are narrowed down to the following four types, considering their corrosion prevention effects, radiation impacts and impacts on the water quality maintenance system by small circulation loops.

- Sodium tungstate (oxide film type)
- Sodium pentaborate (oxide film type)
- Zinc /Sodium carbonate mixed phosphate (precipitation film type)
- Zinc/Molybdenum Sodium mixed Phosphate (oxide layer +precipitation layer type)

There still have been some issues, such as further narrowing-down of the candidates based on the applicability to actual units from the view point of prevention of local corrosion and impact on the existing entire water treatment system and establishing of conceptual design of the corrosion control system with its management procedures applicable to the actual units.

(4) Future course of action

The future actions to be taken are described below based on the evaluation results obtained by FY2015 in order to contribute the decision making on the policy and method of the fuel debris retrieval work.

a. Evaluation of aseismic performance of the Submersion and Partial submersion method

Through performing of the following activities and the results of aseismic performance evaluation by FY2015, the prospects for the seismic performance of the Submersion and Partial submersion method are planned to be obtained.

i) Establishment of safety scenarios against a severe earthquake event

Impact assessments on the important safety functions of the RPVs/PCVs described in (1) against a severe earthquake event need to be performed. If there is any possibility of failure of the SC supports, for example, its consequent impacts (ripple effects) need to be evaluated as well as its countermeasures (preventive and mitigation measures) is to be established (establishment of safety scenarios).

ii) Development of the assessment methods of the seismic safety of the components and evaluation method of impacts of their failures to establish the safety scenarios

The methods to assess the seismic safety of components and to evaluate the impact of their failures are to be developed for the PCVs/RPVs, S/Cs and RPV pedestals in order to establish the safety scenarios described in i).

iii) Improvement of safety scenarios

In order to improve the assessment methods of the seismic safety of the components and evaluation methods of impacts of their failures described in ii), efforts are to be also focused on the verification of the applicability of the developed methods reflecting more realistic conditions such as the restraint conditions of components and the conditions for seismic load evaluation. Furthermore, the detailed analyses and experiments are to be conducted as needed.

b. Development of corrosion control measures for RPVs/PCVs and piping systems and verification of their applicability to actual units

For the future fuel retrieval work, it is necessary to confirm the applicability of the corrosion control measures to actual units in order to prevent the progress of corrosion in the structural material of the RPVs/PCVs and related piping systems. To maintain the current status of them, further narrowing down of candidates of selected rust inhibiter is to be carried out based on the evaluation of the applicability to actual units (e.g. prevention of local corrosion, impact on the existing entire water treatment system, matching with criticality control system). In addition, the conceptual designs of the corrosion control system along with management procedures applicable to the actual units are to be established.

4.3.2.2 Criticality control

(1) Purpose

To prevent workers' exposure and impact on the environment due to re-criticality even when the shapes of fuel debris and water level are changed as a result of the water injection and retrieval works carried out during the fuel debris retrieval.

(2) Major requirements

In order to achieve the purpose, the method of the criticality control for the fuel debris whose signs of criticality are not observed at this moment should be established by being combined with the technology to prevent workers' exposure by shifting the state to subcritical in case of re-criticality through the period of the preparation and fuel debris retrieval work.

The requirements to establish the criticality control method are described below. These are common for the Submersion method and Partial submersion method.

a. Establishment of the criticality control method based on evaluations for possibility of re-criticality and behavior at criticality.

In order to appropriately control the criticality in the current status where the conditions inside the PCV is unknown, it is necessary to evaluate the possibility of reaching the re-criticality under based on the results of the estimation for the amount, location, shape and properties of the fuel debris, and identification of the events that may induce re-criticality. Excessive conservatism may, however, affect the assessment of the feasibility of the criticality control; therefore it is important to set a reasonable level of conservativeness. It is also necessary to accurately evaluate the amount of FPs and exposure dose rate in order to consider the measures for mitigation of environmental impacts in case of re-criticality. In addition, site applicability will need to be studied after the study on the criticality control method based on the evaluation results.

Major requirements that need to be satisfied for the establishment of criticality evaluation and criticality control method are as follows:

- The criticality scenario is to be evaluated based on appropriate conditions assuming multiple methods.
- The information required for the assessment of the reasonable conservatism is to be identified, and a plan for acquisition of such information is to be drafted and implemented.
- The accuracy of the assessment on re-criticality consequences is to be verified.
- The concepts for the systems and facilities to implement criticality control are established and feasibility is evaluated.

b. Technical development to implement criticality control

i) Sub-criticality monitoring methods

Since the neutron multiplication factors are changed if the retrieval work affects the criticality (water level and amount) of the fuel debris; therefore it is necessary to monitor such changes. In addition, if any anomaly is detected, works should immediately be suspended or neutron absorber should be used to maintain the state of sub-criticality.

The following is the requirement to be satisfied for the sub-criticality monitoring methods to be put into practice.

- Partial increase in effective multiplication factor of the widely distributed fuel debris is detectable.

ii) Recriticality detection technologies

Under the current conditions where the distribution of the fuel debris is not fully understood, it is difficult to monitor the neutron multiplication factors, and its technical development is underway. On the other hand, the amount of FPs and neutron and gamma ray dose will increase once the criticality occurs, making it rather easy to detect them. This method may, however, result in the delay in the detection and responses since it may take time as well. Therefore, it is required to take appropriate measures to prevent the workers' exposure and the public.

The major requirements to be satisfied for the re-criticality detection technologies are as follows:

- Safety is ensured by the combination of re-criticality detection, exposure evaluation and mitigation measures.
- Partial re-criticality of the fuel debris widely scattered is detected without having excessive impact of radiation

iii) Criticality prevention technologies

Dissolving the neutron absorber in the coolant or covering the surface of the fuel debris with neutron absorber, criticality must be prevented regardless of the condition of the fuel debris. If this can be achieved, there may be fewer restrictions posed on the fuel debris retrieval method.

The major requirements to be satisfied for the criticality prevention technology are as follows:

- Reactivity required for maintaining the re-criticality in assumed conditions is specified and secured.
- Integrity of the facility is maintained considering the corrosion of materials inside the reactor and

impact on the cooling water circulation system.

(3) Action status and evaluations and issues

a. Establishment of criticality control method based on evaluations for possibility of re-criticality and behavior at criticality.

i) Evaluations for possibility of re-criticality

The criticality scenario that may cause re-criticality during the period from the preparation to fuel debris retrieval work is being identified assuming the multiple methods. Also, possibility of criticality for each location where fuel debris is accumulated is being evaluated in conjunction with the results of criticality evaluation comprehensive analysis of internal PCV condition for each Unit. (Fig.4.3.2-2)

Toward the decision on approaches to the retrieval method, in order to indicate how much the PCV water level can be increased from the criticality control perspective, evaluation are being performed for the boron concentration required for criticality control and possibility of increase in the PCV water level by pure water assuming the remaining fuels in the core region (including stub-like fuels), at the RPV lower plenum and bottom of the PCV. In order to eliminate excessive conservativeness in the analysis, the method to consider gadolinium, as burnable poison and FP, contained in the fuel assembly and evaluation method of remaining fuels in the core region using the detailed composition of fuel assemblies were studied in FY2015.

The setting method of uranium enrichment and gadolinium concentration has been studied for boron concentration required for criticality prevention.

Meanwhile, the current study indicated that the water level up to the RPV lower plenum is unlikely to cause re-critical state if considering the internal structures and FPs as chemical composition of fuel debris within the realistic range. In addition, the increase in the water level up to the core region is unlikely to cause re-criticality, if the height of the fuel assemblies remaining in Unit 1 is smaller than three layers ring and those in Unit 2 is smaller than 5X5. These results are required to be shared for the comprehensive analysis of internal PCV condition and be reflected to the future research plan. It is, however, difficult to completely deny the possibility of criticality in the condition where internal PCV condition is not clear. The policy decision is required to be made with impact mitigation measures assuming the possibility of criticality. In addition, information regarding the possibility of criticality will be absolutely necessary toward the determinations on the policies and confirmation of the method and decision-making on the commencement of the operation. The method to understand the possibility of criticality comprehensively based on the limited data is required to be studied considering the composition of the fuel debris, which is the condition of criticality evaluation in statistic manner.

Since reactivity may be inserted depending on the changes in the shape of the fuel debris, when retrieving the fuel debris, setting of limits of amount of fuel debris per one operation is being studied based on the criticality evaluation.

ii) Evaluation of behavior at criticality

In the development of the impact evaluation method, a thermal-hydraulic model for kinetics analysis code for single-point reactor has been improved in order to assess the neutron response and amount of FPs generated after re-criticality to establish the impact mitigation measure to exposures. Also, FP generation assessment model required for the development of re-criticality detection system using the exposure and gamma-ray at the time of re-criticality was created. In FY2015, the evaluation model assuming the water filling up to the upper part of the PCV was developed and behavior at criticality assuming Unit 2 was performed. As a result, if the number of fuels remained in the reactor does not exceed 380 when the PCV water filling speed is limited to less than 1cm/h, workers' exposure and environmental impact could be controlled to within the standard value for the normal operation even in the criticality state. (Fig. 4.3.2-3) Evaluation results might be changed by reviewing the criticality detection time and further optimization of conservativeness of evaluation condition; however, results of investigation inside the reactor using muon detection system (transmission method) currently performed in Unit 2 will be the data important to confirm the feasibility of the criticality control when filling pure water.

Also, the analysis functions required for the behavior at criticality during the fuel debris retrieval were studied and issues were identified. The setting of the conditions is to be studied in FY2016 to evaluate the behavior at criticality during the fuel debris retrieval.

iii) Establishment of criticality control method

Based on the evaluation results of criticality and behavior at criticality above, the basic concept of the criticality control corresponding to multiple methods are summarized. (Table 4.3.2-1) From criticality prevention perspective, it is desirable to use soluble neutron absorber e.g. boron as coolant for criticality control method when filling the PCV with water. The possibility of filling of pure water is, however, also being studied since there are some issues such as the feasibility of water quality management system including removal of nuclide from the coolant. Also, conceptual studies for the systems and facilities are being performed to conduct criticality control.

Since site applicability of systems and facilities to control the criticality will need to be confirmed, it is important to reflect the studies on the retrieval methods and verification of feasibility of the system to the criticality control method. Also the fuel debris retrieval system and device identified through the conceptual study and the functions required for circulation cooling system are to be indicated.

To evaluate validity of the criticality control, objective of the criticality control is required to be clarified. In this regard, installation of excessive equipment should be avoided in light of the reduction workers' exposure during the field work, realization of fuel debris retrieval and feasibility of other safety requirements. Also, it is important to establish logic based on the data that indicates the validity as well as to set the appropriate and realistic objectives.

b. Development of the criticality control technology

i) Sub-criticality monitoring methods

As a sub-criticality monitoring method towards the application to the liquid-waste process and cooling systems, establishing the concepts of the sub-criticality monitoring system equipped with the neutron detector, gamma spectrum detector, and gamma dosimeter based on the reverse multiplication method, design and prototype of the equipment were created and the feasibility was evaluated by the criticality experiment device. This system is to be installed near detectors outside the tanks or piping of liquid waste process and cooling systems where fuel debris may be accumulated. In the element test, the basic data was obtained regarding the detection characteristics of neutron detectors and gamma ray detectors under the background of high gamma-ray, which is expected to be occurred in the periphery of liquid waste process and cooling systems. Also in the system test, the performance to identify the changes in the sub-criticality state by processing the signals from the detectors was evaluated. It was confirmed that the criticality approach can be monitored in sub-critical condition with effective multiplication factor of approx. 0.5 to 0.7. Moreover, as a different method, usage of the reactor noise in the neutron count rate measurement was studied and possibility of its application was confirmed.

Development of sub-criticality monitoring methods was started since FY2014 in preparation for the application to the fuel debris retrieval work in the PCV/RPV.

Referencing the technologies developed by FY2015, the concepts of the system that combines the methods which may be applicable during the fuel debris retrieval work. This system is to analyze sub-criticality state by measuring the changes in the neutron counter rate during machining the fuel debris. It requires operation check for detector under high gamma ray environment expected in the periphery of the fuel debris.

ii) Recriticality detection technologies

The methods of neutron detection and gamma-ray measurement from short-lived FPs have been studied as re-criticality detection technologies.

Having established the concepts on the neutron detection system to be installed inside the PCV based on the evaluation by the analysis of the neutron dose distribution in and outside the PCV in the event of re-criticality, its equipment design and prototype were created and the feasibility was evaluated by the irradiation testing facilities. As a result, sensitivity data for counting rate of neutron signal discriminated from gamma-ray was obtained under high gamma ray environment expected inside the PCV. The development on the measurement method based on neutron detection was completed in FY2013.

For the gamma ray detection system assuming installation in the gas treatment systems, studding the option for the improved method that has faster response speed of re-criticality detection than the current PCV gas control system, optimized design was created focusing on the difference of FP yield between spontaneous fission and neutron fission. As a result, concurrent counting for Kr-87/88 in addition to the current counting system of Xe-135 was selected, and the element test was conducted for the verification of the principles and the feasibility of the system was confirmed. Also, in FY2015, as a result of the analysis of measured data of gas control system of Unit 1, Kr-87/88 that principally derived from spontaneous fission was confirmed capable for being measured. Although there is a certain level of errors, multiplication factor of neutron source of current status was also confirmed capable of being estimated though the examination of the measured result of multiple short half-lives nuclide. Since the neutron source multiplication factor will be valuable data to understand the level of possibility of criticality in each Unit, its development should be continued in considering the application not only for Unit 1 but also for other Units.

In detection of re-criticality based on the gamma ray, the delay in detection time may have a significant impact on the mitigation and termination of accident progression; therefore approaches need to be taken to quantitatively calculate the detection time required based on the results of evaluation of behavior at criticality and exposure assessments.

iii) Criticality prevention technologies

In the development of the criticality prevention technologies, the study on the possibility of continuous injection of soluble neutron absorber and development of undissolved absorber are being carried out. Sodium pentaborate has been selected as a candidate material that behave as a rust inhibitor for the soluble neutron absorber. The material corrosion test results indicated the prospect that the PCV materials were maintained soundly under the condition of boron concentration of greater than 2,000ppm. (Fig. 4.3.2-4) Also, the evaluation was performed for the boron concentration required for criticality prevention in consideration of the gadolinium contained in the fuel assemblies. As a result, prospect was confirmed that the criticality prevention can be achieved under the condition of around 6,000ppm and boron concentration that can be feasible from the perspective of material integrity and necessary boron concentration evaluation was found. In the future, to confirm the feasibility of continuous injection of sodium pentaborate, conceptual study is required to be performed for the water quality management system, from the perspectives of impact caused on the removal of nuclide in the coolant, environmental impact caused by the sodium pentaborate leakage and amount of boron necessary for the period of fuel debris retrieval.

Undissolved neutron absorber is being developed as a candidate impact mitigation measures when maintaining the sub-criticality during the fuel debris retrieval work and in case of criticality. The requirements such as operability and leaching characteristics have been set and candidate materials were selected to date, narrowing down the selected candidates such as according to the basic physical property test and radiation resistance test. Also the prospects for the durability to the environment close to the fuel debris were confirmed for B_4C/metal sintered body, B/Gd containing glass materials, Gd_2O_3 particles, water glass/Gd_20_3 agglomerated powder materials, and slurry/Gd2O3 particles. Also, confirming the neutron absorption capacity of these candidate materials by the neutronic characteristics test, evaluation for the amount to be placed considering the application to the actual unit and studies on the application method are to be performed. Also, with regard to the application to the actual unit, reactivity reduction effect is required to be confirmed by the placement of neutron absorber, and the method that combined with the measurement of the level of sub-criticality using the criticality proximity monitoring technology is required to be studied.

(4) Future actions

The past achievements and the current studies were compared to the requirements, and the areas that require further studies are identified and summarized below.

a. Establishment of criticality evaluation and criticality control method

- Identify the required information and the timing in the development of various criticality control methods and acquire such information reliably.
- Set sufficient and realistic objective for the criticality control and establish the logic based on the data

of adequacy.

- Evaluate the feasibility of the systems and facilities to perform criticality control in collaboration with studies on the fuel debris retrieval system and device and circulation cooling system.

b. Development of the criticality control technology

i) Sub-criticality monitoring methods

- Although detectors are to be installed in the periphery of the locations for machining fuel debris, possibility of meeting the requirements should be fully examined because the distribution of the fuel debris is yet to be confirmed and there are restrictions on the installation location.
- The development policy is to be amended if requirements cannot be met.

ii) Recriticality detection technologies

- The practical application should be determined with the involvement of the site personnel. Therefore, it is necessary to confirm the location to install the detectors and determine whether the requirements can be met for re-criticality detection technologies using the neutron detection system developed in FY2013.
- The objectives for ensuring safety are reconfirmed, including the reduction of the time for detection by 1/10 of the current time in case of gamma ray detection for re-criticality detection system.

iii) Criticality prevention technologies

- Studying the concept of the water quality management system in order to inject sodium pentaborate solution all the time, the possibility is to be confirmed for the items including the amount of boron to be required for the period of fuel debris retrieval work and the nuclide to be removed from the coolant materials. Also, the extent of the impact caused by sodium pentaborate to the environment in case if coolant is released from the coolant boundary is to be confirmed.
- The criticality control method is to be studied to reduce the boron concentration, such as by combining with the other criticality control methods.
- A method is to be formulated to confirm the amount of neutron absorber absorbed using the binders to the fuel debris and to quantify the reactivity effect so that the insoluble neutron absorber can actually be used on site.

The above mentioned actions should be carried out as early as practicable in preparation for the fuel debris retrieval. It is, however, necessary to provide criticality control for the process that may affect the water levels and fuel debris shapes even before retrieving the fuel debris. Also, its applicability needs to be examined in preparation for the increase in water level after sealing the PCV. The possible technologies that can be applied to such cases include soluble neutron absorber and re-criticality detection by gamma ray or combinations of those technologies, and the development of these technologies must be completed as soon as practicable.

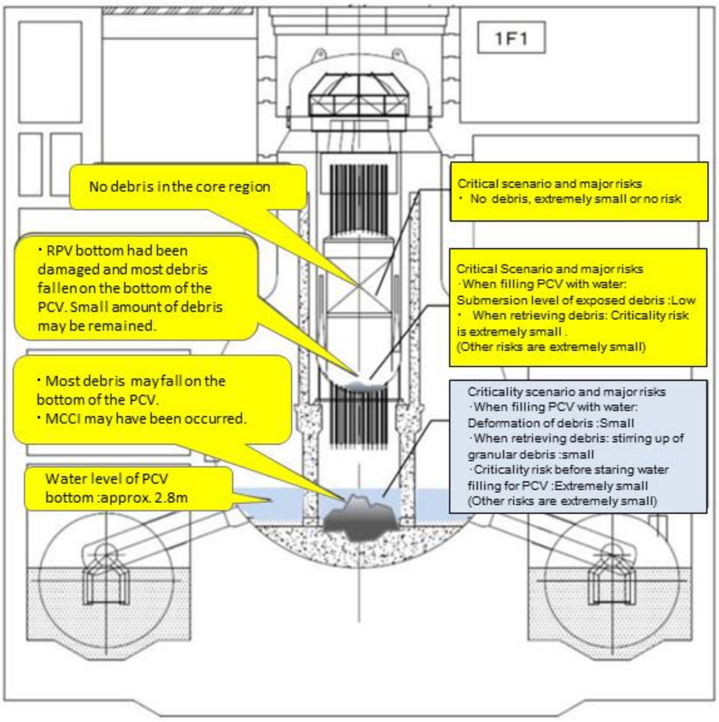


Figure 4.3.2-2 Evaluation of possibility of occurrence of criticality (Provided by IRID)

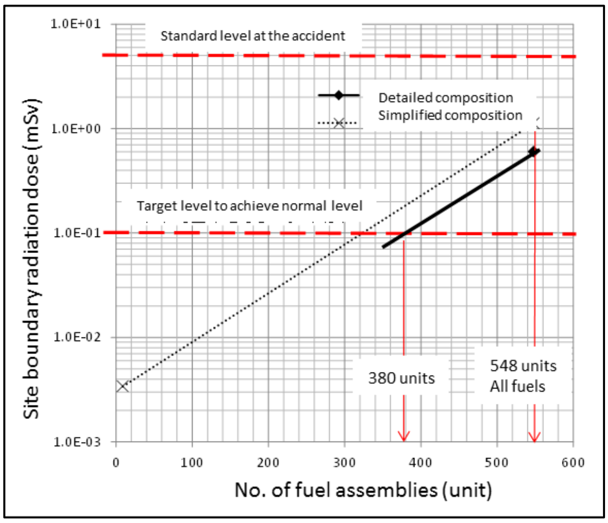


Figure 4.3.2-3 Evaluation results of the feasibility of criticality control when re-submerging the core region (Provided by IRID)

Table 4.3.2-1 Basic concept of the criticality control

	Concept of general management method (sample)	PCV water filling		Debris retrieval
		Pure water	Boric acid water	
Criticality prevention	・Evaluate possibility of criticality ・Limit the reactivity inserted at one time while measuring the degree of sub-criticality in advance of operation ・Suspend the operation by detecting the state close to the criticality	Criticality scenario evaluation Limit of water filling speed Incremental water-filling FP gas behavior monitoring	Criticality prevention by boric acid FP gas behavior monitoring	Criticality scenario evaluation+ limitation of retrieval quantity per criticality prevention using absorber Sub-criticality monitoring by Sub-criticality measurement system
Impact mitigation	(Criticality detection) Identify re-criticality state by monitoring neutron and FP gas behavior (Criticality shutdown) Insert negative reactivity after detecting re-criticality (Behavior at criticality) Evaluate the behavior at re-criticality and study the validity of impact mitigation measures	Criticality detection by FP gas γ-ray End of criticality by boric acid water or decrease in water level	Criticality detection by FP gas γ-ray End of criticality by highly concentrated boric acid water	Criticality detection by FP gas γ-ray +neutron End of criticality by absorber

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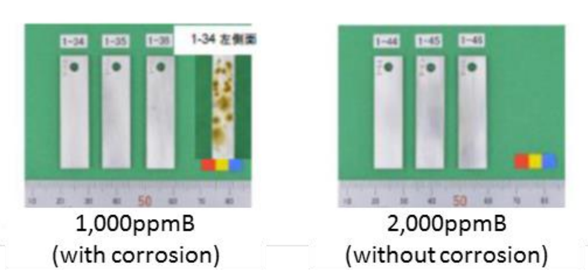


Figure 4.3.2-4 Results of steel materials corrosion test by sodium pentaborate
(1000 times the seawater value for 100 hours) (Provided by IRID)

4.3.2.3 Maintaining the cooling function

(1) Purpose

To maintain the cooling function since decay heat is constantly generated by the fuel debris over a long period time such as for the treatment, PCV repair and fuel debris retrieval.

(2) Major requirements

Basic requirements which should be addressed are as follows:

- Temperature is less than 100 deg. C so as to estimate, manage and record the cooling conditions of fuel debris.
- Alternative water injection system and equipment are installed in case of that a permanent facility cannot cool fuel debris due to the events such as earthquake and tsunami.

Other than above, requirements are different depending on the period (phase) of approach toward the fuel debris retrieval. The following are the major requirements in three phases.

a. Phase 1: Circulation loops during stagnant water treatment

- The reactor vessel is cooled down and removal of Cs and desalting of the contaminated water is possible (Figure 4.3.2-5)
- Operation and management are carried out for stagnant water with different floor height and water levels in series. In the building where the treatment is not completed, operational is controlled so as to keep the level of underground water higher than that of building's stagnant water.

b. Phase 2: Circulation loops during the PCV repair work

- Circulation, collection of excess water and drainage with required flow rate must be possible before the commencement of the PCV repair work.

c. Phase 3: Circulation loops during fuel debris retrieval work

- The required functions (e.g. cooling, cleanup, water level control, criticality prevention) are in place for long-term operation during the fuel debris retrieval works.
- Treatment of the pieces of fuel debris flowing into the circulation loops is studied.

(3) Action status and evaluations and issues

TEPCO is carrying out maintenance and management of equipment for cooling down the fuel debris as well as continuous monitoring of the parameters including the temperatures of the reactors. The main water source of the cooling water injection system for cooling down the fuel debris has been changed from a buffer tank to a CST since July 2013 and operation of CST reactor coolant injection system has been started. By doing so, the risk of losing water injection function due to reduction of a reactor water discharge line length is reduced and improvement of aseismic performance and increase in capacity are achieved for tanks. On the other hand, keeping the cooling water injection, the RPV bottom temperatures and the PCV gas phase temperatures of Units 1-3 are maintained between approx. 15deg. C to 30deg. C during the recent one month in spite of the differences among the Units or locations of the thermometer (Appendix 4.2 Fig.A4.2-1 Trend of temperature around the reactor in the Fukushima Daiichi NPS). Also there is no significant change in the parameters such as the pressure inside PCV and the amount of radioactive materials emitted from the PCV. Also no abnormality in the cooling condition nor the indication of criticality were observed. From the above, it was confirmed that the overall cold shutdown state has been maintained and the reactors are in a stable condition (Reference: Presentation of 63rd Contaminated Water and Decommissioning Issues Team/Secretariat Meeting dated Mar.31, 2016) In addition, the multiple back-up systems for injection of coolant into the reactors are maintained, thereby improving the credibility of the cooling system. (Fig.4.3.2-5 Example of Unit 1)

(4) Action plan

Future actions for each phase are described as follows.

a. Phase 1: Circulation loops during stagnant water treatment

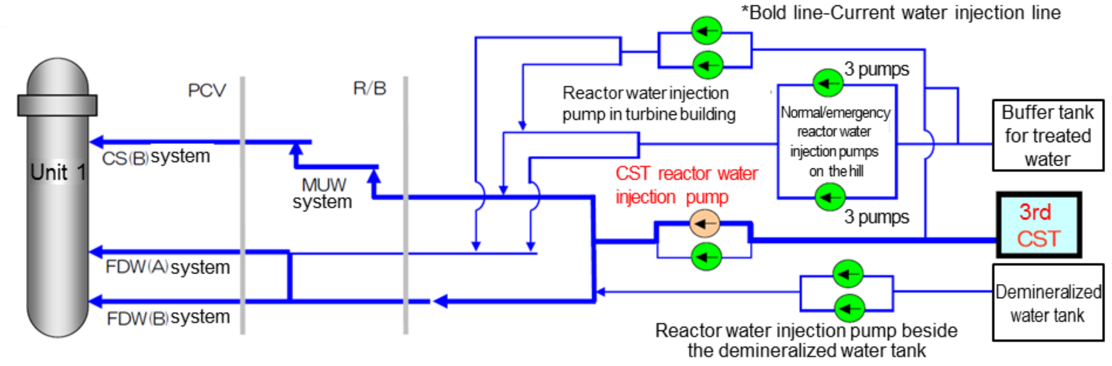
As shown in Figure 4.3.2-6, cooling of the fuel debris and removal of Cs and salt in stagnant water are being performed. Constructions for small loop to improve reliability have been approved and carried out.

b. Phase 2: Circulation loops during the PCV repair work

The flow rates of collection, circulation and drainage from torus rooms must be studied based on the estimated amount and locations of leakage after PCV repair and the circulation loops must be designed. To do so, following the understanding of the actual performance of water sealing and limit value (permissible amount of leakage) of other R&D projects and studies on the locations of water intake and installation method of the water intake lines for circulation system, the cooling system with functions to collect, drain and circulate the cooling water before reaching to the leak locations is to be established before the commencement of the PCV repair work.

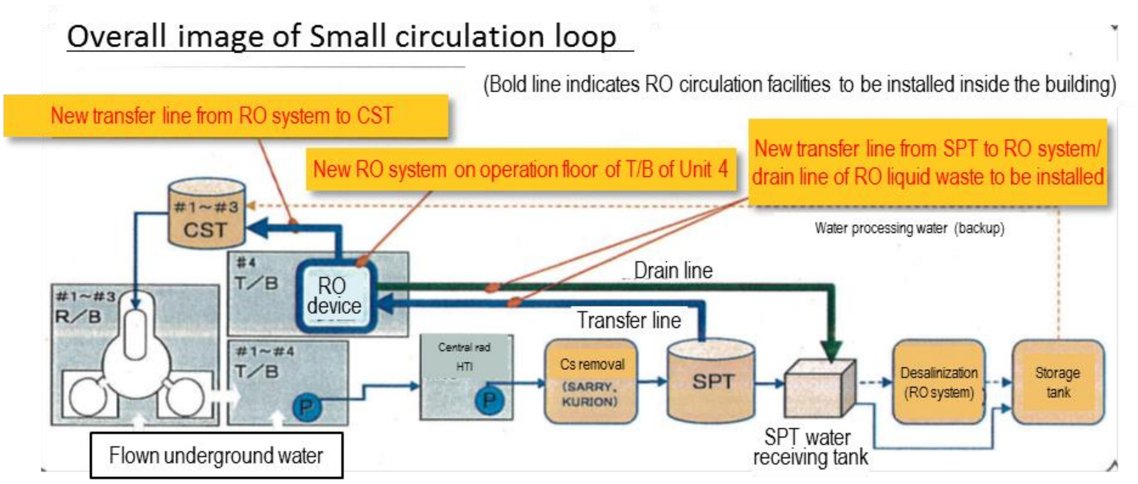
c. Phase 3: circulation loops during fuel debris retrieval work Phase (Refer to Appendix of conceptual diagram)

The circulation loops with required functions that enable the fuel debris retrieval by Submersion method must be established and the conceptual diagram currently planned is shown in Figure 4.3.2-7. As described above, in parallel with the study on the water collection system including the permissible amount of leakage after the PCV repair, coordinating with the concept of boundary, the following should be performed: engineering, R&D, response to regulations are to be performed for the functions of cooling, criticality prevention, removal of radioactive materials, turbidity prevention, water quality management, water level control/monitoring, and interlocks which should be equipped for fuel debris retrieval. Also other than Submersion method, desired circulation loops such as for cooling by pouring water is to be studied.



(Provided by TEPCO)

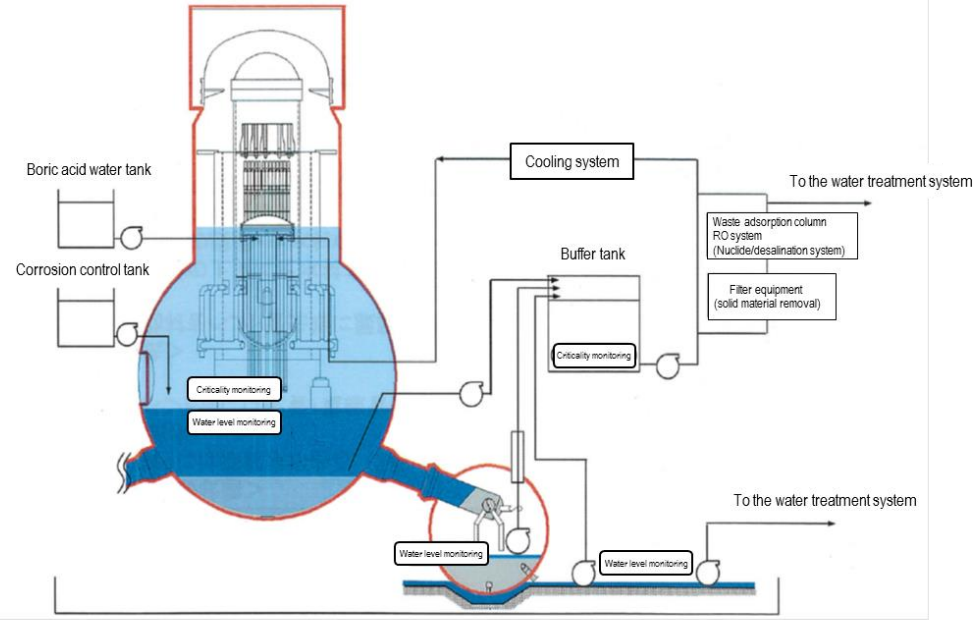
Figure 4.3.2-5 Current water injection line to the reactor (Unit 1)



(Source: 64th the decommissioning and contaminated water management team meeting, Apr. 21, 2016)

(Provided by TEPCO)

Figure 4.3.2-6 Phase 1: Circulation loops during stagnant water treatment



(Provided by IRID)

Figure 4.3.2-7 Phase 3: Conceptual diagram of Circulation loops during fuel debris retrieval

4.3.2.4 Ensuring containment function

4.3.2.4.1 Concept of ensuring containment functions

(1) Purpose

To protect the residents and environment from the impact by confining the radioactive materials including alpha nuclide generated during the fuel debris retrieval work for a long period of time (for normal/abnormal operation) and controlling and preventing its release controlled/managed.

(2) Major requirements

a. Determine concept for ensuring containment functions (boundary) during fuel debris retrieval work

Assuming the current release control of the radioactive materials at the Fukushima Daiichi NPS where containment functions have been lost by the accident, it is necessary to provide an appropriate securement of the containment functions (boundaries) and release control of radioactive materials under the environment which is more severe condition where cutting particles including alpha nuclides are caused by the retrieval work.

In this regard, it is important to consider the release and migration pathway of radioactive materials. There are two major pathways, which is via groundwater from the liquid phase and via atmosphere from the gas phase. For this reason, the study is conducted for each path.

Also, the level of risks, which are the frequency of event generation and degree of impact are required to be studied for the periods during the normal operation and abnormal operation.

In particular, the exposure assessment standards at the time of accident is to be determined as well as the release limit and release control target value during the normal operation.

Although this is originally a regulatory matter, it is important to be provided and shared based on the studies on the realistically achievable containment functions including its concept considering the positioning the Fukushima Daiichi NPS, which is a Specified Nuclear Facility.

b. Establish a containment system for the liquid phase

The detailed containment system for the liquid phase including the release control of the radioactive materials based on the concept of securing the containment functions (boundary) during the fuel debris retrieval works described above. In this regard, it is important to aim at a system which is feasible under a high radiation environment. If necessary, the study is required to be conducted again going back to the concept of containment functions (boundary). Also as the major requirements to be satisfied for the containment system for the liquid phase, the impact of the radioactive materials contained in the liquid phase during the fuel debris retrieval work to the outside is required to be controlled sufficiently, the impact outside must be limited even assuming the accident caused by the PCV such as a large amount of leakage, and the integrity of the system is to be secured under the possible seismic conditions.

c. Establish a containment system for the gas phase

A containment system for the gas phase including the release control of radioactive materials based on the concept for ensuring containment functions (boundary) described above. In this regard, it is important to aim at a system which is feasible under the high radiation environment. If necessary, the study is required to be conducted again going back to the concept of containment functions (boundary)

. Also as the major requirements to be satisfied for the containment system for the gas phase, the impact of the radioactive materials contained in the gas phase during the fuel debris retrieval work to the outside is required to be controlled sufficiently, the impact outside must be limited even assuming the possible accidents, and the integrity of the system is to be secured under the possible seismic conditions.

(3) Action status and evaluations and issues

a. Concept of containment functions (boundary)

At the Fukushima Daiichi NPS, the reasonably feasible measure was implemented on the assumption that the state and the severe environment which the facility damaged in the accident from, post-accident emergency response, and it has resulted in current stable condition. As a standard (goal) to be complied with at this stage (during normal period), NRA sets the impact (entire site) caused by the additional exposure at the site boundary is to be 1mSv/year. This figure is based on the release in controlled area of standard reactor and radiation control. Also, applying the standards of the various kinds of accident, the exposure dose at the site boundary of 5 mSv/accident is assumed by the evaluation for the time of accident.[15]

The increase in the concentration of alpha particle, which are FP and nuclear fuel materials in the PCV are concerned during the fuel debris retrieval. Accordingly the utmost efforts are made aiming at ND which is less than detection limit during the normal period so as to minimize the release of radioactive material containing alpha nuclide to the extent possible. Under the accident such as extremely rare event, 5mSv/accident will be the one of the target value at the site boundaries.

b. Confinement system for the liquid phase

i) Current containment status for the liquid phase

Cooling water injected into the reactor receives PCV as a primary boundary once and the leakage from the PCV makes the building as a secondary boundary. The cooling water injection system to reuse water accumulated in the buildings for the reactor core cooling after the cleanup has been adopted. In this case, the boundaries of the buildings are designed to control the stagnant water in the building under the groundwater level around the building. Accordingly, groundwater is flowing in (inleak) and radioactive materials are prevented from being flown outside the building (outleak). For this reason, since groundwater flows in will be surplus on the water balance and stored as water treated by the water treatment system, contaminated water issue such as storing the tanks continuously within the site has been occurred. In response to this, pumping up the groundwater by the sub-drain systems installed around the building and releasing it in the harbor through the purification system, the amount of inflow is being reduced by lowering the groundwater level. Also, the multi-layered measures are being used to reduce the amount of groundwater itself that comes close to the building by installing the land-side impermeable walls (ice wall). Also, the stagnant water level in the building is designed to be controlled by installing the transfer system for stagnant water in the building. While reducing the groundwater

[15]Reference: "Safety evaluation guide" Partial revision of Former Nuclear Safety Commission of Japan: Regulatory Guide for Reviewing Safety Assessment of Light Water Nuclear Power Reactor Facilities, Mar 29, 2001.

inflow, the difference of the water levels in and outside the building is required to be controlled reliably in any conditions in order to prevent contaminated water from being released.

ii) Approach to the PCV repair (For details, refer to Section 4.3.2.4.)

As for alpha particle, the upper limit of the concentration relating to the internal exposure, which is inhalation in particular is stricter than other nuclides. A special attention should, therefore, be paid for the release from the gas phase in the dust form and careful study will be required for the gas phase boundaries when the fuel debris retrieval. For this reason, it is easier to cut the fuel debris underwater or with pouring water to the extent possible and control alpha particles that are transferred to the liquid phase. In such case, the study will be required to raise the PCV water levels while taking measures to stop the leakage from the PCV (water sealing). This is the one of the major objectives of the R&D for the PCV repair (water sealing).

The technical developments of the PCV repair (water sealing) are being carried out separately for the bottom part (below the torus room ceiling) and upper (above the first floor of the R/B) of the PCV. Since on-site radioactive environment is severe, the element test for the repair technology using the cement-based materials, which allow remote handling, has been carried out. According to the results of the tests, although the leakage can be controlled by the cement-based materials, it is difficult to achieve complete water seal and a certain amount of the leakage is required to be allowed. Assuming this condition, the containment functions are required to be studied even for the PCV repair. (For details, refer to Section 4.3.2.4.2)

iii) Approach to building stagnant water

As a contaminated water management described above, water flowing in the building can be controlled by lowing the stagnant water level inside the building using the installed sub-drain system, land-side impermeable walls and stagnant water transfer system inside the building while lowering the groundwater level. If the groundwater level and stagnant water level inside the building is lowered, T/B can be disconnected from the R/B. If it progresses, treatment of the water currently accumulated in the building will have been completed including the R/B. In the Roadmap, this treatment is planned to be completed by the end of 2020. For this reason, a containment system is required to be established by the time of the fuel debris retrieval work so as to prevent the leakage inside the R/B from the PCV from being leaked outside the building.

iv) Issues to be solved

Since the radionuclide including alpha particle is supposed to be transferred to the liquid phase, regardless of the fuel debris retrieval method, concentration of the radioactive material (risk =Hazard Potential) in the liquid phase will be raised. In this regard, the technical development for the PCV repair (water sealing) has been carried out to establish the containment functions for the liquid phase (boundary). However, in the case where it is difficult to achieve a complete water seal or need to be achieve the inflow of the groundwater and outflow of the contaminated water assuming the accident such as a large amount of leakage, the separation from the groundwater is very important and the studies on the containment system for the liquid phase is required including the treatment of the R/B and impermeable wall. (Refer to Figure 4.3.2-8 Conceptual diagram of boundaries)

c. Confinement system of gas phase

i) Current containment status for the gas phase

Extracting the gas in the PCV and installing the PCV gas monitoring system that extracts the gas in the PCV while deactivating the inside of the PCV by the nitrogen injection system, the gas phase inside the PCV is being maintained slightly positive. It prevents the hydrogen explosion and minimizes the amount of the radioactive material release. Consequently, the results of the evaluation for the additional release amount from the R/B of Units 1-4 during March 2016 indicated that the exposure dose at the site boundaries was 0.00087mSv/year, which is sufficient low.

ii) Issues to be solved

Although the measures are taken to transfer the radionuclide containing alpha particle to the liquid phase during the fuel debris retrieval to the extent possible, the alpha nuclide concentration is expected to be raised also in the gas phase. As for alpha particle, the upper limit of the concentration relating to the internal exposure, which is inhalation in particular is stricter than other nuclides. A special attention should, therefore, be paid for the release from the gas phase in the dust form and careful study will be required for the gas phase boundaries when the fuel debris retrieval.

Since a certain level of negative pressure is required to be maintained to ensure containment of the gas phase, negative pressure control system has been studied under the studies on various concepts for the systems used for the fuel debris retrieval method. (Refer to 4.3.2.9)

The containment functions (boundary) at the Fukushima Daiichi NPS are described in Appendix 4.19" Containment functions (boundary)"including its background.

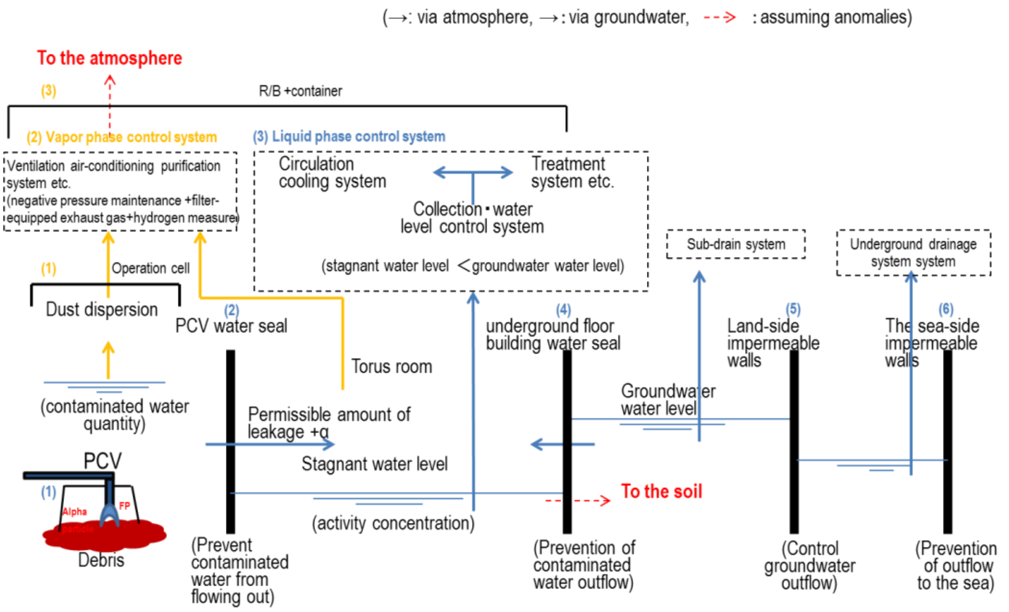


Figure 4.3.2-8 Conceptual diagram of boundaries

(4) Future actions

a. Concept of containment functions (boundary)

The concept is to be established for the containment system of liquid and gas phases which can actually be established during the fuel debris retrieval work based on its study status and the current concept for the containment functions (boundary).

b. Confinement system for the liquid phase

The following is the possible measures to reduce the risks involved in the increase in the concentration of radioactive material in the liquid phase (risk = Hazard Potential) due to the fuel debris retrieval work. (Refer to Figure 4.3.2-8 Conceptual diagram of boundaries)

i) Absorbing the cutting particles in the peripheral area when cutting the fuel debris, the concentrations of alpha particle diffused in the liquid and concentration of the radioactive materials in the contaminated water are to be suppressed so as to reduce the total Hazard Potential to the same or lower level.

ii) The leakage from the PCV is to be prevented by repairing the PCV (water sealing) and the risk is to be reduced by lowering the amount of radioactive materials that transfer to the stagnant water in the building. It is, however, considered difficult to achieve the complete water seal (no leakage).

iii) Even if the risk can be reduced by the method of i) or ii) during the normal operation, the water level of the torus room may be increased and become higher than the groundwater level when an abnormal event is caused by a large amount of leakage in the state where PCV water level is raised. In response to this, the system that lowers the water level by transferring water urgently uses a large scale pump as a countermeasure. It is, however, difficult to prevent temporal inverse of water levels depending on the assumption of outflow velocity. The method which does not raise the PCV water levels can be selected.

iv) Assessing the impact under a rare abnormal condition, the impact to the outside due to the temporal inversion of the water levels is to be confirmed small. Furthermore, risk reduction requires water sealing is to be conducted for the R/B.

v) If complete water seal cannot be achieved for the building, alternative measures are to be studied. The advantages and disadvantages of the exposure, resources, term and interference with other constructions involved in the works are to be evaluated comprehensively while assessing the feasibility of each risk reduction measures and its effect. By do in so, application of the containment system for the liquid phase will be evaluated.

c. Confinement system for gas phase

The cells are to be installed in the upper part of the PCV to minimize the impact to the outside even if alpha nuclide concentration is increased in the gas phase during the fuel debris retrieval work. Also the studies should be conducted for the containment system in the gas phase (primary boundary) including the installation of air conditioning system (secondary boundary) that to control negative pressure inside by installing the container in the R/B. In this regard, estimating the dispersion rate of alpha particle, the impact to the outside is to be evaluated such as by the performance of the filters used in the air conditioning system and leakage rates of primary/secondary boundaries. The evaluation is required to be conducted not only for the normal work but also for possible abnormal event. (Refer to Figure 4.3.2-8 Conceptual diagram of boundaries)

These studies are to be conducted in the conceptual studies of the systems relating to the fuel debris retrieval method. (Refer to 4.3.2.9)

4.3.2.4.2 Construction of boundary (e.g. PCV repair)

(1) Purpose

Based on the boundaries described in the previous section, the PCV, which is the first boundary is required to be repaired to the extent possible in order to prevent the impact to the residents and environment by confining the radionuclides including alpha nuclide caused by the work and minimizing and controlling its release during the fuel debris retrieval work for a long period of time (during the normal operation and abnormal operation). It is, however, difficult to achieve a complete water seal since the cement-based materials will be used for the PCV repair technology which is currently under the development. The establishment of the containment system that allows a certain amount of the leakage should be studied as well.

(2) Major requirements

a. The actual unit with a margin and secure long-term stable water seal from the perspective of ensuring safety is to be applied and regulations relating to the construction work are to be prepared including the construction periods for the fuel debris retrieval.

b. The reliability of water sealing is ensured including the monitoring of the PCV repair work, inspection method after the construction, detection in case of leakage and establishment of the re-repair method

c. The development of the liquid phase management (e.g. containment and criticality prevention) system that generally prevents the leakage to the outside is studied.

(3) Action status and evaluations and issues

The leakages currently identified are the sand cushion drain line at the bottom of the PCV, bellows on the vacuum break line and bellows of the PCV penetrations for MS line. The possible locations of leakage includes suppression chamber, which is shown in Figure 4.3.2-13. Also, the accident event this time is evaluated since the inspection of the upper part of the PCV is less advanced due to the impact of high radiation. The Figures 4.3.2-15 -4.3.2-17 show the possible locations of damage for each unit. Each Unit has about 300 areas, which is quite a few. The areas where boundaries are established are indicated by the bold red lines in Figure 4.3.2-14. Since the leakage may be occurred at the bottom from the suppression chamber connectors of Unit 2, the vent pipes or downcomers located upstream are planned to be closed. Also, the penetrations in the upper part where the leakage are expected to be occurred are also planned to be closed. It is, however, difficult to repair by welding, since the repair areas are of very high degree of difficulty in accessing due to a high radiation and obstacles. Therefore, the grout will be placed in the area where the leakages are occurred using remote-controlled equipment in order to close the leakage. (Under the technical development) However, if the area where welding is considered applicable is found as the on-site inspection progresses, the repair will basically be performed by welding.

The technical developments are being carried out separately for the liquid phase for the bottom part (the torus room ceiling or lower) which is submerged and upper part of the vapor phase (First floor of the R/B or higher) which is barely submerged.

The major issues additionally raised in the technical development last fiscal year are described below.

a. Issues to secure water seal

i) Issues to repair PCV bottom (under torus room ceiling)

1) Water sealing for vent pipes:

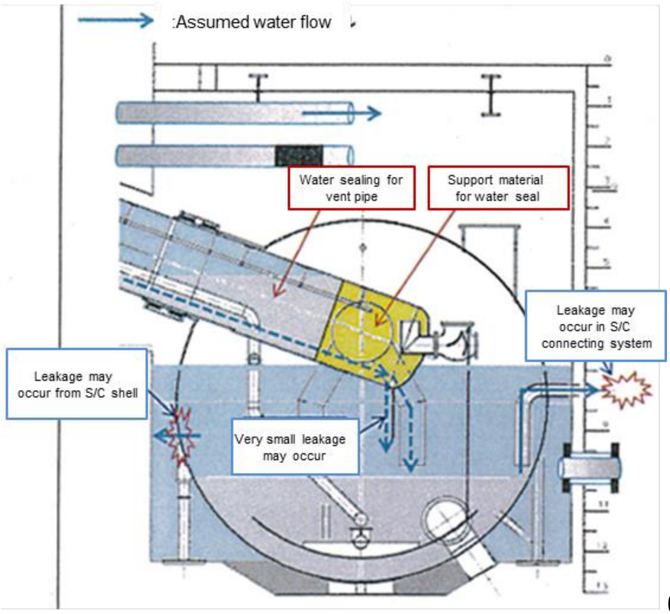
✓ Measures to prevent grout from being flown to the downstream of the vent pipes

Since the water sealing is applied in the middle part, inflatable seals and supporting material for water sealing are currently being developed so as to install staunches. The grout will be injected after blowing up the inflatable seals by the inflatable seal method. The prospect for the technical development is to be confirmed within FY 2016 for the resistance of inflatable seal fabrics (e.g. tension of the suspended portion caused by the weight of the grout), initial setting at the bottom of the vent pipe (on the vent header) to blow the inflatable seal evenly, and the its supporting materials which are packing materials to control expansions and gaps left eventually (including the gaps caused by the obstacles).

Also, zero leakage is aimed under the condition of the water level up to the torus room ceiling as the first objective in the development, and the limit of the water sealing (upper limit of the PCV water filling) is to be confirmed by the test while studying the changes in the water sealing materials including the measures in (2) described below.

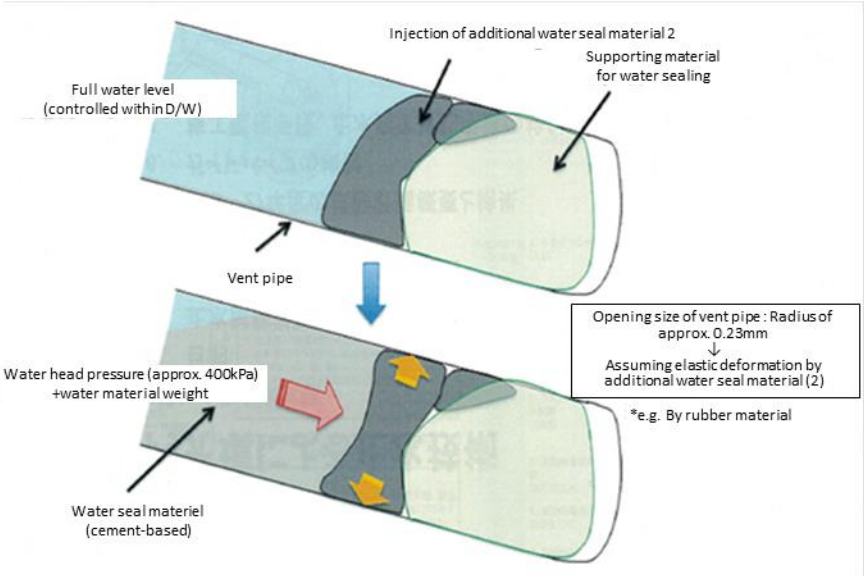
✓ Measures to vent pipe swelling

The development is to be confirmed within FY2016 for the addition repairing materials considering the development and re-repairing of water sealing materials such as rubber materials that conform with the deformation by the pressure so as to prevent the bleeding channel because of that the vent pipe by the increase in the hydraulic pressure by submersion is blown up and water sealing materials come off from the inside of the vent pipe.



(Provided by IRID)

Figure 4.3.2-9 Diagram of water sealing for vent pipes Part 1



(Provided by IRID)

Figure 4.3.2-10 Diagram of measures to the vent-pipe expansion Part 2

2) Water sealing by pouring grout for S/C:

✓ Water sealing for downcomers

Balancing the water levels of the PCV and S/C by installing the guide pipes on the S/C side, the development of the technology to stop the water flow is to be confirmed within FY2016, since it is difficult to implement the repair under the cooling water passing state (hydraulic pressure for the downcomers). However the leakage areas of S/C are not identified for Unit 2 under the current inspection. Although it is appropriate to provide treatment of water sealing before blocking water of the downcomers, if leakage areas are located at the bottom such as expected ECCS systems, if the leakages are located in the upper part of the S/C, effect by the guide pipe cannot be expected. Therefore, the measures such as with pouring water are required to be studied.

✓ Water sealing for vacuum breaker

Since hydraulic pressure will be applied to the vacuum breaker from the vent pipes by the allowable leakage water after the water sealing for vent pipes, there may be some portions in the upper part of the vacuum breaker where the amount of grout applied is insufficient. The development of the water sealing materials is to be confirmed within FY2016.

✓ Impact to the poring of the grout

Impact to the S/C integrity caused by the increase in the load to the S/C by the burial grout is to be confirmed in collaboration with other R&D project, RPV/PCV aseismic performance and impact assessment technical development project within FY2016.

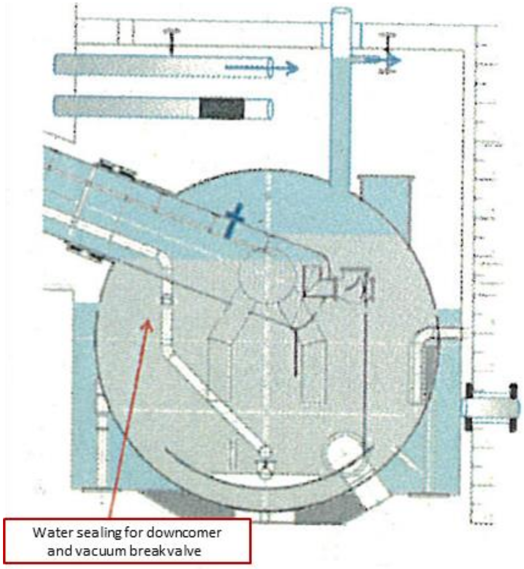


Figure 4.3.2-11 Water sealing for downcomers (Provided by IRID)

ii) Issues on mortar injection to the torus room

1) SC supports reinforcement

- ✓ Mortar reinforcement height

Since a margin for the SC support could not secured by the seismic evaluation conducted for other R&D project (RPV/PCV's aseismic performance and impact assessment technical development), technique to provide reinforcement by applying mortar to the torus room. Therefore, the relationship of current groundwater level > stagnant water level of the torus room may be reversed depending on the height of the mortar applied to the torus room (pin connection points of the column supports in the current plan). No inleak will be occurred and stagnant water may be flown outside the system. Although coordination with the treatment measures to the stagnant water will be required, the measures without reinforcement and measures that will not cause the leakage to the outside the system will be studied within FY2016 in collaboration with other R&D project (RPV/PCV's aseismic performance and impact assessment technical development).

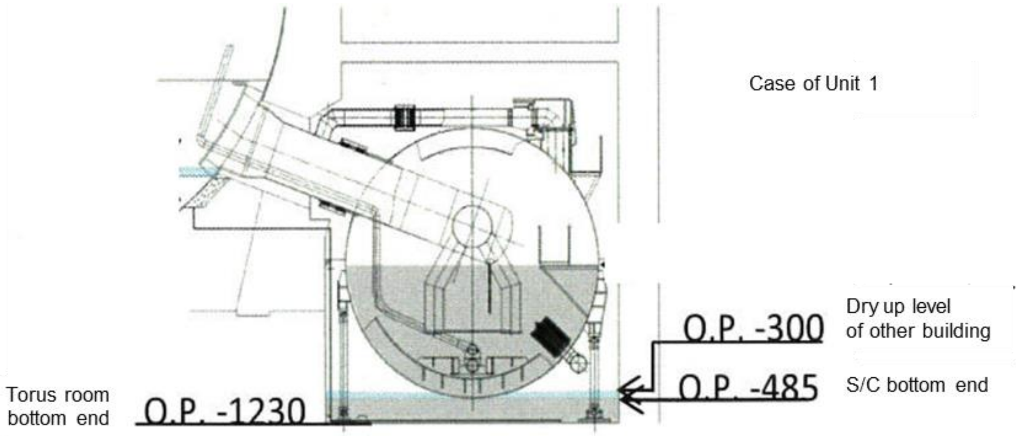


Figure 4.3.2-12 Layout drawing of torus room (Provided by IRID)

Note: To dry up the surrounding buildings, the level of groundwater water needs to be under OP-300mmand fixed water level of torus room to maintain the stagnant water level in the torus room needs to be within the range of OP-1100mm-1300mm considering the margin (approx. 800-1000m) for water level differences control. As described above, the lower end of the torus room of Unit 1 was OP-1230mm. It is very close to the control limit even without mortar, and mortar cannot be placed; therefore it is difficult to be reinforced.

iii) Issues on repair of upper part of the PCV (First floor of the R/B or higher)

The element test will mainly be conducted since inspection has not progressed sufficiently due to high radiation.

Since the difficulties may be involved in the reduction of radiation dose, remote-controlled repair is required to be studied also for the upper part.

1) D/W penetration water sealing

Although the water sealing materials were considered to be applied for the areas of leakage hydraulic at first, it was found that the spray coating was not resistant. To resist the hydraulic pressure, injection operation will be required. Hence, dikes will need to be established. It is, however, not realistic if considering the on-site remote handling and the study on the water filling level of the PCV.

2) Water sealing for equipment hatch

The water seal was studied and it was found that it cannot resist the load up to the full water level of the well. Considering cases where groove welding is difficult, the application of materials with excellent radiation resistance and water-sealing will be studied in parallel with filling the PCV with water.

iv) Water sealing for penetrations on the wall of torus room

Although there is a possibility of conducting the water sealing by injection operation, it is not realistic that the person approaches to the location to build dikes required for the injection operation if considering the remote handling on the site. Accesses from the T/B and Rw/B need to be studied.

b. Issues on repair reliability

i) Impact to the inner surface of the S/C, torus room, and piping (e.g. vent pipes, downcomers)

Conditions of the inner surface are yet to be identified. The impact to the water seal needs to be tested, given the on-site application.

ii) Long-term water sealing

Progression test for the bleeding channel is required to be planned to evaluate the changes in the portions where water leak is prevented such as by the hydraulic pressure, corrosion and earthquake, expansion of bleeding channel. Also technical development which enables leakage detection method and re-repairing (e.g. addition repairing materials and injection position) is also required to be planned.

iii) Constructability

Technical development which enables monitoring of the construction progress status and construction completion status is required to be planned. The plan and evaluation of reproducibility test, plan for the implementation method for the assurance test (e.g. leak test, inspection) and studies on the evaluation method will be required.

c. Issues on establishment of control system for liquid phase

Figure 4.3.2-14 shows the results of the study on the concept of the control system for the liquid phase assuming the leakage at the repair areas. It includes a wide variety of issues, such as cooling of the fuel debris, criticality prevention, corrosion control for reactor internals, PCV water level monitoring and control. As a major issue, it is required to determine the water intake point of the PCV, and inspect and control the water levels of the PCV, S/C, torus room and groundwater. The leak out from the system needs to be prevented while maintaining the relationship of groundwater level >torus room water level and the functions which are complex and require a quick response are to be realized.

(4) Future actions

The following items are to be performed so as to make final judgement on the application of the PCV repair to the actual unit in summer of FY 2017.

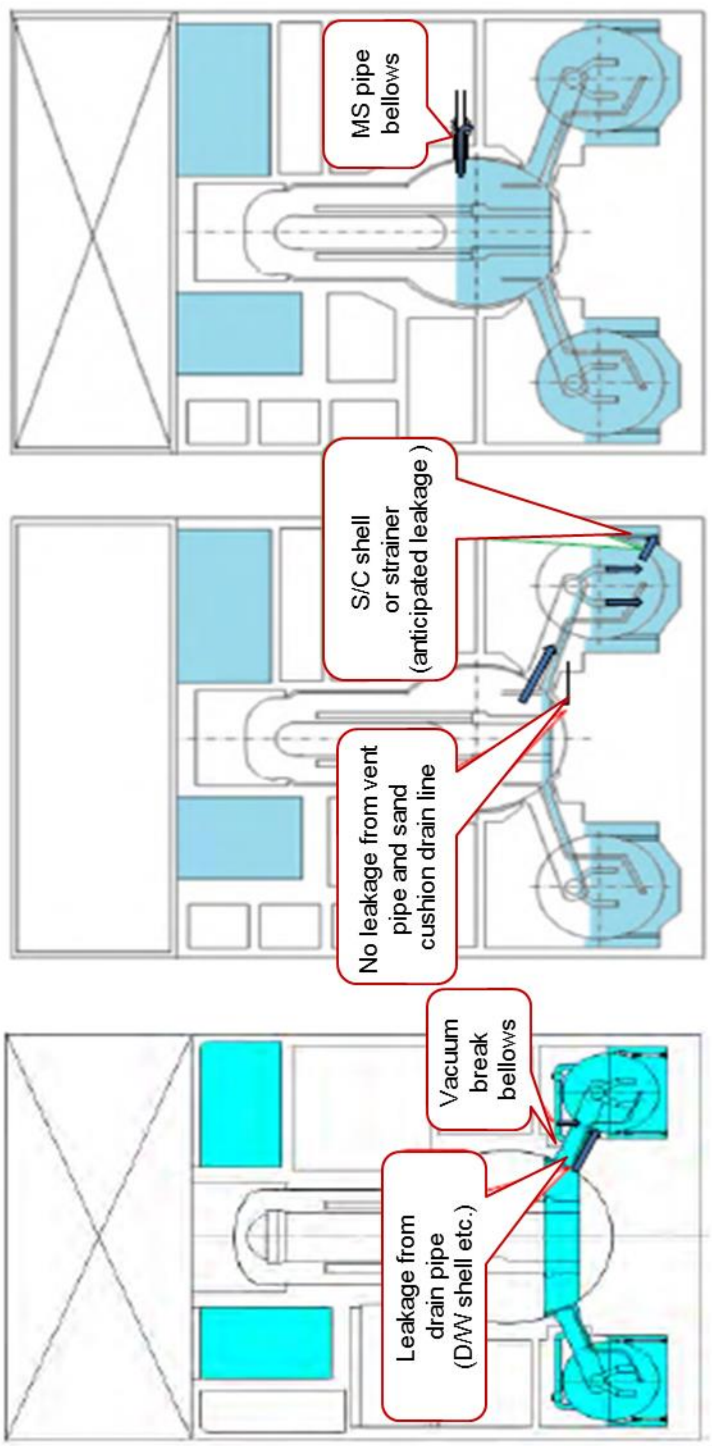
a. Water seal is to be ensured focusing on the resolution of many issues previously described.

In the technical development for the fuel debris retrieval, the control system in the gas phase is to be established to maintain negative pressure inside the PCV from the perspective of prevention of radioactive dust scattering during the fuel debris retrieval. A reasonable repair method for the area in the upper part of the PCV, which are estimated to be damaged, is to be studied considering the prevention of gas leakage.

b. Focusing on the following four items, repair reliability as well as the long-term water seal is to be ensured.
   i) Verification of water sealing performance simulating the conditions of the actual unit (rust and slippage characteristics on the inner surface) and reproducibility and feasibility of re-repairing
   ii) Long-term water seal under a high water pressure including aseismic performance (evaluation such as of bleeding channel development)
   iii) Constructability including monitoring technologies (construction progress and completion verification)
   iv) Studies on the on-site construction considering radiation dose reduction (e.g. decontamination, removal, shielding)

c. Establishment of liquid phase control system

As described above, a complete water seal is very difficult to be achieved by the repair using grout and a certain amount of leakage to the torus rooms needs to be allowed. As described so far, it is important to control the differences in the water levels between the inside and outside(inleak control) in order to keep the water level inside the torus room lower than the groundwater level, currently performed in case of a large amount of leakage at the time of fuel debris retrieval. Also, separation from the groundwater is the most important to prevent the inflow of the groundwater and contaminated water leakage. The study on the control system in liquid phase is required including the water seal for the buildings and outer water shield in case of a large amount of leakage.



(Provided by IRID)

Figure 4.3.2-13 Leak locations currently identified

・Studying the structure that satisfies the basic functions of PCV circulation cooling system, the conceptual system drawing was planed.

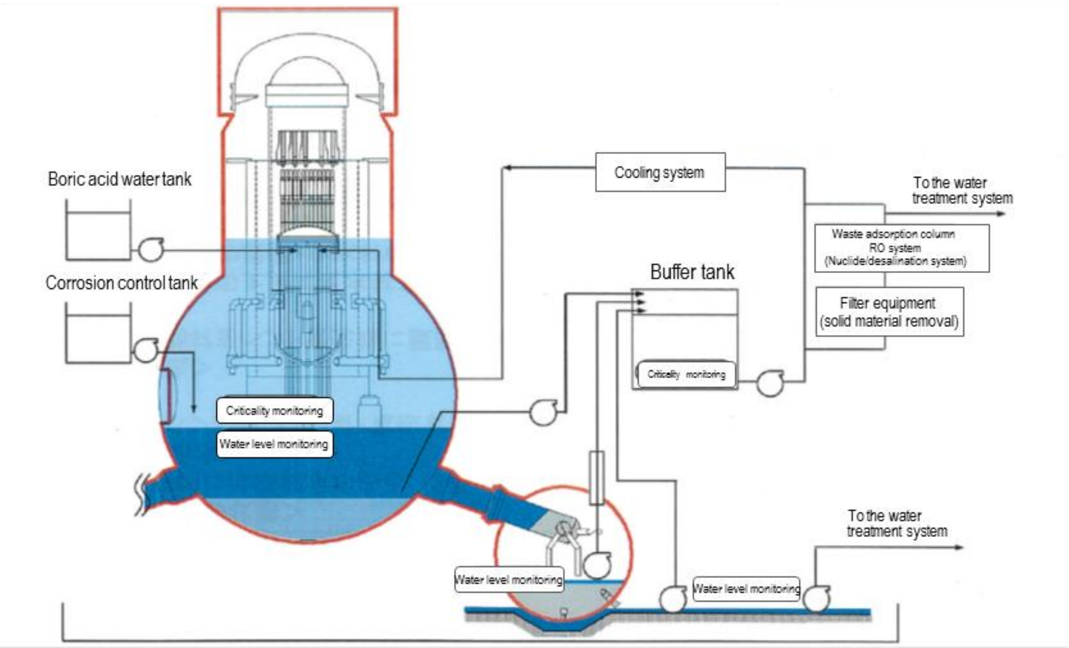
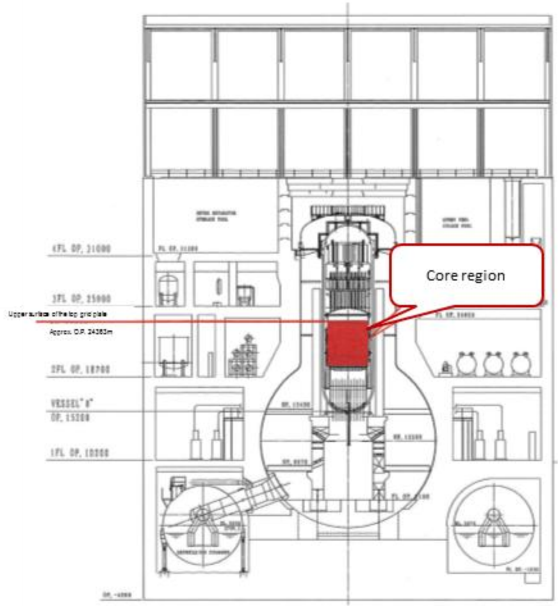


Figure 4.3.2-14 Conceptual diagram of boundaries currently planned (red bold line) and control system in liquid phase (Provided by IRID)

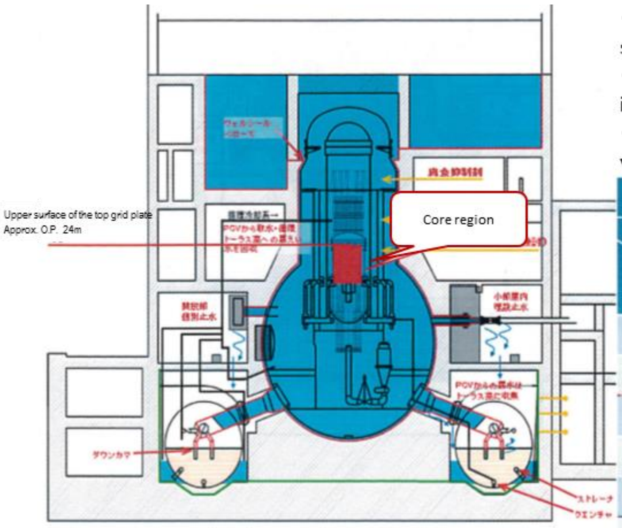


・Number of penetrations required to submerge the core region and isolation valves
・Top end level of upper grid plate is O.P.24363 and it is close to the ceiling of 2nd floor.
・Therefore, the penetrations of 2nd floor and isolation valves are required to be repaired.

Penetrations (including partial PCV body)							Isolation valve
Evaluation / Floor	○	△※		×			
		Earthquake	Molten Fuel	Corrosion	Hyperthermia (seal)	Hyperthermia (elec. penetration)	
4th	0	0	0	1	0	0	0
3rd	0	1	0	2	8	0	30
2nd	4	0	0	7	0	0	74
1st	19	15	0	8	7	17	386
Basement	36	13	3	16	0	0	146
Total	157						636

*△ by hyperthermia of PCV sell was excluded since it will relates to the all penetrations of D/W
Note: Number of isolation value indicates the number that closed at the time of accident.

Figure 4.3.2-15 Possible damage area of each floor (Unit 1) (reference 1) (Provided by IRID)



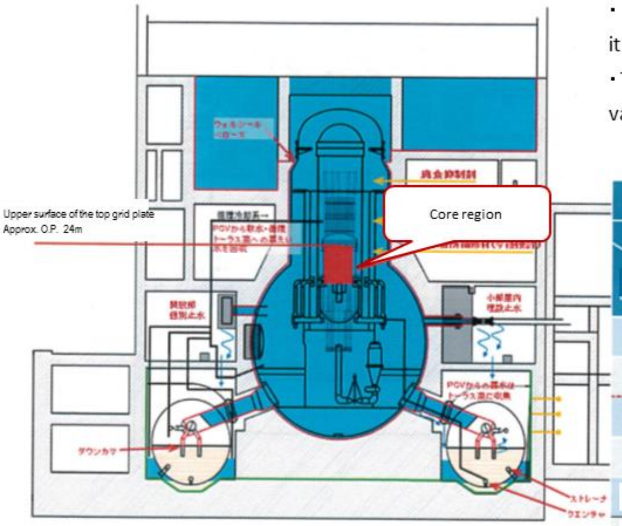
・Number of penetrations required to submerge the core region and isolation valves
・ Top end level of upper grid plate is O.P.24 and it is located at approx. 5m from the 2nd floor.
・Therefore, the penetrations of 2nd floor and isolation valves are required to be repaired.

Floor \ Evaluation	Penetrations (including partial PCV body): ○	△※ Earthquake	△※ Molten Fuel	× Corrosion	× Hyperthermia (seal)	× Hyperthermia (elec. penetration)	Isolation valve
4th	0	0	0	0	0	0	0
3rd	8	3	0	1	0	0	16
2nd	2	3	0	3	0	0	42
1st	35	13	0	12	0	16	204
Basement	17	49	2	8	0	0	157
Total	172						419

(Provided by IRID)

Note: Number of isolation value indicates the number of the valve up to the second isolation vale

Figure 4.3.2-16 Possible damage area of each floor (Unit 2) (reference 2)



・Number of penetrations required to submerge the core region and isolation valves
・ Top end level of upper grid plate is O.P.24 and it is located at approx. 5m from the 2nd floor.
・Therefore, the penetrations of 2nd floor and isolation valves are required to be repaired.

Floor \ Evaluation	Penetrations (including partial PCV body): ○	△※ Earthquake	△※ Molten Fuel	× Corrosion	× Hyperthermia (seal)	× Hyperthermia (elec. penetration)	Isolation valve
4th	0	0	0	0	0	0	0
3rd	0	3	0	1	8	0	21
2nd	2	2	0	3	0	0	43
1st	19	14	0	12	10	22	231
Basement	18	48	2	8	0	0	149
Total	172						444

Note: Number of isolation value indicates the number of the valve up to the second isolation vale

(Provided by IRID)

Figure 4.3.2-17 Possible damage area of each floor (Unit 3) (reference 3)

4.3.2.5 Reduction of worker's exposure during operation

The inside of the R/B is under an extremely high radiation environment because of the impact of the contamination caused by the accident. Since the fuel debris retrieval work and its related works are performed mainly in the R/B, it is not an exaggeration to say that the reduction of the workers' exposure determines the implementation of the work. Although the measures for the dose reduction are taken generally from the perspective of "time, distance and shielding," those are the measures when radiation source cannot be changed. Decontamination (broad decontamination including removal of radiation source) that removes radiation source itself is also important measures at the Fukushima Daiichi NPS.

Also, utilization of remote technology is highly anticipated as measures to secure the distance. The reduction of radiation dose by the appropriate combinations of these measures is what we should aim for. For this reason, the studies on the decontamination are to be conducted to reduce the workers′ exposure in the R/B based on the following basic concepts.

Following the studies focusing on the dose reduction utilizing the remote technologies and combinations of decontamination measures, the radiation exposure control during the work is planned from the perspective of "time, distance and shielding."

- The works are carried out by remote technologies without any access made by human being under extremely high radiation area such as the inside of the PCV and torus room.
- In order to achieve a low cumulative dose for a whole work considering the exposures relating to the decontamination and to the PCV repair works, the optimum combinations such as by the decontamination, shielding, remote technology, and shortening of operation time are to be studied for the inside of the R/B other than those described above (management based on the dose budget).
- The evaluation and studies are to be conducted assuming the installation work of the facilities, maintenance work and response to the trouble will be necessary, even if the remote technologies are utilized.
- The decontamination work, either by remote technologies or manual operation will be determined by the radiation dose rate of the target areas, contamination form, operating space, frequency of use, applicability and development trend of the remote technologies.
- The study is to be conducted on the areas where the work needs is clear. The studies for the areas where the needs are not clear or for the betterment, the reduction of overall radiation dose will not be carried out.

The major issues in the dose reduction during the works related to the fuel debris retrieval are the decontamination of the inside of the R/B related to the preparation work and the shielding used when carrying out the fuel debris retrieval work itself. Those two issues are described below.

4.3.2.5.1 Decontamination of the inside of the R/B

(1) Purpose

Exposure to the workers for the PCV internal survey, PCV repair and preparation for the fuel debris retrieval is to be reduced by the decontamination for the work areas and access paths (shielding and removal of radiation sources).

(2) Major requirements

a. Contamination condition investigation

Considering the needs of the PCV internal survey and PCV repair work, If the data obtained by the inspection conducted to date is not enough, the inspections are to be conducted for contamination situation (e.g. contamination form, contamination distribution and decontamination objects).

b. Radiation dose reduction plan

To secure a required work environment for the target areas using the appropriate radiation dose reduction technologies (decontamination, removal and shielding), the radiation dose reduction plan is to be formulated considering the contamination situation.

The target dose rate of the work areas is to be set so as to fall below the exposure dose limit (50mSv/year and 100mSv/five years) specified by the law though the studies on the work method, operation time, the number of workers. Also, the plan is to be made considering the arrangement of the environment to use decontamination equipment.

c. Radiation dose reduction technology

Information regarding the radiation dose reduction technologies is to be revised as necessary.

(3) Action status and evaluations and issues

a. Contamination condition investigation

- The level of radiation dose inside the R/B was lowered in some areas by decontamination work but still high in most areas. Consequently, the PCV internal survey and inspection in preparation for the PCV repair are not sufficiently conducted. The radiation dose level at the 1st floor of the R/B for each Unit is shown in Figure 4.3.2-18.
- To formulate a radiation dose reduction plan, characterization such as contamination form and contamination penetration depth of target decontamination areas is required to be identified; however, it is difficult to carry out a sampling due to a high radiation.
- In the peripheral area of X-6 penetration (PCV penetration for carrying in and out of the control rod drive (CDR)) on the 1st floor of the R/B of Unit 2, the leakages from the flange seals and highly concentrated contaminants on the floor was found by removing the shielding blocks. After removing the leaked materials from the flange, cleaning and grinding were carried out for the floor surface but the radiation dose was not reduced to the target level. Furthermore, the issues has been raised such as for the understanding of the contamination form and dust scattering prevention since the work environment was deteriorated due to the dust generated by the grinding; therefore the countermeasures are currently being studied. (Refer to Figure 4.3.2-19) Also, since the detailed preliminary inspection was yet to be carried out for the installation structures of shielding blocks and information regarding the precise site conditions was not sufficiently shared, it took time to remove the shielding blocks.
- AC pipe (Atmospheric Control piping) on the 1st floor of the R/B of Unit 1 has been estimated to be a radiation source since the high radiation steam that attached inside the pipe during the S/C ventilation. Also, since DHC pipe (dehumidification system piping for drywell) is connecting with RCW piping (reactor component cooling water system piping) whose whole system is highly contaminated by radioactive materials, the water contained in the pipe is estimated to be a radiation source. To perform the PCV internal survey from X-6, the radiation source is required to be removed to reduce radiation dose.
- During the inspection of contamination distribution on the operating floor of Unit 3, the spectrum, other than radiation dose was measured by collimation. In addition to the detection of nuclides, now the

estimation of the locations of radiation source became possible and the rage to be decontaminated was identified.

b. radiation dose reduction plan

- It is difficult to identify the contamination situation for the areas with high radiation level. Although the decontamination plan will be studied and formulated based on the estimation, it will lead to a delay in its process if the situation becomes clearer after the planning and is found to be severer than expected.
- It is difficult to carry out the works effectively since important facilities and peripheral equipment interface with decontamination work.
- The reduction of radiation dose on the 1st floor of the R/B is contributed to the ducts such as in the narrow area and mid and high places instead of the floors at this point. On the 1st floor of Unit 2, the effect of the radiation dose reduction by decontamination and removal of the ducts has been achieved and decontamination of mid and high places is important.
- Although manual work is be required depending on the target work areas, it will be necessary as a supporting work even in the areas where remote-controlled equipment is used and radiation workers' exposure during the decontamination work is increasing.

c. Radiation dose reduction technology

- Remote-controlled equipment for decontamination inside the R/B (for higher part of 1st floor and upper floors) was developed as R&D (Dry ice blast, Suction and blast, High pressure water jet) and mockup test was conducted (Figure 4.3.2-20 Reference).
- In Unit 3, verifying the actual unit of dry ice blast decontamination equipment for high places, target decontamination performance was achieved by combining suction + dry-ice blast but the issues such as the application for the unevenness of the wall were found.
- Information regarding the radiation dose reduction technologies has been collected through the tendering and published as a technical catalogue. However, since technical findings have been obtained from a result of the actual unit inspection for the remote decontamination equipment for the high places manufactured by the subsequent R&D, it is important to collect those findings and make them utilizable.

(4) Future actions

a. Contamination condition investigation

- The required inspections will be performed after the work environment for the PCV internal survey and PCV repair are identified.
- For the R/B of Unit 1, cleaning of the internal surface of AC pipe and removal of water contained in DHC pipe will be studied.
- Sampling method for high radiation is to be studied considering the reduction of the workers' exposure.
- With regard to the inspection of contamination distribution, the introduction of the method such as spectral measurement with collimator, which can detect the locations of nuclide and radiation source to the inside of the R/B is to be studied.

b. Radiation dose reduction plan

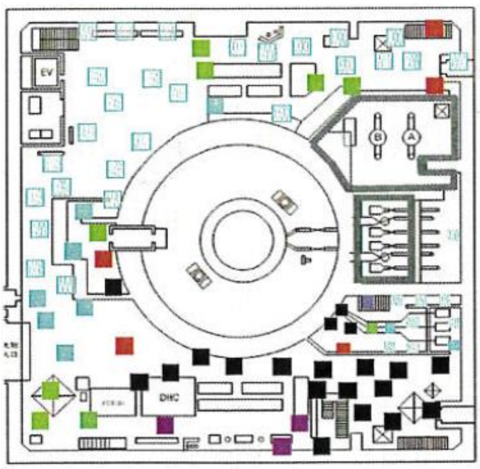
- The target work areas such as of the PCV repair is planned for the representative points considering the construction volume, commencement period, access by people (inspection for plant maintenance/management facilities). After that, a radiation dose reduction plan for the target areas is to be studied through the horizontal development.
- Although it is difficult to understand the contamination form appropriately to the extent possible, the optimum decontamination method is to be selected. Also, it is necessary to study multiple radiation dose reduction plans corresponding to the possible contamination form while securing the required workforce. Even if the situation is found to be different from the assumption, the delay in the process is to be minimized by establishing the system that corresponds to the site situation flexibly.
- Since the spread of contamination by dust and decontamination residue due to the decontamination work are concerned, measure is to be taken on the area management, prevention of radioactive dust scattering and prevention of recontamination.
- In the decontamination for the areas with high radiation level, exposure dose during the decontamination work should fall below the upper limit by using remote-controlled equipment instead of manual work.
- Feedback on the decontamination efficiency, exposure dose and process obtained from the actual unit is to be provided to be reflected to the future radiation dose reduction plan.

c. Radiation dose reduction technology

- The application of the remote-controlled decontamination equipment developed as R&D to the project related to the PCV repair is to be studied. Also, since manual works may be required when collecting the equipment in case of emergency, the measures will be required for the reduction of the radiation workers' exposure according to the site conditions.
- The data on the current decontamination technology and remote technology and information on the results of the practical application is to be added and updated as appropriate. The data base for the technical findings obtained through the actual application at the Fukushima Daiichi NPS is to be established to contribute to the solutions of the issues to be caused during the dose reduction work and R&D so as to be utilized for the decommissioning work over a long period of time.

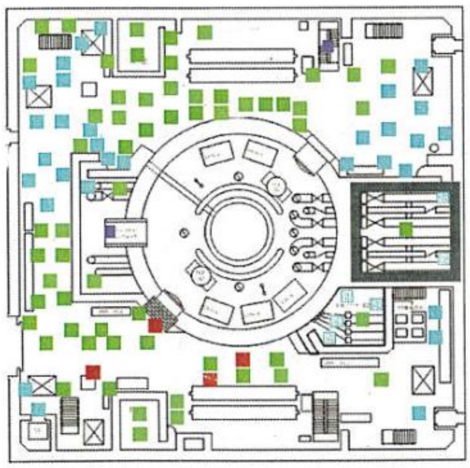
Although decontamination for the work environment to perform the PCV internal survey and PCV repair has been carried out, it is not exactly that the radiation dose reduction for the target has been carried out as planned. The appropriate decontamination is to be carried out while confirming its method and effectiveness. Assuming that there is a limit for the radiation dose reduction in the target work area, it is important to review the method considering the exposure dose involved in the repair work and effectiveness of the range and method of the PCV repair.

Unit 1



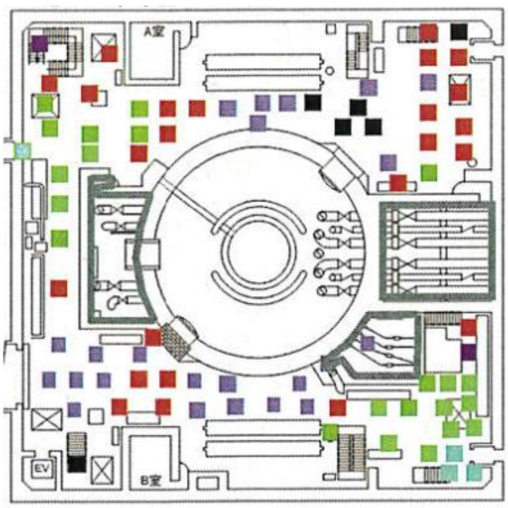
Decontamination performance
(1) Removal of rubbles
(2) Cable arrangement, removing equipment
(3) Decontamination for floor surface and mid and high places
(4) Hot spot shielding

Unit 2



Decontamination performance
(1) Decontamination for floor surface and mid and high places
(2) Equipment removal
(3) Hot spot shielding
(4) Duct decontamination removal

Unit 3



Decontamination performance
(1) Removal of rubbles
(2) Cable arrangement, removing equipment
(3) Decontamination for floor surface and mid and high places

Legend of map radiation dose

: <3mSv/h
: <5mSv/h
: <7mSv/h
: <10mSv/h
: >10mSv/h
: >20mSv/h
: >50mSv/h

(as of Jan. 2016)

Figure 4.3.2-18 Airborne radiation on the 1st floor of the R/B 1500mm from the floor

(Provided by TEPCO)

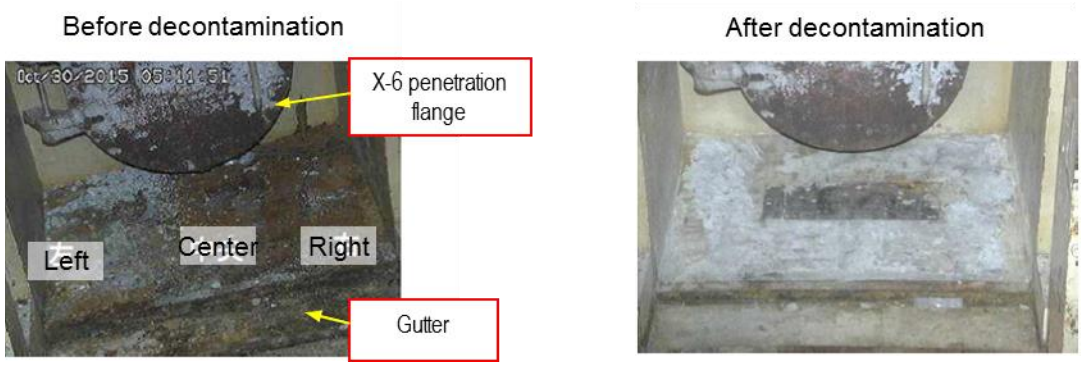


Figure 4.3.2-19 Decontamination states in the periphery of Unit 2 X-6 penetration (Provided by TEPCO)

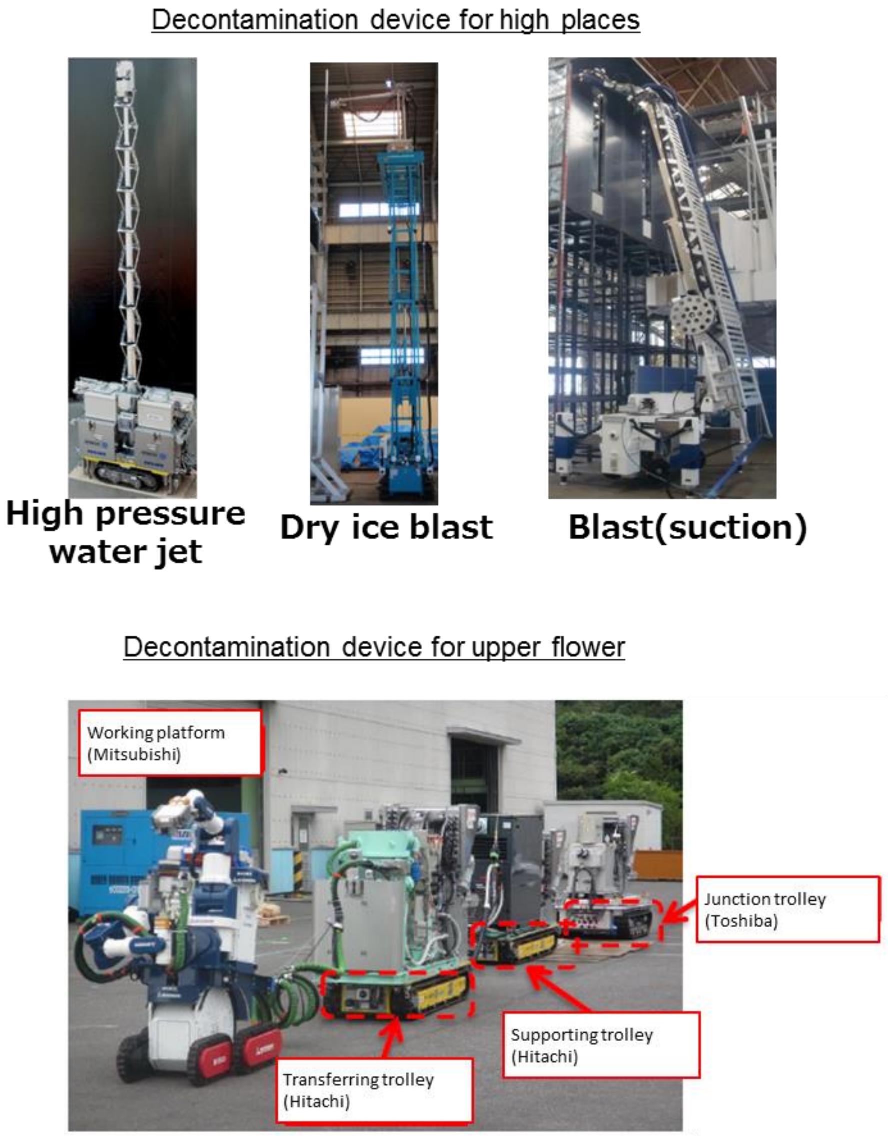


Figure 4.3.2-20 Decontamination equipment for higher places and upper stairs (Provided by IRID)

4.3.2.5.2 Exposure reduction by shielding when retrieving fuel debris

(1) Purpose

Exposure dose in the work areas during the fuel debris retrieval work and the site boundaries is to be reduced as low as reasonably achievable and radiation safety environment is to be maintained.

(2) Major requirements

a. Considering the water shielding effect for the water filling of the PCV based on the investigation and study results of the distribution of the radioactive source such as fuel debris, FP and activation metals, the work area and radiation dose rate at the site boundaries corresponding to the work conditions during the fuel debris retrieval is to be evaluated and appropriate measures such as shielding should be taken.

The system equipment containing radioactive materials, which are radioactive material collection and processing /treatment system, should also be considered in the radiation sources.

b. Appropriate measures should be taken for the scattering prevention and control based on the evaluation of the exposure dose in the work areas and site boundaries which are attributable to scattering of radioactive dust attached to the structures due to the removal and transferring the upper part reactor internals and radioactive dust caused by the fuel debris cutting.

(3) Action status and evaluations and issues

a. A simple evaluation of the water shielding effect on the operating floor and required shielding thickness of the cells was performed (Refer to Figure 4.3.2-21) for the radiation dose, attributable to the radiation dose attributable to the reactor internals and fuel debris, reactor internals and fuel debris. The feasibility of the shielding which can achieve the radiation dose rate on the operating floor of 1mSv/h was obtained even if assuming all fuel debris are located in the core region.

b. With regard to evaluation of radioactive dust generated during the fuel debris cutting, aiming at the laser cutting as a representative of the thermal cutting methods which is relatively likely to generate radioactive dust in the candidate fuel debris cutting method, measurement test is being planned for the properties and amount of dust generated by the cutting using the specimen simulating the mechanical and thermal properties of fuel debris.

To evaluate the radiation dose of the generated radioactive dust, collection of information on the chemical composition of the fuel debris, which will be a radiation source, other than the mechanical properties will be the issue.

Although the composition of fuel debris obtained from the severe accident progression analysis will be the reference at this point, the improvement of the accuracy of information by collecting the actual unit information such as by sampling of the fuel debris will be the issue.

(4) Future actions

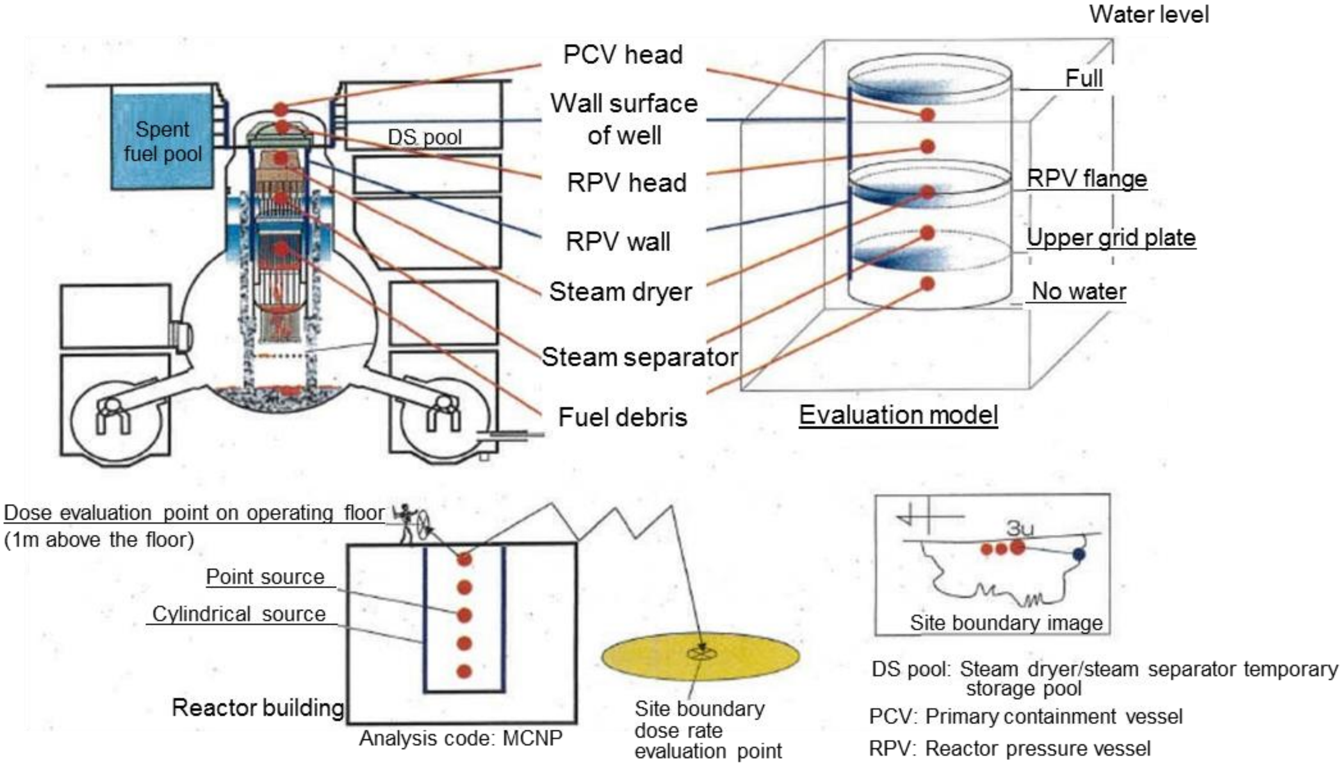
a. The accuracy of information on the fuel debris for each Unit, radioactive source, such as FP distribution is to be improved based on the results of the investigation of conditions inside the actual reactor such as the result of the severe accident progression analysis and PCV internal survey. Also, the detailed dose evaluation is to be conducted based on the requirements on the PCV water level which depending on the methods and reasonable shielding is planned to be designed while studying the required shielding specifications of each Unit for each method to confirm the feasibility.

Dose evaluation on the operating floor is to be studied considering the impacts of radiation dose attributable to the FPs remained in the R/B after the implementation of the predetermined decontamination in accordance with the situation.

b. Improving the accuracy of the evaluation for the amount of absorption of radioactive dust in the upper part reactor internals through the studies on the mechanism of Cs that absorbs the equipment, the understanding of the amount of absorbing by investigation of conditions inside the actual reactor, measures to the decontamination performed in advance of the these transferring and removal.

A measurement test for the fuel debris cutting methods, other than laser cutting is to be planned for the properties and generated amount of the dust caused by the cutting after studying its necessity based on the findings from the past.

The evaluation results of the dust properties and amount generated by each cutting method are to be reflected to plans of the air conditioning system to control negative pressure for the R/B and PCV and filter designs.



[Dose evaluation model]

Maximum dose when equipment is removed

Operating floor dose rate (Sv/h)	Fuel debris *1	0.9
	Cs	6.6
	Co	14.3
	Total	21.9 *2

Can be reduced to less than 1mSv/h with approx. 30cm steel shielding.

*1 The dose rate for fuel debris is calculated considering the self-shielding with the average core burnup of the Units 1 to 3.
*2 Rounded off to the first decimal point
*3 Includes Co

Maximum dose when equipment is removed considering the work steps (Debris source*1 and Cs sources are considered)

	Water level	Max. dose
Operating floor dose rate (Sv/h)	Fully filled	0.28
	Partial submersion	6.6
	No water	7.6 (21.9)*3
Site boundary dose rate (Sv/h)	Fully filled	0.2
	Partial submersion	3.3
	No water	3.6

Effect of Cs adhering to the wall surface of the well Ref.3
Fuel debris submerged Ref.1&2
Fuel debris exposed Ref.4
Can be reduced to less than 1mSv/h with approx. 25cm steel shielding.
Can be reduced to less than 1mSv/yr with approx. 8cm steel shielding.

Maximum dose when components are removed with due consideration of decay

		10 yrs later	20 yrs later	30 yrs later
Operating floor dose rate (Sv/h)	Fuel debris	0.9	0.5	0.3
	Cs	6.6	5.3	4.2
	Co	14.3	3.8	1.0
	Total (Thickness of shielding required to reduce to 1mSV/h)	21.9*2 (approx. 30 cm)	9.6 (approx. 27 cm)	5.5 (approx. 24 cm)

Ref.5

■ The evaluation results contain following conservative points;
- The radiation source intensity of fuel debris is calculated assuming that all nuclides remain except Cs and noble gases. If semi-volatile nuclides is dissolved, the intensity will be about 60%. Increased effect of shielding in the event the fuel debris flowed out of the pedestal is not considered.
- The point source is used for the calculation and the self-shielding effect of the structure in the PCV which is the radiation source is not considered. The fuel debris shape is assumed to be cylindrical although it is unknown. The source intensity of the fuel debris is supposed to be 0.055 times considering the self-shielding effect.

[Results of dose evaluation]

Figure 4.3.2-21 (Reference) Example of dose rate evaluation during the fuel debris retrieval (Provided by TEPCO)

4.3.2.6 Ensuring work safety

(1) Purpose

Most of the operations planned in advance of and during the fuel debris retrieval work will be carried out inside the R/B. The work environment of the R/B is in extremely severe, such as, in limited space, with insufficient lighting, with rubble scattered, under high radiation, with dusts. These operations are carried out for the first time in Japan. Workplace accidents must be prevented even under such a severe condition.

(2) Major requirements

Proactive measures such as the sufficient preparation, planning and training using mockup facility for the first task performed under the extremely poor work environment in the R/B, and study on the contingency measures which are more sufficient than the work safety measures applied to date.

(3) Action status and evaluations and issues

Understanding the current site conditions, work accident prevention is being performed.

- Through the promotion of sharing and horizontal transfer of information regarding OE (operating experience) and enhancement of the systems and structures of safety management, the number of accidents in FY2015 was decreased from FY2014 and the number of accident victims was decreased by 40% (64→38 persons).
- Heatstroke prevention was conducted by prohibiting the work basically during the period of intense heat (14:00-17:00 during July-September) in the heat wave period and applying the standard rules for the heatstroke prevention (the work is prohibited at 30 deg. C or more of WBGT index (index based on three factors of humidity, solar radiation and temperature which have large impact on the heat balance of the human body). As a result, the number of heatstroke incident due to the work in FY 2015 decreased significantly from FY2014.
- In response to the guidelines on the improvement of industrial health and safety standards released by Health, Labour and Welfare Ministry in October 2015, risk assessment, efficient measures for radiation dose reduction has been performed appropriately from the stage of placing an order for the construction in order to achieve further improvement of the management for safety and sanitation.
- The progress has been made in the decontamination outside the buildings. The buildings can be categorized by three areas depending on the contamination level (Anorak area, Cover all area, Regular work clothing area). The safety and workability are being improved by the reduction of work load using different radiological protections for each area.

In order to establish the technical basis for remote-controlled device, mockup test facility was established in Naraha Remote Technology Development Center in FY2015 and 3D virtual reality (VR) system that simulates the conditions inside the R/B at the Fukushima Daiichi NPS (Fig.4.3.2-2-22) was established for the purpose of planning the process plan and pre-operation training.

As a preparation for the retrieval of the rubble and spent fuels from the SFP of Unit 3 which is planned from FY2017, operation training and planning are being carried out using the fuel handling equipment to be installed on site.

The outline of the fuel debris retrieval work has been studied. Labor safety will need to be examined in line with the fuel retrieval work planning, since its details are deeply related to the retrieval policy.
With regard to radiation safety, the internal radiation exposure is required to be monitored immediately after the intake of the radioactive materials. The agreement was therefore closed with Japan Nuclear Fuel Limited, and the organizational structures have been established so as to assess the internal radiation exposure promptly through the internal exposure assessment (bioassay) by the analysis of excrement (urine and feces)

(4) Future actions

a. Reviewing the operations for reducing the radiation dose inside the R/B and investigating the internal PCV condition which has been performed to date to utilize the results of preparation, planning and training of other operations.

b. Improve work environment of the place where people have to access since remote operation cannot be performed, by restoring the lighting (power restoration), improvement of communication environment and removal of rubble, to the extent possible.

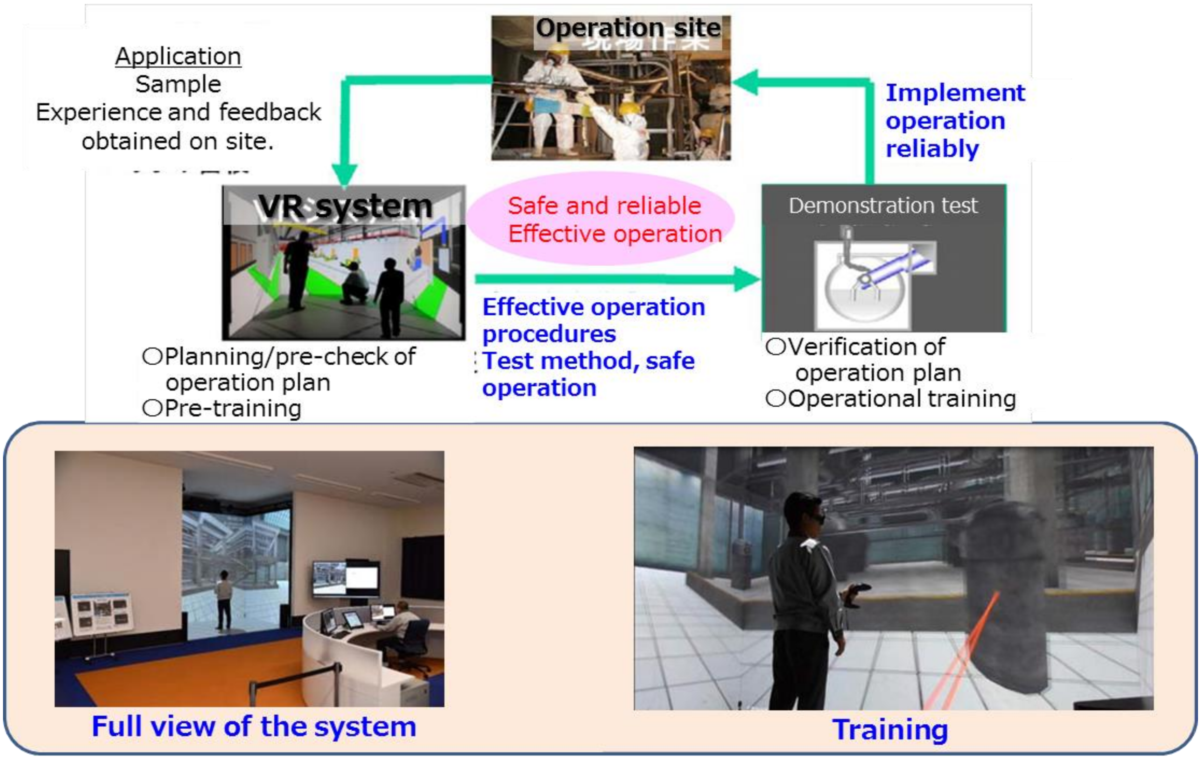
c. Conduct trainings on the mock-up sufficiently for the operations to be carried out for the first time. It is essential to plan, implement and verify effective work procedures and test method. Also, operation time and dose are required to be reduced by making a process plan utilizing VR system and pre-operation training.

d. The on-site work such as the preparation for the retrieval of the fuel debris is expected to include the decontamination inside the R/B, investigation of the leak locations of the PCV, repairing of the PCV bottom/upper part, establishment of system structure equipment, preparation and construction for installation of the fuel debris retrieval equipment and devices, fuel debris retrieval work and collection, transport and storage of fuel debris work. The fuel exposure due to the failure of the cooling function when removing the spent fuels, diffusion/leakage of gaseous radioactive materials and leakage of highly contaminated water might occur during the preparation for fuel debris retrieval work. In the fuel debris retrieval work, hydrogen explosion might occur due to the re-criticality and suspension of nitrogen injection. In the collection, transport and storage work for the retrieved fuel debris, unit tube and storage canister might fall down due to failure of crane. Since these are the works for the first time which no one experienced before, it is very difficult to estimate the accident and trouble in advance. Therefore, possible accident and trouble that may be caused in each work should be identified to the extent possible. Preventive measures are required to be taken against the accident and trouble by performing the risk assessment. Also, a contingency plan such as for securing the maintenance work is required to be studied so as to respond promptly to the accidents and troubles.

e. As for the work inside the R/B as well as fuel debris retrieval work, air sample from the work environment is to be taken continuously using the sampling head installed in the work area to monitor the dust in the work environment. Also, air contamination is required to be monitored continuously by the consecutive measurement using α-ray and β-ray detectors mounted on the sampling head.

f. Internal exposure through the injection of radioactive materials is to be prevented by combining the containment of radioactive materials, decontamination and use of protection equipment. In particular, contamination prevention such as for the protective clothing, protective equipment is important from the perspective of prevention of internal exposure via indirect oral intake of attached radioactive materials or external exposure of the skin. Also, the prevention of oral and nasal intake of radioactive materials into the body using respiratory protective devices is required to be controlled depending on the physical and chemical properties of the nuclide which is subject to the protection. It is also important to perform the protection based on a sufficient understanding of particle diameter that significantly contributes to the collection efficiency of the filters.

g. When the radioactive materials are taken into the body, internal radiation exposure is be monitored by any one of the means, such as whole-body measurements by the whole body counter, bioassay and calculations based on the concentration of airborne radioactive materials. Also, if the worker is (or may have been) exposed to the radiation dose over the limit, temporary medical checkup is required to be provided for the worker and limitation and prohibition to the work area are required to be enforced.



(Provided by JAEA)

Figure 4.3.2-22 Work training flow using virtual reality (VR) system

4.3.2.7 Establishment of the access route to the fuel debris

(1) Purpose

The access route inside the building required for fuel debris retrieval work and the route to access the fuel debris inside the PCV from the operating floor or the side of the building are to be established.

(2) Major requirements

a. Common requirements for all methods

i) Radiation dose at the working areas inside the building is reduced, the obstacles are removed, and the access route inside the building is established so as to carry-in/out and install the equipment/devices as well as transporting the retrieval equipment and fuel debris to be used for the fuel debris retrieval.

ii) Release of radioactive materials from the PCV/RPV is prevented on the access route.

b. Requirements on Submersion-Top access / Partial submersion-Top access methods

i) Since it is a method to retrieve fuel debris from Operating floor, the existing equipment located along the access route to the fuel debris must be removed before establishing the access route from the upper part of the PCV to the fuel debris.

ii) If there is fuel debris outside the core shroud, it will be necessary to remove the core shroud and access the fuel debris; therefore a plan is established according to the conditions.

iii) A plan is established according to the conditions since large opening needs to be created at the bottom of the RPV to access the fuel debris located at the bottom of the PCV. This plan should be established assuming that the fuel debris may be located in and outside the pedestal at the bottom of the PCV.

c. Requirements on Partial submersion-Side access method

i) A plan is established including the creation of an opening in the wall and enlargement of the opening of the PCV required depending on the retrieval method to access the fuel debris at the bottom of the D/W from the first floor of the building.

(3) Action status and evaluations and issues

a. Common action status and evaluations and issues for all methods

Details of the establishment of access route inside the building are not currently studied, and they are required to be made according to timely plan. On the other hand, in the studies of the route inside the PCV to access the fuel debris, the element test has been started for the important technical development to prevent the release of radioactive materials from the inside. Approaches for the method are described in b. and c.

b. Action status and evaluations and issues for the Submersion-Top access / Partial submersion-Top access methods

The access routes inside the building required to carry out the fuel debris retrieval work on the operating floor includes the one which leads from the entrance of the R/B to the operating floor and the other one from the operating floor to the periphery of the RPV. Under present circumstances, radiation dose inside the building is high and its environment is not appropriate as working environment. In response to the timing of the fuel debris retrieval, details and timing of implementation of field works including radiation dose reduction and removal of obstacles located on the route are required to be planned. Although the study on the

access route inside the building is not yet to be studied in detail, it will need to be implemented according to the timely plan.
To determine the feasibility of important step in the establishment of the access route from the operating floor to the fuel debris, the studies on the development including the element test are being performed as a part of technical development of the fuel debris retrieval method.
With regard to Submersion-Top access method and Partial submersion-Top access method, the access route leading from the top of the PCV to the core is structurally secured for the fuel replacement work during the periodic inspection. The access to the inside of the reactor can be secured by removing the well shield plug, PCV upper head, insulator of the RPV upper head and the RPV upper head. The upper grid plate immediately above the core may be reached by removing the steam dryer and steam separator, allowing access to the fuel debris inside the RPV. In the fuel debris retrieval work, the access route for the periodical inspection mentioned above will be used. However, according to the analysis, there is a possibility that the equipment to be removed may have caused creep deformation as the results of exposure to high heat during the accident. Although the damage states of the equipment has yet to be confirmed, the fuel debris may be difficult to be removed by using a normal method depending on the condition of deformation. Figure 4.3.2-23 shows major reactor internals to be removed by Top access method. In Top access method, an access route to fuel debris is to be established by removing contaminated internal structures located over the fuel debris distributed.

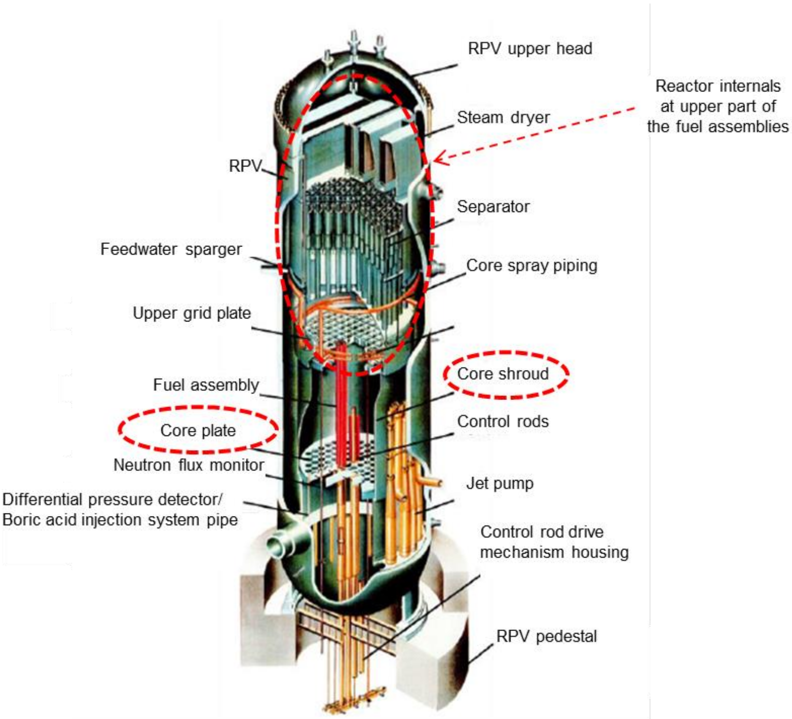


Figure 4.3.2-23 Reactor internals considered when using Top access method (Reference: IRID website)

It is an important and highly difficult issue to be solved that the access route that prevents the release of radioactive materials from the PCV while retrieving fuel debris is to be established.

- Submersion-Top access method

Figure 4.3.1-6 in Section 4.3.1 show schematic drawing of Submersion-Top access method. Fuel debris is to be handled in the primary boundary consists of the PCV and cells installed on the operating floor, and secondary boundary of R/B and containers installed outside the R/B. Technical development to prevent the release of radioactive materials are being carried out. Figure 4.3.2-24 shows the items identified from the concept of Submersion-Top access method which are considered particularly important to establish the access route and prevent the release of radioactive materials on the access route.

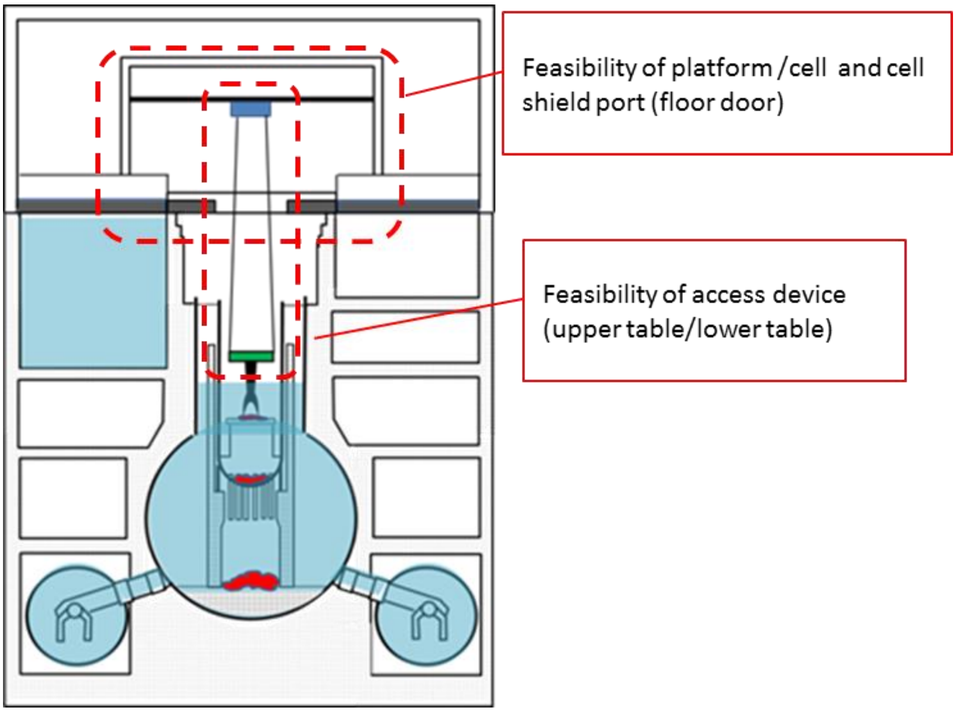


Figure 4.3.2-24 Important item to confirm the establishment of the access route for Submersion-Top access method

The element tests to confirm the feasibility of the important items above are underway.

The FS of platforms and cells, and element test for the shield port to be installed on them, are being planned to confirm the feasibility of shielding of the cells installed on the operating floor and prevention function against scatter of radioactive materials. (For the overview of the element test, refer to (3) of Figure 4.3.2-30 in Sec. 4.3.2.8 and (4) of Appx.4.20.)

Also, element test for Submersion-Top access method is being planned and carried out to confirm the feasibility of device to access the inside of the RPV to retrieve the fuel debris. (For the overview of the element test, refer to (2) of Fig.4.3.2-30 in Sec. 4.3.2.8 and (3) of Appx.4.20.)

These are common development for all methods.

- Partial submersion-Top access method

The section 4.3.1 and Figure 4.3.1-6 show the concept of containment for Partial submersion-Top access method. Fuel debris is to be handled in the primary boundary that consists of the PCV and cells installed on the operating floor and secondary boundary of R/B and containers installed outside the R/B. Since it is the Partial submersion method, technical development has been planned and it is being carried out to prevent the release of radioactive materials assuming stricter conditions. Figure 4.3.2-25 shows the items identified from the concept of Partial submersion-Top access method which are considered particularly important to establish the access route and prevent the release of radioactive materials on the access route.

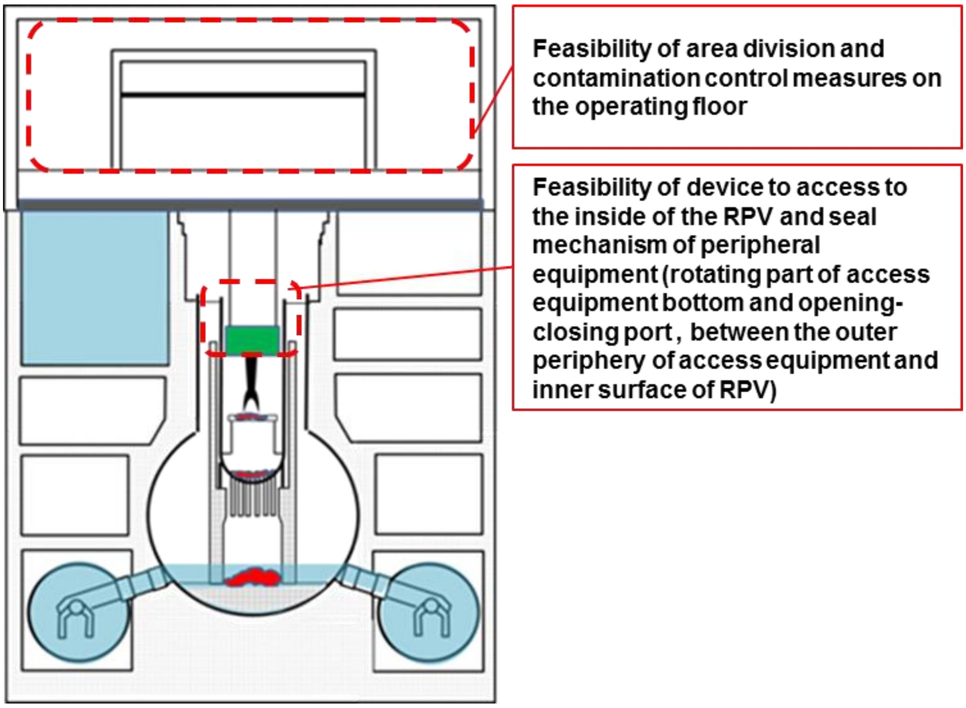


Figure 4.3.2-25 Important item to confirm the establishment of the access route for Partial submersion-Top access method

The element tests to confirm the feasibility of the important items above are underway.
Element test using 1/4 scale model is being planned to confirm the feasibility of the area division of operating floor and of contamination control measures. In Partial submersion-Top access method, the sheets for area division/prevention of spread of contamination is planned to be applied when exporting the reactor internals in establishing the access route from the upper part of the PCV to the fuel debris in the reactor vessel. In the element test using scale models, the feasibility of contamination control method is planned to be confirmed for each work step. (For the overview of the element test, refer to Sec. 4.3.2.8 and (6) of Figure 4.3.2-30 and (7) of Figure A4.20 of Appx.4.20)
Also, the range of high radiation/high contamination area is to be settled for retrieving the fuel debris by using shielding and taking measures against radioactive material containment near the fuel debris. Element test is being carried out for the development of sealing technology for device to access to the inside of the RPV. (For the overview of the element test, refer to (7) of Fig.4.3.2-30 in Sec. 4.3.2.8 and (8) in Appx.4.20)

c. Action status and evaluations and issues for Partial submersion-Side access method

On the lateral side of the PCV, there are equipment hatch, CRD hatch and some other hatches which lead inside the PCV. Although the sizes of the openings of those hatches may be limited, the access route into the PCV is structurally secured. At the bottom of the D/W inside the PCV, there are the PLR pumps, valves, pipes and supports which are located outside the RPV pedestal and there are CRD handling machine, operating floor (grating) inside the RPV pedestal. They will be obstacles to fuel debris retrieval work;

however access to the fuel debris at the bottom of the D/W can be made by cutting off and removed them. In this regard, export route for equipment to be removed should be secured. Figure 4.3.2-26 shows the conditions of obstacles inside the PCV for the Partial submersion-Side access method.

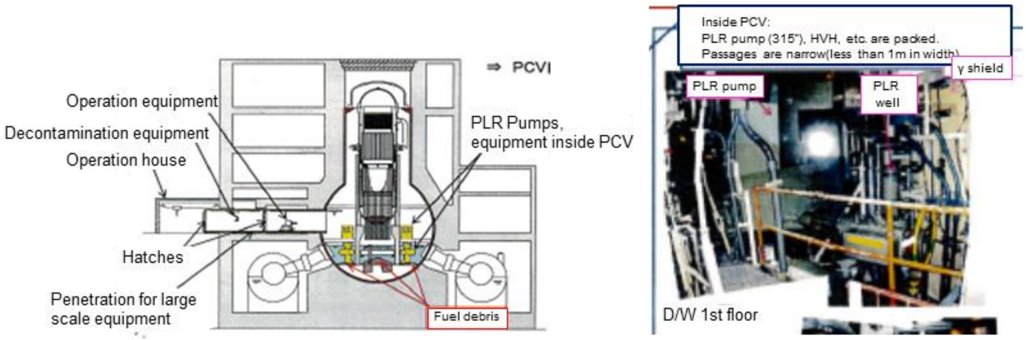


Figure 4.3.-26 Conditions of obstacles for Partial submersion-Side access method

Section 4.3.1 and Figure 4.3.1-6 shows the concept of containment of the Partial submersion-Side access method. Handling is to be performed in the primary boundary consists of the PCV and cells to be installed in the building next to the PCV and secondary boundary of R/B and containers installed outside the R/B. Since it is the Partial submersion method, the plan should be made in consideration of prevention against release of radioactive materials under more severe conditions.

Also, an example of step sequence to access to the fuel debris inside the RPV pedestal by Side access method is shown in Figure 4.3.2-27.

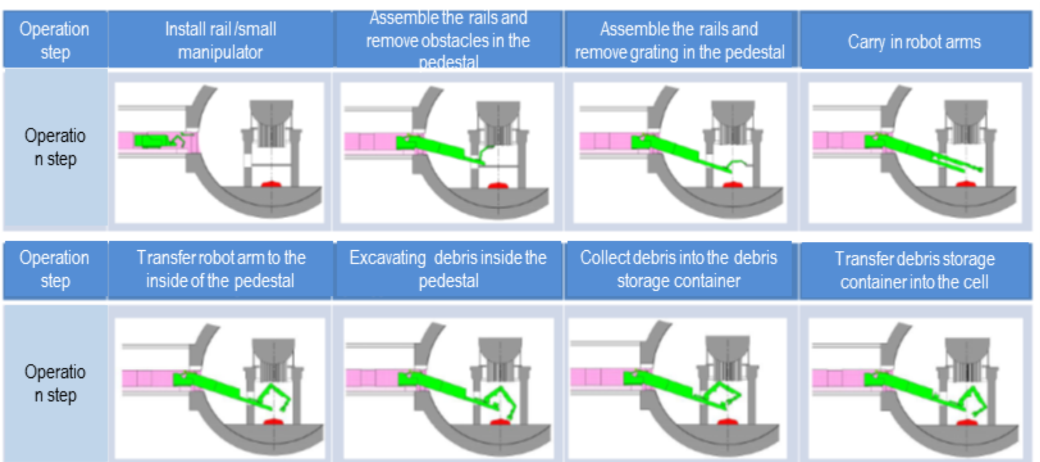


Figure 4.3.2-27 Access steps for fuel debris by Partial submersion-Side access method

Figure 4.3.2-28 shows the items whose feasibility is important to prevent the release of radioactive materials from the PCV in the establishment of the access route for Partial submersion-Side access method.

To determine the feasibility of important step in the establishment of the access route from the side of the R/B to the fuel debris by Partial submersion-Side access method, the studies of development including the element test are being performed as a part of technical development of the fuel debris retrieval method.

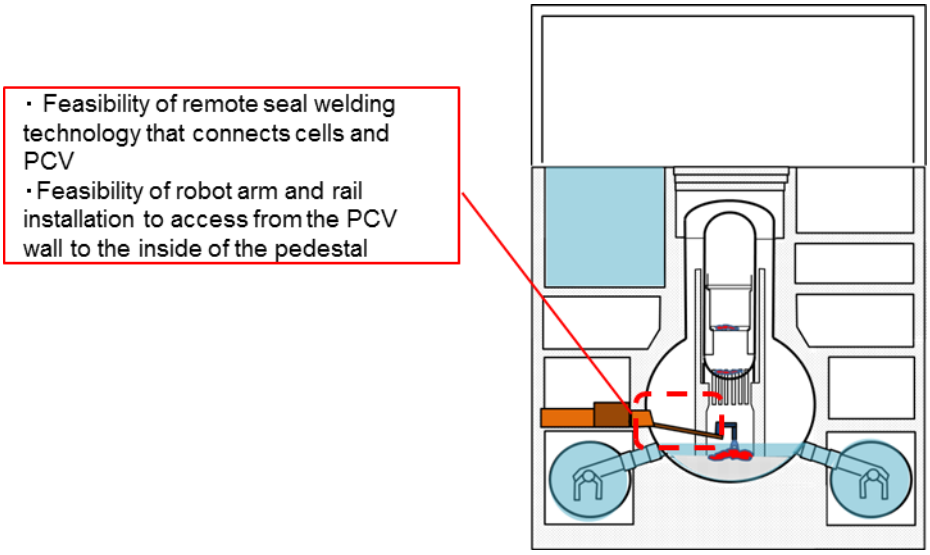


Figure 4.3.2-28 Important item to confirm the establishment of the access route for Partial submersion-Side access method

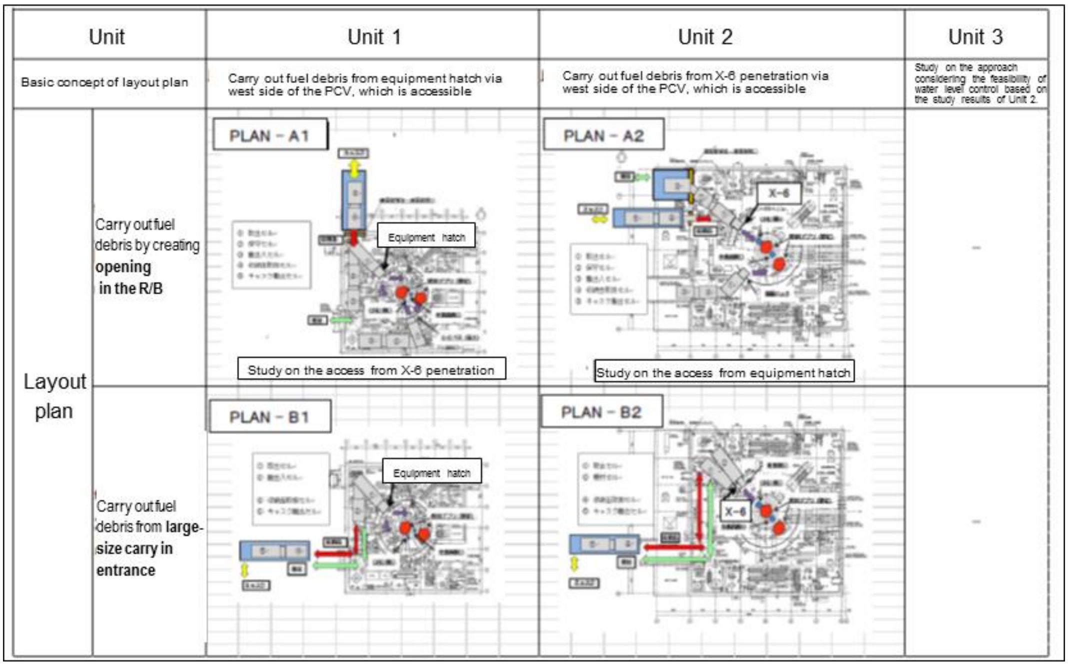
The element tests to confirm the feasibility of the important items above are underway.

The Partial submersion-Side access method requires controlling the radioactive material released by sealing the PCV and cells, which were additionally installed when establishing the access route.

The element tests for remote welding for the PCV and cell adopters are planned and being carried out to confirm the feasibility. (For the overview of the element test, refer to (3) of Figure 4.3.2-30 in Sec. 4.3.2.8 and (4) of Appx.4.20)

Also, the element test is also planned to confirm the feasibility of the robot arms and rails to access the fuel debris inside the pedestal in the PCV. (For the overview of the element test, refer to (8) of Fig.4.3.2-30 in Sec. 4.3.2.8 and (9) of Appx.4.20)

The Partial submersion-Side access method requires the plans including the expansion of openings of the PCV and making opening in the building wall to access from the first floor of the building to the fuel debris at the bottom of the D/W. The layout for each Unit is being studied based on the access route. Figure 4.3.2-29 shows Layout plan for each Unit.



(Reference: IRID Interim report on "Operating grants for Government-led R&D Program on Decommissioning and Contaminated Water Management (Upgrading Approach and System for Retrieval of Fuel Debris and Internal Structures)" in April 2016)

Figure 4.3.2-29 Layout plan for Partial submersion-Side access method

(4) Future actions

The following are the actions to be taken.

a. Common actions for all methods

Actions for three methods focused on are described in b. and c individually.

As shown in Sections 4.2 and 4.3.2.5, radiation dose reduction is estimated to be very difficult issue according to the findings obtained on site to date. The FS on the access route inside the building should be performed commonly for all methods based on the degree of difficulties in radiation dose reduction work required before the commencement of the construction and in removal work of obstacles on site. Also, if necessary, the adjustment should be made for the accessing point to the inside of the R/B which is currently planned with due consideration to the difficulties in the site conditions outside the building.

b. Actions for Submersion-Top access /Partial submersion-Top access methods

Studies on the access route inside the building are required. In specific, the work process should be planned to establish the access route based on the possibilities of radiation dose reduction which can be applied on site. Also, the radiation dose reduction plan for the operating floor should be confirmed from the perspective of the impact on the preparation for the fuel debris retrieval construction. If there is an item in the plan that may cause an impact, some arrangement should be made.

The element test which started to confirm feasibility of the access route established from the operating floor to the fuel debris is required being evaluated as planned. If the issues are found, countermeasure should be planned as soon as possible. Also, when the investigations became necessary to be conducted by additional element tests as the studies on the fuel debris retrieval method progress, the development plan should be revised promptly and measures should be taken so as not to affect the plan for the following fuel debris retrieval work.

Although a large opening is assumed to be created at the bottom of the RPV to access to the fuel debris at the bottom of the PCV, degrees of difficulty in construction would be high. A decision on the policy is required to be made based on the results of the studies on the method to be promoted.

c. Actions for Partial submersion-Side access method

An access route inside the building is required to be made from the horizontal side to the inside of the building under high radiation. Different from Submersion method or Partial submersion-Top access method, to access to the inside of the PCV, installations and piping already installed are required to be removed. A high radiation area may require being an access area and preparation works including the removal of existing obstacles may contains high difficulties. Therefore, the process plan and the measures for decontamination and shielding are required to be developed from the early stage. Also, feasibility of the access route inside the building is to be studied considering the possibility of radiation dose reduction which can be achieved on site.

The element test that started to confirm feasibility of the access routes from the lateral side of the PCV to the fuel debris inside the PCV is require being evaluated according to the plan. If an issue is found, countermeasure should be planned as soon as possible. The studies on the procedures to remove the obstacles for fuel debris retrieval work, such as equipment installed outside the RPV pedestal in the PCV and fallen objects inside the RPV pedestal and expected construction period should contribute to the study on the approaches to the fuel debris retrieval.

When the investigations became necessary to be conducted by additional element tests as the studies on the fuel debris retrieval method progress, the development plan should be revised promptly and measures should be taken so as not to affect the plan for the following fuel debris retrieval work.

4.3.2.8 Development of the fuel debris retrieval equipment and devices

(1) Purpose

To develop equipment and devices corresponding to the site conditions to retrieve fuel debris in safe, proven and effective manner.

The development of equipment and devices relating to the three major methods to be focused on (Submersion-Top Access Method, Partial submersion-Top Access Method, and Partial submersion-Side Access Method) is being carried out.

(2) Major requirements

Fuel debris is envisaged to be inside the RPV and "in and outside" of the pedestal. Following requirements shall be satisfied on the occasion of developing the functions (e.g. access device, visual and measurement, cutting/dust collection, prevention of radioactive dust scattering, shielding and storage) required for the said three methods.

Major requirements on the designs of equipment and devices are as follows:

- Since fuel debris retrieval work will be performed under high radiation environment, remote-controlled retrieval equipment shall be adopted. Although retrieval work is basically being performed by manual but remotely from a safe area, equipment shall have protection mechanism such as automatic stop in case of abnormal conditions (e.g. collision avoidance, overload).
- Equipment and devices for the fuel debris retrieval shall be designed to have high reliability/redundancy and also functions corresponding to the environment of each unit other than high radiation field.
  And, fail safe concept shall be introduced in the design to the extent possible.
- Also, existing proven technologies shall be adapted to the extent possible.
- Although inspection/maintenance works for equipment are basically performed by remote control, manual maintenance works after decontamination and shielding shall be also considered as necessary.
- All necessary repair/recovery work shall be performed without any block caused by malfunction even if equipment/device malfunction happens during the fuel debris retrieval work.
- Specified radiation resistance shall be fulfilled.
  (The required radiation resistance on Dry Method is severer than other methods).
- Equipment and devices shall be designed to stand against dusty environment.
  (The specified dust level on Partial Submersion Method is severer than other method.).
- Waterproof performance corresponding to each PCV water level shall be ensured.
- A view under the muddy water (e.g. suspended matter, cut piece,) shall be secured during the retrieval work.

(3) Action status and evaluations and issues

In the selection of the element test item, priorities are given to "safety" and "site conditions of the Fukushima Daiichi NPS"

Safety: Remote-control, Automation, Shielding and Prevention of spreading contamination for reducing radiation exposure against workers.

Site conditions of the Fukushima Daiichi NPS: The differences among the Units, e.g. contamination level, dose rate and damage level of equipment.

Out of these items, priorities are given to the long-term development items and the items being required to early realization. Also, element test for the establishment of the access route is described in the previous section 4.3.2.7.

a. Current status

Following technical developments are being carried out as the element test. (For details, refer to Appendix 4.20)

Common technical development for all methods

- Test with a hydraulic manipulator
  Element tests such as controllability, repeatability, and operating accuracy are being conducted by using commercially available hydraulic manipulator to obtain basic data of the manipulator being used for fuel debris retrieval.
- Technical developments for cutting/dust collection, visualization and measurement (Fig.4.3.2-30(1))
  Equipment for cutting / machining the fuel debris, dust collection, visualization and measurement being used in high radiation environment are under development.

Technical development of Submersion-Top Access Method

- Development of the devices to access to the inside of the RPV (Fig.4.3.2-30 (2))
  The upper part hanging device installed on the operating floor and the lower work stage used in the RPV (being hanged by the said upper hanging device) are under development.
- Development of platform/cell (Fig.4.3.2-30 (3))
  Platforms/cells to be installed on the operating floor are under development.
- Development of handling equipment for fuel debris canister (Fig.4.3.2-30 (4))
  A device which places collected fuel debris in a storage canister closes a lid of canister and transfer it to the outside is under development. This development can be applicable to other methods.
- Development of light-weight and shape-following shielding (Fig.4.3.2-30 (5))
  Shielding being filled with water only when it is used is under development.
  This development can be applicable to Partial submersion-Top Access Method.

Technical development on Partial submersion-Access Method

- Development of utilizing films and sheets for contamination spread prevention method. (Fig.4.3.2-33 (6))
  Films and sheets for partitioning off working areas are under development to prevent radioactive dust spread.
- Development of sealing technology for devices accessing to the inside of the RPV (Fig.4.3.2-33 (7))
  Sealing system of the access device (which moves up and down inside RPV) is under development to partition RPV into two areas (upper than the device and lower than the device).

Technical development of Partial Submersion-Side Access Method

- Development of device to access inside the pedestal (Fig.4.3.2-33 (8))
  A device to access to the inside of the pedestal though the opening for CRD replacement machine located in the lower part of the PCV is under development.
- Development of flexible structure arm for remote-controlled work ((9) of Figure 4.3.2-33)
  A device to disassemble/remove obstacles in PCV is under development.
- Development of the PCV remote seal welding equipment for cells(Figure 4.3.2-33(10))
  A remote sealing device to connect cells and PCV is under development.
  On the assumption of utilizing the device at X-6 penetration, the welding test in use of the device has been carried out in narrow space.

b. Evaluation and problems to be solved

The first year of the two years element test plan has been finished, and we are now in the stage of the basic study and test device preparation. Since feasibility of devices shall be confirmed through mockup tests from now, final results necessary for making decision of adopting the said three methods have not yet been obtained. It should be noted that the final requirements for actual debris retrieval equipment and device have not entirely satisfied yet in the present development stage, as for element tests. Also, there is a possibility of changing the concepts of actual debris retrieval equipment and device in future. In the development of a radiation-resistant camera for visual (as one of common technical developments for all methods), the results are approaching to the final requirements.

On the other hand, various types of inspection are also carried out in the Fukushima Daiichi NPS. As situation inside PCV, temperature and dose rate are becoming clear. Also, a large amount deposit in sludge form and muddy water have been found in stagnant water in the reactor, and countermeasures for keeping clear view have to be realized consequently.

(4) Future actions

a. Consistency with overall plan for the fuel debris retrieval method

- Coordinating with overall plan, the followings shall be considered.
  i. Assuming the scenario consistent with overall plan that can perform a series of process consecutively
  ii. Picking up the required element technology
  iii. In case that there is any technology which is not addressed currently and/or new issues from the present development of element technology, the plan shall be revised as necessary.
  iv. Also, the development of element technology required to study the feasibility of the scenario shall be carried out.
- In the Fukushima Daiichi NPS, various inspections/analyses are carried out and new information and knowledge have been obtained. These information and knowledge shall be fed back to the development of equipment and device.
- Out of the on-going technical developments, element tests should be accelerated for the items which are significantly delayed as necessary.

b. Applications of existing (proven) technologies

- High reliability shall be required for equipment and device used for fuel debris retrieval. Although development of new technology may be required, existing (proven) technology with high reliability shall be utilized.
- Even if some existing technologies lack in performance, it can be compensated such as by the operation (maintenance). Development of device/equipment shall be decided by the results of comparison between newly developed technologies and existing technologies, from the view point of cost and time perspective.

c. Implementation of mockup test

- Mockup tests shall be required to confirm the on-site applicability of additionally developed equipment and device.
- It is also important to develop human resources who operate the developed equipment and devices. Since the actual site is under high radiation environment, education/training shall be conducted by using mockup.
- Effective use of Naraha Remote Technology Development Center of JAEA as a mockup test site shall be considered from the view point of reducing time and cost.

d. Future development of remote-controlled equipment

- As the result of the PCV internal survey in Unit 1, the information on the radiation dose data and temperature has been obtained, but the incident of sticking the inspection robot had occurred.
  Although such event may be prevented if work procedures are checked one by one, further improvements of reliability and workability can be achieved by ensuring the overall view.
- In the condition investigation on each unit, lack of infrastructure construction, such as reliable communication system and lighting system had been found.
  Along with the development of individual inspection equipment, such infrastructure construction for supporting the inspection work shall be required.
- With regard to the radiation resistance, such data of electronic components mounted on the robot have been obtained through the PCV internal survey. Such data are expected to be compiled in a database and to be utilized for the future equipment development.
- Based on case study of success and failure on the past inspection works in use of robots, besides with careful consideration of actual work performance (e.g. decontamination work) under high radiation environment, actual on-site environment and the robotic technologies in the general industry, the equipment development shall be executed.
- With regard to the technologies developed independently, it is important to create a system of utilizing them in appropriate manner according to their advantages.
- From the perspective of utilizing robotic technology and remote technology, the developments of equipment, functions and elements that can be commonly used shall be enhanced in future.

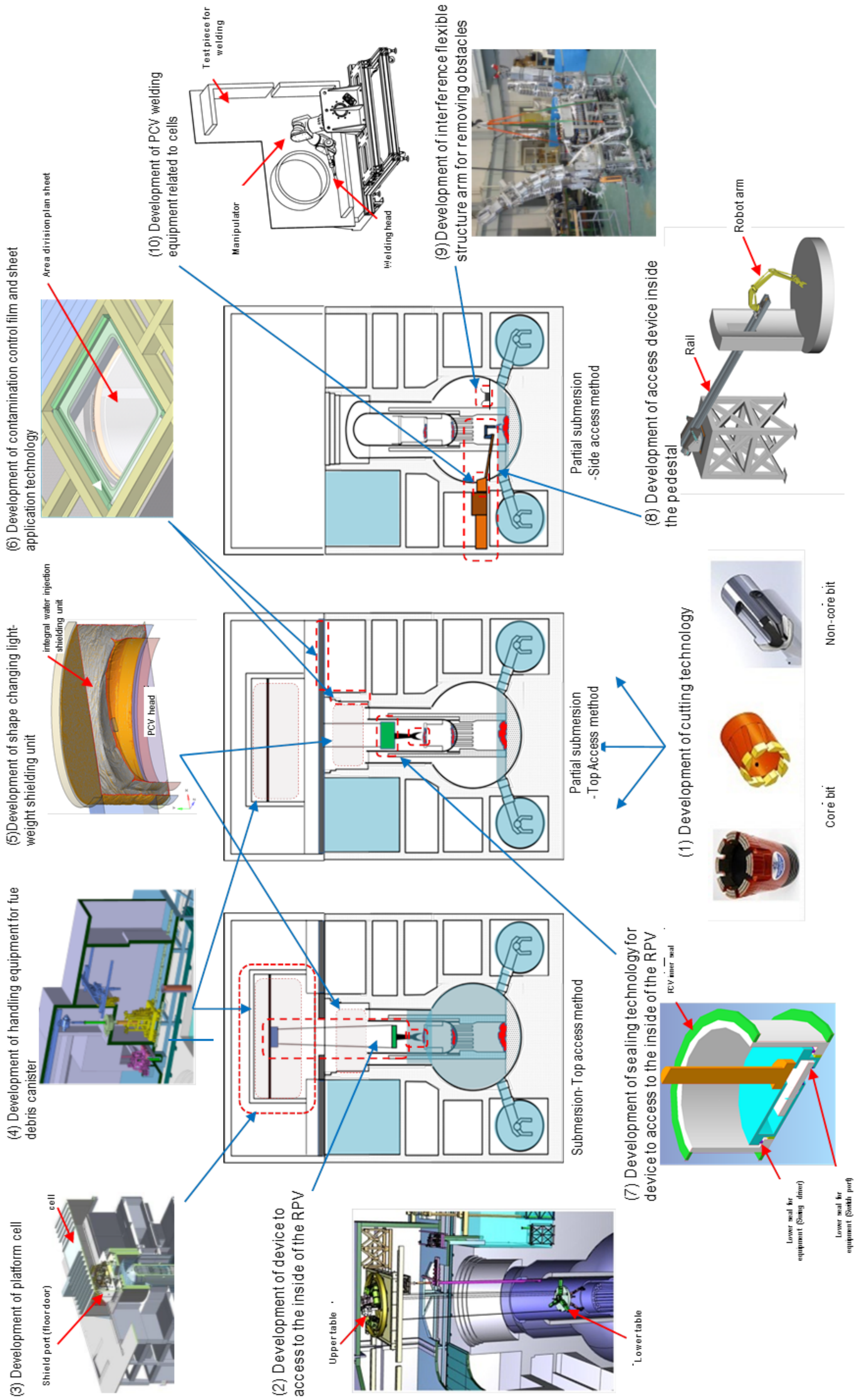


Figure 4.3.2-30 Whole image of element test

4.3.2.9 Establishment of the system equipment and working areas

(1) Purpose

Equipment in the R/B and PCV has been damaged by the impact caused by the accident, therefore a set of the systems including the installations, equipment and devices, and their systems to operate the equipment will be additionally necessary for continuous and safe retrieval and storage of the fuel debris scattered in the PCV. Those pieces of equipment need to be prepared and installed after securing installation space.

(2) Major requirements

a. Identifying the installations, equipment, devices and systems that additionally required design should be developed based the specifications for functional requirements for each item.

b. These installations, equipment, devices and systems are capable of being installed on site from the following perspectives and appropriate operation can be achieved.

- Required structural strength is secured for the additional containers and operating cells to be installed in the building.
- Sufficient installation space is secured for equipment, devices and systems. Necessary environmental conditions are satisfied based on design requirements on equipment.
- Sufficient area is secured for the operation and maintenance of equipment, devices and systems and necessary environmental conditions are satisfied.
- If existing equipment is used for the fuel debris retrieval work, required functions should be secured even if considering the impact on the equipment caused by the accident and ageing degradation.
- Storage area is secured on site for the cut workpiece and severely contaminated structures to be exported over the course of the fuel debris retrieval work.
- Fuels are removed from the SFP, and other stored objects including the control rod, fuel racks and rubble are also removed (if Top Access method is used).

(3) Action status and evaluations and issues

Conceptual studies need to be conducted on the devices and equipment that constitute the system, establishment of a layout and development of a plot plan of the Fukushima Daiichi NPS taking into account a temporary placement area for retrieval equipment and storage area for fuel debris. These activities will be carried out from FY2015 starting with those required for FS on the retrieval method.

a. The systems, equipment and their functions important for ensure safety in fuel debris retrieval work are below. The specifications of their functional requirements are developed through the studies of key issues shown in the Section 4.3.2.2-4.3.2.5 and are varied depending on the fuel debris retrieval method and plant condition for each Unit.

- R/B container and operation cells

  Isolation wall from the outside of the building and its structure that divide the area depending on the contamination level inside the building
- Shielding installations
- Circulation cooling systems

Cooling of the fuel debris and coordination of the PCV water level

- Leakage water collection systems
  Collection of cooling water leaked inside the torus room
- Negative pressure control systems
  Keeping the difference in pressure between the inside and outside of the area division within a certain level and prevent spread of contamination
- Local collection systems for cutting particles of fuel debris
  Most cutting particles are collected near the fuel debris cutting location
- Purification systems for circulation water
  Collect and treat cutting particles of fuel debris flowing into cooling water
- Dust collection/treatment systems
  Collect and treat fuel debris cutting particles and radioactive materials dispersed in the air
- Ventilation and air conditioning systems
  Ventilation air-conditioning and dust collection for each area division
- Neutron absorber injection systems
  Inject neutron absorber into cooling water and adjust its concentration to a certain level (criticality prevention)
- Emergency neutron absorber injection systems
  Inject neutron absorber into cooling water in a short time for emergency (criticality prevention)
- Emergency cooling systems
  Cool duel debris by injecting cooling water for emergency

Conceptual study is being conducted for the system and equipment above.

(4) Future actions

Specifications required for each system and installation are developed based on information regarding plant conditions for each Unit, such as the distribution locations and properties of the fuel debris, damage state of equipment, study status of important technical issues relating to ensuring safety and the fuel debris retrieval method. However, since these pieces of information and study status with high degree of accuracy cannot be expected be obtained in a short term, conservative specifications is to be studied for the meantime. The realistic specifications are to be developed according to the information obtained.

Toward the decision making on the policy of fuel debris retrieval method, following items are to be conducted to confirm the feasibility of fuel debris retrieval method for each Unit.

a. Based on the required specifications developed by the key issue described in 4.3.3.2-5, developing the conceptual design is to be developed for each system and installation, prospect of feasibility of the design as a whole system is to be obtained.
b. Based on the conceptual design for each system and installation, prospect for securing the appropriate installation area is to be obtained.

After the selection of the fuel debris retrieval method, the details of layout plans are examined for the areas used for the installation and operation of the equipment that constitutes the system. Also, the detailed plot plan for the areas of temporary placement and handling of the retrieved equipment, and storing of the retrieved fuel debris is required to be established.

4.3.2.10 Towards the detailed study based on the work steps of each method

In the section 4.3.2.1-4.3.2.9, the current status and future actions for technical requirements on the retrieval of the fuel debris are described. Figure 4.3.2-31 shows the action plans for each technical requirement.

In order to evaluate the feasibility of fuel debris retrieval method, the applicability of the study results of each technical requirement is required to be examined for each work step planned for each method. It is important to identify and clarify the issues and obstructive factors and determined their action policies.

In the studies based on the work steps, those which may cause a serious delay in the process because of the failure and accident during the process of retrieval work and possible troubles that affect subsequent fuel debris retrieval works or make them difficult to continue are the obstructive factors which needs to be identified carefully. Also, attention needs to be paid for the issues related to the work safety and its countermeasure should be studied. The obstructive factors that identified should be evaluated so as to understand the degree of difficulty in handling and level of the impact on ongoing works. To identify the features for each method through these studies is important to determine the fuel debris retrieval policy for each unit.

The items requiring a further study for next stage, the preparation for the construction works and coordination of interface with other field works are also important to be realized.

The points to be considered in the studies based on the work steps are as follows:

(1) The requirements on the environmental radiation dose condition during the work considering the feasibility of work steps including preparation work.
(2) Feasible control of negative pressure including the pressure conditions to minimize the release of radioactive materials during the fuel debris retrieval construction.
(3) Identification of works that requires countermeasure against hydrogen explosion and its countermeasure.
(4) Make a list for equipment required for each retrieval work, and rough volume of the necessary equipment.
(5) Removal work for the deposits at the bottom of the PCV that required in the initial stage of the fuel debris retrieval work.
(6) Estimated period to complete development as a period required to start fuel debris retrieval construction
(7) Estimated period required for fuel debris retrieval work

Works subject to the study are as follows:

- Case of Submersion-Top access method and Partial submersion-Top access method
  Dismantling of shield plug
  Dismantling of the PCV head dismantling work
  Dismantling of base frame for RPV head mirror insulation
  Dismantling of RPV head

Dismantling of dryer/separator

Dismantling of reactor internals

Retrieving fuel debris

(including the estimation of the amount which can be retrieved per day)

- Case of Partial submersion-Side access

  PCV drilling work

  Dismantling of obstacles inside the PCV

  Fuel debris retrieval work from inside the RPV pedestal

  (including the estimation of the amount which can be retrieved per day)

  Fuel debris retrieval work from outside the pedestal

  (including the estimation of the amount which can be retrieved per day)

(8) Studies on the potential obstructive factors of work step are to be identified and its countermeasure, degree of difficulty in handling and impact to the ongoing works are to be studied

(9) Concept of device maintenance, work areas and collection of the fuel debris to the storage canister

(10) Studies on the relative evaluation for the items including man-hour and amount of materials required for the preparation/main work for each method to make an estimation as a whole for each method (including the comparison among some cases in one retrieval method when the difference is caused by setting conditions. i.e. water level for the Submersion method)

* Item (1) requires to be examined through the coordination between development of retrieval equipment and feasibility of on-site environmental condition. Items (2) and (3) require coordination between development of the retrieval equipment and feasibility of system facility.

For the item (5), the Uni1 B1 inspection in FY2015 indicated the possibility of significant amount of deposits at the bottom of the PCV (Refer to (1) Section c. of Appendix 4.3). Measures against those deposits will be necessary in the initial stage of the fuel debris retrieval so as not to affect the subsequent fuel debris retrieval works and system operations. Also, since the deposits to be removed may contain granulated fuel debris, concept of the category after the collection is required to be determined as well as the procedures for collection, transport and storage(Refer to 4.4.1 (4) a.i.).

Items (8), (9) and (10) require coordination between the development of the retrieval equipment and feasibility of on-site application.

Table 4.3.2-2 shows important items to realize work steps based on the results of the studies for each technical requirement.

Table 4.3.2-2 Issues on the technical requirements and action policies

Technical requirements	Major issues identified to date	Details for each work steps of the method Action policy of the studies
Ensuring structural integrity of the PCV and building	(1) Establish the concepts of the R/B evaluation considering the damage, establishment of the evaluation concept for the resistance and stiffness of the RPV pedestal with high temperature history and implement the aseismic performances evaluation for each method based on the concepts. (2) Develop the corrosion control measures such as for RPV, PCV and piping and confirm applicability to actual unit	Study the issues involved in the technical requirements before the selection of the method and their results are necessary to be obtained before the studies on the work steps of retrieval method.
Criticality control	(1) Clarify the management method as feasible method to realize the systems/facilities based on the studies on the criticality evaluation and criticality control method. (2) Clarify the criticality control technology which can be feasible for fuel debris retrieval method based on the studies on the sub-criticality monitoring methods and Recriticality detection technologies. (3) Contents are to be established as a feasible method plan based on the studies on the Criticality prevention technologies.	Clarify the contents to be realized corresponding to the method and reflect them to the system and facilities. Also, clarify the countermeasure against possible criticality during the work step and examine the impact on the subsequent process.
Maintaining the cooling function	The following items need to be performed later. Phase 1: Circulation loops during stagnant water treatment. establishment, Phase 2: Circulation loops during the PCV repair work. establishment Phase 3: Circulation loops during fuel debris retrieval work establishment	Study the issues involved in the technical requirements before the selection of the method and their results are necessary to be obtained before the studies on the work steps of retrieval method.
Securing containment function	(1) With regard to the concept of the containment functions, rational and feasible concept for the Fukushima Daiichi NPS is to be established while checking the status of the studies on the systems. (2) Establish the concept of containment system for the liquid phase (3) Establish the concept of containment system for the gas phase (4) Clarify the possibility of the PCV repair	Study and clarify the containment concept and feasibility of the system study, possibility of the PCV repair in the stage of the selection of the method. Clarify the task that may be caused during the studies on the work steps and examine the possible troubles during the work and impact on the subsequent process.
Reducing workers’ exposure during operation	(1) Elaboration of the plan for the reduction of workers’ exposure during operation assuming the on-site difficult situation and improvement of the accuracy of the prospect of the feasibility of the dose reduction plan. (2) With regard to the shielding during the fuel debris retrieval, improvement of the accuracy of understanding of the FP distribution, development of the shielding plan for each method, estimation for the dust scattering caused by fuel debris cutting, development of measures as a system.	Study the issues involved in the technical requirements before the selection of the method and their results are necessary to be obtained before the studies on the work steps of retrieval method. In the studies on the work steps, estimating exposure dose expected for each step, estimation of overall exposure dose as a whole method is to be summarized.
Ensuring work safety	Preparation is required for ensuring safety for the field work.	This should be included in the plan as a prerequisite for the commencement of the work.

Establishing access routes to the fuel debris	(1) Shielding for the access route and prevention of the radioactive material release (2) Development of the construction plan for the access route to the PCV inside the R/B -FS on the radiation dose reduction -Studies on the procedures to remove existing structures that may impede. (3) Development of the construction plan for the access route to the fuel debris inside the PCV	(1) Planning of the element test for the important items on the left. Correspond to the future progress. Also, when additional test is needed, development should be added as necessary. (2)Evaluation is required to be performed by the detailed study for the dose reduction plan and examination of the impact on the commencement of the construction. Starting the procedures for the removal of existing structures, it is necessary to study together with (1). (3) Studying the procedures for the removal of existing structures is required to be studied as a current development plan. (4) Others ・Acceleration of the equipment development relating to the access to the fuel debris. ・Study on the application of the existing technologies with high reliability. ・Study on the development and securing of the human resource for equipment/device operation.
Developing fuel debris retrieval equipment and device,	(1) Technical development for the shielding, prevention of radioactive material release and remote-controlling and automation. (2) Compliance with the requirements demanded for final debris retrieval equipment/device within the range where they are not examined by the element test. (3) Survey on the Fukushima-Daiichi site showed the temperature, dose, and generation of the mist steam the as environmental information inside the PCV, and indicated that a large amount of deposit in sludge form were accumulated in the stagnant water inside the reactor.	(1) With regard to the important items on the left, technical feasibility is required to be confirmed according to the plan of the element test which has been started. (2) Coordinating with the overall plan for the fuel debris retrieval, compliance with the requirements on the actual unit condition is required to be confirmed. (3) Accelerate the development by feeding back the issues to be solved which is clarified by the new findings, to the equipment development. Same as for the findings obtained by the planned internal survey. (4) Others ・Acceleration of the equipment development relating to the access to the fuel debris. ・Study on the application of the existing technologies with high reliability. ・Study on the development and securing of the human resource for equipment/device operation.
Developing system equipment and working areas	・Performing the conceptual study on the important system and facility important to the fuel debris retrieval method to be focused, it is required to confirm the feasibility of the method.	・Accelerate the verification of the feasibility through the conceptual studies on the systems, study the facilities in detail and its details are examined. ・Confirm the prospect for the storage areas to store the severely contaminated structures and fuel debris that retrieved. ・Study on the plot plan for the temporary storage, treatment and storing of the installation and operation of the facilities and retrieved fuel debris and confirm its prospect.

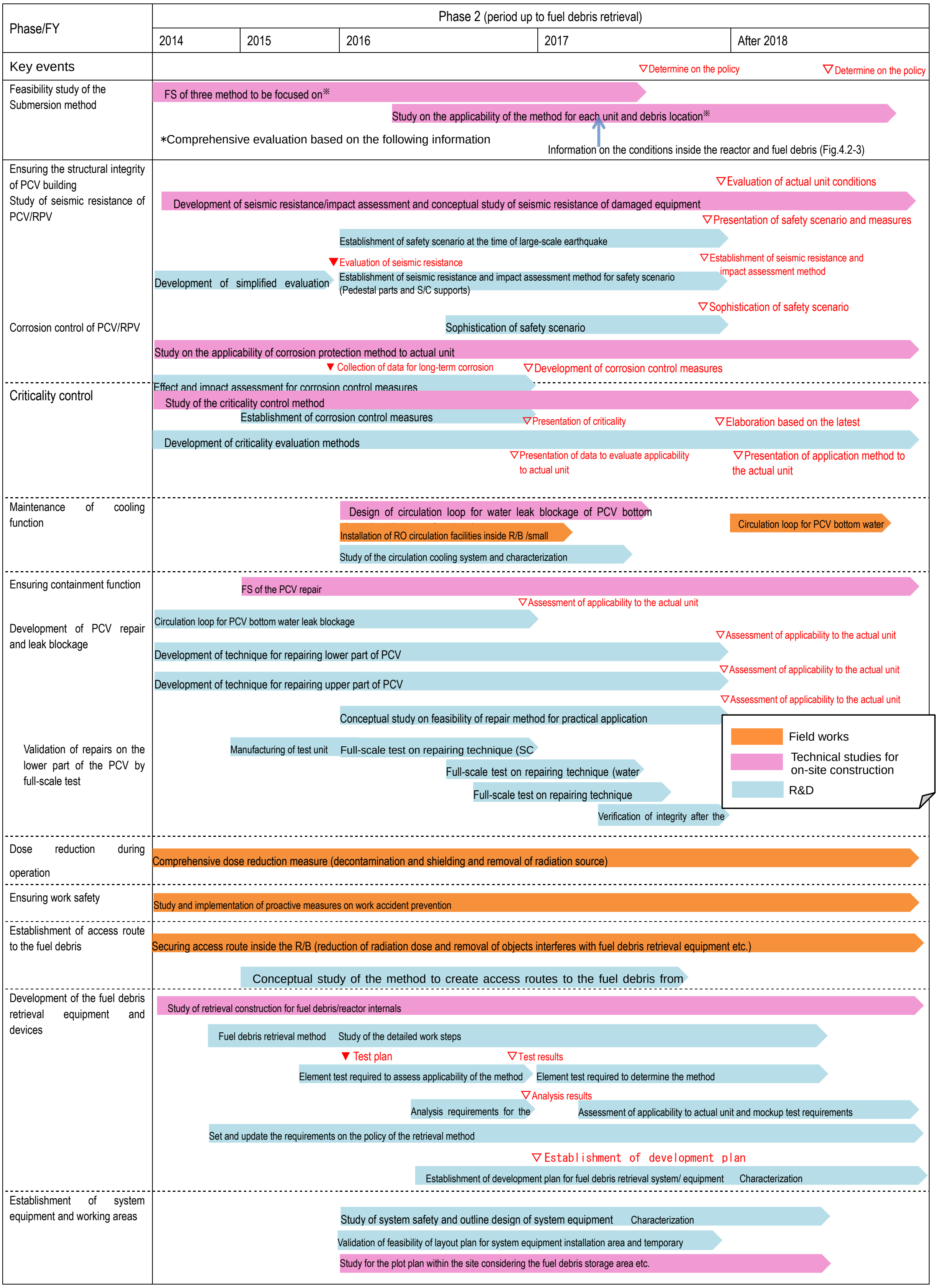


Fig. 4.3.2-31 Future actions for the feasibility study for fuel debris retrieval method

4.4 Study of the treatment for the retrieved fuel debris toward stable storage

This section describes the action status, evaluation/issue and future actions of the technology requirements for the stable collection, transport and storage of the retrieved fuel debris. Key issues in these technologies are as follows:

(1) Establishment of the system for collection, transport and storage of fuel debris

Studying the specifications of storage canister to collect fuel debris with various forms and properties, the system needs to be designed to collect the retrieved fuel debris while ensuring safety during the transportation and stable storage.

(2) Study on the safeguards for the fuel debris

Safeguard measures are required to be formulated so as to ensure reasonable progress of fuel debris retrieval work with transparency.

4.4.1 Establishment of the system for fuel debris collection, transport and storage

(1) Purpose

To establish a system from the designing and manufacturing of canisters to collect retrieved fuel debris to transporting and storing them on site and to secure safety and stable storage within the site.

(2) Major requirements

a. A system by which fuel debris is safely collected, transported and stored is developed.

Although the studies on the system is carried out by referencing the collection, transport and storage of the fuel debris in TMI-2 (Fig.4.4-3), the requirements on the collection works and canisters used for the Fukushima Daiichi NPS are more severe since the retrieval and collection works by remote control under a heavily contaminated environment and fuel debris and MCCI products inside the PCV (in and outside of the pedestal) are expected.

For this reason, the storage canisters and handling technology should be developed based on the requirements set up in accordance with the actual conditions of the Fukushima Daiichi NPS.

In addition, toward the decision on approaches to the retrieval method, the system plans of collecting, transporting and storing the fuel debris are to be developed based on the comprehensive understanding and considerations for the constraints on a fuel debris retrieval method and an actual working place.

Furthermore, the requirements for the handling of the fuel debris are to be established on the basis of "Submersion and Partial submersion" for collection, "Wet transfer or semi-dry transfer" for transport and "Wet storage and dry storage" for storage.

b. Prototype of storage canister and handling equipment and verification by mockup test

Verification should be conducted by using the prototype of the storage canister and handling equipment and mockup test using selected fuel debris retrieval method.

(3) Action status and evaluations and issues

a. A system by which fuel debris is safely collected, transported and stored is developed.

i) Development of overall plan and collection of related information.

The studies have been conducted for the overall technical development plan and its issues by collating the reference information on system development, input from the related projects, and output from other projects. Also, in FY2015, information including the storage canister design, safety assessment technology, fuel debris dry technology is collected from overseas organizations.

ii) Implementation of design of storage canister and study on transport/storage system as R&D

- Study of collection, transport and storage system of fuel debris

The studies on the collection, transport, and storage system are carried out based on requirements assuming various types of fuel debris in light of the conditions of the Fukushima Daiichi NPS. (For the flow plan of export from R/B, refer to Fig.4.3.1-9, and of the storage side, refer to Fig.4-4-2 and 4.4-3 respectively)

- Development of canister technology for storing the fuel debris /safety assessment technology method
  - ✓ The design conditions for storage canisters, basic functions and general shape were determined. (Fig.4.4-1 Reference)
  - ✓ Issues related to the safety assessment required for fuel debris canister design are collated and detailed study items were determined. (i.e. criticality evaluation, structural evaluation and countermeasure against hydrogen generation)

iii) Installation plan for facility storage of the fuel debris and its measures are required to be studied since much of the land within the Fukushima Daiichi NPS is now being used for the contaminated water tanks or temporary waste storage, and the space is limited for the system development.

b. Prototype of storage canister and handling equipment and verification by mockup test

Design is to be developed based on the basic specifications.

As mentioned above, now it is in the stage where development of the system up to storage process and studies on the basic specifications of storage canister are performed. The technical issues identified as the system develops should be handled appropriately. The issues clarified to date are described in the following section.

(4) Future actions

a. A system by which fuel debris is safely collected, transported and stored is developed.

i) Establishment of specifications of storage canister and handling equipment

- Detailed study regarding the safety assessment is to be conducted and the specifications storage canister for mockup test is to be fixed.
- Handling flow for the canister handling equipment is to be studied and its specifications for mockup are to be fixed.

In specific, since the pieces of fuel debris are expected to be collected when collection of the sludge accumulated in the PCV and purification of stagnant water, standards of handling the structures including fuel debris and molten fuel are to be clarified. Also, the specifications of storage canister should be applicable for the system equipment that has been studied separately.

ii) Development of transport and storage facility plan in line with the realization of the system

Preparation must be made for transport and storage before the commencement of the fuel debris retrieval. Although wet-type storage in the pool, vault-type or dry–type storage by metal cask may be considered as a storage method, basic design is to be developed according to the development of the transport and storage.

In specific, measures against drying method for fuel debris and hydrogen and oxygen generated by the water remained in the fuel debris will be required.

iii) Transport and storage plan based on the conditions of the Fukushima Daiichi NPS

Although the space within the site is used for the contaminated water tanks and temporary storage for the spent fuels and waste, the space required for transport and storage of fuel debris needs to be secured in coordination with the related operations.

iv) Safeguards and sampling for analysis are required to be studied according to the flow line from collecting fuel debris to storing within the site (requiring coordination with related projects).

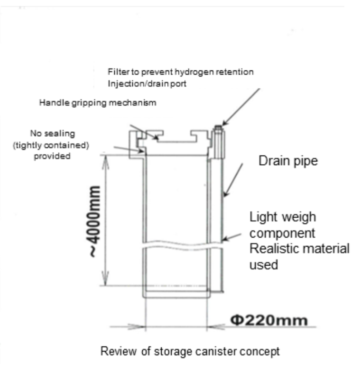
v) Clarification of safety requirements for storage canisters, and transport and storage facilities (including casks) considering the regulatory requirements.

The functional requirements such as the prevention of criticality, shielding and heat removal and the requirements for the structural strength evaluation are to be clarified considering the regulatory requirements.

b. Prototype of storage canister and handling equipment and verification by mockup test

Storage canister and handling equipment for mockup test are designed and manufactured as R&D and mockup test that combined with fuel debris retrieval equipment is to be performed.

Although the fuel debris is supposed to be stored in the storage canister temporality, processing and disposal of fuel debris will be determined in the Third phase in the Roadmap. In the processing and disposal process, transporting to the different container and dispose them is considered to be one of the options.



Safety requirements	Sub-criticality	Assuming diameter of φ220mm based on the maintenance of sub-criticality and storing efficiency
	Heat removal	Assuming fuel debris temperature of 300℃
	Containment	Use transfer container with lid
	Shielding	Use transfer container for weight saving
	Hydrogen generation	Install filters
	Fire prevention	Handling under nitrogen atmosphere with being submerged
Function	-Capable for both wet/dry types	
	-Install drain pipe on the port of water injection and discharge etc.	
	-Install remote control and gripping mechanism (with length up to 4000mm)	

Figure 4.4-1 Basic specification plan for storage canister (Provide by IRID)

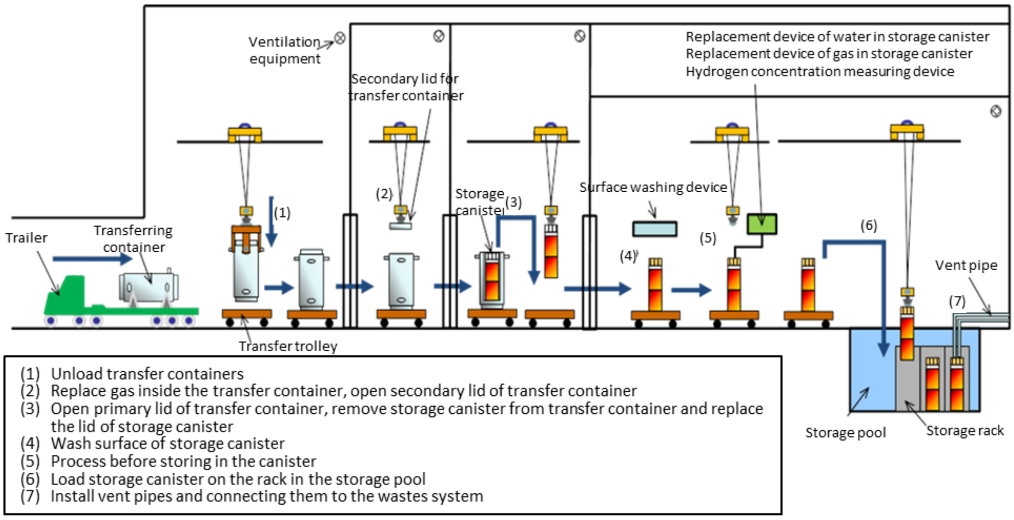


Figure 4.4-2 Flow plan for wet-type storing of fuel debris (Provide by IRID)

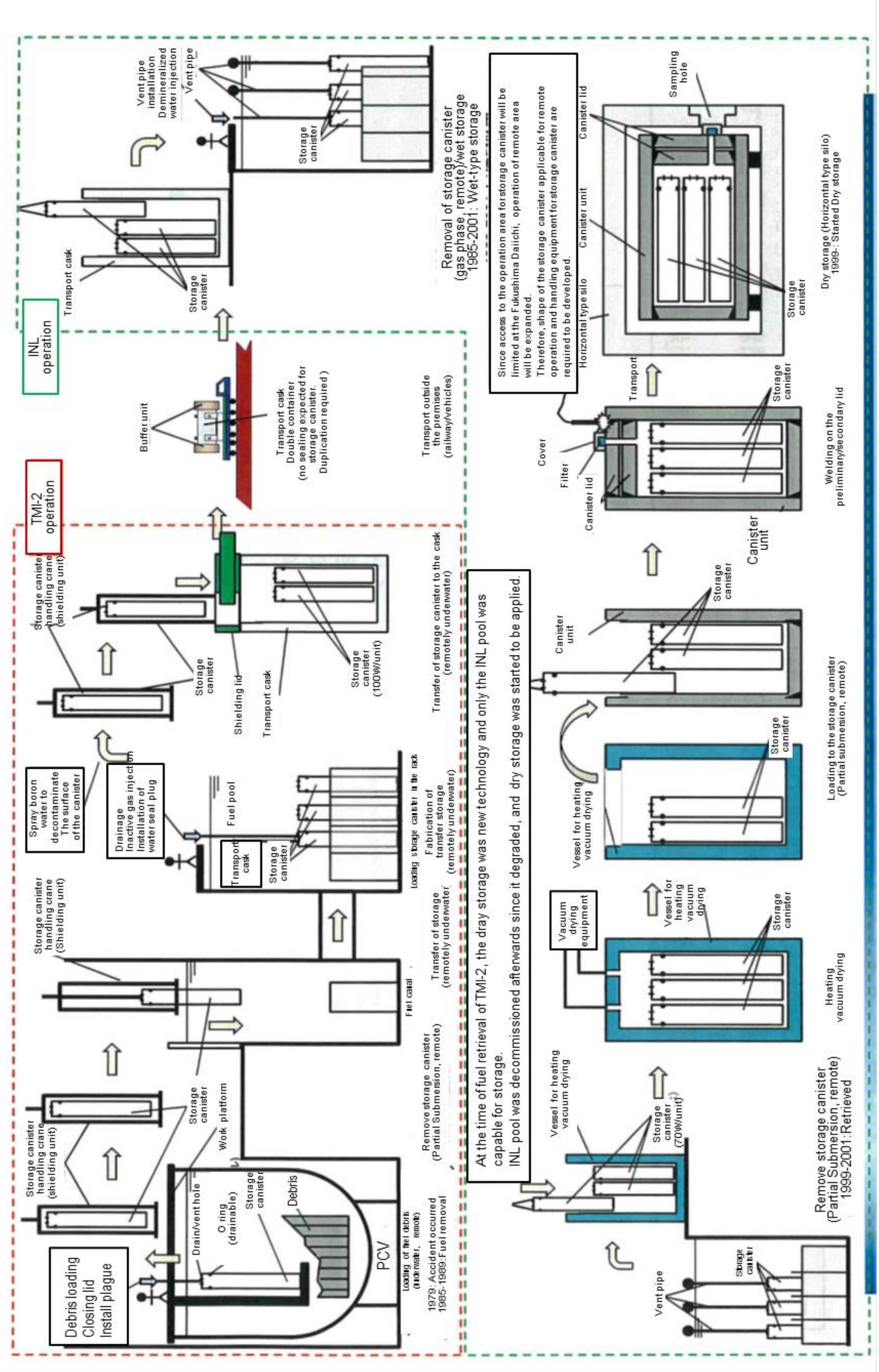


Figure 4-4 Overview of collection, transport and storage of fuel debris at TMI-2 (Reference: Study by IRID)

4.4.2 Study on safeguards approach for fuel debris

(1) Purpose

Technical study must be conducted with the relevant parties so that the transparent safeguards approach suitable to the fuel debris handling is formulated before the commencement of fuel debris retrieval.

The operator is required to set the material control and accountancy (MC&A) procedures and report the inventory of nuclear materials and inventory change to the Government. The Government is required to report the physical inventory of nuclear materials to IAEA based on the Japan-IAEA safeguards agreement and to show that undeclared nuclear materials will not be carried out. Also, the operator needs to accept the inspections conducted by the Government and IAEA and to show that they perform appropriate MC&A.

The fuel assemblies in the reactors of Units 1-3 at the Fukushima Daiichi NPS were melted due to the accident. Since those assemblies are estimated to be in a form of fuel debris, it is considered very difficult for nuclear materials to be diverted to nuclear weapons.

Considering such situations, a realistic safeguards approach is required to be formulated so as to correspond to the anticipated fuel debris retrieval method.

(2) Major requirements

a. Study on safeguards approach for fuel debris

- Study should be conducted so as to formulate the realistic safeguards approach which is agreed with the Government and IAEA and applicable for all the process from the retrieval to transport and storage of the fuel debris.

b. Schedule management for the formulation of safeguards approach

- Identifying items necessary for safeguards approach for fuel debris, a plan needs to be formulated to establish the realistic safeguards approach and control its implementation process.

(3) Action status and evaluations and issues

a. Study on the safeguards approach for fuel debris

- Current status

  ・Installation of remote-control monitoring cameras and radiation monitors in the periphery of the R/B of Units 1-3 at the Fukushima Daiichi NPS and on-site verification activity with short notice operational supports (SNOS) by the Government and IAEA are allowed.

- Evaluation and issues

  ・The fuel debris in Units 1-3 at the Fukushima Daiichi NPS is considered to be difficult to be determined the quantity of nuclear materials by the standard method highly accurately since the heterogeneous mixture of structural materials such as of metal and concrete including the control rod is estimated to have been created.

  ・Therefore, safeguards approach must be drafted considering the fuel debris retrieval method in order for the Government and IAEA to verify that undeclared nuclear materials will not be carried out.

  ・As described above, considering that the quantitative analysis of nuclear materials with a great accuracy is difficult to be performed for the fuel debris in Units 1-3 at the Fukushima Daiichi NPS, the method of MC&A is to be studied so that the Government and IAEA can agree.

・Collaborating with the Government and IAEA in a positive and timely manner, identified technological challenges are to be addressed to avoid leaving major issues in the fuel debris retrieval work.

b. Schedule management for the formulation of safeguards approach

- Safeguards approach needs to be studied and implemented according to the mid- and long- term Roadmap and actual progress status.

(4) Future actions

- Safeguards approach is to be studied and implemented according to the mid- and long- term Roadmap and actual progress status.

4.5 Studies to determine the approaches to the retrieval method for each Unit

The studies on the approaches to the retrieval method for each Unit are to be carried out in order to determine the fuel debris retrieval method for the initial Unit in the first half of FY2018, which is the hold point in the Roadmap.

A feasible fuel debris retrieval method is to be picked up from three methods to be focused on, through the comprehensive analysis and evaluation of internal PCV conditions, from the perspective of the status of the FS for the key issues in the fuel debris retrieval and the applicability of the method to access to the fuel debris. For each Unit, the fuel debris in the area where it is estimated to be located is to be selected as those which are preferentially removed, from the perspective of risk reduction effectiveness. The fuel debris retrieval method which subsequent study is preferentially performed for is to be selected from the feasible fuel debris retrieval methods according to the estimative index based on the Five Guiding Principles. Current study status and detailed approach are described below.

4.5.1 Study status of the internal PCV conditions analysis

The overview of the results of comprehensive analysis and evaluation for the fuel debris locations and amount and the plant survey information described in Section 4.2 is shown in Table 4.5-1.

Table 4.5-1 Fuel debris locations/amount currently estimated and overview for the results of plant information survey

	Location	Representative value of fuel debris weight (Unit:ton)*					
		Unit 1		Unit 2		Unit 3	
Fuel debris distribution estimation	Core region	0	0%	0	(0%)	0	(0%)
	RPV lower head	15	(5%)	42	(18%)	21	(6%)
	Inside the RPV pedestal	157	(56%)	145	(61%)	213	(58%)
	Outside the RPV pedestal	107	(39%)	49	(21%)	130	(36%)
	Total	279	(100%)	237	(100%)	364	(100%)
Plant investigation status	D/W water level	Approx. 3m from the bottom		Approx. 30cm from the bottom		Approx. 6.3m from the bottom	
	S/C water level	Almost full		Near the center of S/C Almost the same as the water level of torus room		Almost full	
	Radiation dose rate inside the PCV	Approx. 10Sv/h		Maximum approx. 73Sv/h		Maximum approx. 1Sv/h	
	Locations for leakage check etc.	Check leakage from expansion joint cover of sand cushion drain pipe in S/C vacuum break line		No trace of leakage was found in the upper part of the torus room.		Check leakage from expansion joint of main steam pipe D	

* The most probable value at this point

At this point, little fuel debris is located in the core region in either unit and small amount of them are accumulated at the bottom of the reactor. It is, however, estimated that most of them had fallen at the bottom of the D/W and reached to the outside of the RPV pedestal.

The uncertainties in the current status are to be reduced through the sensitivity analysis using analysis code. Also, through the PCV internal survey (B2 inspection for Unit 1, A2 inspection for Unit 2, e.g. inspection by swimming robot for Unit 3) and muon detection (Unit 2), accuracy of the comprehensive analysis and evaluation will be improved.

4.5.2 Overview of the approaches to the study on the fuel debris retrieval method

(1) Key issues of the Submersion and Partial submersion methods

The action statuses of the key issues are described for the Submersion method and Partial submersion method respectively.

a. Key issues in Submersion method

i) PCV repair and establishment of water level control system

Conducting the development and study of the method to repair the leakage of the PCV, PCV circulation cooling loops, and leakage water collection/ water level control systems, a system to control the PCV water level in a safe manner is required to be established. It is also required to ensure the construction quality, such as of the PCV repair construction and a long-term reliability and study the prevention of the leakage of contaminated water to the outside.

To prevent the leakage of contaminated water inside the building, keeping the water inside the torus room under the groundwater level will be a possible option and the FS is to be conducted based on the option.

The development has been performed to date focusing on the feasibility of the water sealing technology and method for the vent pipe and downcomer at the PCV bottom (under the torus room ceiling). The solutions of the issues which became apparent to date and tests regarding the construction quality and long term reliability is required to be focused on.

* Since complete water sealing for vent pipes and downcomers by pouring grout materials with remote devices may face great difficulties, some degree of leakage to the torus rooms needs to be taken into account. Not only the control of the differences in the water levels between the inside and outside that maintains torus room water level lower than groundwater level but also the studies will be required such as for the prevention of the leakage from the R/B in the case of a large amount of leakage during the fuel debris retrieval.

The development for the repair technology is to be carried out for the upper part of the PCV based on the on-site radiation dose situation.

ii) Ensuring structural integrity of the PCV and R/B according to its load and aged deterioration when submerged

The evaluation method for the structural integrity in the event of earthquake is to be established considering the deterioration caused such as by corrosion and the load applied to the PCV due to the submersion. Its measures are also studied based on the area requiring the reinforcement.

The evaluation of design basis seismic ground motion Ss is currently being conducted for seismic safety of the RPV/PCV and the peripheral equipment and facilities. The SC supports which are considered to have comparatively small margin will be evaluated through the detailed elastoplastic FEM analysis. Also melted fuels are estimated to have fallen at the bottom of the D/W. Through the analysis of the spread of fuel debris distribution and the internal survey inside the PCV and RPV pedestal, the impact caused by erosion to the RPV pedestal is to be evaluated, as needed.

iii) Establishment of criticality control during fuel debris retrieval work in case of increase in the PCV water level.

Subcritical state shall be maintained even when the water levels were varied and shapes of the fuel debris are changed during the fuel debris retrieval operation. Also workers’ exposure and adverse influences on environment shall be suppressed through shifting the state to subcritical state, in case of re-criticality accident.

Sub-criticality state is desired to be maintained by diluting neutron absorber such as boron with coolant. Studies are carried out for the feasibility of water quality management system including nuclide removal and environmental impact caused by the sodium pentaborate leakage. Current study indicated that water level up to RPV lower plenum may not cause re-critical state because of understanding on the chemical composition of fuel debris and that water level up to RPV core region also may not cause re-criticality if remaining fuel assemblies of Unit 2 is smaller than 5x5 (fuel assemblies). In parallel, mitigation measure of impact by re-criticality has been investigated just in case of criticality accident because conditions of fuel debris inside of the PCV/RPV are still unknown.

The method to maintain sub-criticality state of fuel debris has been studied for each fuel debris retrieval operation step and for the water level rising. Detail specification of sub-critical control method will be studied from the view point of actual applicability based on the results of evaluation of critical accident.

b. Key issues related to the Partial submersion method

i) Shielding for high radiation from fuel debris

It was confirmed that the prospect that the shielding can achieve 1mSv/h of dose rate on the operating floor even if all fuel debris are located in the core region. The accuracy of the information on radioactive source, e.g. fuel debris and FP distribution of each Unit is to be improved based on the result of the severe accident progression analysis and PCV internal survey. Also, studying the detailed dose evaluation based on the conditions depending on the method such as PCV water level, the reasonable shielding specifications are to be studied for the method applied for each Unit.

The same studies are to be conducted for the specifications of the shielding for the cells used for the Side access.

ii) Control of impact on workers and environment caused by dust scattering outside the building

The fuel debris retrieval methods and the dust scattering prevention method have to be established so as to prevent radioactive dust scattering to the outside of the building.

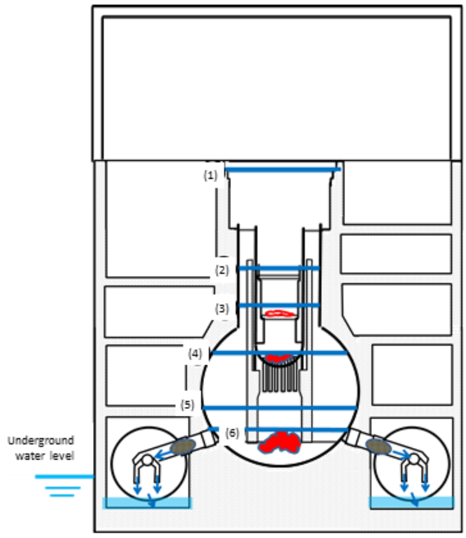
Containment of radiation dust is aimed by installing isolation walls (negative pressure control system) that maintains the negative pressure inside of the isolation walls after establishing a double isolation wall by the isolation wall that consists of the cells for fuel debris retrieval work and PCV and of R/B walls and container to be installed in the upper part. The conceptual studies on the operating cells and negative pressure control system are being performed.

iii) Verification of radiation resistance of fuel debris retrieval equipment

The retrieval equipment and devices will be exposed to the fuel debris with high radiation, and they are required to have radiation resistance that will encourage the smooth progress of retrieval work. The verification of radiation resistance for the manipulator, cutting equipment and camera that will be used close to the fuel debris are being performed in accordance with the conceptual design.

(1) Study on the PCV water level

The PCV water level will vary depending on whether the fuel debris is submerged or not, instead of whether Submersion method or Partial submersion method. If the fuel debris is submerged, shielding of radiation from the fuel debris and measures to the dust during the fuel debris cutting will be easier. According to the definition, Submersion method is used in the condition assuming all fuel debris are submerged (water level is higher than the upper part of the reactor core) and Partial submersion method is used where fuel debris are not submerged. For this reason, setting of the water level is relatively flexible for Partial submersion method. Although the PCV water level can be set consecutively, there are some water levels that will be changed as a function depending on the states. Those water levels as options are described in six stages in Figure 4.5-1. Since the degree of difficulty and advantages in the realization will be varied according to the options for each water level, the evaluation for the trade-off will need to be carried out.



(1) Full water of the reactor well (Full submersion method)

(2) Capable of reducing the radiation dose rate in the upper part of the operating floor since the core region of the reactor core shroud and activation metals of the upper grid plates will be submerged. (Submersion method)

(3) The fuel debris located in the area lower than the core region is possible. Criticality control for the fuel debris in the core region and the RPV lower plenum can be cut in submersion.

(4) The fuel debris located in the area lower than RPV lower plenum can be cut in submersion, but the criticality control of fuel debris in the RPV lower plenum will be necessary when the water level is raised.

(5) Water level where water level of the torus room will not exceed the groundwater level even assuming a large amount of leakage from the PCV.

(6) The fuel debris at the bottom of the D/W can be cut in submersion.

Figure 4.5-1 Options for PCV water level and its features

4.5.3 Study on access direction to the fuel debris

There are the options for the access directions which are Top access and Side access. Based on the concept of option, degree of difficulty in establishing the route to the fuel debris and applicability of access to the fuel debris locations and order of access direction are required to be evaluated. The matters to be considered to establish access routes are described in 4.3.2.7 and points to notice are arranged as below. Also, the matters to be considered in the applicability to fuel debris locations are described in Section 4.5.4.

・In the Top access method, the structures in the upper part, e.g. PCV upper head, RPV upper head, steam dryer and steam separator will be removed from the operating floor and access route to the fuel debris can be secured. However, the temperature of these components reached approx. 1000 deg. C at the time of accident and they may have been deformed. In this case, it will be difficult to retrieve them and could take a long time depending on the situation of each Unit. To access to the RPV lower plenum, cutting and removal of the lower reactor internals such as core support plate will additionally be required. To access to the bottom of the D/W (inside the RPV pedestal), boring on the RPV lower plenum and removal of CRD housings are required and it will takes a long time. Also, measures to the radiation dose reduction and shielding should be taken so as to conduct the works including the preparation on the operating floor.

・For the Side access method, new access route is to be established in the R/B to access to the side of the D/W. The existing equipment and piping are required to be removed under a high-dose radiation environment. The existing structures and fallen objects, e.g. PLR pump, valve and piping inside the PCV are required to be removed since they will be the obstacles for the retrieval of the fuel debris in and outside the RPV pedestal in the PCV. Comparing to the Top access method, dimension of access port is expected to be smaller and long period of time could be required. Also, in the Side access method, regarding the retrieval of the fuel debris located in RPV would be difficult to remove the structures, by creating an opening from the bottom of the D/W and then retrieve the fuel debris by way of the PCV side access route, and it may take a long time.

For both Top access method and Side access method, the environmental conditions including the on-site radiation dose rate is important for the access route establishment inside the R/B in each Unit. The feasibility and degree of difficulty for the methods are to be evaluated according to the possibility of the radiation dose reduction.

4.5.4 Feasibility study of retrieval for each fuel debris location

As described in Section 4.5.1, at current stage, fuel debris is estimated to be located both in the RPV (core region and the RPV lower plenum) and the bottom of the D/W. As described below, feasible fuel debris retrieval method is to be studied from the perspective of the feasibility of key issues to ensure safety during fuel debris retrieval work of fuel debris and applicability of the access to fuel debris of each location.

(1) Fuel debris inside the RPV (core region and the RPV lower plenum)

a. A Perspective of feasibility of key issues in ensuring safety during the fuel debris retrieval

The key issues to ensure safety are varied depending on either fuel debris are submerged or not.

For the cases where fuel debris are submerged key issues correspond to ones described in 2(1) a of Section 4.5. for the Submersion method, while for the cases where fuel debris are not submerged, key issues correspond to ones described in 2(1) b. of Section 4.5 for Partial submersion method.. The keys to the feasibility are considered as follows.

Since the impact by the PCV water level is significant for the feasibility, appropriate water level is required to be studied. (Refer to Figure 4.5-1)

- Case: Submerging the fuel debris
  - ・Feasibility of the repair of the PCV leak locations up to the height level of the core region (Whether it can be performed within realistic construction period and exposure dose).
  - ・Feasibility of the system to collect leakage water from the imperfect PCV repaired areas.
  - ・Feasibility of the system for criticality control during the process of cutting fuel debris underwater or water filling process for the PCV.
  - ・Feasibility of the negative pressure control system to prevent leakages from the imperfect PCV repair areas, operating cells and R/B containers in order to contain the radioactive dust not to migrate into the air when the submerged water level of the fuel debris is low.
- Case: Without submerging the fuel debris
  - ・Impact on the aseismic performance such as of R/B caused by the shielding facilities installed on the operating floor or else.
  - ・Feasibility of the negative pressure control system to prevent leakages from the operating cells, R/B containers and PCV repair areas to contain the radioactive dust not to migrate into the air when cutting the fuel debris in the air

b. Applicability of each retrieval method to fuel debris location

Basically the applicability of Submersion -Top access or Partial submersion-Top access method to the fuel debris inside the RPV (core region and the RPV lower plenum) is high as described in Table 4.3.1-1 of Section 4.3.1 and Appendix 4.18.

However, Partial submersion-Side access method might be used according to the evaluation by the detailed locations and amount of existing fuel debris or reasonability for a whole retrieval work.

(2) Fuel debris at the bottom of the D/W (inside and outside the RPV pedestal)

a. Perspective of feasibility of key issues in ensuring safety during the fuel debris retrieval

The fuel debris at the bottom of the D/W is estimated to be basically submerged with current water level and fuel debris retrieval is possible to be performed under submerged condition. The key to the detailed feasibility is considered as follows.

・Feasibility of the system to collect leakage water from the PCV repair areas.

The degree of feasibility is supposed to be higher by setting the PCV water level to the current level or + α than the case in which the fuel debris inside the RPV is submerged.

・Feasibility of the system for criticality control when cutting the fuel debris underwater

・Feasibility of the negative pressure control system to prevent leakage from the imperfect PCV repair areas and of the operating cells to contain the radioactive dust not to migrate into the air when the water level is not enough for submerged cutting process of the fuel debris.

b. Applicability of each retrieval method to fuel debris location

As described in Table 4.3.1-1 of Section 4.3.1 and Appendix 4.18, applicability to the fuel debris located inside the RPV pedestal is considered to be high for both Submersion-Top access and Partial submersion-Side access methods. The works for the removal of obstacles including the structures in upper part are, however, needed to be considered long when accessing the bottom of the D/W by Top access method as described in 4.5.3. The applicability of the Partial submersion-Side access method to the fuel debris outside the RPV pedestal is considered to be high basically. The Top access method might, however, be used according to the evaluation by the detailed locations and amount of existing fuel debris or reasonability for a whole retrieval construction.

4.5.5 Studies on decision on approaches to retrieval method

As described above, inspection and evaluation for internal PCV condition analysis of each Unit and feasibility for three methods of fuel debris retrieval to be focused on are being studied. The results are aimed to be summarized in FY2016 to contribute to the "Determination of fuel debris retrieval policies for each unit" in summer of 2017.

According to the current estimation of the internal PCV conditions, fuel debris of all Units are scattered in the RPV lower plenum and at the bottom of the D/W (in and outside of the RPV pedestal). All the fuel debris scattered in each Unit are not always retrieved by one method and the policy may be made by combining multiple methods according to the fuel debris locations. In such case, in parallel with the retrieval work for initial location of retrieval, inspections and studies are considered to be performed for the fuel debris in different locations. The work in the next stage may be continued with revising the plan for the fuel debris retrieval method.

In the "Determination of fuel debris retrieval policies for each unit," the results of the studies and findings obtained to date, fuel debris location to be addressed first and the likely method is to be selected for each Unit from the perspective of ensuring safety.

In particular, the following studies will be conducted to evaluate, such as the risks that affect the fuel debris retrieval work.

(1) Evaluate the effect of risk reduction through the resolution of the instability of the inside of the PCV/RPV for each unit and fuel debris location based on the estimation results of fuel debris properties and amount,

(2) Evaluate access route and retrieval method for each Unit and fuel debris location, including PCV water level are nominated and risk regarding ensuring safety, such as criticality that might be caused by the retrieval work and the leakage of radioactive materials based on the features of three methods to be focused on and the results the studies.

(3) Including the evaluation of (1) and (2), select the first fuel debris to be retrieved and its method in a comprehensive consideration of evaluation for the estimative index based on the Five Guiding Principles described in the 4.5.6. In specific, the degree of difficulties of technical development, site constrain including the exposure dose imposed by the work and required areas, and period of time for preparation are important.

    Also, regarding the Unit where first retrieval work is expected, without any experience of construction before, the smaller degree of difficulties of fuel debris retrieval work should be made.

(4) With regard to the fuel debris other than those retrieved at first, the access route and retrieval method depending on the PCV water level are to be studied to make sure that initial retrieval work will not affect the following work.

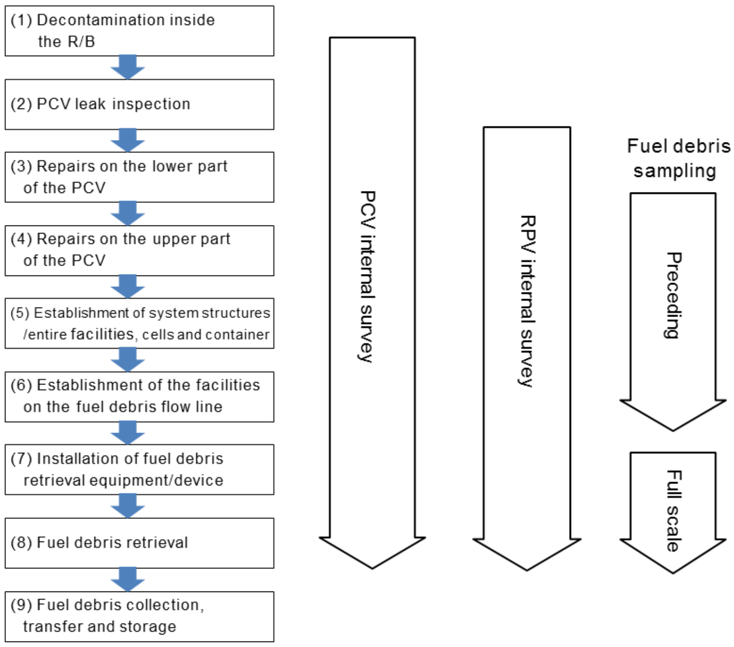
If construction periods of multiple units are overlapped depending on the order and timing of the commencement of fuel debris retrieval for three units, construction risks will be and required human resources will be increasing. On the other hand, if constructions are conducted in series, there is an advantage to reflect the results of the preceding construction to the next construction but overall construction period will be prolonged. Based on the above, the study on the comprehensive optimization will proceed based on the examinations of a whole image of fuel debris retrieval in three Units.

After making a decision on approaches to the retrieval method, the preparation should be made, focused on the fuel debris to be retrieved first for each unit, for the basic designs of system equipment and retrieval equipment for the fuel debris and specific installation area plan while performing the inspections for internal PCV condition analysis. Detailed studies and technical developments need to be accelerated to determine the method to retrieve the fuel debris from the initial unit.

Since the uncertainties in the internal PCV conditions is high, it is important to carry out the studies in stages in parallel with the investigation inside the reactor.

The flow of the on-site works including the preparation regarding the fuel debris retrieval is shown in Figure 4.5-2.

It is required to establish a series of work flow in order to realize the fuel debris retrieval. The preparation for the engineering and permission and authorization will be required for each work items. A series of these project managements will be important.



*It actually does not simply proceed in a series but works to be performed in parallel are considered to be occurred.

Figure 4.5-2 Flow of the on-site work for fuel debris retrieval

4.5.6 Estimative index and perspective of evaluation based on the Five Guiding Principles

The estimative index based on the Five Guiding Principles is shown in Table 4.5-2. Based on this, the examples of the studies from the detailed perspective are described below.

Table 4.5-2 Estimative index based on the Five Guiding Principles

Five Guiding Principles		Estimative index
Safe	Reduction of risks posed by radioactive materials and ensuring work safety	Containment of radioactive materials (environmental impact)
		Radiation workers' exposure (operation time, environment)
		Ensuring work safety
		Effect of risk reduction
Proven	Highly reliable and flexible technology	Level of difficulty of technical development and TRL
		Conformity to requirements
		Flexibility of uncertainties and robustness*
		Alternative plans
Efficient	Effective utilization of resources (e.g. human, physical, financial, space)	Securing human resources (researchers, engineers and workers)
		Reduction of generated waste
		Cost (technical development, design and field work)
		Securing working and storage areas
		Impact on the subsequent processes of decommissioning
Timely	Awareness of time axis	Early commencement of fuel debris retrieval
		Time required for fuel debris retrieval
Field-oriented	Emphasize the Three Actuals (actual field, actual things and actual situation)	Workability (environment, accessibility, and operability)
		Conservativeness (maintenance and actions against troubles)
		Applicability to each unit

*The capability to maintain the robust function even when the condition is changed to a certain extent from what is expected

(1) Safe

a. Containment of radioactive materials (environmental impact)

This is an essential evaluation item to minimize the risks that involved in the fuel debris retrieval. Environmental impacts are required to be assessed by the containment performance of cooling water and radioactive dust. Also, scattering of the radioactive dust should also be considered for the preparatory work stage for the installations of the system equipment and retrieval equipment in advance of the retrieval work.

b. Radiation workers' exposure

Since fuel debris retrieval construction is basically performed by remote control, from the perspective of workers exposure, the radiation exposure caused by the PCV water sealing construction and preparatory work for the installations of the system equipment and retrieval equipment which are considered to include the manual work will be dominant. However, the possible workers' exposure at the time of anomalies during the fuel debris retrieval construction is also required to be considered.

c. Effect of risk reduction

In the state where retrieval of the fuel debris is completed for each Unit, the amount of risk reduction is basically the same. The earlier it reaches this state, the higher the amount of risk reduction is.
The risks of the fuel debris located inside the RPV and at the bottom of the D/W vary depending on its properties and amount in each state of the fuel debris. Therefore, the effect of reduction of the risk over time will be improved by retrieving the fuel debris with a large risk and fuel debris whose stability of current condition is uncertain in advance.

(2) Proven

a. Conformity to requirements

The applicability of the important technical requirements in ensuring safety is to be evaluated based on the number of issues and degree of importance of required approach to the solution of the issues identified from the "feasibility study on the method."

b. Flexibility for uncertainties

It should be confirmed that the concept of the method enables measures to be taken flexibly according to the actual situation or enables fuel debris cutting tools to be changed easily according to the actual fuel debris properties, even if it is out of the scope of the assumption under the condition where information regarding the damaged condition of the equipment that impedes the fuel debris retrieval cannot be obtained sufficiently.

(3) Efficient

a. Reduction of generated waste

The amount and form of radioactive waste expected to be caused by the construction are to be evaluated. The physical weight of the upper reactor internals to be removed by the Top access method is not considered small from the perspective of the comparison between the methods.

b. Cost

Although it is considered difficult to conduct an evaluation in the stage of the decision on approaches to the retrieval method, where the conditions of the fuel debris and circumference environment are uncertain, the conditions of estimation for the fuel debris distribution and rough comparison such as by the features of the concept for the methods. The impact caused by the repair cost to block water leak from the PCV will be large depending on its range.

c. Securing working and storage areas

The fuel debris retrieval construction, outlines of the installation areas for the facilities required for the preparatory work and work areas, and the prospect for securing those are evaluated. The scope includes the system equipment, retrieval equipment and storage facility for storage canisters for the fuel debris and storage facilities for severely contaminated structures.

(4) Timely

a. Early commencement of fuel debris retrieval

The preparation period required before the commencement of the fuel debris retrieval (preparation for the permission and authorization, inspection and construction of the PCV water sealing, preparation for the installation of the system equipment, installation and commissioning of the system equipment and fuel debris retrieval equipment, manufacturing of storage canister and installation of the storage facility for storage canisters) is to be evaluated.

b. Time required for fuel debris retrieval

Although it is considered difficult to conduct an evaluation in the stage of the decision on approaches to the retrieval method, the conditions of estimation for the fuel debris distribution and rough comparison for the period of fuel debris work such as by the features of the concept for the methods.

(5) Field-oriented

a. Workability (environment, accessibility, and operability)

With regard to fuel debris retrieval construction, a concept of the method (retrieval equipment layout, flow line and work procedures) should be planned based on the site conditions of the target Unit. Also the issues, e.g. the interference with the on-site construction such as SFP fuel retrieval construction and contaminated water control is to be confirmed while evaluating the workability including the preparatory work.

b. Conservativeness (maintenance and actions against troubles)

The area division plan such as by the retrieval equipment layout and operating cells are to be evaluated whether it can secure required maintenance area. Also conceptual designs of equipment are to be evaluated, whether the conceptual designs are planed so as to replace the retrieval equipment and maintenance range.

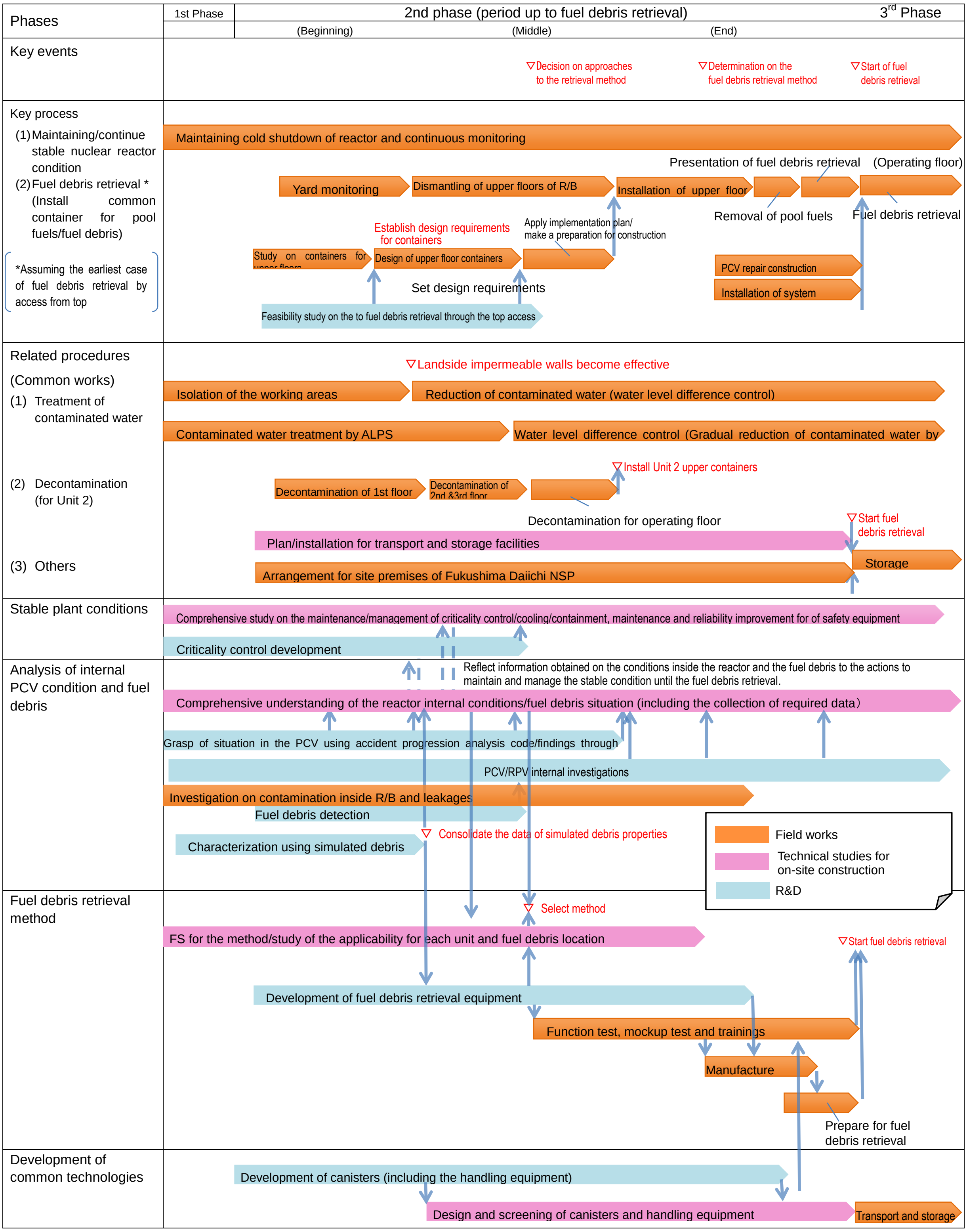


Fig. 4.5-3 Fuel debris retrieval Whole process

5. Strategic plan for waste management

5.1 Study policy on the strategic plan for waste management

The knowledge and experiences on the nuclear facilities that caused accidents are summarized in a number of documents released internationally. The document, NW-T-2.7 “Experiences and Lessons Learned Worldwide in the Cleanup and Decommissioning of Nuclear Facilities in the Aftermath of Accidents”[16] released by IAEA describes the responses should be taken after the accident in chronological order in concrete terms from emergency response, stabilization of the response, the post-accident cleanup, safe enclosure, to the final decommissioning and site remediation. It also describes that the strategic plans had to be established in the initial phase after the accident. The Fukushima Daiichi NPS is in the phase of post-accident cleaning and the Roadmap released by the Japanese Government was recognized as one of the examples. It is important to carry out the radioactive solid wastes management described in the Roadmap[17] in accordance with the Five Guiding Principles (Safe, Proven, Efficient, Timely and Field-oriented) described in Chapter 3.

In the radioactive solid wastes management, it is foremost important to reduce the amount generated, and take realistic measures such as minimization of carry-in materials, reuse, and recycling in accordance with the site conditions.

The radioactive solid wastes generated nevertheless are to be segregated and stored safely depending on the properties to reduce risks for the time being.

In parallel with the storage, characterization of various types of radioactive solid wastes is to be conducted and the wastes are to be categorized by their properties, so that safe and optimal processing and disposal concept and management can be studied based on the proven technologies.

The safety regulations on storage, processing and disposal management of the wastes will be reviewed as necessary, and operations such as decommissioning will be carried out while ensuring the safety.

The waste management is characterized as long-term activities such as developing the necessary programs and systems, and determining the prospect of implementing waste disposal. As part of this, the basic concept of processing and disposal for radioactive solid wastes will be compiled in FY2017 and the prospects of a processing /disposal method and a technology related its safety is made clear by around FY2021.

Radioactive solid wastes from events such as the accident at the Fukushima Daiichi NPS are different from wastes that have been generated by nuclear power plants in that they include radioactive materials released from failed fuel, contain salt, and are generated in large quantities. Therefore, the future plan for processing and disposal is now under study while the waste characterization is continued. The waste storage is also being carried out placing safety as the first priority.

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[16] IAEA, Experiences and Lessons Learned Worldwide in the Cleanup and Decommissioning of Nuclear Facilities in the Aftermath of Accidents, IAEA Nuclear Energy Series No. NW-T-2.7, Vienna (2014).

[17] The “radioactive solid wastes” in the Roadmap refers to “some rubble waste generated after the accident may not be wastes or radioactive wastes due to on-site reuse or other measures. However, these wastes and the radioactive solid wastes that were generated before the accident and have been stored at the Fukushima Daiichi Nuclear Power Station are included in the ‘radioactive solid wastes.”

Although sufficient information on the properties of radioactive solid wastes has not been gathered yet in the present circumstances, it is important to compile the basic concept for ensuring safety in radioactive waste management in general, and to formulate an action policy for the items that may greatly affect the radioactive solid wastes management in future.

As the strategic plan for waste management, studies have been done in the following steps.

(1) Review the general principles compiled internationally for ensuring the safety of radioactive waste disposal and develop an appropriate approach for waste management that emerges from these principles, giving shape to the measures for the disposal of radioactive solid wastes.

(2) Evaluate the status of the solid waste management activities described in the current roadmap, and identify issues that may affect future waste management activities or the plan of waste management.

(3) Describe issues should be addressed or noted from the present on the mid- and long-term solid waste management strategy by taking into account the principles in (1) and the issues identified in (2) above.

(4) Describe future actions on the radioactive solid wastes management based on the (2) and (3) above including the R&D.

This strategic plan is to be reviewed and elaborated in accordance with the future development.

5.2 International safety principles on radioactive waste management

Summarized below are the principles for ensuring safety in implementing general radioactive waste management developed by international organizations such as the IAEA and the International Commission on Radiological Protection (ICPR).

5.2.1. Principles for ensuring the safety of radioactive waste disposal

ICRP describes the concept of the radiological protection in relation to the radioactive waste disposal in Publ.46 (1986)[18], Publ.77 (1998)[19] and Publ.81 (1998)[20] in systematic manner. Publ.81 supplemented and revised the recommendation provided in Publ.46 in light of the international progress and was released to clarify the concept further. Although ICRP released Publ.103 (The 2007 Recommendations)[21] and then Publ.122 (2013)[22] in which Publ.103 is applied for geological disposal, Publ.81 above is said to be still valid in each document. The introduction of ICRP Publ.81 states that waste disposal strategies can be divided into two conceptual approaches; ‘dilute and disperse’ or ‘concentrate and retain’, and both strategies are in common use and are not mutually exclusive. The main part of Publ.81 deals with the radiological protection

[18]ICRP, 1985. Principles for the Disposal of Solid Radioactive Waste. ICRP Publication 46. Ann. ICRP 15 (4).
[19]ICRP, 1997. Radiological Protection Policy for the Disposal of Radioactive Waste. ICRP Publication 77. Ann. ICRP 27 (S).
[20]ICRP, 1998. Radiation protection recommendations as applied to the disposal of long-lived radioactive solid wastes. ICRP Publication 81. Ann. ICRP 28 (4).
[21]ICRP, 2007. The 2007 Recommendations of the International Commission on Radiological Protection. ICRP Publication 103. Ann. ICRP 37 (2-4).
[22]ICRP, 2013. Radiological protection in geological disposal of long-lived radioactive solid wastes. ICRP Publication 122. Ann. ICRP 42(3).

of members of the public following the disposal of long-lived radioactive solid wastes using the ‘concentrate and retain’ strategy.

On the other hand, IAEA’s “Disposal of Radioactive Waste”[23] (Specific Safety Requirements No. SSR-5, paragraph 1.10) states the aims of disposal of radioactive solid wastes, and Safety Guide WS-G-2.3[24] describes “Regulatory control of radioactive discharges to the environment” on radioactive gaseous and liquid materials as waste management instead of waste disposal, which corresponds to ‘dilution and dispersion’ of ICRP.

Based on a comprehensive understanding of the above, the safety principles on disposal of waste including gaseous and liquid wastes are as follows:

Waste management is to be addressed based on one or combinations of these safety principles in order to prevent significant effects on health.

i) To contain the waste;

ii) To isolate wastes from the accessible biosphere and reduce substantially the likelihood of, and all possible consequences of, inadvertent human access to the waste;

iii) To inhibit, reduce and delay the migration of radionuclides from the waste to the accessible biosphere at any time;

iv) To ensure that the amounts of radionuclides reaching the accessible biosphere due to any migration from the disposal facility are such that possible radiological consequences are acceptably low at all times;

v) To control the release of radioactive materials to ensure that their concentrations are at the level that will not cause significant effects on health.

The institutional systems for processing and disposal of radioactive waste in Japan have been formulated, and the safety regulations requires safety of the facility by means of the above i) to v) and specifies the radiation dose rate and concentration that will not cause significant effects on health. There are some radioactive wastes generated by the operation of normal nuclear facilities whose policies and regulations on disposal have not yet been established. The current status of safety regulations in Japan regarding processing and disposal of radioactive waste is shown in Table 5.2-1.

[23]IAEA, Disposal of Radioactive Waste, IAEA Safety Standards, No. SSR-5, IAEA, Vienna (2011).

[24]IAEA, Regulatory Control of Radioactive Discharges to the Environment, IAEA Safety Standards, No. WS-G-2.3, IAEA, Vienna (2000).

Table 5.2-1 Summary of status on studies regarding radioactive waste disposal

Table 5.2-1 Summary of Current Status of Safety Regulations in Japan regarding Processing and Disposal of Radioactive Waste

Reported: Reports of the advisory board are submitted / Established: Necessary regulations are established

Law	Categories of radioactive waste			Atomic Energy Commission	Former Nuclear Safety Commission, etc.*3						Laws on safety regulations, etc.		
				Disposal policy	Approaches to safety regulations			Upper limit of radioactive concentration		Safety guideline	Cabinet order	Licensing Criteria Rule*4 & Explanation	Disposal Rule*5
Reactor Regulation Law*1	High-level radioactive waste			Reported (May. 1998)	Provisionally reported (Nov. 2000)	Commonly important issues Reported (Jun. 2004)				To be developed	Established (Dec. 2007)	To be established	Established (Mar. 2008)
Reactor Regulation Law*1	Low-level radioactive waste	Wastes from the nuclear power station	Comparatively high-level waste (Intermediate depth disposal)	Reported (Oct. 1998)	Reported (Sept. 2000)		Reported (Jul. 2007) (Except for uranium waste)	Reported (Sep. 2000)	Reported (Jul. 2007)	Reported (Apr. 2010)	Established (Dec. 2007)	Under consideration	Established (Mar. 2008)
Reactor Regulation Law*1	Low-level radioactive waste	Wastes from the nuclear power station	Comparatively low-level waste (Near surface disposal)	Reported (Aug. 1984)	Reported (Oct. 1985)			Reported (Feb. 1987, Jun. 1992)		Reported (Mar. 1988)	Established (Mar.1987, Sept.1992)	Established (Dec. 2013)	Established (Jan. 1988, Feb. 1993, Mar. 2008)
Reactor Regulation Law*1	Low-level radioactive waste	Wastes from the nuclear power station	Very low-level waste, e.g. concretes (Near surface trench disposal)					Reported (Jun. 1992)		Reported (Jan. 1993)	Established (Sep. 1992)		Established (Feb. 1993, Mar. 2008)
Reactor Regulation Law*1	Low-level radioactive waste	Wastes from the nuclear power station	Very low level waste, such as metals (Near surface trench disposal)					Reported (Sep. 2000)		Reported (Mar. 2001)	Established (Dec. 2000)		Established (Mar. 2008)
Reactor Regulation Law*1	Low-level radioactive waste	Long-lived low heat-generation radioactive waste		Reported (Mar. 2000, Apr. 2006)	Reported (Apr. 2006)			(Except for uranium waste)		Reported (Apr. 2010, partially to be developed)	Established (Dec. 2007)	Established (Dec/2013, to be partly established)	Established (Mar. 2008, partially to be established)
Reactor Regulation Law*1	Low-level radioactive waste	Uranium waste		Reported (Dec. 2000)						To be developed		To be established	To be established
Reactor Regulation Law*1	Low-level radioactive waste	Waste from research institutes,etc.*6		Reported (Jun. 1998)						Reported (Jan. 1993*9 partially to be developed)		Established (Dec/2013, to be partly established)	Established (Mar. 2008, partially to be established)
RI Act*2	Low-level radioactive waste	RI waste			Reported (Jan. 2004)			Under consideration			Established (May. 2005)		Established (Jun. 2005, partially to be established)

Law	Categories of radioactive waste			Atomic Energy Commission	Former Nuclear Safety Commission, etc.*3			Laws on safety regulations, etc.	
				Disposal policy	Clearance level		Law	Cabinet order	Regulation*8
Reactor Regulation Law*1	Wastes not necessary to be treated as radioactive waste	Waste from reactor facilities	Major reactor facilities (including research reactor)	Reported (Aug. 1984)	Reported (Mar. 1993)	Reported (Dec. 2004)	Established (May. 2005)		Established (Dec. 2005)
Reactor Regulation Law*1	Wastes not necessary to be treated as radioactive waste	Waste from reactor facilities	Heavy water reactor, Fast reactor		Reported (Jul. 2001)				
Reactor Regulation Law*1	Wastes not necessary to be treated as radioactive waste	Waste from nuclear fuel facility	Nuclear fuel use facility (Facilities to handle irradiated fuels and materials)		Reported (Apr. 2003)				
Reactor Regulation Law*1	Wastes not necessary to be treated as radioactive waste	Waste from nuclear fuel facility	Facilities using nuclear fuel, nuclear fuel fabrication facilities (Facilities using uranium only)		Reported (Oct. 2009)				Established (Feb. 2014)
Reactor Regulation Law*1	Wastes not necessary to be treated as radioactive waste	Waste from nuclear fuel facility	Other facilities regarding nuclear fuel		To be considered				To be established
RI Act*2	Wastes not necessary to be treated as radioactive waste	Waste from RI facility	Facilities subject to Radiation Hazard Prevention Act*7		Reported (Jan. 2010)		Established (May. 2010)	Established (Mar. 2012)	Established (Mar. 2012)

Source: Website of the Agency of Natural Resources and Energy (http://www.enecho.meti.go.jp/category/electricity_and_gas/nuclear/rw/gaiyo/gaiyo04-1.html); modifications were made to the content.

*1 Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors (hereinafter, "the Reactor Regulation Act")

*2 Act on Prevention of Radiation Health Impairment Due to Radioisotopes, etc. (hereinafter, "the RI Act")

*3 Including the Radiation Safety Regulation Committee concerning the RI Act of the Ministry of Education, Culture, Sports, Science and Technology

*4 Rule on Standards for the Location, Structure and Systems of Category 1 Waste Disposal Facilities, Rule on Standards for the Location, Structure and Systems of Category 2 Waste Disposal Facilities

*5 In the Reactor Regulation Act, the Rule on Category 1 Waste Disposal Activities for Materials Contaminated by Nuclear Source Material or Nuclear Fuel Material, the Rule on Category 2 Waste Disposal Activities for Materials Contaminated by Nuclear Source Material or Nuclear Fuel Material, the Rule on the Location, Structure and Systems of Waste Disposal Facilities; In the RI Act, the RI Act Enforcement Ordinance

*6 Radioactive waste from research facilities that is subject to the Reactor Regulation Act.

*7 Approval notified users, notified distributors, notified leasing companies, licensed waste disposal service providers

*8 In the Reactor Regulation Act, the Rule on Confirmation of Activity Concentrations in Construction and Other Materials used in Factories in Refining Business, etc., the Rule on Confirmation of Activity Concentrations in Reactors, etc. for Testing and Research; In the RI Act, the RI Act Enforcement Ordinance

*9 There are other reports in March 1988 and March 2001.

5.2.2. Examples of application on safety principles for radioactive waste disposal

For the specific radioactive waste disposal, measures are taken based on one or combinations of these safety principles in order to prevent significant health effects.

(1) For the near surface disposal of low-level radioactive solid wastes, safety is ensured by isolating the waste from biosphere by, for example, enclosing or solidifying them in a container which meets the requirements of transport, safe handling and dispersion prevention, as well as combining conditioned waste form with engineered barrier and surrounding natural barrier.

In this case, leakage of radioactive materials is prevented or reduced by the containers and the engineered barrier, and as a result, the migration of radioactive material to the natural barrier is reduced, and furthermore, the function of the natural barrier delays the migration of radioactive materials to the biosphere. Thus the concentration of radioactive material is reduced, so that there will be no effect on health even if the radioactive materials reach the biosphere. In addition, when the concentration of radioactive materials is extremely low, it is possible to dispose the waste in a disposal facility without an engineered barrier (i.e. trench disposal) so as to prevent any effects on health.

(2) For the near surface disposal, safety regulation is in place to impose institutional control, such as restriction of specified acts until the concentration of radioactive materials would become lower than safe and acceptable level so as not people to access inadvertently or excavate the site.

(3) For radioactive solid wastes with high concentration of radioactive materials, the measure is taken to reduce concentration of radioactive material by disposing the waste deep under the ground to secure a longer migration pathway of radioactive material to delay migration of radioactive material to the biosphere. In addition, the safety is ensured by a greater disposal depth so that it is not necessary for depending on the institutional control, such as restriction of specified acts.

(4) For gaseous radioactive waste, concentration and amount of radioactive materials are reduced as much as reasonably achievable by using a treatment system, and then, the gaseous waste is released to the environment from a stack with a diffusion function. For example, a gaseous waste disposal facility in a nuclear power station is equipped with activated carbon type noble gas hold-up system. When off-gas goes through the activated carbon filled adsorption column, the noble gas repeats adsorption and desorption in the activated carbon while it moves through the system, and the radioactivity is attenuated over the course of time before reaching the stack. The exposure is controlled by reducing the concentration of radioactive materials to below the standard value defined in the regulations.

(5) For liquid radioactive material, concentration and amount of radioactive materials are reduced to the lowest level reasonably achievable using a treatment system so as to be released to the environment where dilution effect can be expected. For example, as for the liquid waste at nuclear power station, the concentration and amount of radioactive materials released are reduced as much as reasonably achievable by means such as of accumulation, filtering, evaporation treatment and ion exchange. Treated liquid waste is released from the outlet of the condenser cooling water after being confirmed that the concentration of radioactive materials is below the level specified in the regulations.

5.2.3. Appropriate radioactive waste management

In IAEA's Safety Requirements GSR-Part 5[25], predisposal is positioned as a management of radioactive waste covers all the stages from generation to disposal including processing, storage and transport. The processing of radioactive waste is divided into pretreatment, treatment and conditioning. The terms related to radioactive wastes management released by the IAEA are shown in Figure 5.2-1. Processing is performed so that the radioactive waste is in a form that is suitable for the selected or likely disposal option. In waste management, the waste may be stored and needs to be in a form that is suitable for transport and storage.

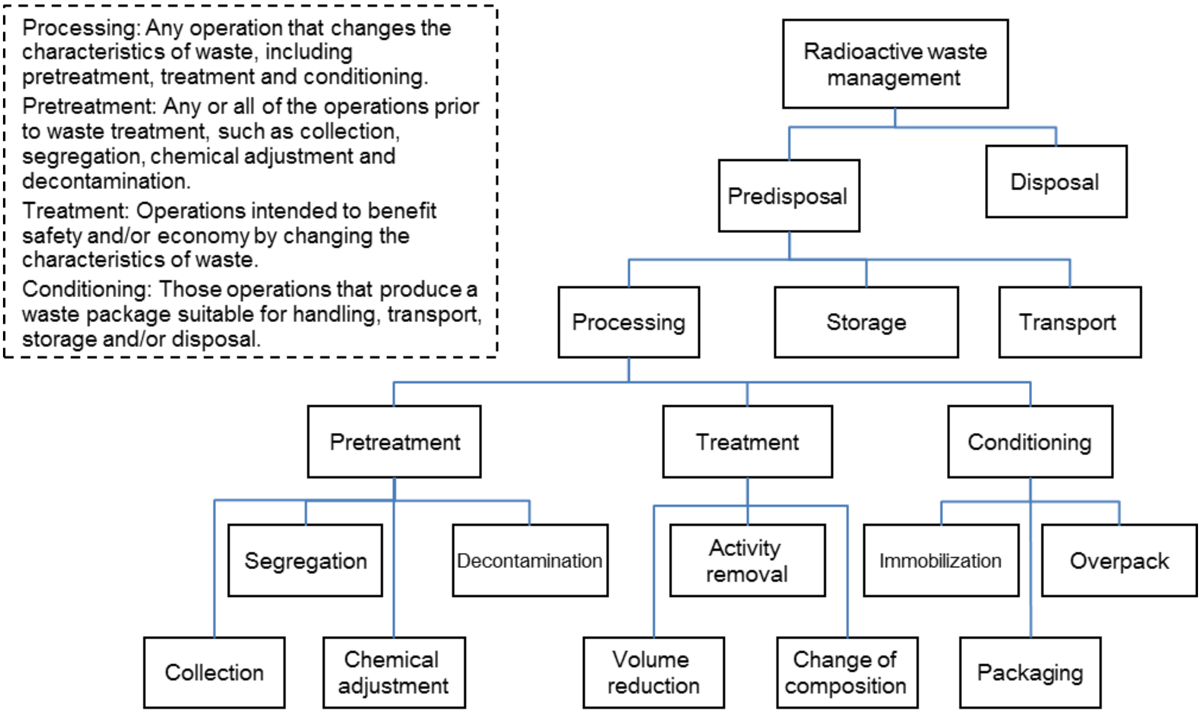


Figure 5.2-1 Terms regarding radioactive waste management (IAEA)[26]

The IAEA's draft safety guide DS448[27] provide specific measures to meet the safety requirements set out in GSR-Part 5 for the pretreatment management of radioactive waste from nuclear power plants and research reactors.

The following are the provisions based on the IAEA's safety requirements GSR-Part 5 and draft safety guide DS448 for appropriate radioactive waste management regarding the principles for ensuring the safety of radioactive waste disposal described in Section 5.2.1.

[25]IAEA, Predisposal Management of Radioactive Waste, IAEA Safety Standards, No. GSR Part5, IAEA, Vienna (2009)

[26] IAEA Safety Glossary, Terminology Used in Nuclear Safety and Radiation Protection 2007 Edition, p216

[27] IAEA, Predisposal Management of Radioactive Waste from Nuclear Power Plants and Research Reactors, Draft Safety Guide, DS448, IAEA, Vienna (2015). (The IAEA's draft safety guide DS448 is in the process of being published as an updated version of the current safety guide WS-G-2.5 "Predisposal Management of Low and Intermediate Level Radioactive Waste (2003)")

i) The radioactive waste needs to be characterized and separated into categories in all stages of radioactive waste management, from waste generation, processing (pretreatment, treatment and conditioning), storage and transport to disposal.

ii) The main purpose of processing radioactive waste is to produce waste in a form that meets the criteria for the safe processing, transport, storage and disposal of waste and thus to increase the safety of radioactive waste management and ensure the safe disposal of waste.

iii) The processing shall be based on appropriate consideration of the characteristics of the waste and of the demands imposed by the different stages in its management (pretreatment, treatment, conditioning, transport, storage, and disposal). The anticipated needs for any future stages in radioactive waste management have to be taken in to account as far as possible in making decisions on the processing of the waste. Various factors are to be considered not only effects on health of radiation, but also other factors such as environmental impact that may result from the content of non-radioactive materials, and social and economic impacts.

iv) Quantities, activity and physical and chemical nature of the radioactive waste to be treated, the technologies available, the storage capacity and the availability of a disposal facilities are taken into account in determining the level of waste processing.

v) If waste processing is performed before the waste disposal requirements are set, it must be remaining possible to process the waste in a way which meets those requirements once they have been set.

vi) Storage is an option that should be considered in the waste management strategy. Proper storage should be provided at all stages in waste processing prior to disposal to ensure waste isolation and environmental protection. Storage is used to facilitate the subsequent stages in radioactive waste management; to act as a buffer between and within waste management stages; to allow time for the decay of radionuclides prior to clearance[28] and other activities or is used to store waste in cases where the management has yet to be determined.

vii) Waste shall be stored in such a manner that it can be inspected, monitored, retrieved and preserved in a condition suitable for its subsequent management and under passive condition to the extent possible. Due account shall be taken of the expected period of storage, and, to the extent possible, passive safety features shall be applied. For long term storage in particular, measures shall be taken to prevent the degradation of the waste containment.

The ultimate goal of radioactive waste management is the safe disposal of radioactive waste. As described above, to reduce the risk of leakage and scattering of radioactive materials from the perspective of improving safety, the need for the processing of radioactive waste should studied in a way that allows for flexibility in making it consistent with the disposal measures. Following guidance described in the Draft Safety Guide DS448 may be applicable to the waste from events such as the accident at the Fukushima Daiichi NPS.

[28] In Japan, clearance is defined as the removal of materials used in a nuclear facility from regulatory control by the Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors (subject to the regulations on the laws concerning waste and recycling) if the Government determines that the concentration of the radioactive materials contained is below a level at which the effects on health of radiation are negligible.

- Liquid waste is often converted into a solid form by solidifying it in a suitable matrix in accordance with the waste acceptance criteria, and should be contained in a waste container as a solid waste form. Solidification may also be achieved without a matrix material, for example by drying.
- It should be noted that transporting slurry and sludge generated as secondary waste from water treatment can produce radiation hot spots due to pipe clogging. Radiolysis or chemical reactions in the slurry and sludge may occur, generating combustible gases, causing physical degradation or exothermic reaction. Special care should also be taken to long-term storage, including the function and performance of the collection and storage containers.
- Organic waste requires management stages that taken account not only of its radioactivity, but also of its chemical organic content, since this can also have detrimental effect on the environment.
- It should be taken into account that certain metals could react with water to produce hydrogen. The behavior of chelating agents or oil and salt content in liquid waste may also be of concern in the conditioning process.

Volume reduction should be studied from the perspective of limitation in storage capacity and economic rationality while ensuring consistency with the disposal measures. The draft safety guide DS448 states that the following points should be taken into account for volume reduction.

- Processes such as volume reduction to be employed should be selected on the basis of the characteristics of the waste concerned.
- Processes that achieve high volume reduction factors and that use proven techniques, such as compaction or incineration, should be employed if possible, taking account the concentration of the contained radioactive materials and the subsequent handling of the waste.
- Incineration of combustible solid waste normally achieves the highest reduction in volume as well as yielding a safe and stable waste form.
- Compaction is a suitable method for reducing volume of certain types of waste.

5.3 Assessments and issues on the action status based on the Roadmap

In order to reduce risks caused by radioactive solid wastes, it is important to study the processing and disposal management from the mid- and long-term perspective while carrying out an appropriate storage for the radioactive solid wastes involved in the progress of decommissioning. (Refer to 3.3.1)

As for the storage, approaches to the reduction of generated amount of radioactive solid wastes and ensure safety will be absolutely necessary. TEPCO is formulating the storage plan for the time being coordinating such as with NRA and local governments. R&D is being carried out focusing on the characterization, since processing and disposal management requires characterization of radioactive solid wastes, disposal management based on features of radioactive solid wastes and approaches related to the processing that consistent with disposal management is necessary. The issues which may have impact on the action status, their evaluations and future approaches to the waste management are described below.

5.3.1. Storage

(1) Reduction of waste generation

a. Action status

The structure of the waste management department of TEPCO has been improved, and waste management has been driven forward by the involvement of the department from the stage of developing construction project for decommissioning work. Measures have been taken and driven forward to reduce amount of radioactive solid wastes generation by control of bringing packing, materials and equipment into the site, reuse and recycling.

b. Assessment and issues on action status

Measures on minimizing amount of radioactive solid wastes generation such as control of bringing things into the site and on-site recycling have been taken by involvement of waste management department from the stage of developing construction project for decommissioning work, and achieving certain results.

It is important for TEPCO to study and take the measures continuously to reduce further amount of waste generation.

(2) Storage

a. Action status

Solid radioactive wastes are classified roughly into rubble waste and secondary waste generated from the water treatment. Rubble waste is separated into rubble, cut-down trees, used protective clothing and others, and stored accordingly. Rubble is classified according to surface dose rate and stored separately in the storage area and the form respectively from the perspective of shielding and preventing scattering. Cut-down trees are separated into two categories: ‘trunks and roots’ and ‘branches and leaves’, and stored accordingly from the perspective of fire risk and dose rate. The amount of stored rubble waste has been increased as a result of the removal of rubble waste through decommissioning activities. The amount of stored rubble and cut-down trees increased to about 185,000 m^3 and 84,000 m^3 respectively. The rubble accounts for 67% of the storage capacity and the cut-down trees for 79%. (As of April 30, 2016) [29]

For the volume reduction of rubble waste, a solid waste incinerator was installed. Incineration of combustible rubble waste, such as used protective clothing started from March 2016[30]. On the other hand, secondary waste from water treatment is separated into adsorption columns, waste sludge and concentrated liquid waste, and stored accordingly. The adsorption columns are stored in a form (rack, box culvert) suitable for the type of column. With the progress in the treatment of contaminated water, the number of adsorption columns stored as secondary waste from water treatment reached 3,165 and accounts for 51% of the storage capacity. (As of May 19, 2016) [30]

Management status of rubble and secondary waste generated from the water treatment is shown in A.5-1-1 of Appendix 5.1.

Slurry from the pretreatment system for the multi-radionuclide removal system, out of waste adsorption columns is stored in high-integrity containers (HICs) in a box culvert. Stagnant water was found in the

[29] TEPCO Document 3-4: Processing and Disposal of Radioactive Waste “Status of the Management of Rubble and Cut-down Trees (as of April 30, 2016), Meeting of the Secretariat of the Team for Countermeasures for Decommissioning and Contaminated Water Treatment (30th Meeting), May 26, 2016.

[30] TEPCO, Document 3-4 “Starting the Installation of Incinerators for Solid Waste in the Fukushima Daiichi Nuclear Power Plant on a Full Scale,” page 6, Meeting of the Secretariat of the Team for Countermeasures for Decommissioning and Contaminated Water Treatment (28th Meeting), March 31, 2016.

periphery of upper lid of HIC. As a countermeasure, the surface water in the HICs was removed using simplified water removal system while lowering water level of the slurry in the HIC. Installing a water removal system, water removal of the surface water in the HICs will be carried out on a full scale to accelerate the process.

At the meeting of the Secretariat of the Team for Countermeasures for Decommissioning and Contaminated Water Treatment, TEPCO presented a schematic of rubble storage for up to 2027 (Refer to Figure A5.1-2 of Appendix 5.1).

In the development of a storage management plan, TEPCO projected the amount of solid waste from the construction and other activities described in the - and long-term roadmap. The prediction showed that solid waste generated in the next 10 years or so would increase and exceed the current storage capacity for solid waste. TEPCO plans to build a storage facility with the function of shielding and preventing scattering and store the waste properly with continuous monitoring.

In the facility to be built was determined based on the projected processing amount that correspond to the solid waste to be generated, which will be reduced to the extent possible. TEPCO plans to implement this storage management plan to reduce the volume of solid waste stored temporarily outside the solid waste storage facilities and the solid waste to be generated to the extent possible; to store the waste from the rubble with high radiation dose indoors, and eliminate the temporary storage area outside the solid waste storage facilities, and thus to further reduce the risks associated with waste management.

A 9th solid waste storage facility with a capacity of about 30,000m^3 is being built. Additional ten (FY2020-) to 13 storage facilities will be built in phases to increase the total storage capacity by about 140,000 m^3. With the incineration/compaction system TEPCO started to burn used protective clothing in two incinerators, each with a capacity of 7.2 tons per day. In 2020, TEPCO plans to start to burn combustible materials comprising cut-down trees and rubble in one additional solid waste incinerator with a capacity of 95 tons per day. A system to reduce the volume of metal and concrete in the rubble will be constructed (to be completed in FY2020) to increase the volume reduction capacity. This system can reduce the waste temporarily stored outside the solid waste storage facilities, which is projected to increase to about 750,000 m^3 by around FY 2027 under the current storage conditions, to about 200,000 m^3 (mainly waste with a dose rate of less than 0.005 mSv/h).

The secondary waste from water treatment will be stored in buildings, and the temporary storage area will be eliminated to reduce risks. Measures of volume reduction will be developed when the waste is stored inside.

b. Assessment and issues on action status

It is important to implement storage of the generated radioactive solid wastes appropriately until management of the processing and disposal are specified.

Temporary measures to prevent fire are in place for the cut-down trees (trunks and roots) that are piled up outside. Additional solid waste incinerators will be installed as a permanent measure. It is important to implement the measures in accordance with the plan.

Considering the integrity of storage container, the need for countermeasures for further risk reduction has to be studied continuously for the secondary waste generated from the water treatment system.

5.3.2. Processing and disposal

(1) Waste characterization

a. Action status

On the solid waste, radiological analysis on rubble, cut-down trees, soil and others, characterization on secondary waste from water treatment and development of radiological analysis technique for difficult-to-measure nuclides have been implemented[31,32]. Table 5.3-1 shows the status of the radiological analysis of solid waste. Through the radiological analyses conducted so far, correlation between activity concentrations of nuclide such as Sr-90 and Cs-137 was observed in the specimen inside the R/B, and contamination state was found different depending on the locations in Units and buildings.

The number of current analysis specimens has reached about 70[33] through cooperation with institutions joined.

In addition, a method of sampling for the wastes under a high radiation environment (e.g. rubble of high radiation area inside the building and secondary waste generated from the water treatment) was studied and some of the wastes were actually sampled.

The radiological analysis plan has been developed focusing on the period until around FY 2021, when the prospects of a processing and disposal method and a technology related its safety be made clear.

While compiling the results of the analyses for water and rubble conducted so far, data is being updated by adding the data on the rubble inside the building and on-site soil.

Also, waste information catalog are being prepared based on the evaluation of characterization.

Nuclides contaminate waste are derived from such as fuel and activation, and their migration pathways are through circulating water and atmosphere. Based on that, the results of the inventory evaluation have been updated while studying transport ratio and the evaluation methods for inventories.

Uncertainties in the rate of migration of nuclide to stagnant water were reduced by introducing observed value of radionuclide analysis to the studies on the analytical model for inventory evaluation.

The results of estimation for waste inventories have been reflected to the studies on the safety assessment of the disposal.

[31] IRID supplementary budget for 2014 “Management Water Contaminated and Decommissioning of Project(R&D for treatment and disposal of solid waste) Interim report April 2016 http://irid.or.jp/_pdf/201509to10_12.pdf?v=4

[32] IRID/JAEA, P.2 of Attachment 3-4 ”Rubble sampled within the Fukushima Daiichi NPS,” Decommissioning and contaminated water management team meeting/ secretariat meeting (the 29th) April 28, 2016

[33] Nuclear Regulation Authority, 2nd Meeting of the Study Committee on the Regulation of Radioactive Waste from Specified Nuclear Facilities, Document 4, TEPCO “Capability to Analyze Waste from the Fukushima Daiichi Accident,” February 12, 2016.

Table 5.3-1 Status of radiological analysis of radioactive solid waste[34]

FY	Specimen		Number of specimens
2011-2014	Water treatment system outlet/inlet water	• Stagnant water in Unit 1 to 4 T/B • High level contaminated water on the basement floor of the central radioactive waste processing facility • Desalination system concentrated water • Stagnant water on the basement floor of the High Temperature Incinerator Building • Treated water (cesium absorption system, 2nd cesium absorption system)	25
	Rubble in the building; Boring core	• Rubble from 1st floor of Units 1 and 3 R/B • Boring core from 5th floor (floor surface) of Unit 2 R/B • Boring core from 1st floor (floor surface and walls) of Unit 1 R/B • Boring core from 1st floor (floor surface) of Unit 2 R/B	13
	Rubble and cut-down trees	• Rubble around Units 1, 3 and 4 • Cut-down trees (branches, leaves), live trees around Unit 3 (branches)	24
	Live trees, fallen leaves, soil	• Live trees (branches, leaves) at various locations on the plant site, fallen leaves, soil	121
2015	Water treatment system outlet/inlet water	• High level contaminated water on the basement floor of the central radioactive waste processing facility, stagnant water on the basement floor of the High Temperature Incinerator Building • Treated water (cesium absorption system, 2nd cesium absorption system, decontamination system, multi radionuclide removal system)	26
	Slurry	• Multi radionuclide removal system slurry and/or (existing and additional system)	4
	Rubble	• Rubble in Unit 1, 2 and 3 R/B • Rubble collected from the temporary soil covered storage facilities • Sand in Unit 1 T/B	33

Source: IRID/JAEA, P.2 of Attachment 3-4"Rubble sampled within the Fukushima Daiichi NPS," Decommissioning And contaminated water management team meeting/ secretariat meeting (the 29th) April 28, 2016.

b. Assessment and issues on action status

Characterization of solid waste is extremely important in developing processing and disposal measures. Since FY 2015, the number of samples that can be radiologically analyzed per year has increased from about 50 samples to 70. Radiological analysis was performed on samples that were collected from only a limited number of locations due to the high dose rates, studies on the method for collecting highly radioactive samples have started. Knowledge on the properties of waste has been accumulating from the result of radiological analysis conducted so far, but analytical capability (facility, technology and human resource) is still not sufficient to conduct studies on the process and disposal management for radioactive solid wastes and collection of the data on the radiological analysis of waste required for developing regulatory system. For this reason, it is essential to establish radioactive-material analysis and research facilities as planned.

(2) Study on processing and disposal management

a. Action status

Measures for the processing and disposal of solid waste have been developed taking into account its characteristics, with a focus on the applicability of the existing processing techniques and disposal concepts.[31]

[34] IRID/JAEA, an edited version of Document 3-4 "Analysis of Rubble collected on the Site of the Fukushima Daiichi Nuclear Power Plant," page 2, Secretariat of the Team for Countermeasures for Decommissioning and Contaminated Water Treatment (29th Meeting), April 28, 2016.

On the processing technology, compilation of existing technology including conditioning focusing on the secondary waste generated from the water treatment with few experiences for processing and basic test on conditioning technology have been done, and evaluation of conditioning techniques including kneading processes such as cementation and heating processes such as the melting and solidification process have been studied. Through the survey and the test, the data needed to evaluate processing technologies, such as the mechanical strength and chemical properties and waste contents of candidate cemented and melt-solidified waste forms, has been obtained and developed.

In developing disposal measures, the existing disposal concepts and safety assessment methods have been reviewed and discussed. Current disposal concept, site model and safety assessment method (e.g. scenario and nuclide migration model) applied for radioactive solid wastes have been examined and the disposal classification and important nuclide based on the classification have been provisionally studied.

In addition, conducting the study of sensitivity of the disposal system for nuclide migration parameter, the impact on the nuclide migration parameter has been collated in consideration of the differences in the properties of waste form depending on the process technologies as a study on the properties of radioactive solid wastes affecting disposal process study.

Furthermore, including overseas disposal facilities in the scope, the issues in establishing new disposal concept have been studied from the perspective of reasonable disposal based on the features of radioactive solid wastes. Appendix 5.2 shows the sample of disposal facilities in Japan and abroad.

A series of these waste stream handling processes has been studied to ensure the safety and rationality of the entire radioactive solid waste management process, from waste generation, storage and processing to disposal, and to take an overall view of the entire process from a broader perspective and conduct R&D

b. Assessment and issues on action status

The applicability of processing techniques to radioactive solid waste has been evaluated. Solidification techniques and information and test data necessary for characterizing waste form have been steadily accumulated. From the perspective of safe and rational processing, it is desirable to perform sufficient survey for technical information including the knowledge from overseas, obtain test data and collate the requirements on the evaluation.

Important nuclides in the classification of waste into disposal categories have been studied as part of a study on the applicability of the existing disposal methods. It is important to conduct studies toward the establishment of new disposal concepts from the perspective of establishment of safe and rational disposal method considering the overseas disposal method and technologies.

Based on progress of studies on the characterization, processing and disposal, it is important to conduct necessary studies on the concept and method to narrow down the wastes stream as well as coordinating items and priority of waste stream to be detailed. It is also important to put technical information together for establishing basic policy of waste processing and disposal of radioactive solid waste by conducting above-mentioned activities.

5.4 Mid- to long-term action policies for waste management of the Fukushima Daiichi NPS

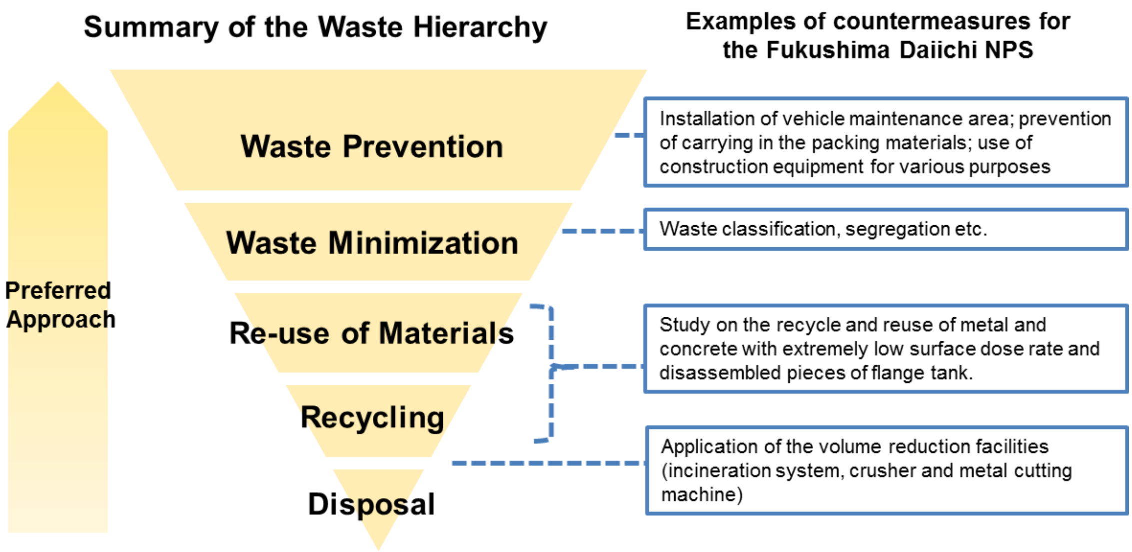
Taking into account the international principles in Section 5.2 for ensuring the safety of radioactive waste management and the issues identified in evaluating the current status of the activities in Section 5.3, the following sections describe the policy for addressing issues that may significantly affect waste management in the future, such as those require a particular focus or those being addressed but still require particular attention, due to the necessity of implementing the entire waste management process as planned in the mid- and long-term radioactive solid waste management strategy for the Fukushima Daiichi Nuclear Power Plant.

5.4.1. Storage

(1) Reduction of waste generation

a. Waste hierarchy

In the UK, the priority given to the measures be implemented (in order of 1. waste prevention, 2. waste minimization, 3. reuse, 4. recycling, and 5. disposal) is shared to minimize waste generation and disposal to reduce the environment load, and the amount of waste disposal has been successfully reduced by implementing waste management in line with this policy. In the UK and the US, it is indicated that it is important for the waste management department to be involved in the process of developing a plan for decommissioning work in order to actually deploy waste hierarchy. As shown in Figure 5.4-1, approaches for waste hierarchy have already taken. It is important to enhance the awareness for the minimization of the radioactive waste by disseminating the concept throughout the Fukushima Daiichi in future.



(a) Concept of the waste hierarchy in UK NDA[35] (b) Measures in the Fukushima Daiichi Nuclear Power Plant

Figure 5.4.1 Summary of the waste hierarchy

[35] Nuclear Decommissioning Authority (NDA), an edited version of the figure on page 34 of "Strategy Effective from April 2011."

b. Considerations for secondary waste

To reduce the volume of waste to be stored, a volume reduction system has been installed in accordance with the plan. If a radioactive solid waste incinerator is installed, exhaust system filters and other consumable parts are generated from equipment as secondary waste during a certain period of operation. Also after the end of its operating life, incinerator itself will become wastes. Therefore, if a volume reduction system is installed, it is necessary to consider its effectiveness in reducing the volume of all waste generated, including secondary waste, and the processing of the secondary waste.

The need for decontamination in preparing for a reduction in radiation exposure and the retrieval of debris is expected to grow in the future. If a wet decontamination process is used for concrete surfaces, contamination from water penetration may occur and result in an increase in the waste generated. Due to the use of a decontamination agent, organic materials and hazardous materials that affect the barrier performance of the disposal facility may enter the waste.

While the priority is to achieve the target decontamination factor, it is important to discuss the decontamination techniques to be applied and the secondary waste that may be generated from decontamination with the waste management department and to select appropriate decontamination techniques in consideration of their effect on the disposal of radioactive solid waste.

(2) Storage

a. Storage planning

The amount of radioactive solid waste from the construction and other activities described in the mid- and long-term roadmap was projected, and a storage management plan was developed based on the projection. The plan is to significantly reduce the increase in radioactive solid waste by installing a volume reduction system as the decommissioning process proceeds, and to eliminate the temporary storage area by transporting radioactive solid waste to store in the storage facilities, thereby it is appropriate for reducing the risk of leaking and scattering of the radioactive solid waste. Although it is important to implement the storage plan and reduce the risk attributable to the radioactive solid wastes, the plan should be revised flexibly according to the changes such as the progress status of the project.

b. Stabilization of stored waste

The prospect of R&D for a dehydration method to stabilize the slurry from the pretreatment system for the multi-radionuclide removal system is in sight at the basic stage. To achieve more stable storage as soon as possible, it is necessary to move from the R&D stage to the practical application stage. Storage management of concentrated liquid waste, flammable gas and sludge with exothermic reaction should be implemented with due consideration given to the physical degradation from the perspective of risk reduction of secondary waste generated from the water treatment, and the study of a more stable form of storage in the future should be accelerated.

c. Solid waste generated from the retrieval of fuel debris

Once the retrieval of fuel debris starts, radioactive solid waste such as removed objects surrounding fuel debris, materials and equipment is expected to be generated in large quantities. The radioactive solid waste includes heavy objects with a high dose of radiation and radioactivity concentration. To retrieve fuel debris safely, smoothly and efficiently, it is important to study the location and method of storing the fuel debris and the removed objects surrounding it, materials and equipment, as well as the method of putting them in storage containers including the storage containers themselves and the cutting process and transporting them to the storage area prior to the removal. It is important to store and manage the waste generated in a safe and stable way.

5.4.2. Processing and disposal

(1) Waste characterization

a. Radiological analysis plan

To develop specific measures for the processing and disposal of radioactive solid waste, it is extremely important to characterize the waste by performing radiological analysis based on the characterization analysis plan.

Radiological analysis has been carried out to characterize the waste and a lot of data has been obtained. It is important to improve the efficiency of radiological analysis by selecting nuclides to be measured based on an evaluation of the radiological analysis and analytic results, and by developing evaluation and analysis methods that can be applied to a limited number of radiological analysis samples.

As the decommissioning process proceeds, the number of samples collected from locations where sample collection was difficult and those of high radiation dose has increased, resulting in an increase in the number of radiological analysis samples and the possibility of obtaining new knowledge. To obtain data under these circumstances efficiently, top priority should be given to obtaining data that can contribute to promoting the decommissioning process and developing measures for the processing and disposal of radioactive solid waste.

In developing a plan to collect analysis samples under the radiological analysis plan, it is important to consider the capacity of the existing radiological analysis facility and the new radioactive-material analysis and research facilities to be built, to examine the appropriateness of the sampling locations, the sampling method, the number of samples, the time of sampling, and other sampling conditions, and reflect the results of the examination back into the plan, where appropriate.

A study of the method of collecting samples in areas that are difficult to access due to high dose of radiation has started, and important to be continued.

It is important to characterize wastes generated as the decommissioning process proceeds, such as the incinerator ash from the radioactive solid waste incinerator which went into operation in 2015, radioactive solid waste generated during the fuel debris retrieval, secondary waste generated from decontamination, in preparation for processing and disposal in the future. It is important to comprehensively evaluate the mid- and long-term trends of the variations in the measurements of the radioactivity and dose rate of the waste which are implemented for waste storage management, with considering complement of the characterization under the radiological analysis plan.

In developing the analysis plan, it is important to consider improving the efficiency of analysis and quality assurance based on the progress in waste characterization. It is important to update the plan in a flexible way as the decommissioning process proceeds and as the capacity of the radioactive-material analysis and research facilities changes.

In addition, it is important to standardize the radiological analysis methodology improved and developed, with disclosing them proactively.

b. Radiological analytical capability for waste characterization

The radiological analysis for characterization of waste will provide important information not only as a foundation of a radioactive solid wastes disposal measure but also to develop plan for dismantling the facility, measures for reducing the exposure of workers, and plans for radioactive solid wastes disposal and storage. Radiological analysis for characterization requires advanced capabilities such as enhancing types of samples can be analyzed, increasing number of samples, improving accuracy of radiological analysis.

To meet this requirement, it will be required to use the existing radiological analysis laboratory more effectively, to build a new radioactive-material analysis and research facility and establish and enhance the management system of the facilities. In establishing and enhancing the management system of them, it is important to develop a streamlined and continuous management system and to perform radiological analysis under the system. It is important to develop radiological analysis methodology by improving analyzing techniques (including pretreatment techniques), developing it for difficult-to-measure nuclides, developing it suitable for highly radioactive samples, and the standardizing of it; and to appropriately reflect the results of the development back to the new radioactive-material analysis and research facilities to be built and thus to improve capability of the radiological analysis. In addition, it is important to appropriately reflect the status of improvement and development of the radiological analysis methodology and of study on the radioactivity verification method for the waste form to the specifications and operation method of radioactive-material analysis and research facilities.

Currently, there is a lack of personnel who may develop analysis methodology and perform the radiological analysis. It is important to appoint and develop personnel for these tasks in the near term, as well as to develop personnel on a continuous basis, including those who can evaluate the entire analysis process from a broader perspective.

(2) Study on processing and disposal management

a. Processing and disposal management in accordance with the characteristics of the waste at the Fukushima Daiichi NPS

In the current Roadmap, the basic concept of processing and disposal for radioactive solid wastes is to be compiled in FY 2017 and the prospects of a processing/disposal method and a technology related to its safety is to be made clear by around FY 2021 as a target.

As for the measures for processing and disposing of radioactive solid waste, it is necessary to determine the characteristics of the waste, such as its attributes, chemical properties and concentration of radioactive materials, and to develop measures for disposal that take these characteristics into account, and then to develop an appropriate processing technique based on the measures. For processing technique, it is important to improve the necessary data to evaluate the conditioning technique. Also, utilizing the technology with high reliability and flexibility, it is important to build information and test data on the safety of the candidate processing technique with the aim of applying a safe and rational processing technique and to evaluate the applicability of the processing technique and characterize the waste with the aim of treating several types of waste with a minimum number of processing processes. In developing and designing a treatment process, it is desirable to share tasks with R&D institutions in the appropriate way, to examine the reliability of the processing technique by performing basic tests and engineering-scale tests.

On the disposal management, it is important to study on new disposal concept considering the particularities such as large quantity of radioactive solid wastes generated by the decommissioning and contamination by the radionuclide attributable to the failed fuel, and bring the R&D issues to realize the new disposal concept into shape and deal with it while utilizing the experience and findings of disposal in Japan and abroad.

For developing feasible waste stream and presenting its basis, it is important to refine waste stream and to study on narrowing down the processing technique including its consolidation while reflecting the progress made in the studies. Also it is important to continue linkage for the studies related to the characterization of waste, and processing and disposal.

It is extremely important for the organizations involved to strengthen collaboration, conduct a comprehensive study and give shape to the processing and disposal measures.

Based on the results of these activities, the NDF will play the central role in putting measures for processing and disposal of waste into practice.

In addition, it will be necessary to actively provide the regulatory authorities with information related to progress of studies so that the regulatory system on processing and disposal would be established in a smooth manner.

b. Waste segregation and history information management

Since the amount of radioactive solid wastes is increasing due to decontamination of facilities, fuel debris retrieval, and progress in dismantling the facilities, it is important to implement the appropriate waste segregation for implementing safe storage and disposal in a smooth manner. Therefore, it is important to retain and manage information on the attributes of radioactive solid waste, such as the history of waste generation, the history of contamination, and the concentration of radioactive materials in the waste, and to separate the waste into categories and manage it accordingly.

The information on the origin of the waste will be important when processing and disposal are actually carried out; therefore it is useful to develop a database containing information on the properties of the waste, technical information on treatment and conditioning, and management information on waste disposal.

c. Regulatory system

The current mid- and long-term roadmap states that the prospects of a processing/disposal method of radioactive solid waste and technology related to its safety is to be made clear by around FY 2021 as a target and the preparation for the necessary regulatory system will be studied.

To develop a regulatory system for radioactive solid waste in a smooth way, it is important to share an understanding of the measures with the regulator body. Therefore, it is important to provide the regulatory body with the required information on the radioactive solid waste characterization and the development of processing and disposal measures.

In addition, it is important to share with the regulatory body the principles of radioactive waste management described in this strategic plan, because they are related to the principles applied to the development of a regulatory system and standards.

In the Study meeting for the regulations on the radioactive waste of Specified Nuclear Facility at Nuclear Regulation Authority, issues related to the stable management of the radioactive solid wastes at the Fukushima Daiichi NPS has been studied since FY 2015 bearing in mind the decommissioning work over the long period of time.

5.5 Future actions of waste management for the decommissioning of the Fukushima Daiichi NPS

Toward the decommissioning of the Fukushima Daiichi NPS, the necessary measures for risk reduction and optimization of overall facility have to be carried out promptly and effectively. Safe and steady storage of radioactive solid wastes caused by the accident is required for waste management, and it is important to study the processing method and disposal concept from the mid- to long-term perspective.

In the basic concept of processing and disposal for radioactive solid wastes, to be compiled in FY 2017, it is important to describe the specific direction of the resolution of the issues attributable to the features of the wastes at the Fukushima Daiichi NPS, and its outline is aimed to be indicated in the Strategic Plan 2017.

Figure 5.5-1 shows the future actions to be taken under the Strategic Plan for waste management including R&D activities that are in line with the principles for ensuring safety in Section 5.2 and the policy for taking action from the mid- and long-term perspective in Section 5.4.

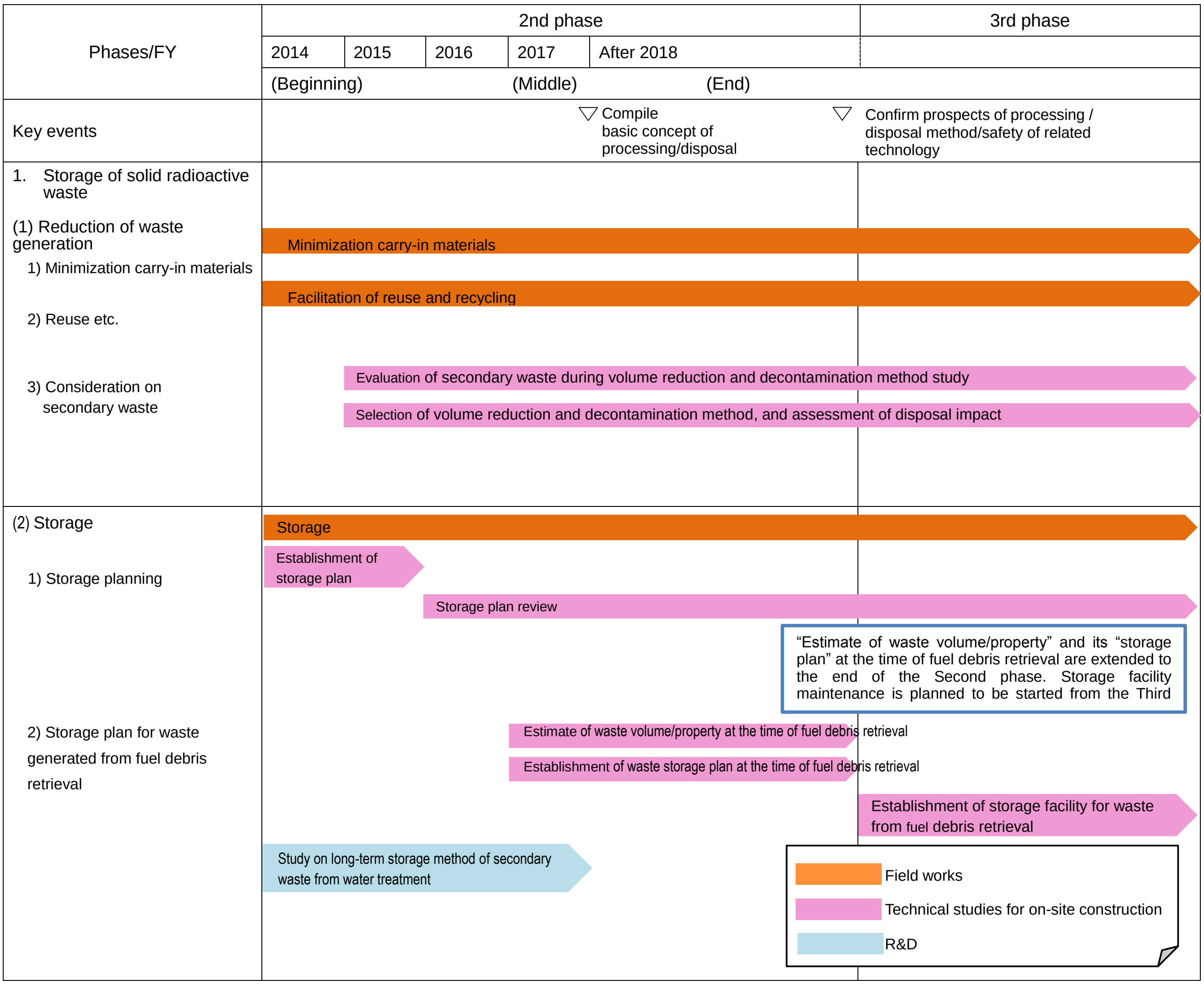


Figure5.5-1 Future action in the strategic plan for waste management (1/2)

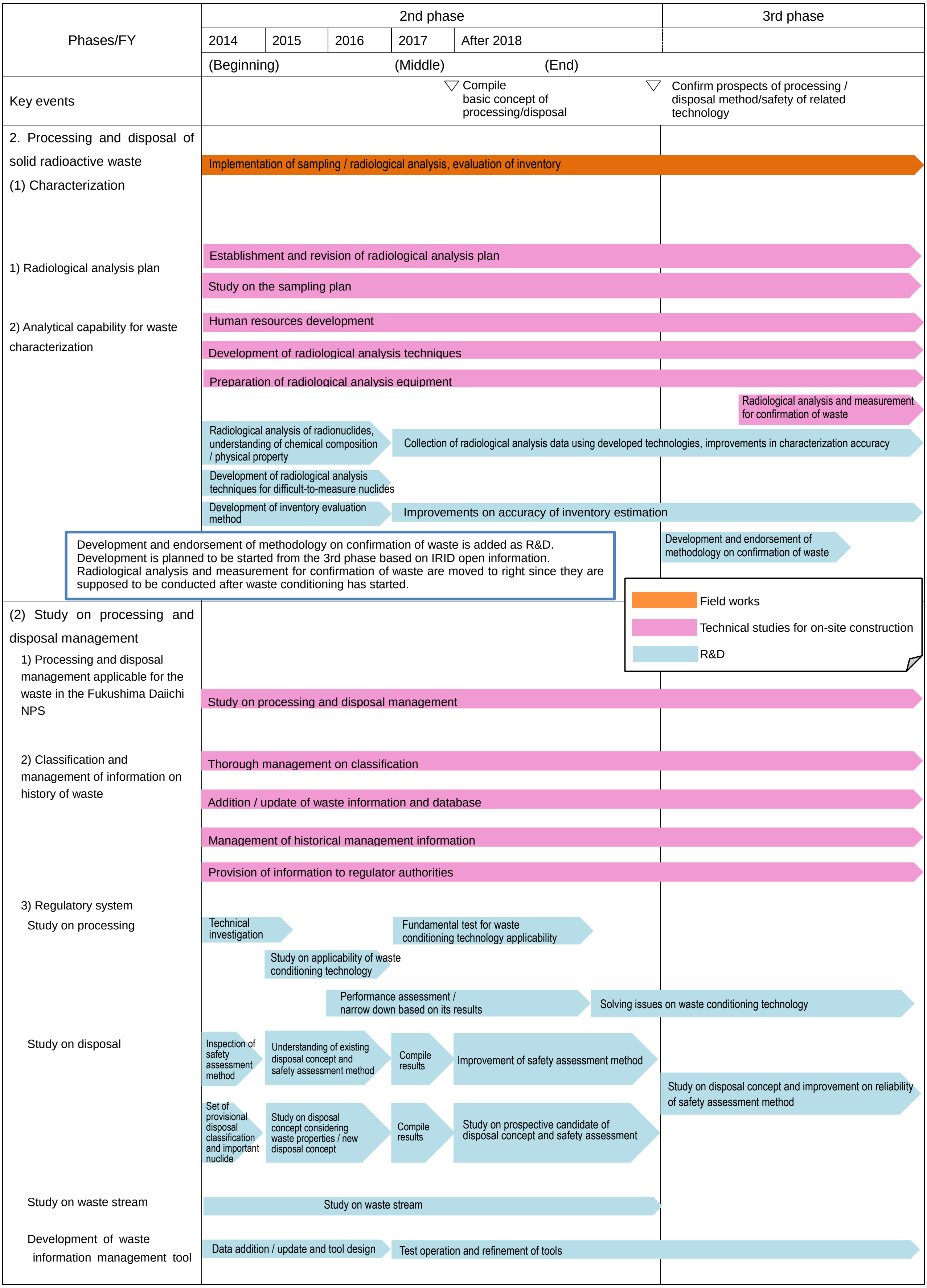


Figure5.5-1 Future action in the strategic plan for waste management (2/2)

6. R&D Initiatives

6.1 Basic Policy for R&D and its overview

6.1.1 Basic policy

The decommissioning of the Fukushima Daiichi NPS involves many highly technical challenges. Through the governmental subsidies for research projects and facilities, multiple R&D projects aimed at practical application of technologies are being led by IRID (established August 2013), and JAEA facilitated R&D sites and equipment. Basic/generic and applied researches are conducted by research institutes, such as JAEA, and universities as well.

Furthermore, the NDF was reorganized in August 2014 and a statutory mission for research and development for technologies required for decommissioning was added. To fulfill this mission, the NDF has determined a policy for execution of the R&D for the technologies required for decommissioning (hereinafter, "R&D duties execution policy") to clarify approaches to R&D project planning, coordination and management. Based on the above, the Strategic Plan 2015 presents a comprehensive R&D plan that reflects R&D agendas derived from in the areas of fuel debris removal and waste management to strengthen the management of R&D and boosting its effectiveness. The NDF also has been struggling with strengthening partnerships between universities and research institutes that are conducting human resources development and basic/generic researches, IRID (which deals with practical development), and TEPCO (which is engaged in decommissioning work at the site) through the Decommissioning R&D Partnership Council[36].

In the decommissioning of the Fukushima Daiichi NPS, there are some successes appeared in measures to deal with contaminated water and the focal point is now gradually moving towards mid-and-long-term efforts such as fuel debris retrieval. In order to deal with the challenges in uncharted territory, it is required to further promote effectiveness of R&D activities, collaboration between relevant institutions, cooperation with organizations overseas, utilization of R&D facilities, and human resources development.

6.1.2 Overview

To deal with the decommissioning of the Fukushima Daiichi NPS that involves many technically challenges, R&D activities on a variety of fronts is being conducted by a number of organizations. Figure 6-1 shows stages of R&D activities and the leading organizations.

The roles of key organizations and details of their R&D are described as follows in Figures 6-1 to 6-5. The NDF comprehensively reviews such R&D activities in order to promote effective and efficient R&D approaches, and seeks the overall optimization of these activities through promoting further clarification and adjustment of roles allocation based on their special characteristics and the expected results of the R&D they are engaged and also through close cooperation with related organizations.

[36] The Team for Countermeasures for Decommissioning and Contaminated Water Treatment decided to set up at its meeting held on May 21, 2015.

(1) Government

To clarify and present governmental policy concerning the decommissioning of the Fukushima Daiichi NPS through the Mid-and-Long-Term Roadmap, and as part of this effort, presents R&D plans for highly-difficult technical challenges and provides funding to organizations engaged in R&D activities.

- Ministry of Economy, Trade and Industry (METI): Project of Decommissioning and Contaminated Water Management
- Ministry of Education, Culture, Sports, Science and Technology (MEXT) : Acceleration Plan of Reactor Decommissioning R&D for the Fukushima Daiichi NPS, TEPCO

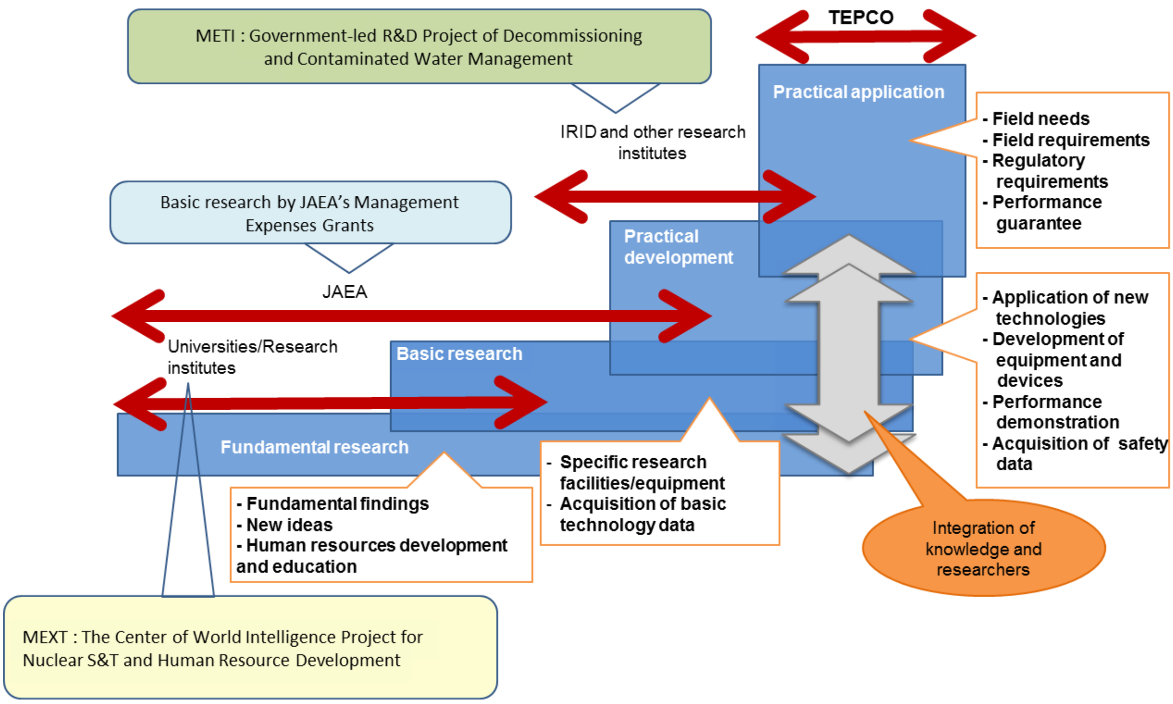


Figure 6-1 Overview of R&D activities related to the Fukushima Daiichi NPS

(2) NDF

To offer guidance and advice regarding the decommissioning initiatives undertaken by TEPCO and R&D plans of research institutes, through the presentation and implementation of the Strategic Plan, which embodies the mid-and-long-term strategies related to the decommissioning of the Fukushima Daiichi NPS. Also to engage in overall coordination and management through operating the Decommissioning R&D Partnership Council to strengthen collaboration among the diverse and varied R&D efforts being undertaken towards the decommissioning of the Fukushima Daiichi NPS.

(3) TEPCO

To engage in on-site work involving the decommissioning of the Fukushima Daiichi NPS, conducts engineering and R&D required for those activities, and seeks such as regulatory approval.

(4) IRID

To build a cooperative and collaborative framework with research institutes and companies in Japan and abroad through each individual project, and carries out R&D projects required for the practical application of technologies and systems such as for the “Project of Decommissioning and Contaminated Water Management.”

(5) JAEA

To promote basic/generic and applied researches, and promotes the establishment and management of R&D sites. Also, through the “Collaborative Laboratories for Advanced Decommissioning Science (CLADS),” draw knowledge together in Japan and abroad and strengthen the efforts to establish the research center for basic/generic and applied research.

(6) Private sector/research Institutes

In addition to R&D carried out by IRID, there are many companies involved in the practical development of technologies focusing on the decommissioning of the Fukushima Daiichi NPS (e.g., development of robots by various manufacturers). Furthermore, in addition to IRID and JAEA, other R&D facilities, such as Central Research Institute of Electric Power Industry (CRIEPI), conduct basic/generic and applied research focusing on contributing to the decommissioning of the Fukushima Daiichi NPS.

(7) Educational institutions (e.g. universities and technical colleges.)

Educational institutions, such as universities and technical colleges, are engaged in basic and applied research with a focus on contributing to the decommissioning of the Fukushima Daiichi NPS, and the development of human resources through provisions of special courses or learning opportunities.

(8) Academic societies

To offer advices and to make proposals on various initiatives that considers a variety of views from an independent standpoint concerning the decommissioning of the Fukushima Daiichi NPS. Even from academic societies that are not related to nuclear energy, initiatives for the decommissioning of the Fukushima Daiichi NPS are implemented.

6.2 Management of R&D for application to decommissioning

In order to manage the continuous improvement of the effectiveness of R&D initiatives, it is important to take a broad overview of R&D process from basic and generic studies through to applied and practical research under the Decommissioning R&D Partnership Council, such as based on the R&D Execution Policy as shown in Table 6-6.

Considering that the focus in the effort to decommission the Fukushima Daiichi NPS is shifting towards mid-and-long-term efforts such as the removal of fuel debris, TEPCO, IRID, and NDF should develop closer cooperative relationships, and review and strengthen R&D efforts being promoted such as by IRID, that are in line with the current status of decommissioning work and have direct application at the site in mind to enhance effectiveness. It is important for related organizations including research institutes and universities to make a shift towards developing closer ties based on decommissioning work requirements as well.

6.2.1. Application management for decommissioning

In regard to R&D needs that are presented in the strategic plans for fuel debris removal and waste management, it is important to act swiftly and accurately toward the objectives to be achieved based on the importance and difficulty of each challenge with considering the division of roles and system of implementation among related organizations[37].

It is also important to set appropriate objectives that are directly related to an achievement of the objectives of each R&D effort in order to ensure a steady advanced. On this occasion, an R&D management mechanism that takes an overall view of these efforts including on-site construction work and technical considerations is needed. In case that potential R&D seeds that meet R&D requirements exist and taking an application of them in mind, it is important to achieve full mutual coordination between R&D needs and seeds, and to have such aspects as goals and development periods shared to further promote R&D activities as well.

In regards to the specific management of R&D, including those which undertaken as parts of the Project of Decommissioning and Contaminated Water Management, in addition to R&D planning in an appropriate manner, it is necessary to periodically check the tasks and objectives to be achieved and make adjustments if required even after the commencement of R&D activities. It is particularly important to review and check the followings in the course of practical development efforts:

(1) R&D Project Planning Stage

a. To identify tasks from targets indicated in fuel debris retrieval and waste management objectives, and to establish implementation plans and milestones for the approaches to resolve issues, taking associated risks into account. Based on this plan, to set individual R&D targets and division of roles, and have these reviewed by related institutions and those be reflected.

b. To confirm if needs arising from on-site construction and related technical studies are fully reflected when setting a division of roles of each R&D project.

   i) Scope of R&D (e.g. equipment/devices, systems, assessment method, and data acquisition.)

   ii) Technology Readiness Level (TRL) and developmental stages based on it, criteria for determining stage transition.

   iii) Establishment of a common base of equipment and devices, optimum technical experimental method, methodology for confirmation and endorsement of safety evaluation technologies, etc by third parties.

[37] Related organizations are expected to undertake the following efforts:
METI: To continue to improve effectiveness of R&D, and to ensure the continued availability of resources.
MEXT: To develop work that addresses the needs derived from decommissioning work, such as the Acceleration Program of Reactor Decommissioning R&D.
JAEA: To conduct research that addresses the needs derived from decommissioning work and to ensure resources required for ongoing research.
Universities/Research Institutes: To actively participate in research related to decommissioning of the Fukushima Daiichi NPS.

c. To study a feasibility of each R&D project and to identify risks and alternative solutions if needed.
d. To create interface control documents (documents on information transmitted) across R&D projects.

(2) Implementation stage after startup of R&D projects

a. To periodically monitor achievement and challenges of each R&D project, site conditions, needs at every milestone, and to conduct expert review on transition to the next stage.
b. In case of critical issues are identified above, to determine if each R&D objective is required to be changed based on “Common objectives” (the relationship between R&D projects needs to be considered as well). To reconsider based on “Common objectives” if there could be an impact on the objectives.

(3) Stage where R&D results started to appear

a. To confirm and review the validity of research results considering the latest situation of the decommissioning site, and identify/specify problems.
b. Through evaluation by relevant institutions and the activities such as of the Decommissioning R&D Partnership Council to identify R&D needs that need to be given priority and reflect these in the next strategic plan.
c. Based on the above, to present the direction for each R&D project and reflect this in the research.

6.2.2 Comprehensive approach for R&D

In addition to practical development of which the necessity is clear, it is required to study technical development that is considered to be required in future. For example, fuel debris retrieval should be promoted as follows.

For "Determination of fuel debris retrieval policies for each unit " (summer of FY2017) and " Determination of fuel debris retrieval methods for the first implementing unit" (FY2018), it is important to continuously monitor progress and achievements of the "Project of Upgrading Approach and System for Retrieval of Fuel Debris and Internal Structures" carried out in the "Project of Decommissioning and Contaminated Water Management." As a part of this, it is important to determine further R&D needs and respond in an agile manner to critical issues regarding access to fuel debris, containment of radioactive material, and minimization of exposure, or have requirements reflected in new R&D initiatives.

In addition, through broadening perspectives to include technology development challenges from a strategic point of view and comprehensively searches for and collects basic and generic technology that is considered to be required in future, it is important to develop strategies for basic/generic studies through to applied and practical research based on an overall perspective."

6.2.3 Utilization of knowledge overseas

The countries, such as United States, the United Kingdom and France have an abundant knowledge and experiences through their activities of decommissioning and environmental reclamation of nuclear facilities including Nuclear Power Plants. It is considered that expertise and experience from such countries either directly or indirectly should be useful for the decommissioning of the Fukushima Daiichi NPS. Therefore, relevant domestic institutions are proactively promoting cooperation between overseas bodies in such as an advancement of internal PCV condition analysis using accident codes (OECD/NEA BSAF). It is important to continue to actively incorporate knowledge gained from overseas according to the needs derived from the decommissioning work. (Refer to Table 6-7)

6.3 Strengthening research cooperation

The Decommissioning R&D Partnership Council has been set up in the NDF, and that strengthened efforts to reflect the results and knowledge obtained through basic and generic research on decommissioning technologies to decommissioning work and development works. (Refer to Figure 6-2)

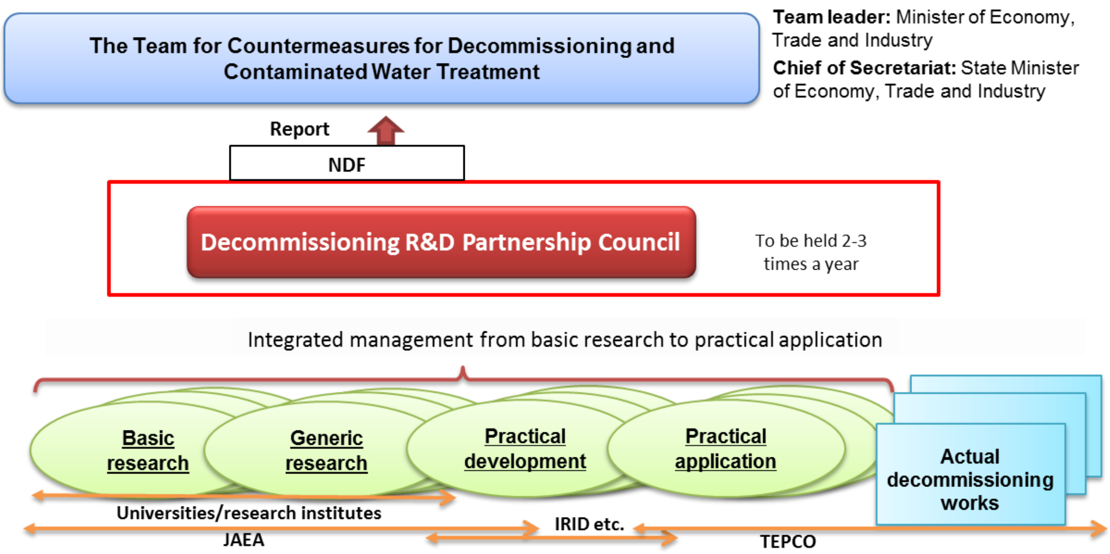


Figure 6-2 Functions of the Decommissioning R&D Partnership Council

The main roles of the Decommissioning R&D Partnership Council include as follows:

- Sharing information on R&D needs at each institution.
- Sharing information on promising R&D seeds.
- Coordinating R&D based on needs from decommissioning work.
- Promoting cooperation in R&D between each institution.
- Promoting cooperation in human resource development of each institution.

The council provides opportunities for discussion to promote communication between all parties, by integrating the efforts that had been undertaken independently by industries, universities, and academic society to date, and also by facilitating a share of the information about decommissioning work needs, situations of research institutes, and outcomes and knowledges gained from researches undertaken by research institutes and universities according to each specialized field between researchers of research institutes and universities and stakeholders those who directly engaged in decommissioning efforts. Through these activities, the diverse results of R&D efforts are expected to be applied to actual decommissioning work.

The Decommissioning R&D Partnership Council has agreed with the “Basic Policy on Strengthening R&D Cooperation” and the “Direction of Future Efforts” policy initiatives, as outlined in Table 6-8.

6.3.1. Mutual sharing of information on R&D needs, seeds and infrastructure construction

In order to strengthen cooperation in R&D, through basic, applied research to practical development, it is important to offer and share information such as regarding R&D seeds held by related parties, R&D needs, and decommissioning work needs. With regards to the information sharing in basic and generic research areas, JAEA established the Collaborative Laboratories for Advanced Decommissioning Science (CLADS) in April 2015. It aims to promote exchanges such as between personnel of domestic and international universities, research institutes, and industry, and to promote R&D and human resource development in an integrated manner by industry, academia, and government. In addition to the existing works of the Atomic Energy Society of Japan (AESJ) to strengthen partnerships with academic societies in various fields, it launched the Liaison Network of Academic Societies Contributing to Revitalization of Fukushima and Promotion of Decommissioning (with 33 participating organizations) in December 2015 to foster an even wider range of partnerships (see Table 6-9). It is important that the relevant institutions make full use of such opportunity to offer and share information.

6.3.1.1 Development of integrated information platform

Through cooperation with relevant organizations, the NDF has launched a web-based portal site (first version) that offers an access to R&D-related information on from R&D seeds, decommissioning work requirements, basic and generic research, through to applied and practical research. Not only being as a platform for offering and sharing information, following contents must be integrated and uploaded so that the portal can strengthen two-way cooperation, including the creation of match-ups of needs and seeds, and contribute to the greater participation of researchers from a wider field, such as from academia which had been less likely to be involved in the decommissioning of the Fukushima Daiichi NPS.

- On-site challenges and R&D efforts
- R&D results (e.g. photographs/video of actual field demonstrations at the Fukushima Daiichi NPS)
- Relevant R&D efforts (e.g. basic and generic research)
- Relevant data/environmental conditions at the site

6.3.2. Strengthening mutual cooperation and increasing participation of researchers from a wider field

In order to apply a variety of R&D outcomes at the decommissioning site, on-site decommissioning needs should to be matched with R&D seeds owned by universities and research institutes. However, as TEPCO that is supposed to know the needs derived from decommissioning work has to put precedence on dealing with on-site issues, R&D needs that sees beyond the mid-and-long-term tends to be hard to be sufficiently collated. Meanwhile, research institutes and universities find it difficult to fully comprehend decommissioning needs by themselves. Even if research institutes and universities have good technology or potential, R&D projects for visible R&D needs are already underway mostly led by reactor manufacturers that are familiar to the decommissioning sites, so it is difficult for them to join into these ongoing projects. Additionally, there are few channels for them to propose technologies or their potential.

To clear such situation, it is critically important to search for R&D seeds that meet R&D needs, and seek out R&D seeds that are missed in order to meet R&D needs through initiatives undertaken by the Decommissioning R&D Partnership Council, and visualize efforts that are required to meet decommissioning work needs in order to effectively utilize R&D resources. It is important to accurately incorporate these required efforts based on their TRL into the activities of relevant institutions.

6.3.2.1 Mutual ooperation functioning as tangible and effective bridge

As decommissioning work needs are not always the same as R&D needs, they do not simply correspond to the R&D seeds held by researchers. Furthermore, decommissioning needs are based on the situation of the site and the status of related R&D at the in time. Therefore, it is important to pick out "appropriate R&D needs" from decommissioning work needs prior to searching for R&D seeds. In addition, in order to search for and to pick out basic and generic technologies that would be required in the future from a wide area, it is needed to have efforts to grasp an overall image of the challenges associated with the decommissioning of the Fukushima Daiichi NPS.

To promote the search for R&D seeds among the selected R&D needs, it is essential to understand a greater range of R&D seeds, and it is also important to actively provide easily understandable information to researchers who are engaged in R&D basic and generic technologies in universities and other institutes in home and overseas including information that would elicit their interest in the decommissioning of the Fukushima NPS. In addition to providing opportunities to make actual matching, it is needed to provide information to researchers who have potential and have interests in the technology for which there is a clear need through the "Information Portal for the R&D for the Fukushima Daiichi Decommissioning" and the Platform of Basic Research for Decommissioning set up by JAEA as a consultative body to promote basic and generic research for decommissioning, and to encourage their participation in R&D related to the decommissioning of the Fukushima Daiichi NPS as well.

Taking these efforts into account, a task force has been established to identify additional R&D needs for the decommissioning of the Fukushima Daiichi NPS that should be addressed strategically and in a prioritized manner, to prioritize issues with matching R&D needs and seeds, and also to accurately deploy R&D at relevant institutions based on TRLs and so on. The NDF and JAEA will lead efforts that continue to actively promote activities of the taskforce and the Platform with cooperation with TEPCO and other related institutions.

6.3.3. Development of R&D centers

It is important to construct a framework that brings together human resources from varied fields with different roles and levels of expertise referencing the types of functions required of an open innovation hub to accomplish research efficiently and effectively in the course of developing and operating R&D centers.

Furthermore, basic and generic research, and the development of human resources should also be undertaken in an integrated manner, working closely such as with universities.

In addition to cooperating with efforts concerning the revitalization of Fukushima and surrounding areas and R&D center projects (e.g. environmental recovery, health management, regional economic development), it is important to consider activities being undertaken at existing facilities (e.g. academic research on debris from Three Mile Island (TMI) at JAEA Tokai and the Oarai Research and Development Center, sharing workload or providing a backup system related to the decommissioning of the Fukushima Daiichi NPS). The status of development of R&D centers is shown in Table 6-10. Also, the functions of mockup test facility of JAEA, Radioactive Material Analysis and Research Facility and Collaborative Laboratories for Advanced Decommissioning Science are as follows.

Mockup test facilities (“Naraha Remote Technology Development Center”) -Research management building -Test building	・ A facility to conduct development/demonstration of remote control equipment/devices ・ Simulating the work environment in the building of the Fukushima Daiichi NPS by using mockup facility, virtual reality and robotics simulator. ・ Demonstration test of equipment required for decommissioning work by using the mockup and training for workers using virtual reality system and so on.
Radioactive Material Analysis and Research Facility (“Okuma Analysis and Research Center”) -Facilities administration building -Building No. 1 -Building No. 2	・ Analysis facility for radioactive wastes, fuel debris, etc. ・ Installed analysis devices to obtain data that can contribute to the appropriate processing of radioactive waste. ・ Installed equipment that can obtain basic data of fuel debris, which are high level radioactive waste.
Collaborative Laboratories for Advanced Decommissioning Science (CLADS) -International Collaborative Research Building	・ Promotes R&D and human resource development in comprehensive manner while establishing a network that facilitates the communication among universities, research institutes, and industry of multiple fields in Japan and abroad and the nuclear industry. ・ R&D for characterization, storage, processing and disposal of radioactive wastes. ・ R&D for characterization, handling, analysis, etc. of fuel debris ・ R&D for clarification of chemical and migration behavior of substances inside the reactor ・ Inspections of fuel debris and R&D for the visualization of the radiation

6.3.4. Development and recruitment of human resources

In regard to development and recruitment of human resources required to continue the decommissioning of the Fukushima Daiichi NPS over the long term, it is important to specify models of human resources required and priority technical fields in which training should be focused, taking an overall view of the entire decommissioning process in the date ahead and human resources involved in it into account.

Meanwhile, efforts of the nuclear power industry as a whole are also required in order to recruit necessary human resources continuously. In addition to industry and educational institutions cooperatively promoting activities that makes students deepen their understandings on nuclear industry and conveying attractiveness of nuclear energy industry to students continuously, it is important to show paths of researchers and technicians to success in the process of decommissioning of the Fukushima Daiichi NPS, such as conveying the “attractiveness” of the decommissioning of the Fukushima Daiichi NPS that is extremely high-level technical challenges never seen or presenting diverse clear career paths that researchers and engineers can play an active part in.

Relevant institutions are currently engaged in the following human resource development efforts based on their standpoints respectively. It is important to effectively and efficiently continue to make the relevant institutions to cooperate with each other as well.

MEXT	In addition to strongly promoting the cooperation in tackling the 7 projects adopted in FY2014/FY2015 through the Platform of Basic Research for Decommissioning, promotes initiatives that contributes to human resource development (see Table 6-11 and 6-12 for details). a. In FY2014, three projects by three institutions, including Tohoku University (Theme: Basic research and core human development program on maintaining the reliability of plant facilities including PCV and buildings, and treatment and disposal of nuclear fuel debris for the accidental nuclear power plant decommissioning) were adopted. b. In addition to three institutions selected in FY2014, additional four projects by four institutes were adopted such as the one by the University of Fukui (Theme: Research and human resource development for analysis of fuel debris and decommissioning technology of Fukushima Daiichi nuclear power plants). c. New initiatives are being considered in order to contribute to human resources development, including holding a Creative robot contest for decommissioning to have young students get interested in decommissioning work (National Institute of Technology, Fukushima College).
TEPCO	Promotes education and human resources development through deepening understandings of situation of the Fukushima Daiichi NPS and decommissioning work. a. Field tours b. Internships c. Dispatching lecturers to universities and technical colleges d. Participation in various conferences
JAEA	In order to strengthen its function on mid-and-long-term human resource development, establishes collaborative lectures in conjunction with institutions adopted such as in the MEXT's "Human Resource Development and Research Program for Decommissioning of the Fukushima Daiichi NPS," and integrates different field analysis technology and implements human resources development initiatives. In addition, using cross-appointment system to bring a diverse range of human resources together. Promotes human resource development through practical training during summer vacation, involvement in research from a few months to one year, and programs to accept students.
IRID	Contributes to the securing of human resources and their development through offering information related to the R&D projects of the decommissioning of the Fukushima Daiichi NPS and through joint research initiatives, etc. a. Collaborations with MEXT's human resource development programs (cooperation for workshops) b. Holding workshops in conjunction with universities (sharing status of R&D efforts of each, discussions on particular themes) c. Lectures/exhibitions at events d. Dispatching lecturers (to universities, technical colleges, etc.)
Hitachi-GE Nuclear Energy / Toshiba / Mitsubishi Heavy Industries	As a measure to deal with the decline in public acceptance of atomic energy due to the accident at the Fukushima Daiichi NPS and accompanying drop in students aspiring to enter the nuclear power industry, develops various activities for maintaining and expanding numbers across the nuclear power industry as a whole (For details, refer to Table 6-13). a. Activities through the network for development of human resources in nuclear power field such as the "Nuclear Dojo" (Nuclear energy education in Japan and abroad in collaboration with 16 universities in Japan). b. Activities through the Atomic Energy Society of Japan (AESJ) such as participating in the Young Researcher's Meeting. c. External PA/educational support activities such as dispatching lecturers to universities, etc.
CRIEPI	In addition to develop younger researchers through a variety of researches such as the one focused on the treatment of contaminated/sea water, the Permeable Reactive Barrier, and so on outlined in Table 6-5, takes charge of MEXT's "Human Resource Development and Research Program for Decommissioning of the Fukushima Daiichi NPS" entrusted by the Japanese Geotechnical Society as a society activity.

<IN FOCUS>

Applied research of particle physics and the decommissioning of the Fukushima Daiichi NPS

The muon36 was firstly observed in cosmic radiation37 by Carl D. Anderson and Seth Neddermeyer in 1936.

The muon is the most common charged particle on the Earth's surface among secondary cosmic radiation that is generated by a hit of atmosphere by cosmic radiation.

Muons have a 207 times higher mass of an electron, have relatively lower energy loss in matter, and are much more deeply penetrating than electrons. While muons are constantly passing through every matter on the Earth due to this fact, they are totally harmless. The vertical muon flux is about 1 muon penetrates an area of 1 cm2 per minute. That means millions of muon particles are passing through a human body over one night.

Taking advantage of greater penetrating ability of the cosmic muons, many researchers have attempted to experiment with radiography38 of massive objects. In the 1960s, Luis W. Alvarez of the University of California, Berkeley used muons to search for possible hidden chambers in the Pyramid of Khafre in Egypt. More recently, a joingt research team of the Earthquake Research Institute of the University of Tokyo and the High Energy Accelerator Research Organization (commonly known as “KEK”) successfully analyzed changes in density of the volcanic interior before and after eruptions of Mt. Asama (in Nagano Prefecture, Japan) with muon radiography (“Muography”) in 2009.

This shows that the internal structure of objects, such as pyramids or volcanoes, can be investigated by printing tracks (of intensity and direction) of muons that passed through the target object onto photographic plates and reconfiguring the density distribution in the object. Because the horizontal incident energy of muons (i.e. penetrating power) is larger than the vertical one, they are more suitable for examining taller structures like pyramids.

Having learned that muon tomography techniques are likely to be applicable to the investigation into the inside of the damaged reactors at the Fukushima Dai-ichi nuclear power station (NPS), KEK in April 2011, shortly after the nuclear accident at the site, proposed TEPCO (Tokyo Electric Power Co., Inc.) to perform examination, with a simple equipment installed onsite near the reactor buildings to evaluate the applicability of such techniques. TEPCO agreed to carry out the examinations as long as the onsite decommissioning work would not be affected, considering the growing demand for capturing the status inside the reactors.

Based on the findings from TEPCO’s examiniations, KEK made improvements and upgrades for accuracy to the equipment and conducted a demonstration test between 2013 and early 2014 at the Tokai Daini Power Station of the Japan Atomic Power Company to provide muon imaging of nuclear fuels inside a reactor.

Later, the IRID in Japan and KEK, working together as a part of government aided projects, set out to develop a new muon tomography method intended for a demonstration test at Unit 1 of the Fukushima Daiichi NPS. Based on the relevant data obtained from the test conducted in January 2015, the damaged inside of Unit 1 was successfully imaged.

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(36) The muon, with an electric charge of −1 e equal to the elementary charge and a spin of 1⁄2, has a static mass of 105.6 MeV/C2 (about 206.7 times of an electron) and a mean lifetime of 2.2×10-6 sec.

(37) High energy cosmic rays, which are mostly generated and emitted by a supernova explosion and then undergo disturbance due to magnetic fields while they propagate in interstellar space, arrive at the Earth's surface isotopically in both vertical and horizontal directions.

(38) A type of non-destructive imaging inspection method that allows radiation to pass through a target object irradiated to be printed onto films for visualization, in order to see what it's like inside the object.

Reference: http://www.eri.u-tokyo.ac.jp/koho/press/asama2009/

https://ja.wikipedia.org/wiki/%E3%83%9F%E3%83%A%%E3%83%BC%E7%B2%92%E5%AD%90

Table 6-1. METI

- Measures for decommissioning of the reactor and treatment of the contaminated water (R&D)

Title	Objectives and contents
R&D for an understanding of the quantity, locations and properties of fuel debris and the distribution of fission products (FPs)	
Upgrading level of comprehensive condition analysis (Fig. 6-3)	To contribute to checking of the plant's stability and the determination of fuel debris retrieval policies and methods, comprehensive analyses and assessments shall be carried out to precisely grasp the situation of debris and fission products inside the reactor.
Development of technology for analysis of debris properties (Fig. 6-4)	To contribute to comprehensive analyses and assessments of situation inside the reactor, determination of methods for retrieval from inside the reactor and development of technologies for packing, transferring and storing fuel debris, the properties of fuel debris will be analyzed and assessed. Furthermore, mock debris will be used for experiments, and technologies necessary to analyze and measure actual debris to be retrieved will be developed.
Development of technology for PCV internal survey (Fig. 6-5)	To contribute to determination of fuel debris retrieval policies, equipment for investigating and detecting fuel debris locations and its spread and for checking the situation inside and outside the pedestal shall be developed and tested. Furthermore, remote devices for more detailed visual inspection will be developed to establish fuel debris retrieval methods.
Development of technology for RPV internal survey (Fig. 6-6)	To contribute to establishment of methods for fuel debris retrieval, investigative technologies utilizing remote devices for getting a good grasp of the debris situation inside the RPVs shall be developed.
R&D for fuel debris retrieval methods	
Upgrading approach and system for retrieval of fuel debris and Internal structures (Fig. 6-7)	To contribute to the establishment of fuel debris retrieval policies and methods, the development of the technologies and system for retrieving fuel debris and structure inside the reactor will be carried out. Submersion and partial submersion methods as well as retrieval from top and from side will be studied.
Development of fundamental technologies for retrieval of fuel debris and internal structures (Fig. 6-8)	To contribute to assessment of the feasibility of retrieval technologies for fuel debris and structures inside the reactor, element tests shall be carried out on the relevant devices. Submersion and partial submersion technologies as well as retrieval from above and from the sides will be tested.
R&D for ensuring safety during fuel debris retrieval work	
Development of corrosion inhibition technology for RPV/PCV (Fig. 6-9)	To contribute to determination of fuel debris retrieval policies, the applicability of structural material corrosion preventing measures for the RPVs and PCVs, into which seawater was once injected, shall be assessed.
Development and management of evaluation method of RPV/PCV seismic performance/impact (Fig. 6-10)	To contribute to deciding on fuel debris retrieval policies, safety scenarios in case of a large-scale earthquake shall be established for important equipment inside RPVs and PCVs and technologies for assessing measures for preventing and mitigating its impact will be developed.
Development of Criticality Control Technologies of Fuel Debris (Fig. 6-11)	To contribute to checking plant stability and determining fuel debris retrieval policies and methods, criticality assessment technologies shall be established. At the same time, with multiple technologies for retrieving fuel debris in mind, criticality control technologies such as a criticality detection technology and a criticality prevention technology using neutron absorber shall be developed.
Development of repair technology for PCV leak locations (Fig. 6-12)	To contribute to determination of fuel debris retrieval policies and methods and in terms of radioactive material scatter and expansion prevention, a feature for confining water inside the PCVs for the purpose of radiation shielding and cooling will be established and a technology for repairing leaking parts to maintain stable confinement function will be developed. These shall be verified for their applicability to actual equipment.
Full-scale test of repair technology for PCV leak locations (Fig. 6-13)	To contribute to determination of fuel debris retrieval policies and methods, a full-scale test shall be implemented to check the validity of a developed technology for repairing bottom part of PCV.

R&D for stable storage of fuel debris after its packing and transfer		
	Development of fuel debris packing/transfer/storage technology (Fig. 6-14)	To contribute to determination of fuel debris retrieval policies and methods, a system will be developed that can safely and reliably pack, transfer and store retrieved fuel debris.
R&D for studying waste storage, processing and disposal measures		
	R&D for solid waste processing and disposal (Fig. 6-15 and 6-16)	To smoothly promoting decommissioning activities, a set of feasible waste management and handling methods covering waste generation, storage, processing and disposal shall be presented with substantial reasons. For this, the properties of radioactive waste will be analyzed and assessed. As for difficult-to-measure nuclides, their analytical methods shall be developed. In addition, measures for integrity for long-term storage of post-water-process secondary waste shall be laid out. Characteristics of wastes shall be taken into account, and safe disposal concepts and their preconditions shall be clarified.
R&D for fuel stored in spent fuel pools		
	Assessment of long-term integrity of fuel assemblies retrieved from SFP (Fig. 6-17)	Integrity assessments shall be performed for fuel assemblies affected by hydrogen explosion and seawater injection and retrieved from spent fuel pools in a shared pool and dry storage facility for the long-term safe storage.

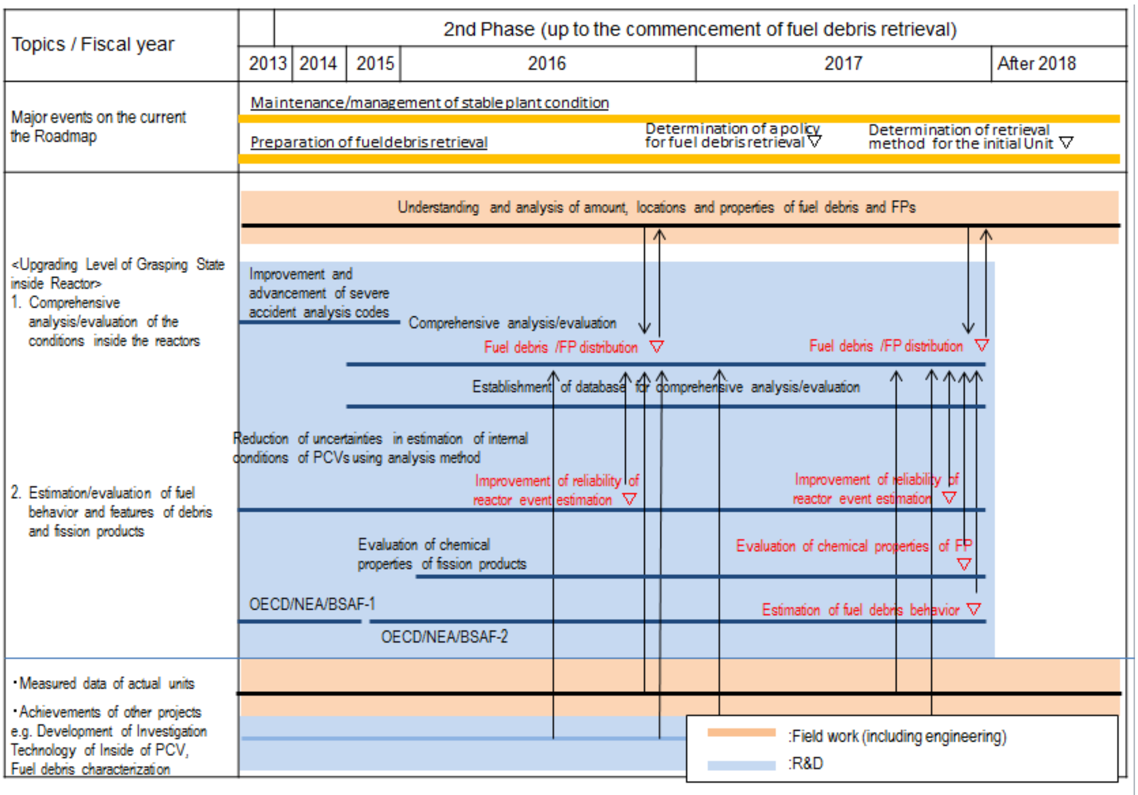

Figure 6-3 Upgrading level of comprehensive condition analysis

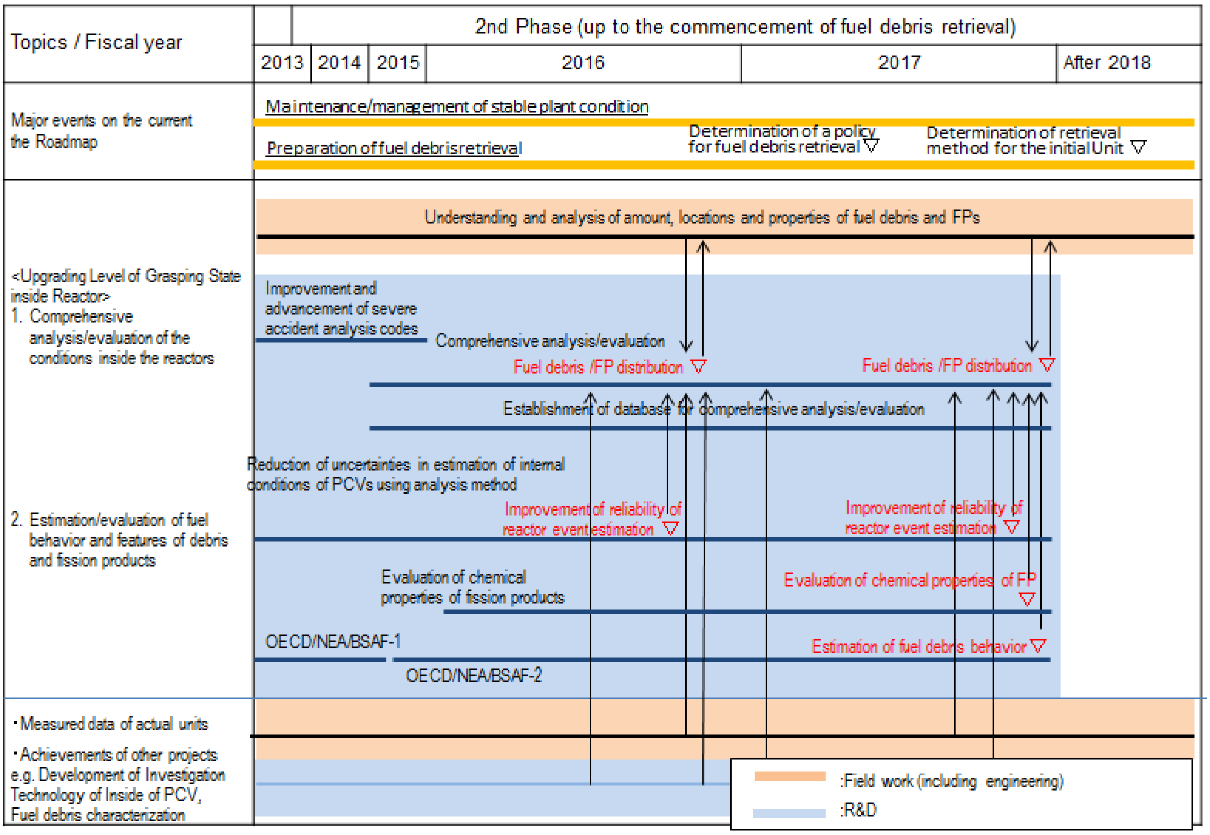

Figure 6-4 Development of technology for PCV internal survey

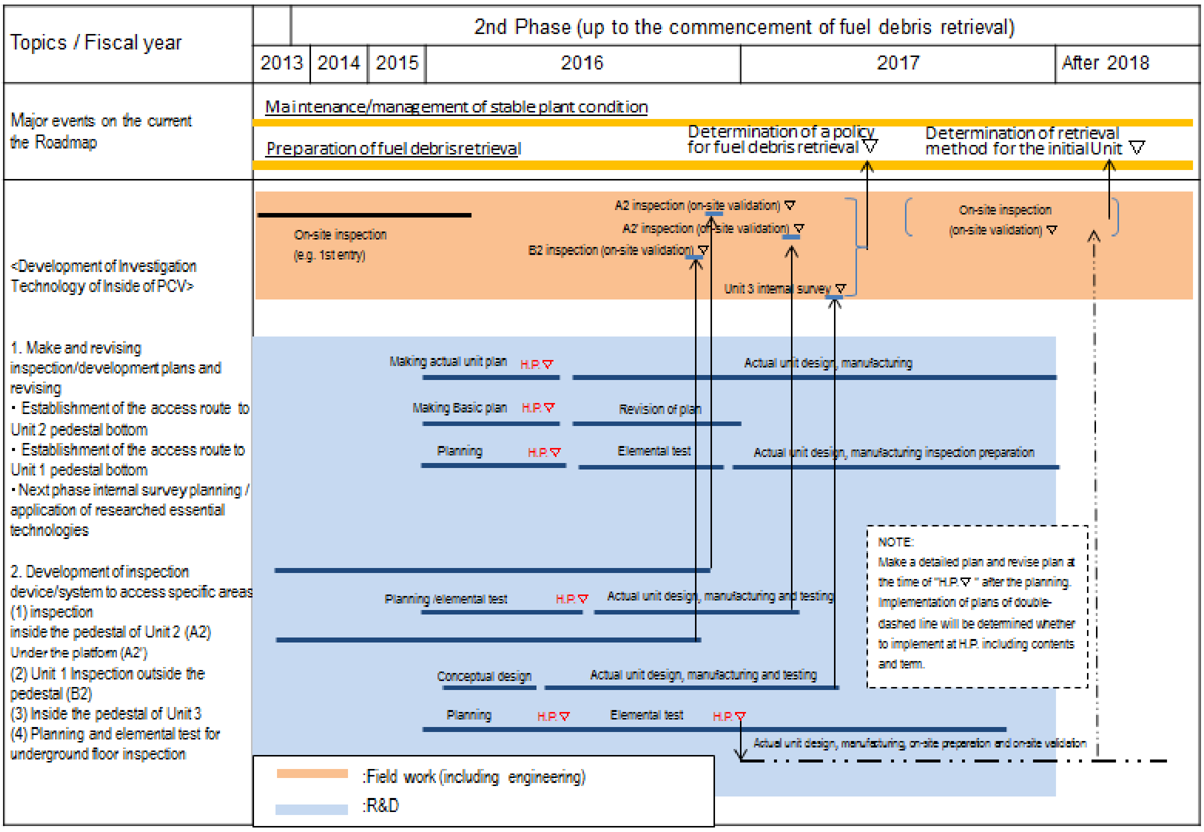

Figure 6-5 Development of technology for PCV internal survey

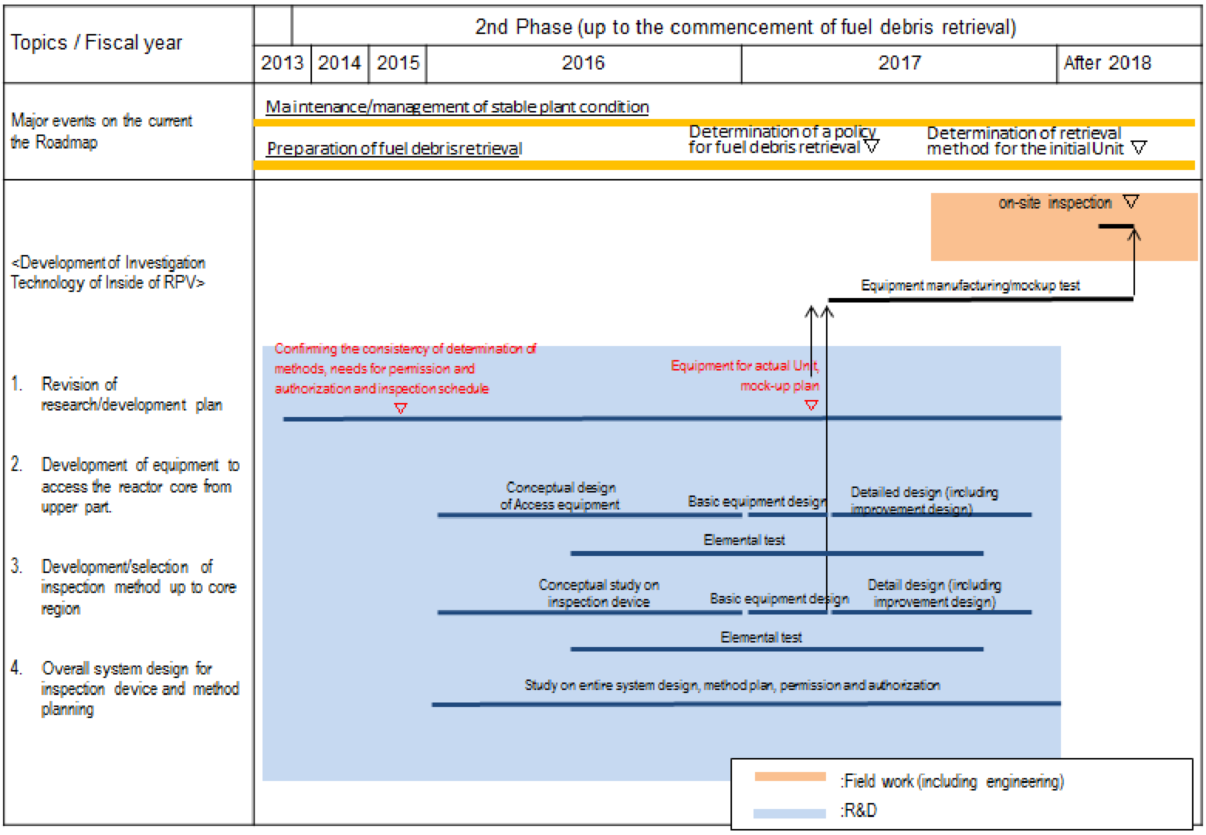

Figure 6-6 Development of technology for RPV internal survey

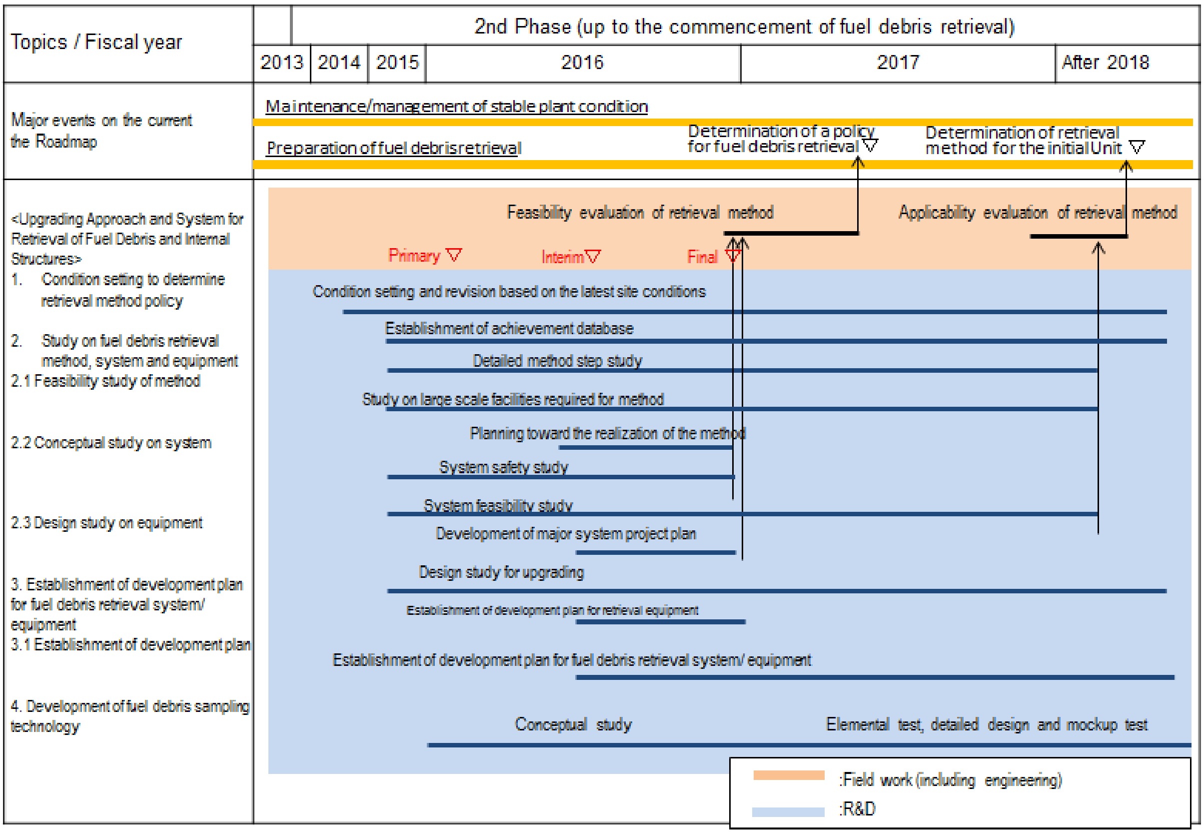

Figure 6-7 Upgrading approach and system for retrieval of fuel debris and internal structures

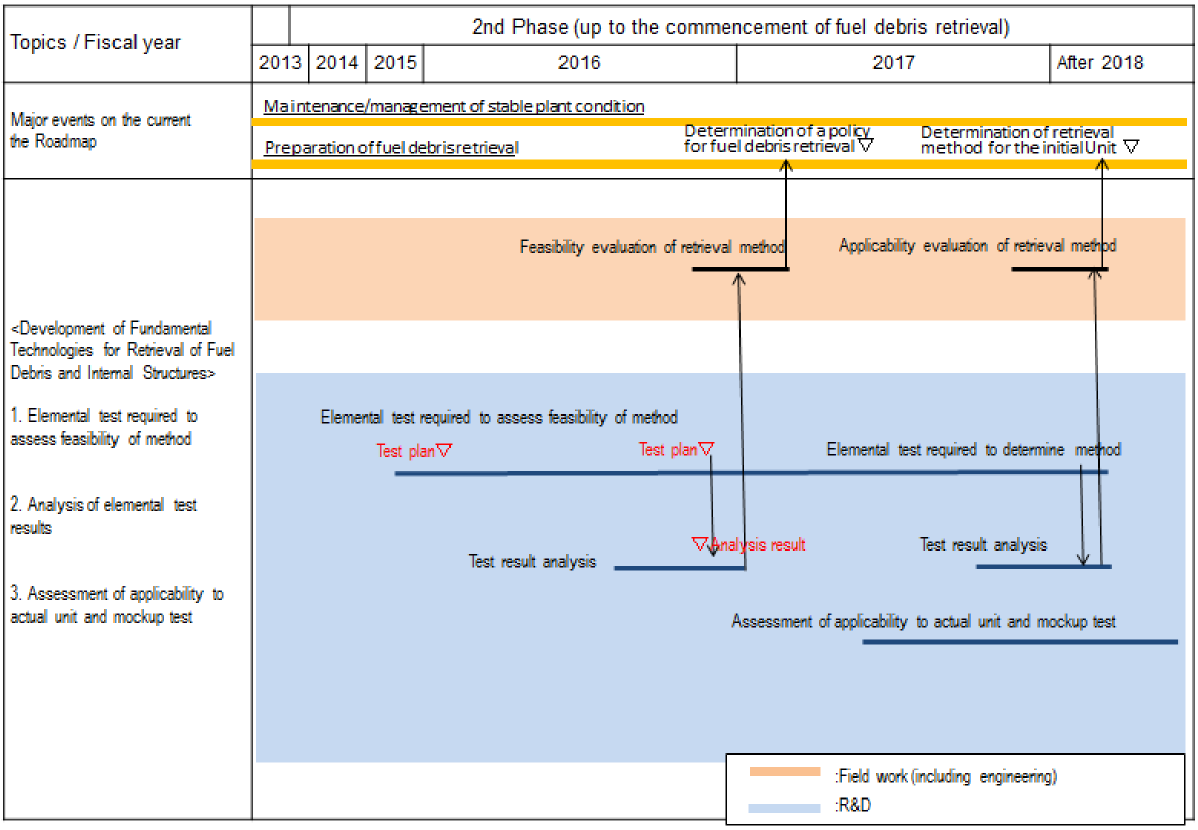

Figure 6-8 Development of fundamental technologies for retrieval of fuel debris and internal structures

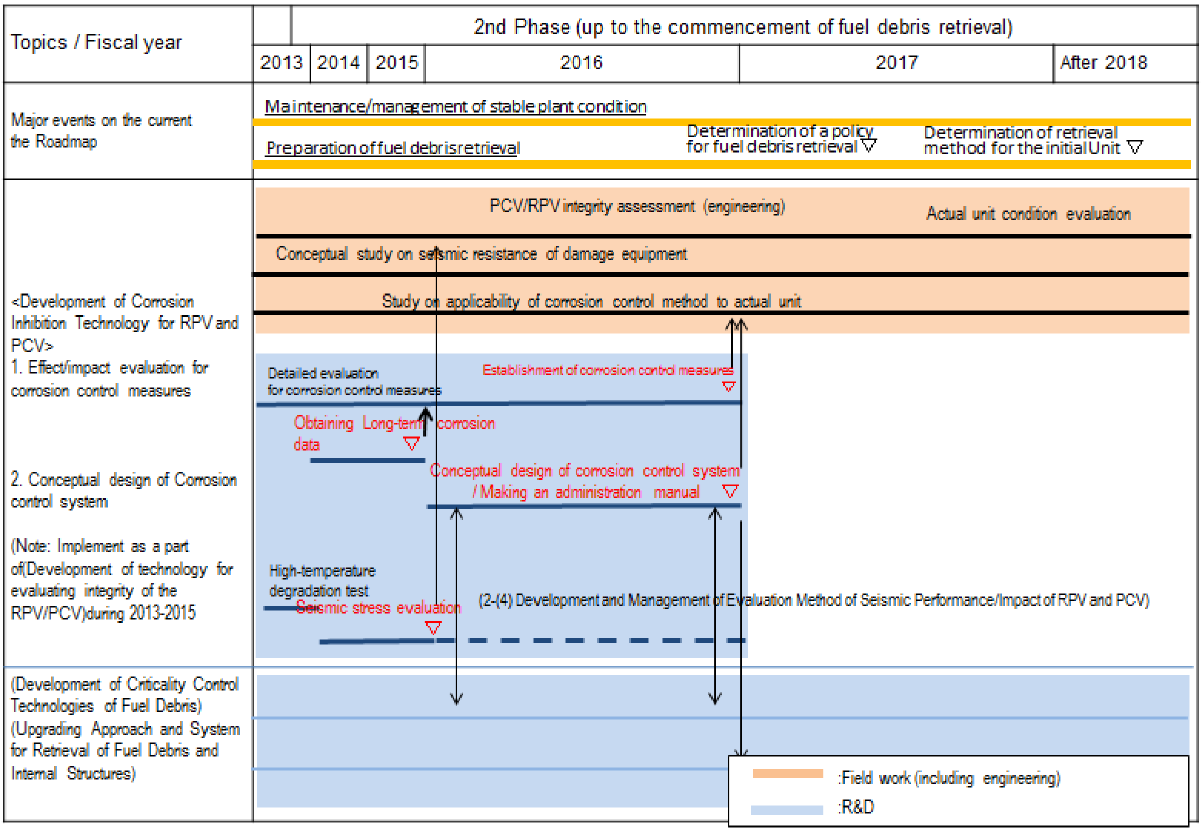

Figure 6-9 Development of corrosion inhibition technology for RPV/PCV

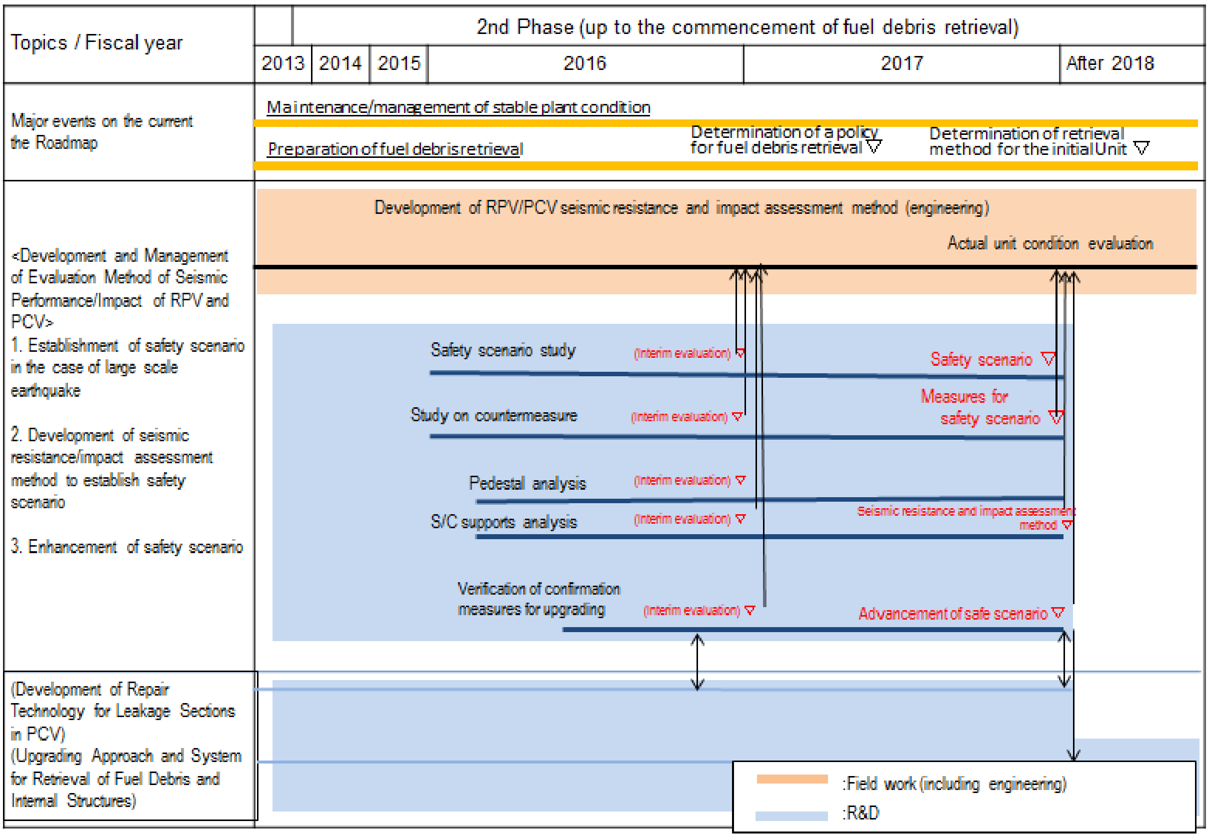

Figure 6-10 Development and management of evaluation method of RPV/PCV seismic performance/impact

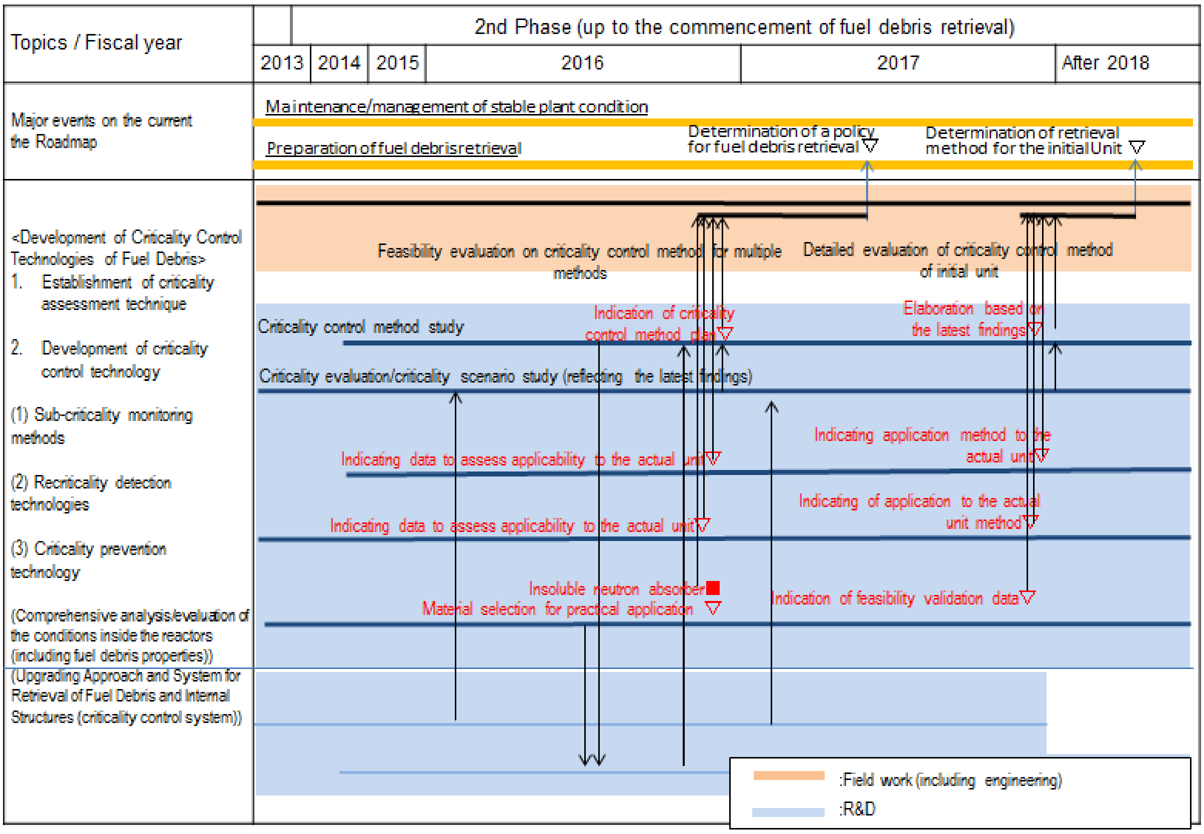

Figure 6-11 Development of fuel debris criticality control technologies

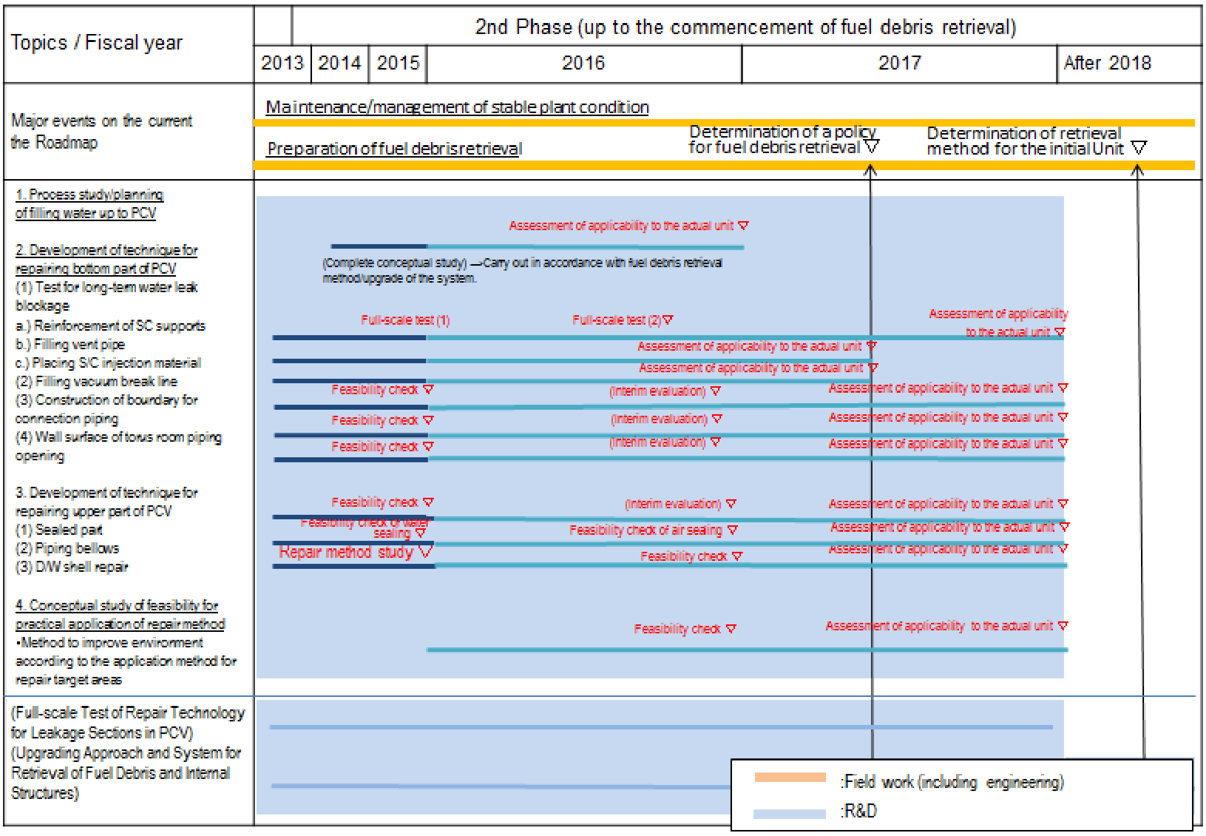

Figure 6-12 Development of repair technology for PCV leak locations

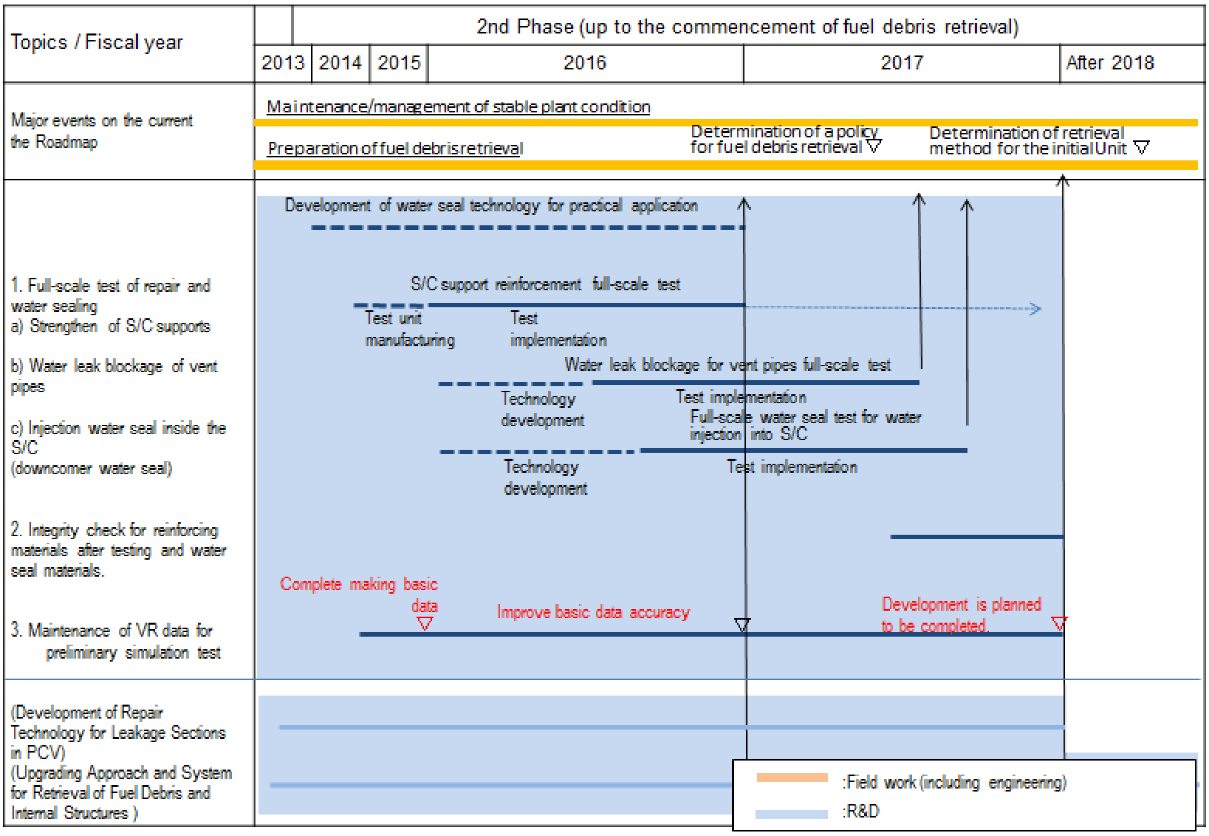

Figure 6-13 Full-scale test of repair technology for PCV leak locations

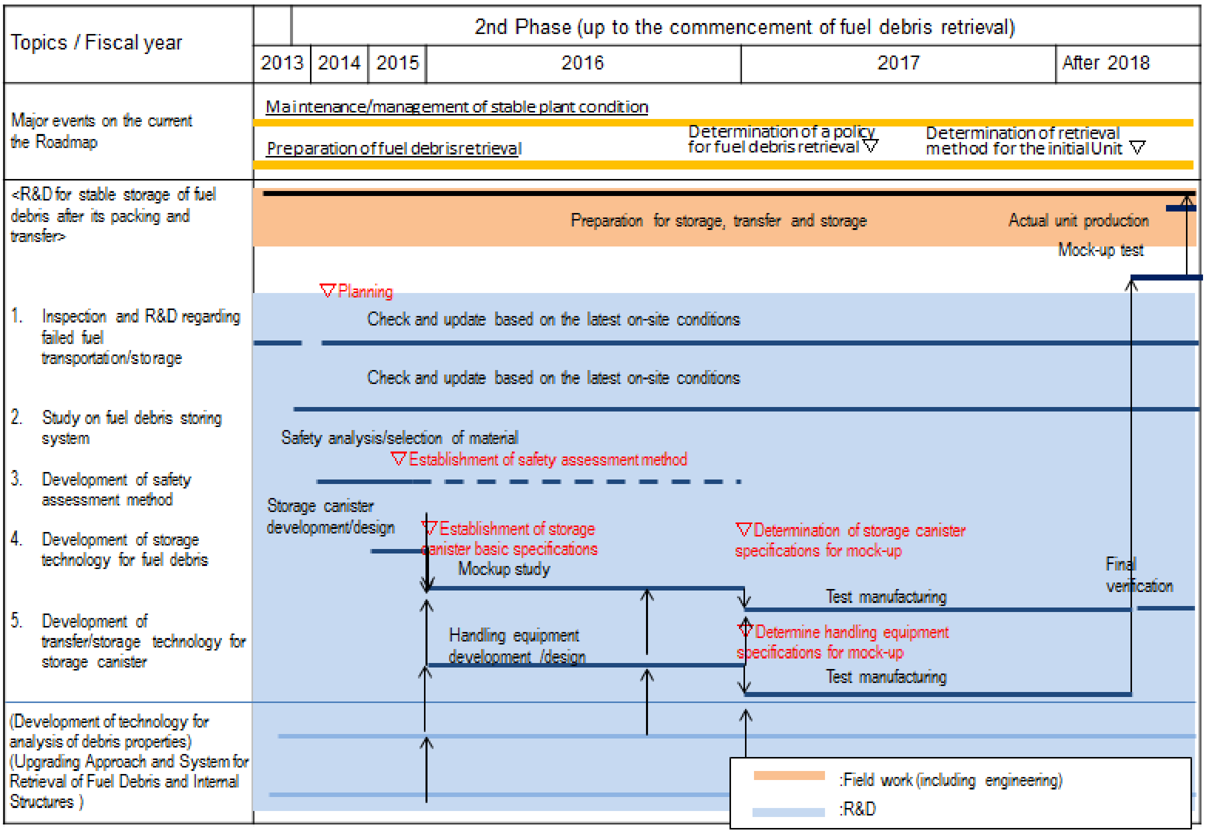

Figure 6-14 Development of fuel debris packing/transfer/storage technology

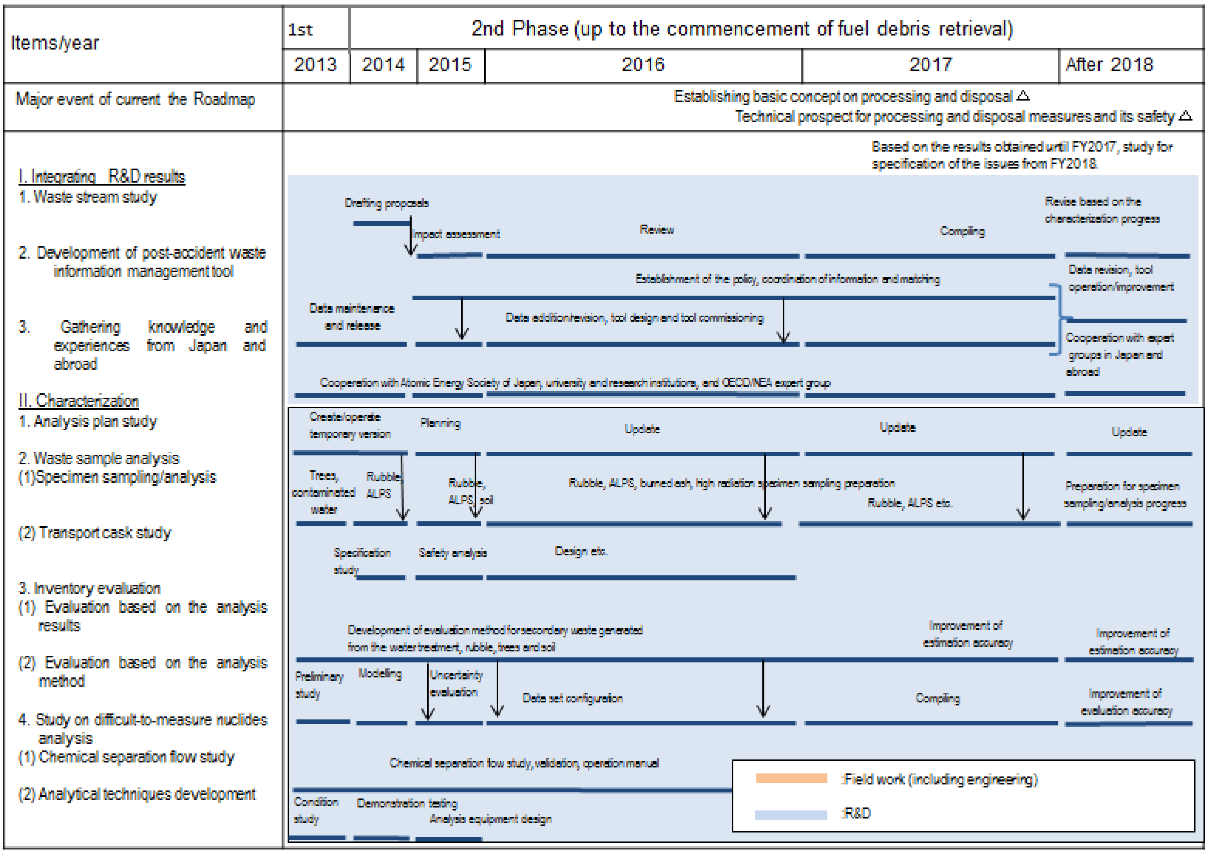

Figure 6-15 R&D for solid waste processing and disposal

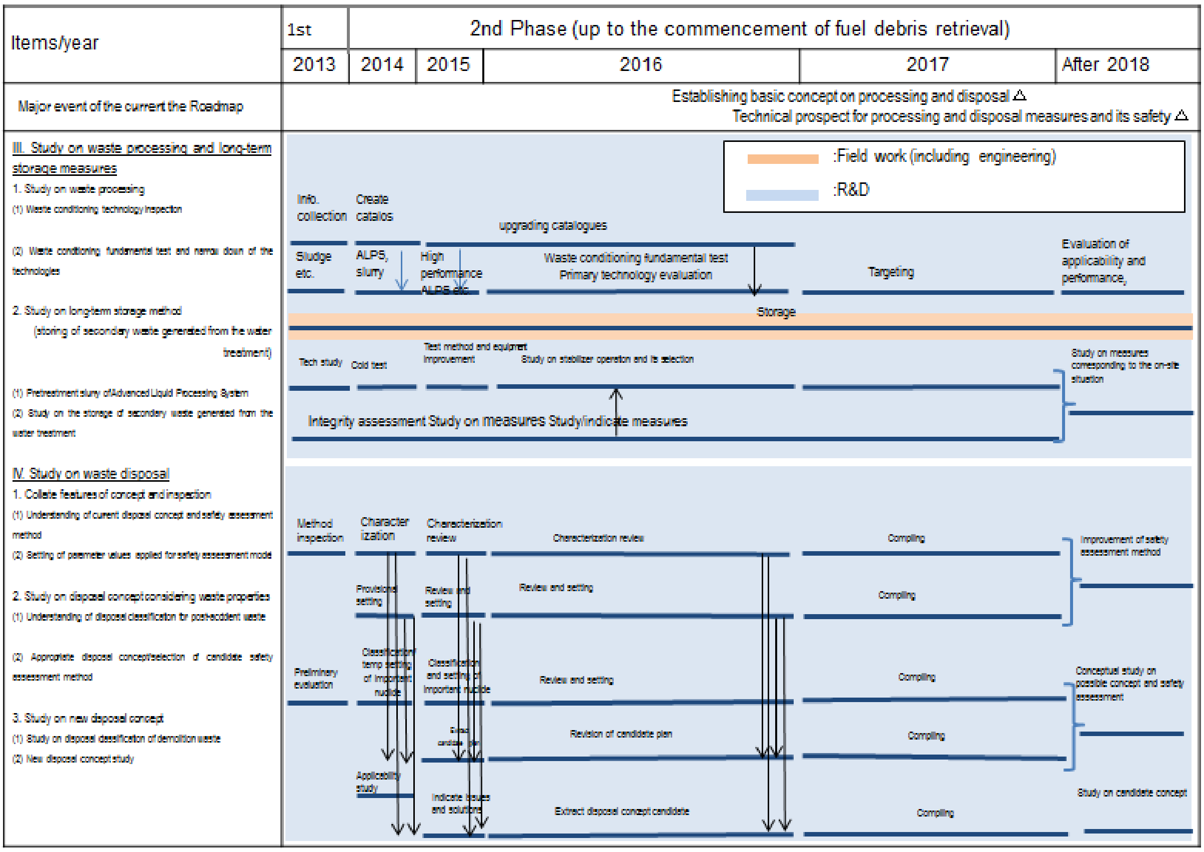

Figure 6-16 R&D for solid waste processing and disposal

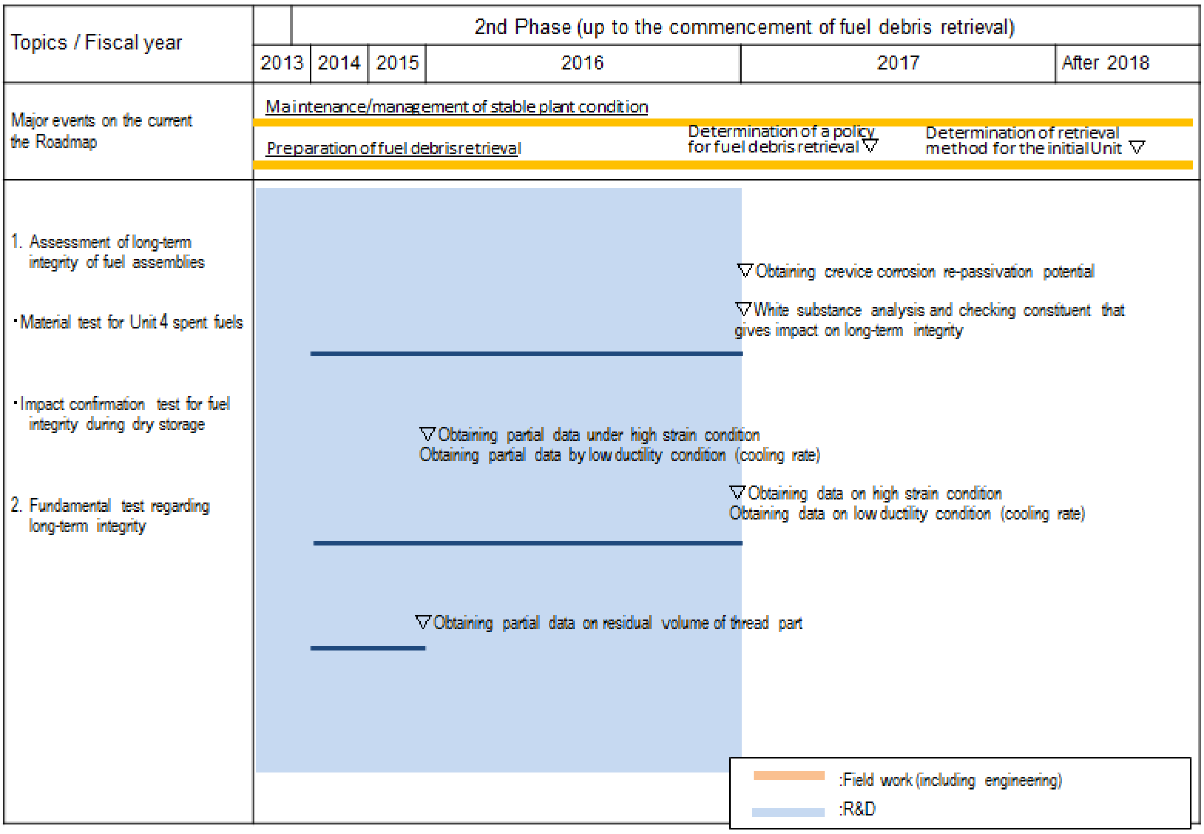

Figure 6-17 Assessment of long-term integrity of fuel assemblies retrieved from SFP

Table 6-2 MEXT

Acceleration Program for Reactor Decommissioning R&D (Domestic research) – Projects adopted for FY 2015

Core institutes	Tasks
Theme 1 Research on fuel debris retrieval	
Tokyo Institute of Technology	Accessibility for removal of fuel debris in BWR plant after severe accident
Hokkaido University	Development of a high efficiency multi-nuclide aerosol filters for radiation protection during a process of cutting core debris
JAEA	Advanced study on remote and in-situ elemental analysis of molten fuel debris in damaged core by innovative optical spectroscopy
Theme 2 Research on environment countermeasures including those against waste	
Shinshu University	Decontamination of radiation-tainted water with newly-designed metal oxide nanomaterials
JAEA	Study on evaluation of radionuclides inventory on the 1F site based on the behavior observed in the environment off the 1F site

Table 6-3. MEXT Acceleration Program for Reactor Decommissioning R&D

(Joint Research between Japan and the UK) – Projects adopted in FY 2015

Core institutes	Institutes representing the U.K.	Tasks
Theme 1 Research on fuel debris retrieval		
Nagaoka University of Technology	Lancaster University	Technology development to evaluate dose rate distribution in PCV and to search for fuel debris submerged in water
Tokyo Institute of Technology	University of Bristol	An ultrasonic measurement system and its robotic deployment into vessels for the combined assessment of debris condition and water leakage
Theme 2 Research on environment countermeasures including those against waste		
Kyushu University	University of Sheffield	Advanced Waste Management Strategies for High Dose Spent Adsorbents
JAEA	University of Sheffield	Development of solidification techniques with minimised water content for safe storage and disposal of secondary radioactive aqueous wastes in Fukushima

Table 6-4 Activities by JAEA

Area of research	Research subject
Processing and disposal of waste	Since there are various types of radioactive wastes generated by the accident and their properties are unknown, appropriateness of disposal technologies to be applied and safety during disposal contain uncertainties. For this reason, in addition to developing the technologies related to the current processing and disposal, more advanced and reasonable technology are being developed for processing and disposal in order to minimize uncertainty. ・Technical development related to improvement of reliability of safety assessment of disposal ・Technical development for the improvement of artificial barriers ・Technical development for the improvement of waste packages ・Development of a method for analyzing difficult-to-measure radionuclides ・Assessment of integrity of the container of spent absorption tower of cesium absorption system ・Development of safe technology for storing post-accident waste ・A study for disposal according to properties ・A study for a new disposal concept
Fuel debris handling and analysis	For fuel debris retrieval, it is needed to recognize conditions of distribution and attachment of fuel debris inside the reactor and radioactive materials, their mechanical, chemical and thermal properties and radiation exposure during retrieval process. For this reason, R&D related to characterization and developments of analysis/measurement technology are underway. ・Development of an in-situ analytical probe inside the reactor using radiation-proof optical fiber ・Development of a remote spectral analysis technology using an element-isotope quantitative analysis laser ・Development of X-ray surface analysis technology ・Assessment of properties of debris inside the reactor ・Development of debris analysis technology ・Development of a technology for assessing dose distribution inside the reactor ・Development of a rational accounting (measuring control) technology applicable to fuel debris ・Development of a technology for heat generation and cooling of fuel debris
Assessment of accident progression	Assessment using severe accident analysis code, improvement of accuracy of accident progression scenario analysis requires detailed analysis and experimental data on various kinds of phenomena and behavior in the course of accident. For this reason, analyses and experiments required to conduct evaluation of failure behavior of structural materials and thermal flow behavior. ・Evaluation of failure behavior of structural materials ・Evaluation of thermal flow behavior ・Evaluation of chemical behavior of FP nuclides ・Evaluation of migration behavior of reactor core materials
Remote control technology	Virtual reality training system to improve standard test method for disaster response robot, control technology for remotely controlled device and robot simulator utilized for development and modification of robot are being developed and maintained to improve utilization of Naraha Remote Technology Development Center. Also, R&D on inspection of fuel debris inside the reactor and improvement of safety for decommissioning work and visualization of the radiation to improve efficiency is underway. ・Development of standard test method to disaster recovery robot ・Technical development to improve accuracy of virtual reality system ・Development of a robot simulator ・Development of a gamma ray imaging technology ・Development of a technology to measure gamma-ray and neutron ray under high-dose radiation environment ・Development of a SLAM technology for recognizing work environment and a highly permeable underwater laser measurement technology ・Development of a technology for establishing environmental structure models based on remote-sensing data and development of a system for storing and maintaining spatial-temporal data ・Development of a technology for utilizing spatial-temporal data by combining virtual reality interface and robot simulator ・Development of a technology for the 3D visualization of radiation source distribution ・Development of a contamination source analysis and estimation technologies based on inverse estimation on the basis of monitoring data ・Development of an impact and risk assessment technology by collecting, integrating and analyzing monitoring data

Table 6-5 Efforts by CRIEPI

Research subject	Area of research
Contaminated water/ seawater treatment and permeable reactive barrier	- Nuclear chemistry and water chemistry - Groundwater flow and diffusion - Groundwater organisms - Geological condition
Assessment of diffusion and impact on the environment	- Atmospheric diffusion - Marine diffusion - Bioaccumulation
Assessment of fuel debris properties and development of accounting and control technologies	Reactor physics and fuel
Material integrity assessment (pipe corrosion)	Material corrosion and water chemistry
Accident analysis code (BSAF Phase-2)	Thermal hydraulics
Waste management (processing and inventory assessments)	- Nuclear chemistry - Waste disposal

Table 6-6 "R&D duties execution policy" and issues to be focused for the R&D project management

No.1 Basic policy on the operations of the NDF regarding the R&D for the technologies required for appropriate and steady decommissioning

1 Implementation of the operations aiming at the practical application

(1) Multi-layered approach

Policy	Additional issues to be focused on
In order to deal with the highly uncertain situation with a number of unknown factors in the site conditions, take a multi-layered approach taking the risk assessment results into account as well.	(1) Develop multi-layered approach for the investigations and access into the reactor and the fuel debris.

(2) Prioritization of objectives and flexible review considering on-site needs

Policy	Additional issues to be focused on
Determine the order of priority of the objectives based on the short-term and mid- to long-term needs and issues of the site and develop an R&D plan. In addition, review the objectives flexibly based on the feedbacks such as the latest findings or the knowledge obtained from the actual decommissioning process.	(1) Before starting the R&D project, establish objectives to be achieved and the order of priority considering site needs, and share the objectives among the relevant organizations. Take note of the highly uncertain situation as well. (2) Since the needs for R&D will change according to the latest site conditions, it is important to flexibly and promptly review the objectives and the order of priority. Therefore, it is important to periodically follow the progress of R&D as well as to arrange venues to confirm the necessity of reviewing the issues to be addressed or the objectives to be achieved.

(3) Realization of efficient R&D

Policy	Additional issues to be focused on
By conducting efficient R&D and appropriate division of roles, eliminate unnecessary work and achieve results so that the developed technologies can be applied in the decommissioning process.	(1) Considering the target for the fuel debris retrieval from Units 1-3, it is important to promote efficient R&D of technologies and systems applied commonly for all plants. In specific, considering the operation and maintenance of equipment and devices to be required over a long-term, use of common parts and interface and modularization will be important as well. (2) Also, it is necessary to establish the procedures of performance evaluation for development and demonstration of equipment and devices, to clarify the required mock-up test, possible training model for operators and concept for on-site demonstration and to carry them out properly.

(4) Approaches contributing to the establishment of the standards

Policy	Additional issues to be focused on
In order to carry out proper and steady decommissioning of the damaged reactors, it is important to ensure the safety and reliability of the newly developed technologies and to establish the standards necessary for actual application of the technologies in a timely manner. To realize this, clarify the concept on the standards which will be necessary in the future decommissioning process of the damaged reactors, and carry out necessary activities for technical R&D that will serve the establishment of new standards.	(1) The standards for maintenance and operation of equipment and facilities and for ensuring the work safety at the Fukushima Daiichi NPS, which are the Specified Nuclear Facilities, must be established by the site licenser, TEPCO, and it is necessary to carry out the operation and works in accordance with those standards. It is important to develop equipment and devices and the safety assessment method that will contribute to the establishment and application of these standards. (2) Methods of confirmation and endorsement by a third party should be considered for development of the safety assessment method.

2 Approaches focusing on ensuring the safety

(1) Prevention of high risk event

Policy	Additional issues to be focused on
In order to prevent risks such as recriticality, leakage of highly contaminated water, re-dispersion of radioactive materials in the actual decommissioning and contaminated water management, assess these risks appropriately in the R&D planning to minimize them.	(1) At the time of establishment of R&D issues and objectives, it should be arranged so that it will not lead to an event of high risk and priorities should be given to studies on activities which will contribute to risk reduction.

(2) Risk reduction of exposure to the workers

Policy	Additional issues to be focused on
Establish a plan for R&D so as to reduce exposure risk involved in the operation,	(1) Prioritize the measures to reduce risks of exposures, such

considering the safety for the workers for actual decommissioning and contaminated water management as well as for the workers to implement R&D.	as decontamination of radioactive materials and dose reduction and formulate and implement a plan considering the safety of workers as much as possible when conducting on-site demonstration for equipment and devices.

3 Implementation of appropriate management (coordination and control)

	Additional issues to be focused on
Establish a close cooperation among the Japanese and international organizations, such as the decommissioning licenser and R&D implementing organizations, and act as a coordinator in the field of R&D. Determine appropriate division of roles among the decommissioning licenser and R&D implementing organizations including JAEA, and a competitive relationship may be established as necessary.	(1) Effective and efficient methods and measures should be introduced to manage a number of various R&D projects integrally, and an appropriate system should be established as well. (2) For sharing and dissemination of information with the other R&D projects and technical studies on the site construction works by TEPCO, it is especially important to have a documentation system to ensure dissemination of information.

4 Gathering knowledge and experiences from Japan and overseas experts for smooth facilitation of decommissioning work

	Additional issues to be focused on
By collecting information necessary for addressing highly technical issues and collaborating with Japanese and overseas research institutes, incorporate the latest findings and technologies from all over the world including the field other than nuclear engineering and gather knowledge and experiences from wide range of fields.	(1) It is important to conduct R&D by incorporating the technologies, knowledge and experiences which are already utilized around the world and to establish collaborative relationships with related companies, research institutes and experts. Increase the opportunities to share the information to encourage these activities. (2) Especially for the development of equipment and devices, it is important to utilize the technologies with comparatively high TRL (the best available technology) and reliability by the technical investigations and the international RFI/RFP (Request for Information/ Request for Proposal). (3) For acquisition, analysis and evaluation of the data on basic technology, insights from research institutes and universities should be taken into account.

No.2 Other important items in the R&D for the technologies required for proper and steady decommissioning

1 Approaches to securing human resources

	Additional issues to be focused on
To secure the human resources to achieve long term decommissioning work, promote human resource development of researchers and engineers.	(1) It is important to enhance the development and securing of human resource through the promotion of basic research in collaboration with the nuclear industry, research institutes and universities.

2 Creation of an archive and dissemination of information obtained from decommissioning of damaged reactors and research achievements

	Additional issues to be focused on
From the perspectives of utilizing the information and research results in decommissioning process of facilities other than damaged reactors, in responsive measures to the similar accidents occurred in and outside of Japan, as contribution to the investigation of the accident for the enhancement of the safety at nuclear facilities and in the human resource development as well, compile and archive the information and study results obtained from the decommissioning work of damaged reactors in collaboration with the decommissioning licenser and R&D implementing organizations including JAEA, and disseminate them within Japan and overseas appropriately.	(1) In multiple R&D, collection of data and information on the internal condition of reactor vessels, condition of fuel debris, contamination inside the R/B, radioactive waste analysis and inventory evaluation has been started. Continue these measures effectively and establish an integrated system to compile and disseminate the information. (2) Creation of an archive of literature and bibliographic information is currently carried out mainly by JAEA, and it is important to collaborate with this activity in the future.

Table 6-7 Examples of cooperation with overseas organizations

Activities	Contents	Related organizations
OECD/NEA BSAF	・11 countries and 22 organizations have come together to conduct benchmark experiments using extreme accident analysis codes developed by these organizations to find out how the accident in The Fukushima Daiichi NPS progressed and how the fuel debris and fission products spread inside the reactor. Knowledge and findings relating to the modeling of phenomenological issues, obtained by member countries' organizations, are being utilized. ・Data measured during the accident and information database regarding the post-accident radiation levels are shared.	-Institute of Applied Energy (IAE) - NDF -NRA -IRID -TEPCO -JAEA -CRIEPI
OECD/NEA SAREF	・Based on information obtained from The Fukushima Daiichi NPS and in order to contribute to the promotion of its decommissioning through safety studies, related knowledge and information on ongoing activities are being collected and organized. ・Before actual fuel debris retrieval, preparatory research with international cooperation is under consideration, international cooperation for research related to technologies and methods for characterization of debris properties (using mock debris), assessment and analyses of radiation doses and exposure, etc. are under consideration.	NDF NRA TEPCO
IAEA/DAROD	・Experiences regarding decommissioning management (regulation, technologies, systems and strategies) at the post-accident nuclear facilities are being shared between all the countries.	NDF NRA
OECD/NEA EDFWMD	・Expansion of knowledge for waste management and decommissioning at The Fukushima Daiichi NPS ・Advice to Japan's R&Ds regarding waste in The Fukushima Daiichi NPS	NRA JAEA TEPCO ANRE (METI) NDF IRID
Bilateral government-based programs	・Japan-U.K. joint nuclear research in FY2015 and FY 2016 Theme: Research on fuel debris retrieval Research on environment countermeasures including those against waste ・Japan-France joint nuclear research in FY 2016 Theme: Research on remote controlling technology under severe environment conditions ・Japan-U.S. joint nuclear research in FY 2016 Theme: Research on environment countermeasures including those against radioactive waste	MEXT MEXT MEXT
Cooperation with overseas organizations at the stage of IRID's practical application research	・Test on the melt fracture of RPV penetration pipe (Republic of Korea) ・Development of in-reactor fuel debris detection technology Debris measurement using muon (U.S.) ・Detection of fuel debris properties MCCI (France) and mock debris (Asia) ・Designing and development of fuel debris canisters (U.S., Hungary, IAEA, Russia, and U.K.) ・Development of debris retrieval technology (U.S.) ・Development of criticality control technology (Hungary and Russia) ・Development of a technology for investigating inside RPV (U.K.)	IRID IAE IRID IRID IRID IRID IRID IRID
International Special Advisor	Information exchanges are made to obtain appropriate support and advice from experts from the U.S., U.K., and France for the efficient decommissioning of the Fukushima Daiichi NPS.	NDF
International Advisor Meeting	Meetings for obtaining advice from experts from the U.S., U.K., and Spain regarding overall IRID management – advice based on knowledge and experiments in those countries	IRID
International Experts group	To obtain proper supports and advices for effective decommissioning of the Fukushima Daiichi NPS, information exchanges were held with experts from U.K., U.S., France and Ukraine.	TEPCO

Table 6-8 Practical activities and action plan for decommissioning-related R&D

Basic policies	Activity orientation
1. To understand and share the contents of ongoing R&D activities by organizations and institutes under various systems and recognize differences in the characteristics of such organizations and institutes and differences in the characteristics of R&D (differences in objectives, methodologies, periods, etc.)	(1) Interactive information dissemination and sharing regarding R&D needs and seeds and establishment of related infrastructure ・Sharing of R&D needs ・Sharing of R&D seeds ・Establishment of infrastructure for integrated information sharing
2. To continuously carry out R&D activities in various fields by transmitting information such as site situation, needs and seeds and by ensuring cooperation and coordination between the decommissioning site and research sites. To this end, an integrated coordination function and an open platform function shall be established.	(1) Enhancement of interactivity and expansion of researchers' participation ・Expansion of researchers' participation by utilizing existing facilities ・Matching for strengthening interactive cooperation and coordination ・Cooperation with decommissioning infrastructure research platforms ・Expansion of participation by researchers engaged in various fields ・Establishment of international forums and a mechanism for appropriately evaluating researchers (2) Enhancement of cooperation among research facilities and research sites ・Cooperation among research facilities established and planned by JAEA and their shared use ・Sharing of information on facilities owned by related organizations and institutes
3. To promote activities for development of human resources such as researchers and engineers (education, training, procurement, mobilization, etc.) for the purpose of long-term continuous R&D activities	(1) Reinforcement of efforts for human resource development, procurement and mobilization ・Enhancement of sharing and cooperation in the efforts for human resource development

Table 6-9 Organizations participating in the liaison network of academic societies contributing to revitalization of Fukushima and promotion of decommissioning

Japan Society of Energy and Resources	Society of Chemical Engineers, Japan	Society of Remediation of Radioactive Contamination in Environment
Society of Instrument and Control Engineers	Mining and Materials Processing Institute of Japan	Japanese Geotechnical Society
Japan Society of Civil Engineers	Japan Radioisotope Association	Japan Society of Engineering Geology
Oceanographic Society of Japan	Japan Society of Mechanical Engineers	Institution of Professional Engineers, Japan
Japan Weather Association	Atomic Energy Society of Japan	High Pressure Institute of Japan
Japan Concrete Institute	Japanese Society for Multiphase Flow	Seismological Society of Japan
Japan Association for Earthquake engineering	Japanese Society of Fisheries Science	Ceramic Society of Japan
Japan Electric Association	Japanese Society of Soil Science and Plant Nutrition	Japan Society for Bioscience, Biotechnology and Agrochemistry
Physical Society of Japan	Japan Society of Nuclear and Radiochemical Sciences	Japan Radiation Research Society
Japan Society of Maintenology	Robotics Society of Japan	Japan Society of Corrosion Engineering
Japan Society of Plasma Science and Nuclear Fusion Research	Laser Society of Japan	Japan Health Physics Society

Table 6-10 Development status of research facilities

Facility name	Outline	Start of operation	Location
Naraha Remote Technology Development Center	This facility is used to verify the validity of remote control equipment and has a water leak blockage test area for full-scale tests on equipment and devices for repairing and blocking leakage from PCV bottom, an element test area for the real and full-scale simulation of the work environments in The Fukushima Daiichi NPS buildings (mock stairs and water tanks and motion capture technology are used), and a research activity promotion area used for analyzing and organizing experimental data on remote control equipment repairs and modifications.	September 2015 (partially) April 2016 (fully)	Naraha-machi, Fukushima Prefecture
Okuma Analysis and Research Center	A facility for analyzing radioactive materials the development of which is ongoing in Okuma-machi consists of No. 1 building that mainly handles low radiation-level samples such as rubble samples, No. 2 building that handles high radiation-level samples such as fuel debris samples, and administrative building with offices rooms. No. 1 building has equipment such as iron cell, glove box and fume hood and the installation of analyzing devices such as radioactivity measurement device and inductively-coupled plasma mass spectrometer is scheduled. No. 2 building has equipment such as concrete cell, iron cell and glove box and the installation of analyzing devices such as radiation measurement device, mass spectroscope, scanning electron microscope, and electron beam microanalyzer is scheduled.	1st term: within FY 2017 2nd term:in 2021	Okuma-machi, Fukushima Prefecture
Collaborative Laboratories for Advanced Decommissioning Science (CLADS)	In addition to existing facilities owned by JAEA in Tokai and Oarai regions in Ibaraki prefecture, JAEA will make use of a mock test facility in Naraha-machi of Hamadori region in Fukushima prefecture and a radioactive material analysis and research facility and others, and develop CLADS as a hub (utilizing planned international joint research buildings) for inviting researchers from domestic and overseas universities, research institutes and companies from various fields.	In FY 2016	Tomioka-machi, Fukushima Prefecture
Fukushima Environmental Creation Center	Established by Fukushima Prefecture, this center is a comprehensive hub aimed at environment recovery and creation, in which monitoring, researches, information gathering and dissemination, education, trainings and association are provided under cooperation between JAEA and NIES.	Environmental Creation Center-Research building: FY 2016- Communication building: FY 2016 Environmental Radiation Center : November 2015 Inawashiro Water Environment Center : April 2016 Wildlife Cohabitation Center : April 2016	Miharu-machi, Fukushima Prefecture Minami Soma City Inawashiro-machi Ohtama-mura

Table 6-11. MEXT

Human Resource Development and Research Program for Decommissioning of the Fukushima Daiichi NPS - (Projects adopted in FYs 2014 and 2015)

Core institutes	Tasks
Tasks adopted in FY 2014	
Tohoku University	Basic research and core human development program on maintaining the reliability of plant facilities including PCV and buildings, and treatment and disposal of nuclear fuel debris for the accidental nuclear power plant decommissioning
University of Tokyo	HRD for The Fukushima Daiichi Decommissioning based on Robotics and Nuclide Analysis
Tokyo Institute of Technology	Advanced Research and Education Program for Nuclear Decommissioning (ARED)
Tasks adopted in FY 2015	
University of Fukui	Research and human resource development for analysis of fuel debris and decommissioning technology of the Fukushima Daiichi NPS
National Institute of Technology, Fukushima College	Human resources training on the decommissioning of nuclear power plant, based on study for graduation --- interdisciplinary challenge from Fukushima
Fukushima University	Development of analytical specialist by multi-phases educational system and the rapid measurement system for hard-to-measure nuclides in the decommissioning support techniques
Japanese Geotechnical Society	Geotechnical perspectives towards the solution for Fukushima No. 1 Nuclear Power Plant

Table 6-12 MEXT

Examples of new efforts for human resource development

Core institutes	Efforts
National Institute of Technology, Fukushima College	Creative robot contest for decommissioning to raise students' awareness and interest in decommissioning
University of Fukui	Promoting a human resource development aimed at decommissioning, based on synergy effects gained from seminars and practical trainings in the *Nuclear Safety Engineering Course* and the *Decommissioning-Related Research and Human Resource Development Program*, which were newly introduced in FY 2016 April
Fukushima University	Promoting a cooperation with TEPCO through advancement of strontium 90 analysing device
Seven institutions adopted for the *Human Resource Development and Research Program for Decommissioning of the Fukushima Daiichi NPS*	Conference for R&D Initiative on Nuclear Decommissioning Technology by the Next Generation (NDEC), as an opportunity of research presentations, association and friendly competitions among graduates, undergraduates, and technical college students studying decommissioning of the Fukushima Daiichi NPS

Table 6-13　Various activities for maintaining and expanding human resources in the entire atomic industry

Activities	Contents
Activities through the network for development of human resources in nuclear power field	・Nuclear education seminars (16 domestic universities are cooperatively carrying out nuclear education at home and abroad) ・Observation tours in nuclear facilities for promising people ・Nuclear industry seminars (to support job hunting and employment)
Activities through Atomic Energy Society of Japan	・ Participation to workshops for young researchers and ・ Supporting activities planned by the society such as facility observation tours and students' presentations
PA and education support activities outside companies	Sending lecturers and instructors to universities, support for activities of various institutes (e.g., delivery lectures, robot exhibitions, overseas education support for students, hands-on trainings utilizing projects aided by MEXT,and internship) ・ Endowed courses

7. Future Actions

The risk reduction strategy is to be reviewed considering the changes of the situation due to the progress of the decommissioning work and internal PCV condition analysis. Also, the countermeasures against various types of risks including the project risk are studied to steadily advance the decommissioning.

The milestones indicated in the Roadmap, such as "Determination of fuel debris retrieval policies for each unit" and "Establishment of basic concept of processing/disposal for solid radioactive wastes" will be reached respectively in FY2017.Therefore, the next one year will be an important period for the fuel debris retrieval and waste management.

Based on the investigation and study results obtained from all the activities performed so far, the results of the R&D for the important technical issues to be obtained in the future are to be reflected while collaborating with related institutions. Also, the "spiral-up" of the strategy is aimed to be achieved by repeating the evaluations and reviews in order to contribute to the determination of fuel debris retrieval policies to be made in summer of 2017.

The studies on the strategy are also to be carried out towards the subsequent determination of the methodology for fuel debris retrieval and the steady progress of the decommissioning work including actual fuel debris retrieval.

With regard to the radioactive waste management, the overview of the basic concept of the processing and disposal that specifies the direction of the resolution of the issues attributable to the features of the wastes at the Fukushima Daiichi NPS is aimed to be incorporated in the Strategic Plan 2017.

In the light of the fact that the decommissioning of the Fukushima Daiichi NPS is a challenge which has never been experienced before, the approaches to the practical application of the R&D results are to be carried out through the improvement of the R&D effectiveness, enhancement of cooperation with relevant organizations, cooperation with the organizations overseas and further utilization of the R&D facilities.

APPENDIX

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APPENDIX

Appendix 1: Framework of the Japanese Government for decommissioning of Fukushima Daiichi NPS

In response to the accident at the TEPCO's Fukushima Daiichi NPS on March 11, 2011, the Japanese government established a Nuclear Emergency Response Headquarters according to the Act on Special Measures concerning Nuclear Emergency Preparedness (Act #156 of 1999). The headquarters was aimed to press ahead with emergency measures to address the nuclear emergency caused by the accident.

The government established a Council for the Decommissioning of Fukushima Daiichi NPS so that under the leadership of the Nuclear Emergency Response Headquarters, it would focus all its energies on fundamentally solving the issues of contaminated water and the decommissioning of the Fukushima Daiichi NPS, without leaving these issues to TEPCO. The council is deliberating and deciding on important matters associated with the mid- and long-term road map to the decommissioning of the power station.

The framework of the Japanese government to decommission the Fukushima Daiichi NPS is shown below.

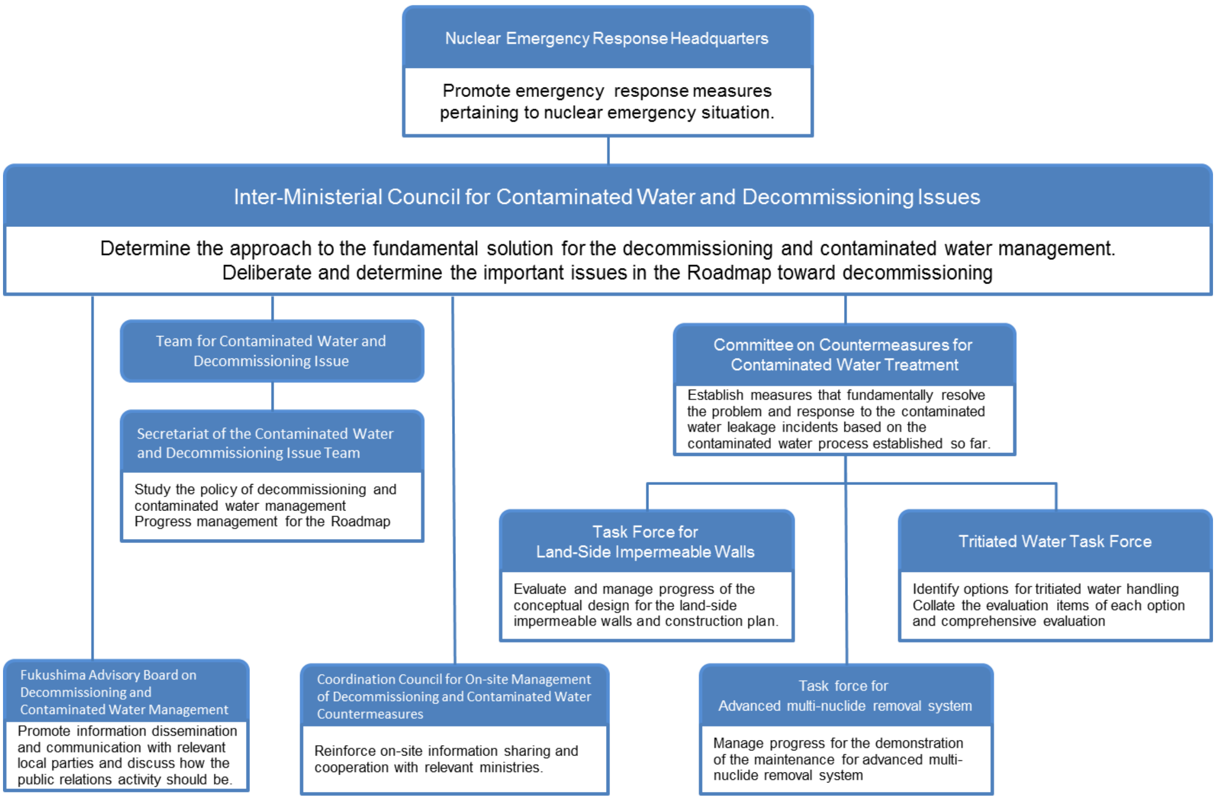


Figure A1-1: Framework of the Japanese Government for decommissioning of Fukushima Daiichi NPS

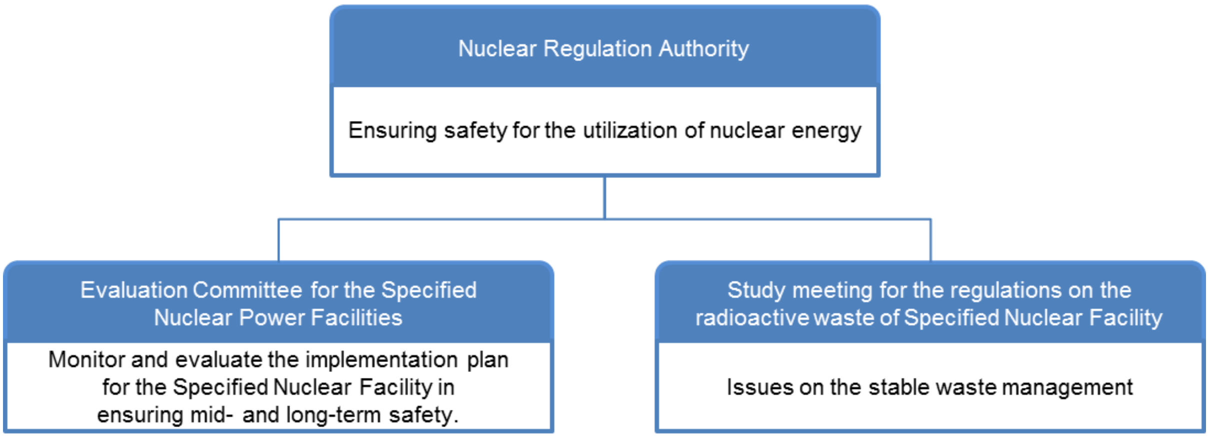


Figure A1-2: Regulatory Framework for the Fukushima Daiichi NPS

Appendix 2: About the Strategic Plan

(1) History of the Roadmap

Table A2-1 shows the history of the Roadmap. The Roadmap is being reviewed based the on-site situation and R&D results.

Table A2-1 History of the Roadmap

Dec 7, 2011 Results of Deliberation to Formulate a Mid- and Long Term Strategy for Cleaning Up the Fukushima Daiichi Nuclear Power Plant	Report issued by Advisory Committee for mid- and long-term measures to TEPCO's Fukushima Daiichi NPS. The first document to study Fukushima Daiichi and explains its mid-and long-term plan.
Dec. 21, 2011 The Roadmap (The 1st edition)	The Government and TEPCO's Mid-to-Long Term Countermeasure Meeting developed the first edition of the Roadmap on December 21, 2011 in response to achievement of the Step 2 objective*[1] of "the Roadmap towards Restoration from the Accident at Fukushima Daiichi NPS." This Roadmap was developed by TEPCO, Agency for Natural Resources and Energy, and the Nuclear and Industrial Safety Agency based on the report of the Advisory Committee. *1The objective of Step 2 is "release of radioactive material has been controlled and the radiation dose has been significantly reduced."
Jul. 30, 2012 The Roadmap (The 1st revised edition)	The first revised edition of the Roadmap was issued on July 30, 2012. It reflects the progress of the actions taken and the detailed mid-and long-term plan on the priority issues for improvement of reliability (hereinafter referred to as "Reliability Improvement Program"). The Reliability Improvement Program was set out by TEPCO after completion of Step 2.
Jun. 27, 2013 The Roadmap (The 2nd revised edition)	The Council for the Decommissioning of TEPCO's Fukushima Daiichi NPS was established in the Nuclear Emergency Response Headquarters on February 8, 2013. In order to accelerate the decommissioning project, it was decided that the progress of field work and R&D will be managed in an integrated manner by the Japanese government and TEPCO as well as with the participation of relevant organizations. This Council formulated the second revised edition of the Roadmap on June 27, 2013.
Jun. 12, 2015 The Roadmap (The 3rd revised edition)	Appropriate measures on risk reduction to protect people and environment from the risks caused by radioactive materials were decided to be continued by categorizing the risks, giving priority to them. This Council formulated the second revised edition of the Roadmap on June 12, 2015 (the third revised edition).

(2) About"the Strategic Plan"

As described in Chapter 2, the Strategic Plan developed by the NDF focuses on considerations from the technical viewpoint according to the role of NDF, namely technical support. It is a total plan that covers not only on-site operations but also necessary research and development activities and technical considerations associated with site work.

The plan provides not only a "strategy" but also concrete policies and requirements necessary to carry forward the strategy, review on available resources, indicating the plans on the on-site work and R&D activities required.

Figure A2-1: Strategy, Policy, and Plan Included in the Strategic Plan

Appendix 3.1: Overview of SED score

SED score, which is referenced for the risk analysis in the Strategic Plan is explained as follows. The NDA uses this index as one of the indexes for prioritizing the many facilities located at the 17 sites that the NDA owns[1].

SED score is expressed in following formula. The first term and second term represent Hazard Potential of the risk source and Safety Management, respectively. Factors are explained as follows. CHP is a hazard potential of chemical materials but no explanation is provided since it is not used in the Strategic Plan.

$$SED = (RHP + CHP) \times (FD \times WUD)^4$$

Radiological Hazard Potential (RHP) is an index that represents hazard potential of radioactive material. The following formula expresses the impact that given to the public when all amount of radioactive materials is released.

$$RHP = \frac{Inventory \times Form\ Factor}{Control\ Factor}$$

Inventory is represented as the product of radioactivity of risk sources and specific toxic potential (STP), as shown in the following formula, and this corresponds to effective dose[2]. STP is the amount of water needed to dilute the radioactive materials of 1TBq so that the ingestion of such diluted water throughout the year would not exceed the radiation dose of 1mSv. It corresponds to the dose coefficient. SED score conservatively uses either ingestion or inhalation as the dose coefficient, which is larger.

$$Inventory(m^3) = Radioactivity(TBq) \times Specific\ \ Toxic\ \ Potential\ (m^3/TBq)$$

Form Factor (FF) is a factor that represents how much radioactive materials will specifically be released depending on the properties of gas, liquid and solid and is shown in Table A3.1-1. A hundred percent of gas and liquid and ten percent of powder will be released when a containment function is completely lost according to the measured data. There is no clear basis for solids, and the small scores are set to express it is unlikely to be released.

Control Factor (CF) takes into account pyrogenicity, corrosivity, combustibility, likelihood of hydrogen generation, reactivity with air and/or water, criticality, and others as the characteristics of risk sources. They are the factors that indicate the time allowable until recovery in the event of a loss of safety function for maintaining the current stable condition. CFs are shown in Table A3.1-2.

Facility Descriptor (FD) is a factor that represents whether containment function of facility is enough or not. It prioritizes the risk source based on the combinations of elements such as integrity of facility, multiplicity of containment functions, safety measures situation. As shown in Table A3.1-3, dividing the state of facility into ten categories, scores are set for each category.

Waste Uncertainty Descriptor (WUD) is a factor that represents how an impact is caused when the retrieval of risk source is delayed. It prioritizes the risk source based on the combinations of elements such as

1 NDA Prioritization - Calculation of Safety and Environmental Detriment score, EPGR02 Rev.6, April 2011.
2 Instruction for the calculation of the Radiological Hazard Potential, EGPR02-WI01 Rev.3, March 2010.

degradation of risk source, activation level, and state of packaging and monitoring. As shown in Table A3.1-4, dividing the state of risk sources into ten categories, scores are set for each category.

Table A3.1-1 Form Factor (FF)

Properties	Score
Gas, liquid	1
Sludge, powder	0.1
Discrete solid	0.00001
Large monolithic solid, activated component	0.000001

Table A3.1-2 Control Factor (CF)

Category	(In hours)	Score
hour	1 h	1
day	24 h	10
week	168 h	100
month	730 h	1,000
years	8,760h	10,000
10 years	87,600 h	100,000

Table A3.1-3 Facility Descriptor (FD)

Category	Definition	Score
1	Design life of the building was expired. Single containment with critical defect. Insufficient arrangement in the event of abnormality and the latest design basis is not satisfied.	100
2	Same as Category 1 but without critical defect.	91
3	Same as Category 2 but with sufficient arrangement in the event of abnormality.	74
4	Same as Category 3 but with double containment.	52
5	Same as Category 4 Design life is not exceeded but will be expired at the time of retrieval.	29
6	Same as Category 5 but design life will not be expired at the time of retrieval.	15
7	Same as Category 6 The latest design basis is satisfied but implementation of safety case is limited.	8
8	Same as Category 7 Safety case is fully implemented but has impact from adjacent buildings.	5
9	Same as Category 8 but affects adjacent buildings.	3
10	Same as Category 9 but not have impact from adjacent buildings and not affect adjacent buildings.	2

Table A3.1-4 Waste Uncertainty Descriptor (WUD)

Category	Definition	Score
1	Unpackaged radioactive waste that degrades*. Not under monitoring and management.	100
2	Same as Category 1 but packaged.	90
3	Reactive** unpackaged radioactive waste. Unclear presence, quantity and position. Not realistic to be verified.	74
4	Same as Category 3 but capable of being confirmed by sampling.	50
5	Same as Category 1 but under monitoring and management.	30
6	Same as Category 2 but under monitoring and management.	17
7	Not reactive but unpackaged radioactive waste that degrades. Not under monitoring and management.	9
8	Same as Category 7 but packaged.	5
9	Not reactive and nor degraded unpackaged radioactive waste. Under monitoring and management.	3
10	Same as Category 9 but packaged.	2
* Degrade：a nature, which will cause possibilities of change of the retrieval method, increase in radiation exposure dose, and occurrence of criticality by dissociation and dispersion. ** Reactive：a nature that causes rapid changes, such as generation of heat and explosion.		

Appendix 3.2: Selection of nuclides for evaluation

With consideration given to the quantitative changes during the evaluation periods, nuclides were selected that have significant effects on people.

Figures A3.2-1 and A3.2-2 show the effective (ingestion) doses of the heavy nuclei and FPs during the 100 years after the accident occurrence, including a few decades before the nuclear reactor is completely decommissioned, for the reactor core and the fuels in SFP of Unit 2 (source: JAEA-Data/Code 2012-018 "Estimation of fuel compositions in Fukushima-Daiichi Nuclear Power Plant" for radioactivity and ICRP Publication 72 for dose coefficients). Each of the nuclide shown in these charts causes a dose that makes up more than 1% of the total effective dose.

With attention focused on the contribution during the period of a few years to a few decades after the accident occurrence, the risk analyses are conducted on heavy nuclei such as Pu-238, Pu-239, Pu-240, Pu-241, Am-241 and Cm-244 and FPs such as Sr-90, Cs-134 and Cs-137.

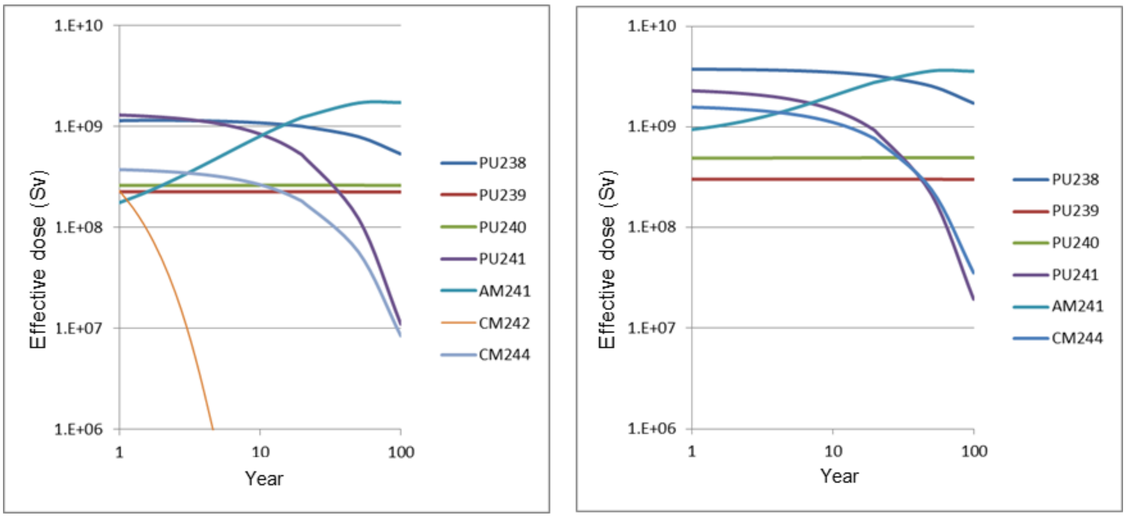


Figure A3.2-1: Effective Doses Caused by Heavy Nuclei
(Left: Reactor Core of Unit 2, Right: Fuels in SFP of Unit 2)

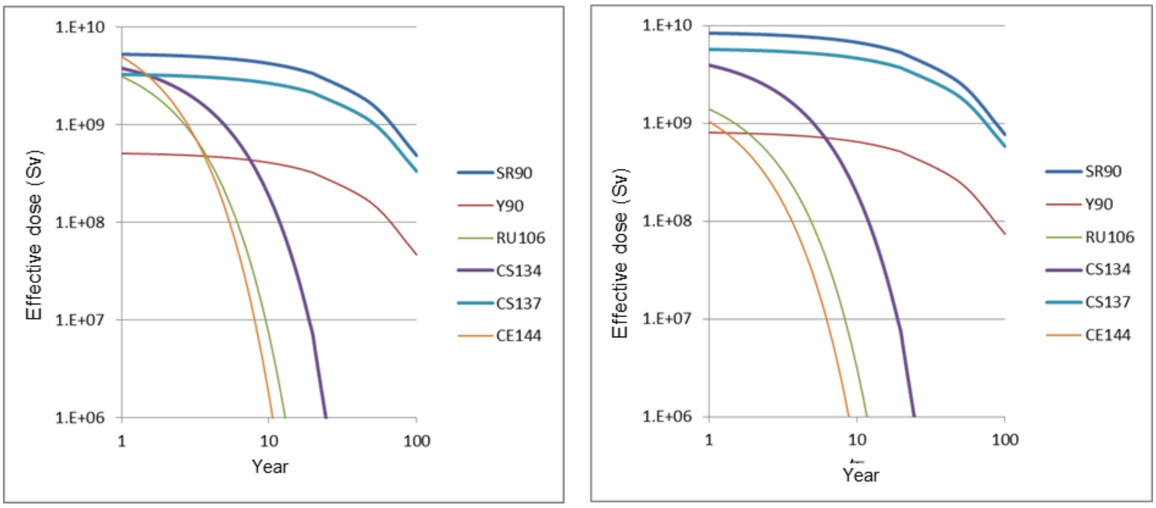


Figure A3.2-2: Effective Doses Caused by FPs
(Left: Reactor Core of Unit 2, Right: Fuels in SFP of Unit 2)

Appendix 3.3: Details of the risk analysis

3.3.1 Hazard Potential

Figure A3.3-1 shows the inventory.

The amount of radioactivity of each nuclide in the fuel debris that derived from the fuels loaded on the reactor core at the time of accident has been evaluated as a function of time since the accident (Reference: "Estimation of fuel compositions in Fukushima-Daiichi Nuclear Power Plant" JAEA-Data/Code 2012-018). All amounts of heavy nuclei are assumed to be remained inside the PCV in the form of fuel debris.

Existing similar data was used for spent fuels. Estimations for fuels in the common pool and dry casks were based on the data on the pooled fuels in SFPs. For highly volatile Cs, an assumption was made that 5% of Cs had been released into the cladding tube based on the conservative settings for evaluation of power-reactor accidents (source: NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants"). On the other hand, no Sr is assumed to be released. Form factors of Cs and Sr are varied by being considered solid material in the fuel debris or released in gas or powder form.

For the contaminated water, the secondary wastes and radioactive solid wastes estimations were based on the data found on the websites of the Ministry of Economy, Trade and Industry and TEPCO.

The contamination on the surfaces of internal structures and radiated materials were estimated based on the data on a boiling water reactor with a power output of 1,100 MW from Handbook of Evaluation on the Impact of Decommissioning on the Environment (3rd edition), with the power generating capacity taken into consideration. The surface contamination includes the amount of the Cs released from the fuel at the time of the accident.

The uncertainty was determined with consideration given to the uncertainty estimated from the information obtained and the variation between measurements. Since the data obtained for the radioactive solid waste including various types of concentrations of radioactive material is limited, a large uncertainty is set.

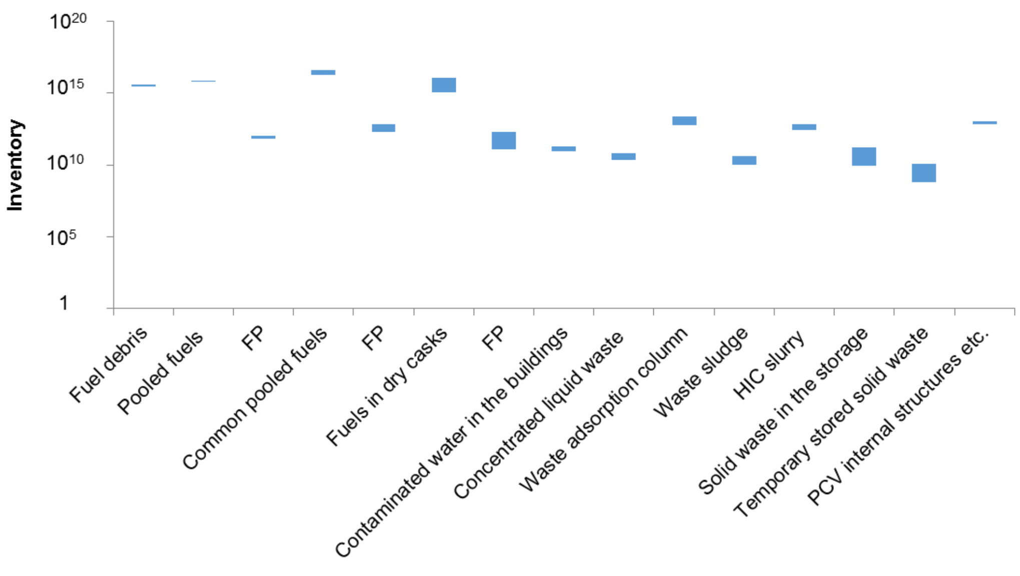


Figure A3.3-1 Inventory

Form Factor (FF) is shown in Figure A3.3-2.

The fuel debris was regarded as discrete solids although it may include large monolithic solids. The spent fuel is discrete solids. However, the Cs released from it was regarded as powder. The contaminated water, secondary wastes from the water treatment, and radioactive solid wastes were regarded as liquid, sludge, and powder, respectively. In the PCV internal structures, the radiated materials were considered as activated components and the surface contaminants as powder or stuck material on the surface.

The form factors are categorized into four groups. This categorization, however, does not always appropriately represent the characteristics of the various risk sources in the Fukushima Daiichi NPS. Each risk source was categorized as a group that seemed to be most appropriate; depending on the risk source, a neighboring FF was assigned as uncertainty. In specific, the uncertainties in the properties are considered for the fuel debris and uncertainties in the degree of fixation of FPs are considered for PCV internal structures.

Figure A3.3-2 shows Control Factor (CF).

As for fuel debris, time margin is assumed for suspension of nitrogen injection and cooling shutdown referencing the implementation plan. As for re-criticality, stable sub-criticality state is being maintained without any specific arrangement.

The time margin for cooling shutdown was estimated for the pooled fuels in SFPs and fuels in the common pool referencing such as the implementation plan. The fuel in dry cask is not required to be cooled.

No special treatment, such as cooling, is required for contaminated water, radioactive solid wastes, and PCV internal structures. However, when fuel debris is no longer cooled and hence becomes hot, FPs adhering to the surface such as of the PCV internal structures may evaporate. For this reason, it was taken into consideration as a time allowance for the PCV internal structures.

The secondary wastes from the water treatment are so designed to be retained for longer than 10 years by natural cooling and ventilation. The HICs, however, underwent drip of water overflown by the hydrogen generated by radiolysis of water, and hence limiting the storage quantity, conducting the drainage, and monitoring the impact of hydrogen generation that are currently performed were considered. Waste sludge is stirred to prevent fixation and the time allowance was estimated assuming a halt of this stirring.

The uncertainty was determined as one order of magnitude, which is the difference between neighboring scores. No uncertainty is set for the risk source that does not require a specific arrangement such as cooling.

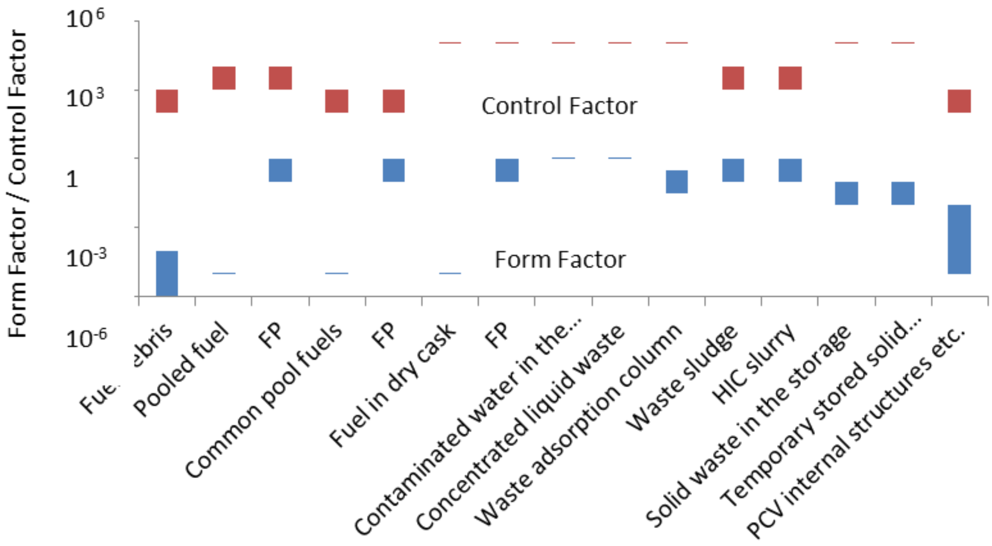


Figure A3.3-2 Form Factor and Control Factor

3.3.2. Safety Management

Table A3.3-1 shows the overview of containment functions, safety equipment, management and monitoring state for each risk source. Based on this, relative comparison was conducted, and risk source was graded into 10 categories and the scores are set for modified FD and modified WUD.

The uncertainty is assumed as the score of neighboring category. Since the score is a straight line on the logarithmic scale, uncertainties will become nearly constant regardless of the category. A various types of storage form were considered as uncertainties in the tentatively stored solid wastes.

Table A3.3-1 Features of Risk Sources related to Safety Management

Risk sources	Features
Fuel debris	No significant damages were found on PCVs, and criticality control, cooling and prevention of hydrogen explosion are implemented in multiple ways. Also, important parameter including Xe concentration, temperature, and hydrogen concentration are being monitored.
Spent fuel	SFP of each Unit is designed to maintain sub-criticality and equipped with redundant cooling system. The fallen rubbles and heavy objects, damaged ceiling of the building, and seawater injection have been occurred in some Unit. No damage on the common pool and dry casks as well as the buildings by the earthquake or tsunami.
Contaminated water	The contaminated water inside the buildings and trenches is kept confined by balancing the water level with that of ground water. Concentrated liquid waste was made from condensing concentrated salt water by evaporative concentration apparatus and its concentrations of radioactive material and salt are high. It is stored in the welded type tank and the tank is placed in the dike.
Secondary waste generated from the water treatment	A waste adsorption column contains zeolite—a substance that adsorbs Cs housed in a carbon-steel shielding container. They are placed on the box calvert or storage racks. No management such as decay heat removal is required. Waste sludge is stored in agglomeration pits in the main process building, and leakage monitoring, decay heat removal and hydrogen discharging are carried out. An HIC slurry is contained in a polyethylene container, which are housed in a reinforced SUS structure and stored in box calvert. Since slurry contains water, hydrogen is released into atmosphere from the vent. Although decay heat removal is not necessary but monitoring is being carried out continually, since water drip occurred due to the hydrogen generation.
Radioactive solid waste	As for stored radioactive solid waste, the highly radioactive rubbles are collected in the containers and stored in the radioactive solid waste storage building. No special management is required. Tentatively stored solid waste is the waste with various concentration levels of radioactive materials that are stored outside in various forms. It requires monitoring.

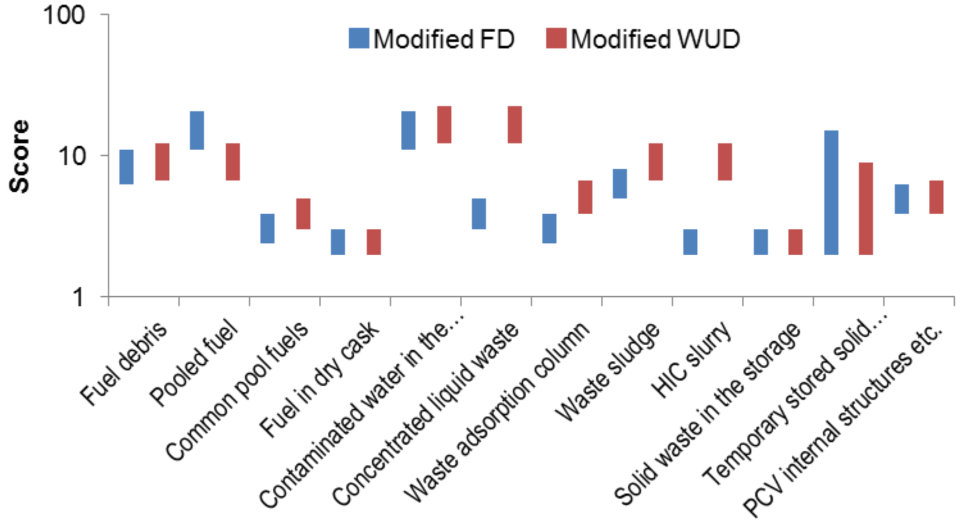
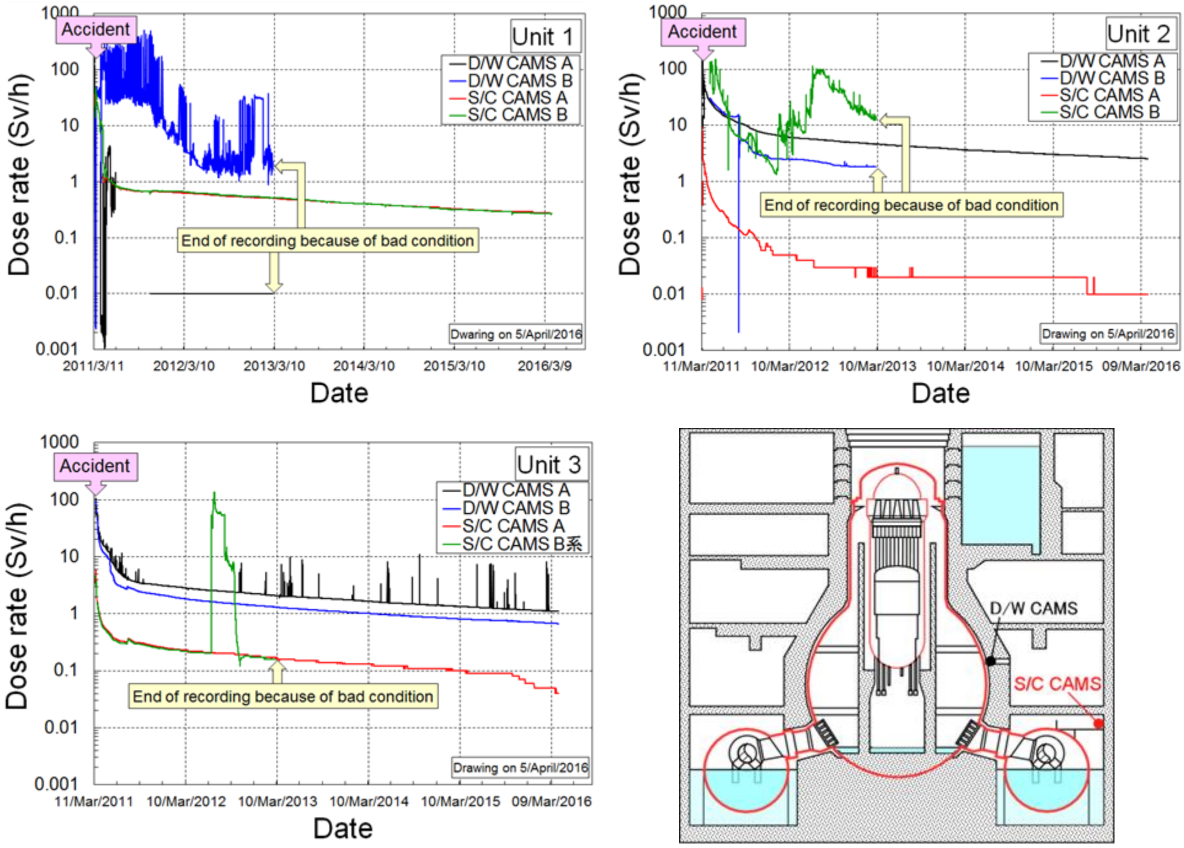


Figure A3.3-3 Factors of Safety Management

Appendix 4.1: Radiological environment inside the PCVs

The following charts show temporal changes in the radiation dose rate inside the PCVs since the accident occurrence. The radiation dose inside the PCV of each unit is still high even though it is lower than that right after the accident.



[Based on data published by TEPCO]

Figure A4.1-1: Radiation Dose Rates inside the PCVs at the Fukushima Daiichi NPS

Appendix 4.2: Periodic measurement of plant data

From plant data on the temperature, hydrogen concentration, pressure, and other items inside the PCVs that have been continuously obtained since the accident occurrence, it is estimated that Units 1-3 in the plant are kept in a stable cold shutdown condition.

(1) Measurement of Basic Plant Data inside the PCVs

Here is a summary of the temporal changes in the plant data about the conditions inside the PCVs, which reveal that the fuel debris is in a stable state.

a. Temperature

At the time of accident, a power outage occurred, disabling decay heat from fuel assemblies to be immediately removed and thus causing the temperature to rise to higher than 1,000°C. Then, a rapid oxidation reaction occurred between the fuel cladding tube (zircaloy) and steam ($Zr + 2H_2O \rightarrow ZrO_2 + 2H_2 + 586$ kJ/mol), which generated additional heat. This situation caused the fuel melting.

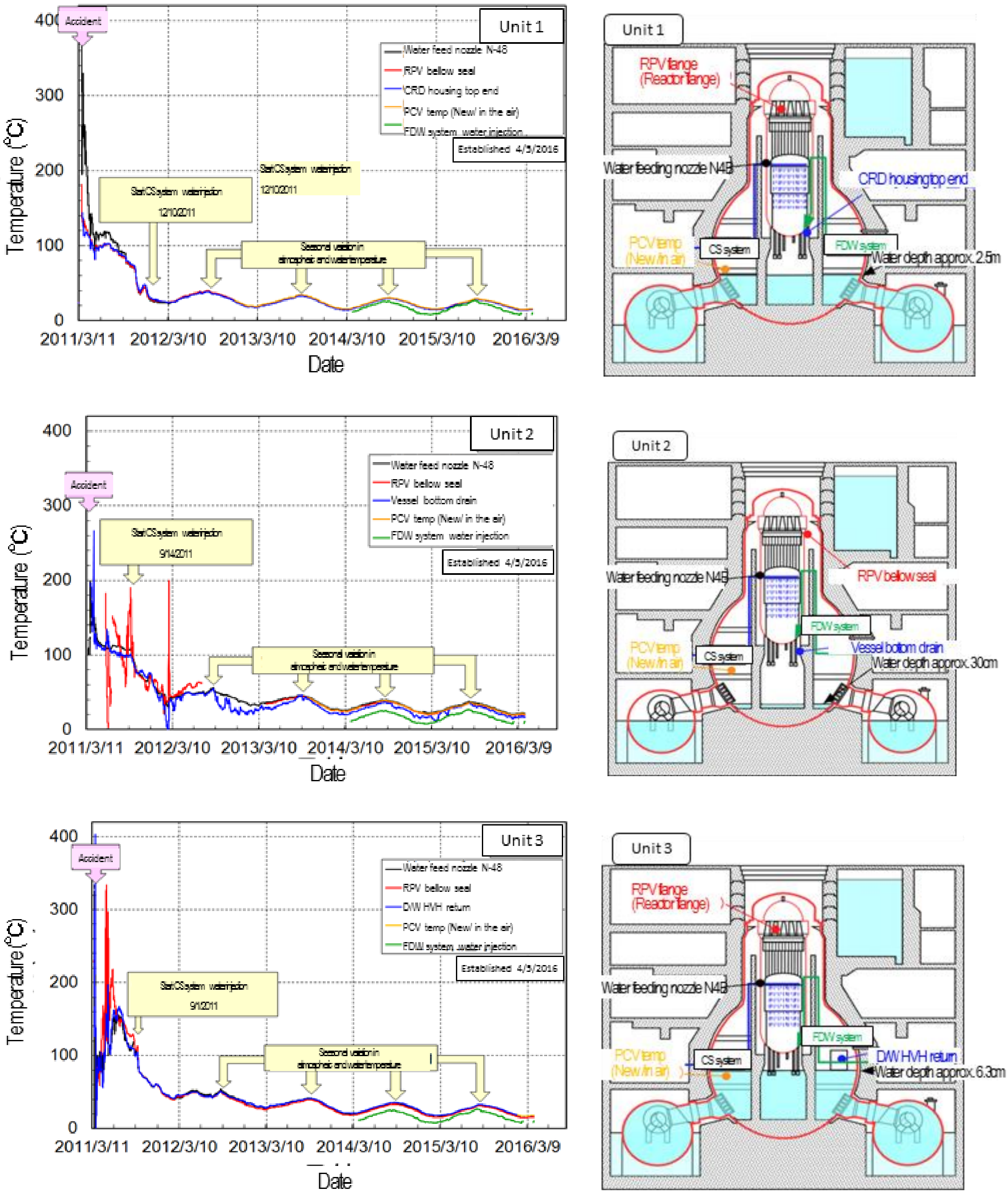
Figure A4.2-1 shows the temporal changes in the temperature data around the reactors summarized based on the information published by TEPCO. After the accident, the temperature inside the PCVs started dropping and decreased to 100°C in six months. After that, the temperature has been gradually dropping every year while following the seasonal variations in air and water temperatures. The temperature is staying at a level lower than 50°C at each section inside the PCVs without showing a sharp peak. Since zircaloy does not react with water at low temperatures, it is estimated that no oxidization reaction occurs and thus no additional heat is generated.

Figure A4.2-2 shows the (decay) heat from the elements that make up the fuel assemblies loaded at the time of accident. Immediately after the halt of the nuclear reactors, short-lived nuclides generated much decay heat; in five years, the amount of the heat generated decreased to lower than one thousandth of the heat at the time of the accident. Now, only long-lived nuclides are surviving. Since they have long half-lives and therefore decay slowly, decay heat is also expected to gradually decrease. It is estimated that the temperature will further drop in the future over time.

b. Hydrogen Concentration and PCV Pressure

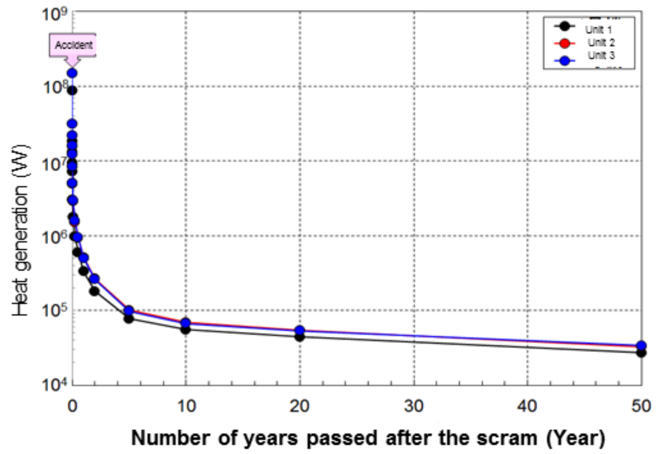
Irradiating water with gamma rays causes hydrogen to be generated by radiolysis. The PCVs are filled with water to cool the fuel debris inside them. In addition, the doses inside them are high as Appendix 4.1 shows. With these factors, there is a fear that hydrogen may be generated inside them. Based on the fact that hydrogen has a lower combustible limit of as low as 4%, the PCVs have been filled with nitrogen since the accident occurrence to dilute hydrogen to prevent a hydrogen explosion. Figures A4.2-3 and A4.2-4 show changes in hydrogen concentration and pressure inside the PCVs, respectively. The hydrogen concentrations are low enough, indicating that hydrogen has been effectively diluted by the inclusion of nitrogen. From the viewpoint of confining FPs, it may be effective to remove hydrogen out of the PCVs. However, doing so may cause the pressure inside the PCVs to be lower (negative) than the (normal) atmospheric pressure. This may allow air that includes oxygen to enter the PCVs through, for example, sealed sections of them, resulting in mixture of hydrogen and oxygen. For this reason, the pressure inside the PCVs is kept slightly higher (slightly positive) than the atmospheric pressure as Figure A4.2-4 shows.

With these facts, it is estimated that Units 1-3 are kept in a stable cold shutdown condition.



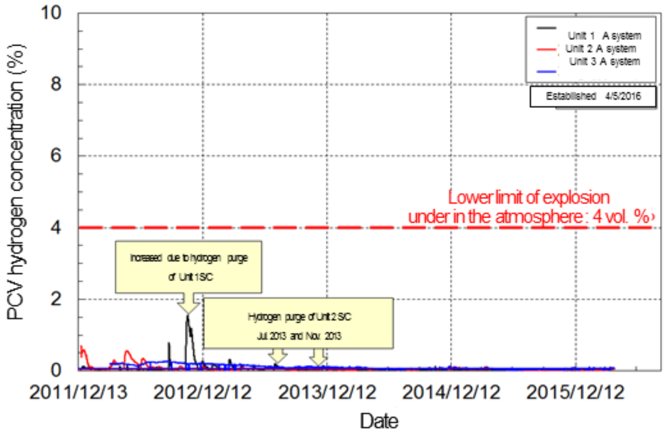
[Based on data published by TEPCO]

Figure A4.2-1: Changes in the Ambient Temperature of the Nuclear Reactors at the Fukushima Daiichi NPS



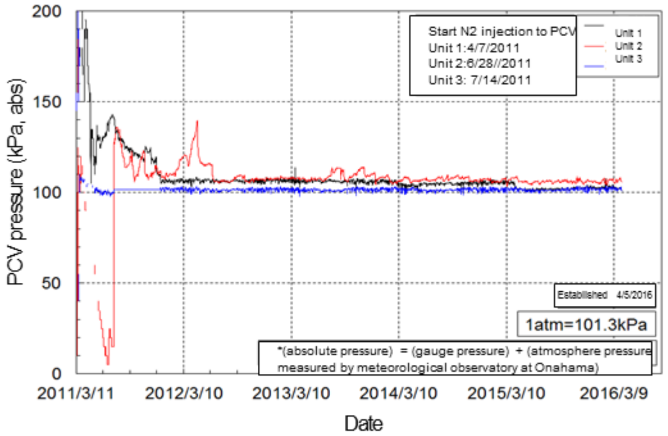
[Based on data published through JAEA-Data/code 2012-018]

Figure A4.2-2: Heat from the Fuel, FPs, and Radiated materials inside the Reactors



[Based on data published by TEPCO]

Figure A4.2-3: Changes in the Hydrogen Concentration inside the PCVs



[Based on data published by the Meteorological Agency and TEPCO]

Figure A4.2-4: Changes in the Pressure inside the PCVs

Appendix 4.3: Information obtained through past PCV internal survey

Here is a summary of the examinations conducted so far on the PCV internal survey.

(1) Unit 1 PCV internal survey

a. Objective: Inserting inspection device from PCV penetration of Unit 1 (X-100B penetration), conduct survey for the "information on the first floor grating in the PCV."

b. Method: In October 2012, the inside of the PCV was examined using a CCD camera and accumulated water was sampled. In April 2015, the outside of the pedestal was examined using a shape deformation robot (B1 examination).

c. Information obtained: The following information was obtained. Figure A4.3-1 shows part of the examination results.

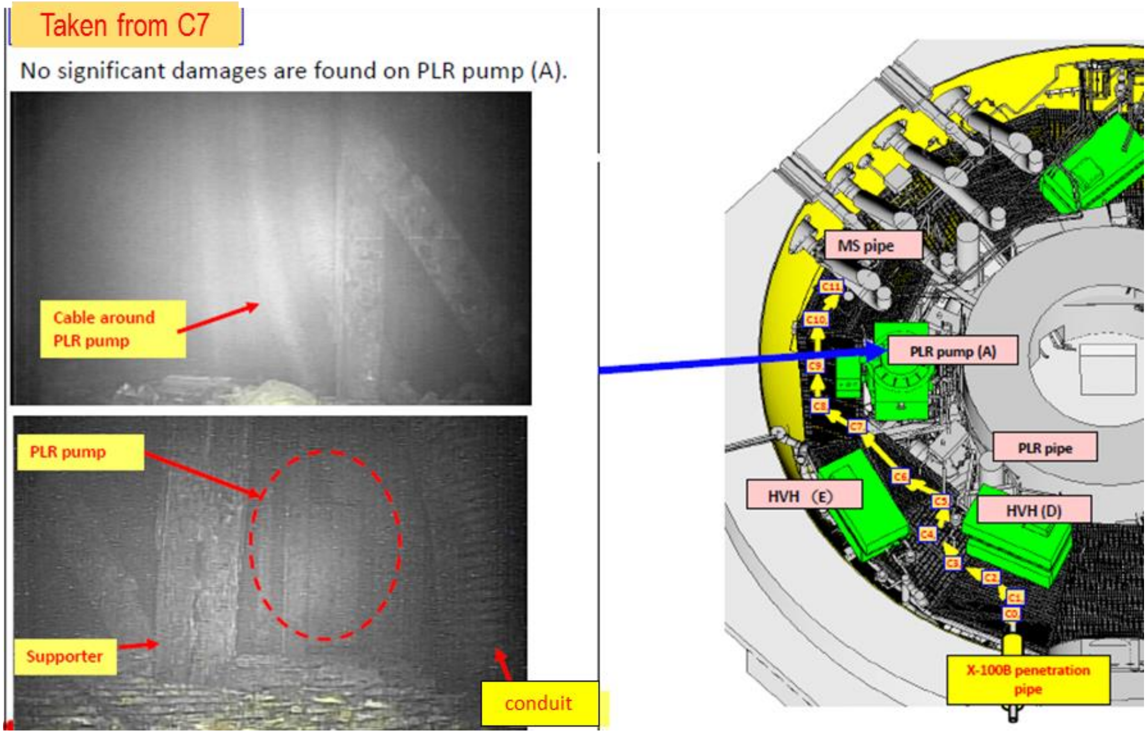
(1) No large damage on the existing facility (e.g. PLR pump, PCV inner wall and HVH)

(2) Dose rate was approximately 10 Sv/h.

(3) PRL piping shielding unit confirmed fallen.

(4) The access route to the bottom of the D/W was confirmed but the deposits are scattered over a wide range.

Since Information (4) above had revealed that measures against deposits would be required when subsequent examinations are made or fuel debris is taken out, the procedure for examinations on the outside of the pedestal at the bottom of the PCV (B2 inspection) was reviewed and as a result, the examinations were postponed to fiscal 2016. No fuel debris was found in the survey this time.



[Source: Results of the field verification testing on the condition above the grating on the first floor of the pedstal from *Development of technology for investigation inside the PCV* from TEPCO]

Figure A4.3-1: Results of Examinations on the outside of the Pedestal at the bottom of the PCV (Unit 1 B1 inspection)

d. Considerations: It can be estimated that the temperature at the periphery of the grating on the 1st floor might exceeded 328℃, which is the melting point of lead since PLR piping shielding units (lead wool mattress) have fallen.

e. Challenge: Back and forth motion should be used while checking the crawler portions. During the counterclockwise inspections, inspection crawler robot was stuck in the gaps between grating bars in the area between PLR pump and air-conditioning unit.

(2) Unit 2 PCV internal survey

a. Objective: To verify fallen objects on the platform, damage states, and access route to the periphery of the bottom of the PCV using the internal survey robot.

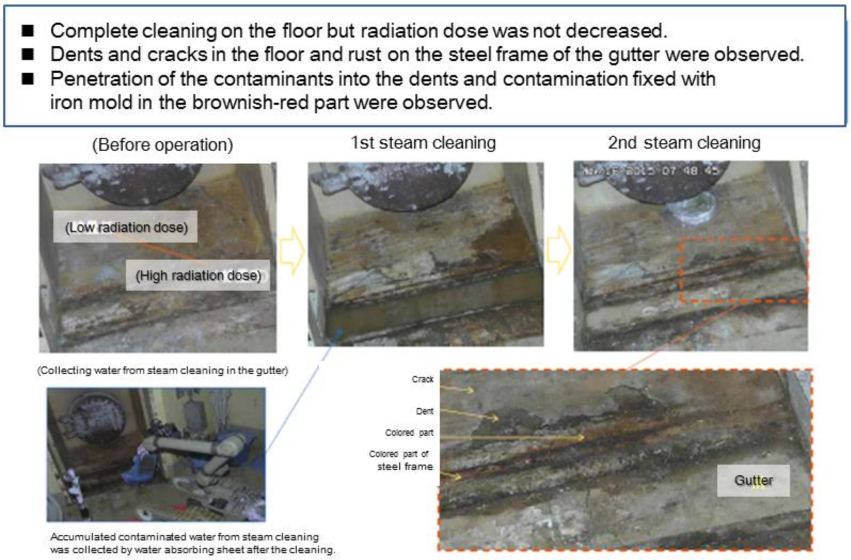
b. Method: In March 2012 and August 2013, dose rate measurements were made, the inside of the PCV was examined using a CCD camera, and accumulated water was sampled through the PCV penetration (X-53 penetration). An Inspection was scheduled in August 2015 based on an internal survey robot through the X-6 penetration.

c. Information obtained: The following information was obtained. Figure A4.3-2 shows some of the preparation processes of the examination.

(1) The dose rates on March, 2012 and August 2013 were 31 and 73 Sv/h respectively. The does rate varies depends on the area and.

(2) Although internal survey for the pedestal inside the PCV (A2 inspection) was planned, eluted materials were confirmed near the CRD hatch (X-6 penetration) and peripheral dose rate exceeded the assumption significantly. Consequently the scheduled examination was postponed to fiscal 2016 in order to take countermeasures to decrease the dose before conducting the examination.

d. Challenge: Timing of the inspection was postponed to FY2016 since the measures are required to reduce the radiation dose around the X-6. The future scope of the PCV repair including some peripheral areas will be required since low temperature history for the X-6 during the progress of the event is assumed, instead of leaching from X-6.



[Source: Resutls of the Reexamination on the Inside of the PCV of Unit 2 from TEPCO]

Figure A4.3-2: Conditions Revealed during Preparations for the PCV internal survey (Unit 2)

(3) Unit 3 PCV internal survey

a. Objective: The inspection were aimed to check, in particular, that the inside of the PCV is properly kept cooled inserting inspection instruments (a camera, thermometer, and dosimeter) from the PCV penetration of Unit 3 as well as to obtain information that would be useful in reviewing the examination method to be used in the future.

b. Method: In October 2015, dose rate measurements were made, the PCV internal survey using CCD camera, and accumulated water was sampled.

c. Information obtained: The following information was obtained. Figure A4.3-3 shows part of the examination results.

(1) No damage was found on the examined sections of structures or walls.

(2) No damage was found on the examined sections of the X-6 penetration and CRD rail.

(3) Deposits were found on the CRD rail and the grating on the first floor.

(4) Sediments were observed on the CRD rail and gratings on the 1st floor. (transparency under the water inside the PCV was fine). .

(5) Water level inside the PCV was OP: approx. 11,800mm. Almost consistent with the estimated value.

(6) The maximum radiation dose detected in the gas phase inside the PCV was approximately 1Sv/h.

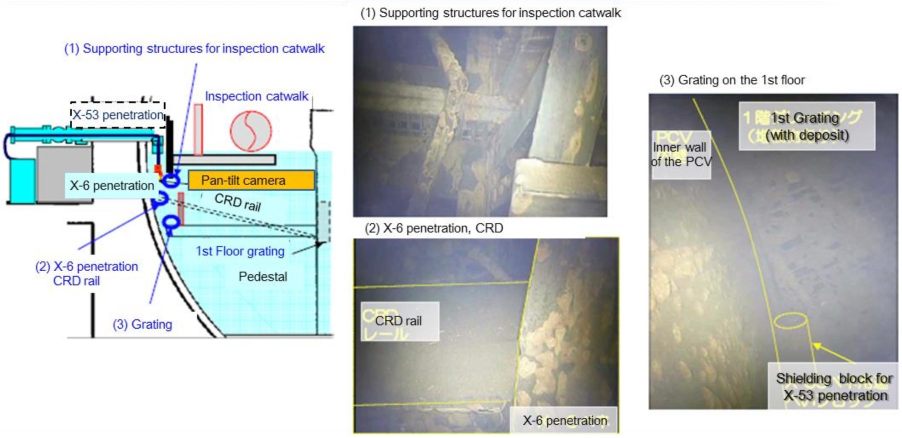
(7) The quality of the accumulated water inside the PCV indicated that the inside of the PCV is not in a severely corrosive environment; rather, it is in a less corrosive environment.

At present, an examination of the inside of the pedestal of the PCV is being considered that is based on such as underwater swimming device put into the PCV through the X-53 penetration.

d. Consideration: The radiation dose inside PCV is the lowest among the Units 1-3. This is considered to be because of shielding due to high stagnant water level.

e. Lesson Learned: Water level coordination or waterproof equipment will be required for PCV internal survey since stagnant water level is high.

- No damage was observed in the structures underwater in the PCV (conduit, supporting structures, X-6 penetration, CRF rail),
- Deposits were observed on the CRD rail and grating of the first floor.

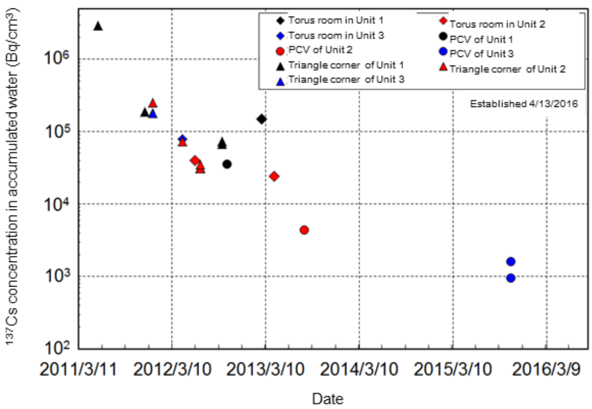


[Source: Results of the PCV internal survey of Unit 3 at the Fukushima Daiichi NPS of TEPCO (Quick Report on the Examinatin Conducted on October 22)]

Figure A4.3-3: Results of (Preliminary) PCV internal survey (Unit 3)

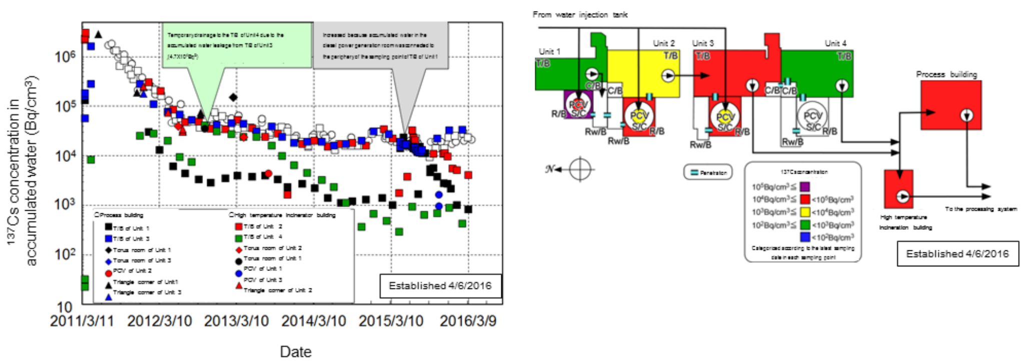
(4) Results of the Sampling of the Accumulated Water

Figure A4.3-4 shows the analysis results of the concentration of Cs-137 in the accumulated water sampled during the PCV internal survey. Since the samples were taken from different units at different timings, it is difficult to determine the trend for a specific unit. However, as a whole, the concentration of Cs-137 is on the decline because water treated with an adsorption system for Cs and Sr multi-nuclide removal system has been injected into the PCV as cooling water. Figure A.4.3-5 shows the analysis results of the concentrations of Cs-137 in the accumulated water sampled in the T/B, building for processes, and high-temperature incinerator building. Although the T/Bs have penetration and affected by the connections of drainage from neighboring facilities, the concentrations in the accumulated water in the T/Bs for Units 1, 2, and 4 are gradually decreasing. For all of Units 1, 2, and 3, the concentrations of Cs-137 in the accumulated water are lower inside the PCVs than in the torus rooms. For each unit, at the time of the accident, the SR valve was in action and the volatile FPs inside the reactor were introduced to the cooling water inside the S/C. In addition, for Units 1 and 3, the volatile FPs were introduced to the cooling water inside the S/C even before ventilation. With these facts, it is assumed that Cs-137 has been released from the S/Cs although no measurement has been made of the accumulated water inside the S/Cs.



[Based on data published by TEPCO]

Figure A4.3-4: Concentrations of Cs-137 in the Sampled Water

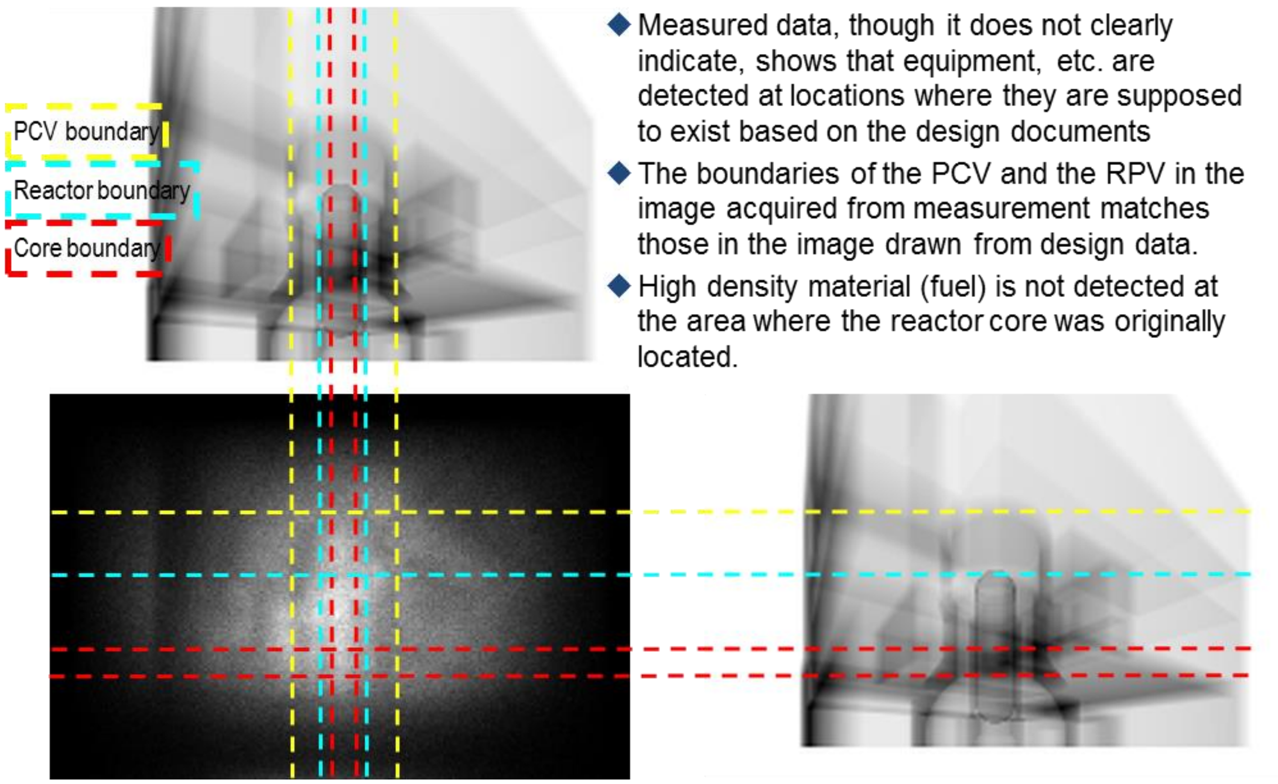


[Based on data published by TEPCO]

Figure A4.3-5: Concentrations of Cs-137 in the Water Sampled in the Turbine Buildings and Water Injection System Diagram

Appendix 4.4: Estimation results of fuel debris location by Muon detection

For Unit 1, fuel debris distribution measurements based on penetration muon detection of transmission method were made in the periods from February to May and from May to September 2015. With the measurement results, it is reasonable to assume that no fuel fragment larger than 1 m, which is maximum muon detection capacity by transmission method or no water exists at the original reactor core region.



[Source: Quick Report on the Measurement Results for Unit 1 from *Development of Technology for Detecting the Fuel Debris Location inside Nuclear Reactors* from TEPCO]

The results obtained by the evaluation by the comparison between analysis value by muon detection and measured value indicate that the survival rate of the fuel in the core region was (9-36)±51%.

*Joint research by TOSHIBA and Nagoya University

Appendix 4.5: Overview of the MAAP and SAMPSON codes

Here is a summary of the characteristics of MAAP and SAMPSON codes, which are severe accident analysis codes.

Code	MAAP code	SAMPSON code
Developed by:	Electric Power Research Institute (EPRI) in the US	Nuclear Power Engineering Corporation (NUPEC) in Japan (development activities are continued by the Institute of Applied Energy (IAE)).
Analysis object	In+Ex Vessel	In+Ex Vessel
User adjustment factor	Many	None (dependent on mesh division)
Computation time	Short	20 to 30 times the actual time
Characteristics	-Capable of obtaining the user-desired results by combining adjustment factors. - Analysis results often depend on the user.	-The analysis results do not depend on the user because the code is based on the theoretical physical model built around mechanical logics.
Verification of individual models and codes	- The code is continuously improved by the user group.	-Presented in the OECD/NEA International Standard Problem (ISP) program and well received. -Verified by many other experimental analyses

Source: Information from the Expert Committee on SA Evaluation Studies, Atomic Energy Society of Japan by Takashi Okamoto

Appendix 4.6: Major improvements to MAAP and SAMPSON codes and their results

Tables A4.6-1 and A4.6-2 show major improvements made to the MAAP and SAMPSON codes and their results.

Table A4.6-1: Improvements to the MAAP Code

Improvements in FY2014 Improvements in FY2015

Purpose	Details of model improvement	Improvement effect
Improvement of core damage progression evaluation	Improvement of radiation model between shroud /RPV wall	-Result showed that the shroud was unlikely to be melted when being contacted by debris due to the shroud heat conduction and two-dimensional radiation between shroud/wall in the RPV.
	Improvement of thermal hydraulic model for primary system	-Improving the evaluation of the reactor water level from the exposure to the damage of the reactor core and response to the pressure, evaluation accuracy for the generated amount of hydrogen after the core damage was improved.
Improvement of the evaluation for the debris distribution/composition inside the RPV	Model improvement for the debris layer inside the lower plenum	-Cooling effect for debris inside the lower plenum became larger due to the modeling of the solidified accumulated state of the debris that falls from the core region over time and additional heat abstraction model from the CRD housing.
	Add heat conduction mode to the CRD housing outside the RPV and radiative model between housings.	
	Improvement of RPV bottom head damage model	-Large scale damage is unlikely to be caused by changing the model that housing will not fall considering the limits of CRD housing support. (Unit 2 PCV internal survey implies the possibility of no major damage was caused to the CRD.)
Improvement of the evaluation of the debris distribution/composition	Add melting model of lower structures when debris are flown out of the RPV.	-Accuracy of estimation for the quantity of the metal that mixed into the debris that fell inside the PCV was improved.
	Add model of debris flowing into the sump pit and erosion	-Became capable of evaluation for concrete erosion quantity considering actual unit shape.
	Add model of debris flowing into pedestal sump pump	-Evaluation became available for the amount of the debris transferred to the drywell side through the pedestal sump piping and accuracy of the estimation of debris distribution inside the PCV was improved.
	Improvement of the heat transfer model at the time of water injection during the core concrete reaction	-Evaluation became possible for the concrete interactions that reflect the latest findings (OECD/MCCI plan).
	Improvement of the model of the core concrete mixture properties	-Evaluation became possible for the solidification of the debris that reflects the latest findings.
Estimation of PCV damage location (temperature distribution evaluation)	Add local flow (stratification) model inside the PCV.	-Utilized for the locations where PCV damage due to the overtemperature was estimated for each Unit.
Improvement of fission product distribution evaluation	Improvement of the deposit model of FP	-Estimation of the amount of FP attached to the PCV vent pipe became available and reference information regarding the decommissioning work was obtained.

Reference: IRID Completion report for "Improvement of recognition regarding the internal PCV condition of the reactor vessels using severe accident progression analysis and actual plant data"

Table A4.6-2: Improvements to the SAMPSON Code

Improvements in FY2014 Improvements in FY2015

Purpose	Details of model improvement	Improvement effect
Improvement of core damage progression evaluation	Detailed modeling of heat release form RPV	-Improving the heat rereleased from the RPV and PCV sides so as to be considered in detail, temperature of the inside the RPV and PCV became more accurate.
	Improvement of the model of a fall of the reactor core structure	-Model was improved so as to drop the upper part of the defect part such as fuel rod without intermediate and bottom part and control rods. The result indicates that it is unlikely that the most of the debris in the Unit 1 is not located in the core region.
	Add eutectic reaction model with the released B_4C and the structures	-Timing of the damage caused to the reactor core structures becomes earlier considering the eutectic reaction with released B_4C and the structures. Possibility of remaining reactor internals, such as shroud was increased.
Improvement of evaluation of the debris distribution /composition inside the RPV	Improvement of the damage model of the penetration pipe into the bottom of the RPV breakage	-Adding model of molten materials entering into penetration pipe, evaluation of more realistic damage on the RPV became available.
	Add evaluation model of the melted debris attached to the lower structures inside the RPV.	-Estimation of amount of debris attached to the CRD housing, which are the lower structures outside the RPV became available.
Improvement od the debris distribution/composition evaluation inside the PCV	Add thermal interaction model with debris and lateral side of the PCV.	-Temperature evaluation for the lateral side of the PCV became available.
	Add evaluation model of the melted debris attached to the lower structures outside the RPV.	-Estimation of amount of debris attached to the CRD housing, which are the lower structures outside the RPV became available.
	Improvement of the core concrete interaction model	-Feeding debris into the pipe between sumps in the 3D MCCI response model, and adding the model of the shaft of the fuel debris properties during the MCCI response, the information on the three-dimensional expansion of debris and concrete corrosion erosion were obtained.
Estimation of PCV damage location	Improvement of PCV damage model	-Dividing the D/W and R/B node more finely, estimation of temperature, which can be utilized for the estimation of the locations of overtemperature damage, became available.
Improvement of fission product distribution evaluation	Improvement of fission product (FP) migration model	-More accurate FP distribution evaluation became available by dividing the node more finely and revising the scribing model of the suppression pool.

Reference: IRID Completion report for "Improvement of recognition regarding the internal PCV condition of the reactor vessels using severe accident progression analysis and actual plant data"

Appendix 4.7: Examples of sensitivity analyses using the MAAP code

In FY 2015, the analysis was performed focusing on the event specific to each unit.

In this analysis, plant behavior mechanism is being clarified using the sensitivity analysis, reducing the uncertainties in the analysis.

For Unit 2, it is not known how the three spiking pressure behaviors occurred that were recorded after the RPV decompression and how much water was injected with a fire engine. Since there have a significant impact on the analysis results for the damage on the RPV and amount of fuel debris that transferred, clarification was conducted by the analysis using the severe accident analysis code.

The results are as follows.

a. Analysis of reactor pressure behaviors after the decompression of the RPV of Unit 2

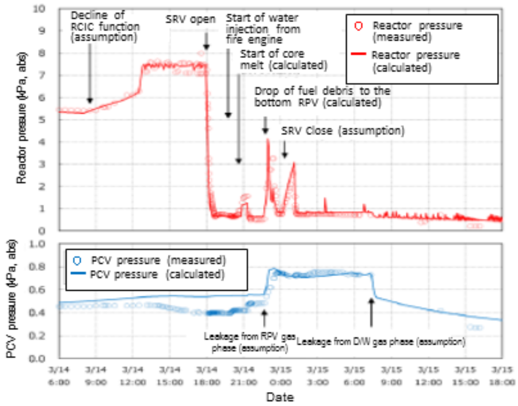
With the analysis results, it is assumed that the pressure spikes occurred as follows:

-First spike: Water injected from fire engine partially submerged the damaged reactor core, generated hydrogen and steam, raising the pressure. The water powering was suspended and the SRV was opened to decrease the pressure.

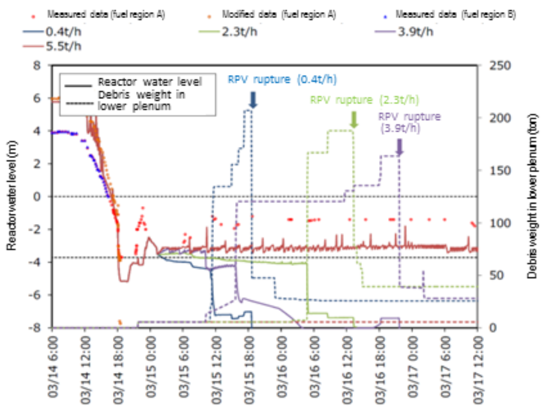
-Second spike: Debris dropped into the lower plenum, generated hydrogen and steam. This raised the pressure and caused air to leak from the RPV to the PCV, raising the pressure inside the PCV. After that, the SRV was opened to decrease the pressure.

-Third spike: The SRV was (assumedly) closed, which raised the pressure. After that, the SRV was reopened to decrease the pressure.

It turned out that during these processes, the amount of the debris that dropped onto the bottom of the RPV was relatively low and therefore the amount of water injected with fire engines probably governed the degree of the damage to the RPVs.



Reference: IRID Completion report for "Improvement of recognition regarding the internal PCV condition using severe accident progression analysis and actual plant data"
Figure A4.7-1 Pressure behavior analysis for depressurizing RPV for Unit 2 (MAAP)



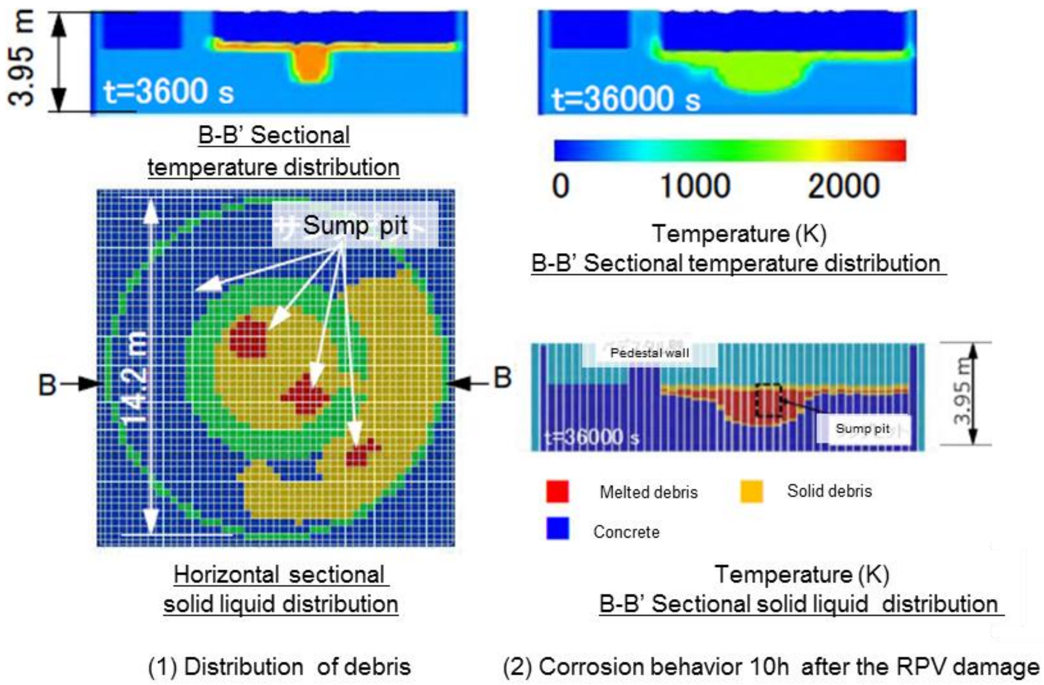
Reference: IRID Completion report for "Improvement of recognition regarding the internal PCV condition using severe accident progression analysis and actual plant data"
Figure A4.7-2 Sensitivity analysis for water injection amount by fire engine for Unit 2 (MAAP)

b. Sensitivity analysis of the amount of water poured from a fire engine to Unit 2

Figure A4.7-2 shows the results of the sensitivity analysis for water injection amount by fire engine for Unit 2. The analysis used four different water injection rates from 0.4, 2.3, 3.9, 5.5t/h. Three rates of 0.4, 2.3 and 3.9 t/h caused damage on RPV and no damage by 5.5t/h.

Appendix 4.8: Analysis results of the MCCI of Unit 1 using MCCI evaluation module

At the Fukushima Daiichi NPS, the pedestals, including their sump pits, are complicated in shape and it is important to evaluate the erosion of concrete and the generation amount of MCCI. For this reason, with an advection/diffusion model for eroded concrete added to the MCCI evaluation module within the SAMPSON code, an evaluation was made of the spreading/erosion behavior of fuel debris for Unit 1. Figure A4.8-1 shows the revaluation results. They indicated that for Unit 1, 40% of the D/W floor area was covered with fuel debris.



Reference: IRID Completion report for "Improvement of recognition regarding the internal PCV condition of the reactor vessels using severe accident progression analysis and actual plant data"

Figure A4.8-1: Spreading/erosion Behavior of Fuel Debris for Unit 1 (MCCI Advanced model)

Appendix 4.9: FP distribution analysis results by severe accident analysis code

The FP distributions inside the RPV, PCV, and R/B were evaluated using MAAP and SAMPSON codes. Table A4.9-1 shows the analysis results of distribution of Cs and Sr, which are the representative FP nuclide. The large uncertainties in the analysis results of both codes are caused by variations of FP models and chemical form of FP nuclide considering evaluation model.

Table 4A.9-1 FP (Cs and Sr) Distribution analysis results (deposition amount at the time of 6 days after the accident) (unit: kg)

Area	Unit 1		Unit 2		Unit 3	
	Cs	Sr	Cs	Sr	Cs	Sr
Inside the RPV	10-90	0.4-0.7	20-150	13-33	38-110	11-46
Inner wall of D/W	6-29	0.2-0.3	Approx. 0.9	0.2-0.4	less than 0.1- 1	less than 0.1
S/P underwater	26-74	0.4-2	27-130	3-4	57-140	1-6
Release to environment	less than 0.1-4	less than 0.1	less than 0.1-6	less than 0.1-0.4	less than 0.1-4	less than 0.1

-Sr was basically exists in debris, other than those described above.
-FPs in gaseous form is estimated to be released by being replaced by nitrogen gas.
Those of Units 1 and 2 that exist in the S/P water were estimated to be released as contaminated water.
Those of Unit 3 in the S/P water may remain as it is or release by flowing back to the D/W side from the leak points as contaminated water.

Reference: IRID Completion report for "Improvement of recognition regarding the internal PCV condition of the reactor vessels using severe accident progression analysis and actual plant data"

Appendix 4.10: Evaluation criteria for degradation of major internal structures and equipment under a high-temperature environment

The conditions of the equipment inside the reactors were estimated according to temperature evaluation results based on severe accident analysis (MAAP and SAMPSON codes). Evaluations were made of degradations due to high-temperature deformation, creep rupture, and corrosion. The table below shows the evaluation criterial for each degradation event.

Degradation event	Evaluation criteria
High-temperature deformation	In general, the tensile strength of material decreases as the temperature rises. When the temperature is higher than 1,000°C, stainless steel and other metal materials exhibit almost no stress resistance. This means that structures made of such materials are no longer serviceable and therefore determined to be likely to collapse.
Creep rapture	In general, when the temperature is higher than a certain level, stainless steel and other metal materials develop plastic deformation over time and finally break (develop creep rapture) even under a load condition under which they do not deform or break at normal temperature. It is determined that under an environment with a temperature higher than 500°C, which is higher than normal operating temperature, they may undergo creep deformation or break.
Corrosion	Immediately after the accident, seawater was injected into the RPVs to cool the hot reactor cores. In general, stainless steel and other passivated materials have their passive films destroyed by chloride ions and others contained in seawater and when the temperature is higher than a certain level, they develop SCC cracks and/or undergo pitting corrosion. At present, the water is purified and the temperature is low and thus they have no chance to crack or corrode; however, it is determined that it was highly possible that they had undergone cracking and corrosion immediately after the accident. It is not clear how many SCC cracks were developed and how deep (long) pit corrosion was immediately after the accident and thus it is difficult to evaluate the impact of corrosion on the structural soundness of equipment. Since it is difficult to evaluate the impacts on individual pieces of equipment, corrosion is not considered when the conditions of the individual pieces of equipment inside the reactors are estimated.

(Provided by IRID)

Appendix 4.11: Estimation results of major structures and equipment conditions

Through evaluation of degradation events, the conditions of the equipment inside the reactors were estimated according to temperature evaluation results based on severe accident analysis (MAAP and SAMPSON codes). The following table shows the evaluation results.

Note that the use of the following evaluation results requires consideration of the fact that the accident development analysis (MAAP and SAMPSON codes) includes uncertainty.

.	Unit 1	Unit 2	Unit 3
Steam dryer	For both of MAAP and SAMPSON code, the temperature is lower than 1,000°C and the dryer may have been deformed by creep.	For both of MAAP and SAMPSON code, the temperature is between 750°C and 1,000°C and the dryer may have been deformed by creep.	For SAMPSON code, the temperature is higher than 1,000°C for certain duration and the dryer may have been excessively deformed. For MAAP code, the temperature is approximately 800°C and the dryer may have been deformed by creep.
Steam separator	For both of MAAP and SAMPSON code, the temperature is lower than 1,000°C and the dryer may have been deformed by creep. For MAAP code, the temperature is higher than 1,000°C for certain duration at the shroud support and head and the appropriate sections of the separator may have been excessively deformed.	For MAAP code, the temperature is higher than 1,000°C for certain duration and the separator may have been excessively deformed. For SAMPSON code, the temperature is approximately 700°C and the separator may have been deformed by creep.	For SAMPSON code, the temperature is higher than 1,000°C for certain duration and the separator may have been excessively deformed. For MAAP code, the temperature is approximately 800°C and the separator may have been deformed by creep.
Upper grid plate	For MAAP code, the temperature is higher than 1,000°C for certain duration and the plate may have been excessively deformed. For SAMPSON code, the temperature is lower than 1,000°C but the plate may have been deformed by creep.	For MAAP code, the temperature is higher than 1,000°C for certain duration and the plate may have been excessively deformed. For SAMPSON code, the temperature is approximately 700°C and the plate may have been deformed by creep.	For both of MAAP code and SAMPSON code, the temperature is higher than 1,000°C for certain duration and the plate may have been excessively deformed.

Estimated Degradation Levels of the Equipment inside the Reactors at High Temperatures (continued)

	Unit 1	Unit 2	Unit 3
Shroud	For MAAP code, the temperature is higher than 1,000°C for certain duration at the shroud head. Fusion penetration may be occurred in the side surface (middle height position) of the shroud when molten pool is formed in the core region. For SAMPSON code, the temperature is higher than 1,500°C and the shroud has been melted middle height position.	For MAAP code, the temperature is higher than 1,000°C for certain duration and the shroud may have been excessively deformed. For SAMPSON code, the shroud is melted at the intermediate height. It is, however, assumed that the shroud is less likely to be remarkably damaged judging from the primary-system behavior against the changes in water injected from the water supply system in the actual plant.	For both of MAAP and SAMPSON code, the temperature is higher than 1,000°C for certain duration and the shroud may have been excessively deformed.
Fuel assembly /control rod	They are assumed to have melted and dropped.	Same as the left	Same as the left
Reactor core support plate	The plate is assumed to have contacted fuel debris, melted, and remarkably deformed.	Same as the left	Same as the left
Shroud support	For MAAP code, the temperature is higher than 1,000°C for certain duration and the shroud support may have been excessively deformed. The part that had contacted fuel debris may have melted and deformed.	Same as the left If the shroud support is remarkably damaged, the shroud and other structures may incline; however, the shroud is assumed to be less likely to remarkably incline judging from the primary-system pressure behavior against the above-mentioned changes in water injected from the water supply system.	Same as the left
Control rod guide tube and nuclear instrumentation guide tube	The guide tubes are assumed to have contacted fuel debris, melted, and remarkably deformed.	Same as the left	Same as the left

Estimated degradation levels of equipment inside the reactors at high temperatures (continued)

	Unit 1	Unit 2	Unit 3
Stub tube	The stub tube is assumed to have contacted fuel debris, melted, and remarkably deformed.	Same as the left	Same as the left
CRD and ICM housings	The housings are assumed to have contacted fuel debris, melted, and remarkably deformed. The welding region with the RPV may have melted and penetrated.	Same as the left	Same as the left

(Provided by IRID)

Appendix 4.12: Overview of heat balance method and estimation results (details)

The heat generation is still continued due to decay heat from the fuel debris and water injection into the RPV to cool it down. As temperature of water injection goes up to the RPV temperature and from RPV temperature to PCV stagnant water temperature, cooling has been continued with increase in temperature.

Heat balance method is a method that estimates the fuel debris proportion inside the RPV and PCV assuming it maintains (balance) the state that enables decay heat (heat generation) = Sum of temperature rise of cooling water (heat release), That is, the heat balance method means to estimate the proportion between the fuel debris in the RPV and PCV based on the assumption that the temperature of the cooling water injected into the RPV is raised to the temperature of the accumulated water by the heat source (fuel debris) inside the RPV and PCV, namely the assumption that the heat input (heat of the injected cooling water and decay heat) comes into balance with heat radiation (heat radiated out to the building or into the air through the PCV walls and cooling-water temperature rise caused by fuel debris). Figure A4.11-1 is a conceptual rendering of evaluation based on the heat balance method.

Shown below is the fuel debris distribution estimated based on the heat balance method for each unit.

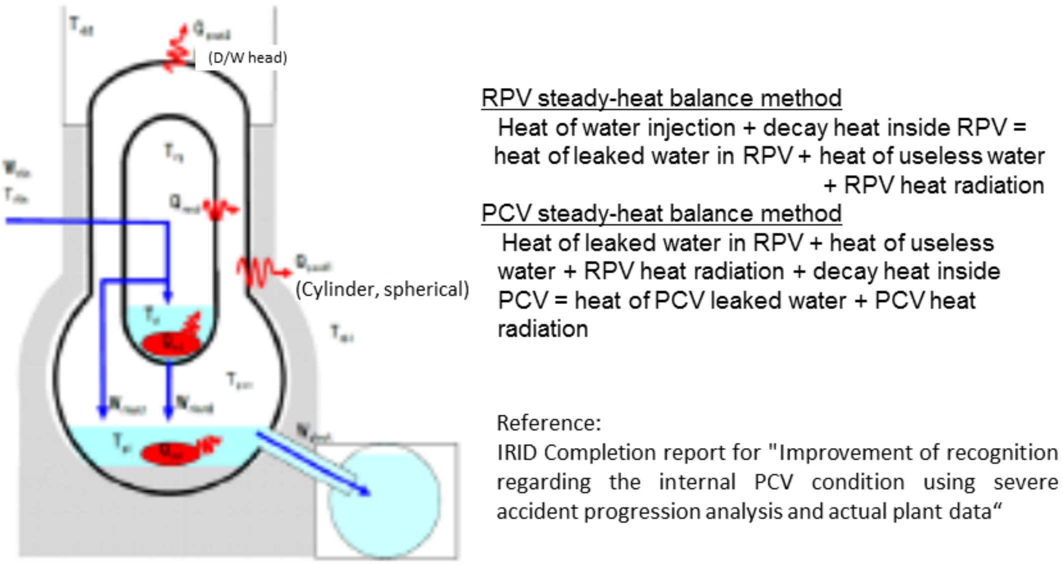
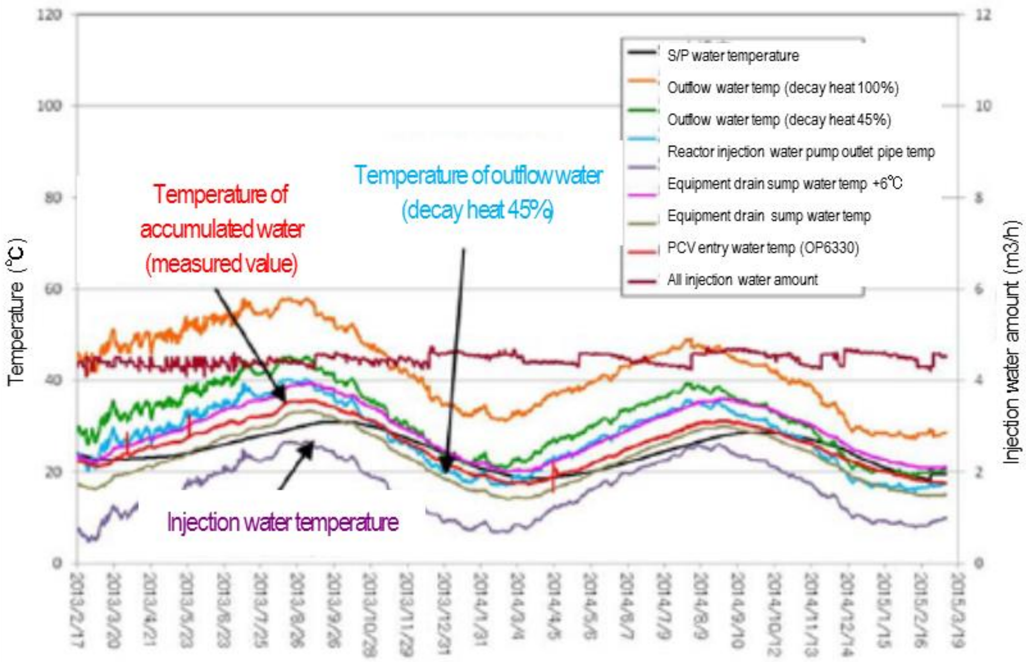


Figure A4.12-1: Conceptual Rendering of Evaluation Based on the Heat Balance Method

a. Evaluation results for Unit 1

For Unit 1, an evaluation based on the heat balance method was performed on the assumption that no heat source exists inside the RPV (i.e., the RPV decay heat in the equation above = 0) according on the analysis results based on the MAAP code. The evaluation used the quantity of the heat of the injected and leaked water calculated from actually measured temperatures of both water, with the degree of contribution of decay heat to the rise of the accumulated water temperature used as an evaluation parameter.

The evaluation results indicate that if it is assumed that the heat source is equivalent to 45% of decay heat, the actually measured changes in accumulated water temperature are almost reproducible as Figure A4.12-2 shows; it is determined that a significant heat source exists inside the PCV. It is, however, deemed that the fuel debris distribution is significantly affected by the uncertainty about the decay heat of the fuel debris that fell at the bottom of the PCV (according to evaluation by JAEA, the decay heat decreases approximately 60% if all of the highly-volatile nuclides are released), possibility of heat radiation from fuel debris to the floor concrete, and uncertainty about the evaluation of the transfer rate of the heat radiated from the PCV into the outside air.



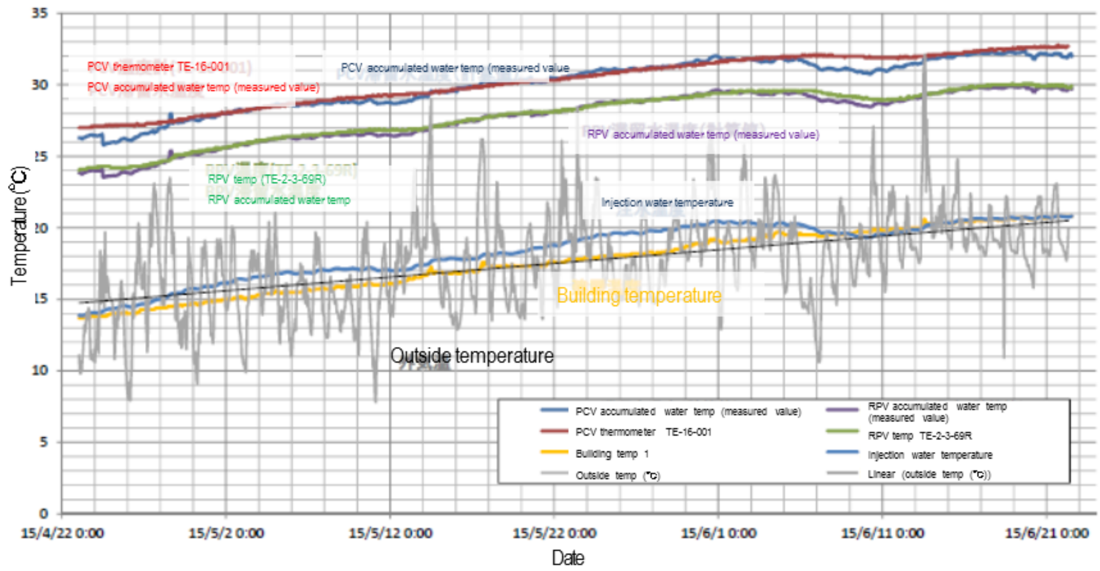
(Provided by IRID)

Figure A4.12-2: Example of an Evaluation Based on the Heat Balance Method for Unit 1

b. Evaluation results for Unit 2

The evaluation was made based on the quantity of the heat of the injected and leaked water calculated from actually measured temperatures of both water, using as evaluation parameters the degree of contribution of decay heat to the rise of the accumulated water temperature and the ratio between the amounts of the heat sources (fuel debris) inside the RPV and PCV.

Figure A4.12-3 shows an evaluation result example. With the ratio between the quantity of the heat sources (fuel debris) inside the RPV and PCV used as a parameter, the evaluation results indicate that if it is assumed that 30 to 60% of the heat source remains inside the RPV, the changes in accumulated water temperature inside the RPV and PCV are almost reproducible.



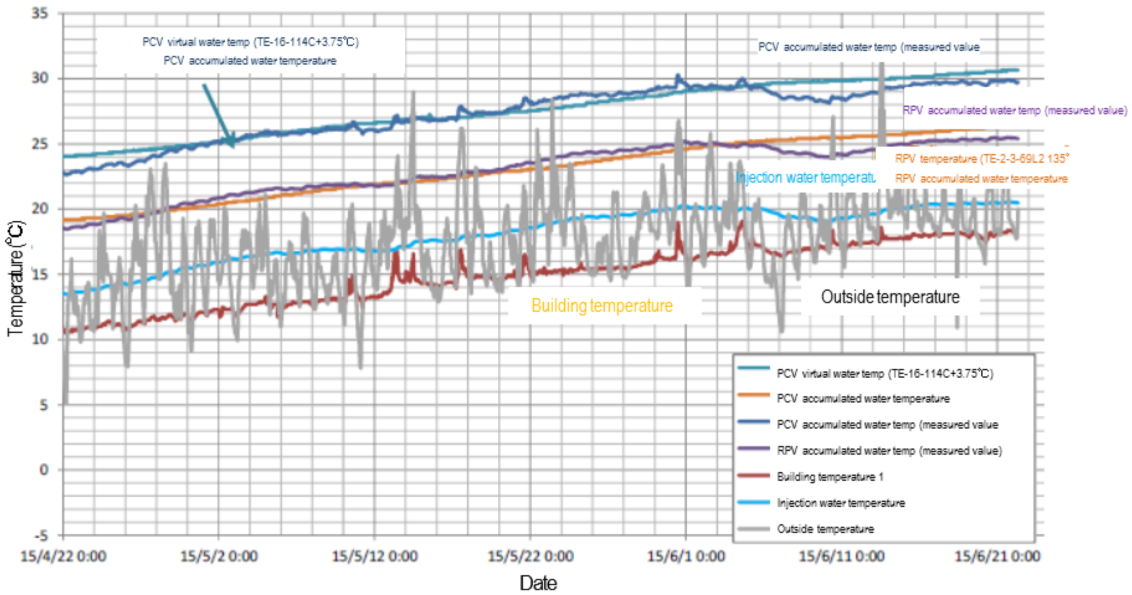
(Provided by IRID)

Figure A4.12-3: Example of an Evaluation Based on the Heat Balance Method for Unit 2
(Percentage of Heat Source inside the RPV: 60%)

c. Evaluation results for Unit 3

As with Unit 2, the evaluation was performed based on the quantity of the heat calculated from the actually measured water temperatures.

Figure A4.11-4 shows evaluation result examples. The evaluation results indicate that if it is assumed that 20 to 70% of the heat source remains inside the RPV, the changes in accumulated water temperature inside the RPV and PCV are almost reproducible. They also indicate, however, that since the changes in the temperature of the accumulated water in the RPV do not agree with that of the injected water, the quantity of the fuel debris existing inside the RPV as a heat source is a lower than estimated.

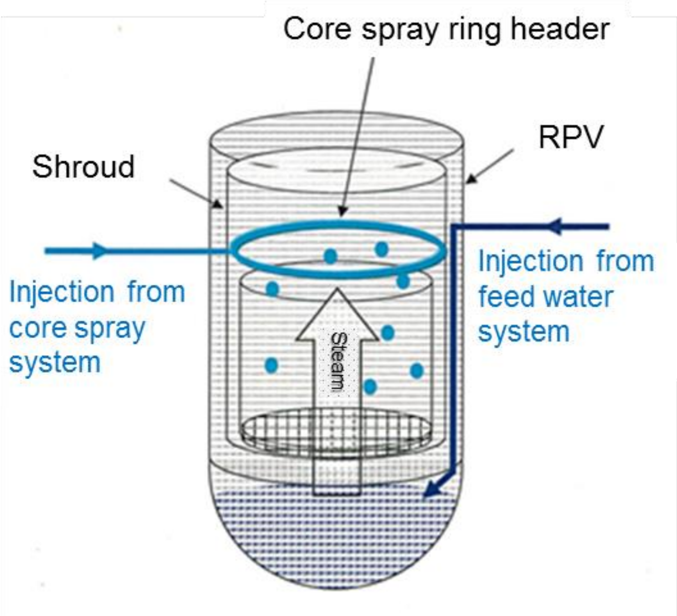


(Provided by IRID)

Figure A4.12-4: Example of an Evaluation Based on the Heat Balance Method for Unit 3
(Percentage of Heat Source inside the RPV: 40%)

Appendix 4.13: Location of fuel debris estimated from plan parameter trend

The heat source of RPV (fuel debris) was estimated based on the trend of the temperature around the RPV in the post-accident condition, water temperature of S/C, water injection amounts of feedwater (FDW) system and reactor core spray (CS) system. Figure A4.13-1 shows difference between flow paths of the FDW system and CS system. FDW system is a system that introduces the cooling water during the normal operation of BWR into the RPV. After flowing into the RPV, cooling water is accumulated in the space between reactor core shroud and the RPV (annulus), it will be flowing into the jet pump at the time when the water level reaches to the upper part of the jet pump mixer. If the integrity of the bottom of the RPV is maintained, the cooling water flowed in will accumulated inside the RPV and water level of the accumulated water will be raised. However, since increase in the water level was not observed, the bottom of the RPV has been damaged and the cooling water is estimated to be flown down from the damaged portion to the inside of the pedestal. That is, FDW system water injection cannot cools down the BWR reactor core portion but the bottom part of the RPV, even after the accident. While on the other hand, CS system is a core spray system during the coolant loss accident and is installed along the walls immediately above the core reactor core shroud. In the CS system water injection, the cooling water is flowing down the space from the reactor core to the bottom of the RPV and the scape can be cooled down. Based on the above, the fuel debris locations were estimated for each Unit.



Reference: Public data released by TEPCO

Figure A4.13-1 Flow path of FDW and CS systems

Shown below is the fuel debris distribution for each unit estimated from the trend of the plant parameters.

a. Evaluation results for Unit 1

Figure A4.13-2 shows temperatures at several positions inside the PCV of Unit 1 along with changes in the quantity of injected water and measurement locations. In response to the changes in the quantity of injected water, the following characteristic changes in temperature were observed.

(1) Compared with Units 2 and 3, the ambient temperature of the RPV decreased at a fast rate, which decreased to below 100°C five months after the accident.

(2) The ambient temperature of the RPV did not rise at a rate that corresponds to the decrease in the amount of the injected water for the FDW system.

(3) With increases in the quantity of injected water for the FDW system, the ambient temperature of the RPV dropped to below 50°C and the S/C water temperature rose.

(4) With decreases in the amount of injected water for the FDW system, the ambient temperature of the RPV rose.

With the characteristics shown in (1), (2), and (3) above, it is estimated that the heat source is probably is small inside the RPV. From (3) and (4), it is assumed that a heat source may exist in the water injection channel for the FDW system and the heat removed in response to water injection has transferred to the S/C.

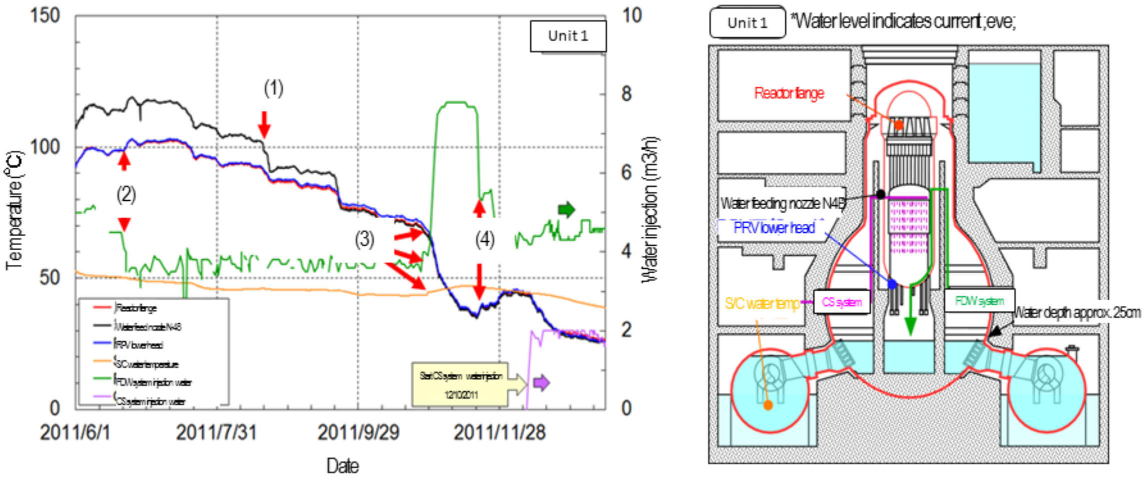


Figure A4.12-1: Changes in Plant Parameter and Measurement Locations for Unit 1

b. Evaluation results for Unit 2

Figure A4.12-2 shows the temperatures at several positions inside the PCV of Unit 2 along with changes in the amount of injected water and measurement locations. In response to the changes in the amount of injected water, the following characteristic temperature changes were observed.

(1) Compared with Unit 1, the ambient temperature of the PRV is high, which was higher than 100°C even six months after the accident.

(2) The temperature of the lower RPV head sensitively responded to the decreases in the amount of injected water for the FDW.

(3) With the start of water injection for the CS system, the ambient temperature of the RPV decreased and the S/C water temperature rose.

(4) With decreases in the amount of water injection for the CS system, the ambient temperature of the RPV rose, around the lower RPV head, in particular.

(5) With increases in the quantity of injected water for the CS system, the ambient temperature of the RPV dropped.

With the characteristics shown in (1), (3), (4), and (5), it is estimated that a certain quantity of heat source may exist inside the RPV; from (2), it is assumed that the lower RPV head is closer to the heat source than the water supply nozzle (N4B). From (3), it is deemed that the removed heat had transferred to the S/C.

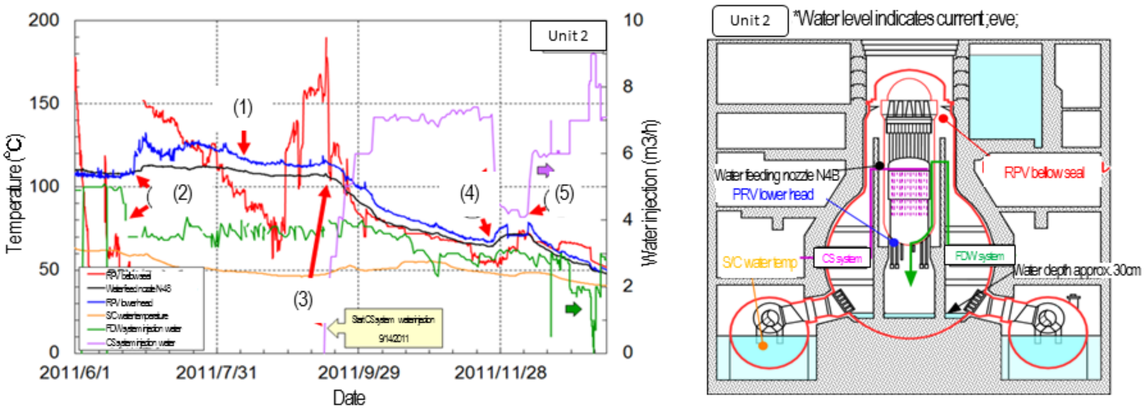


Figure A4.13-3: Changes in Plant Parameter and Measurement Locations for Unit 2

c. Evaluation results for Unit 3

Compared with Unit 1, the ambient temperature of the PRV is high, which was higher than 100°C even six months after the accident; as with Unit 2, it is assumed that a certain percentage of fuel debris exists in both of the RPV and PCV. The procedure for the above estimation is shown below:

(1) Compared with Unit 1, the ambient temperature of the PRV is high, which stayed at a level higher than 100°C even six months after the accident.

(2) Although the amount of injected water is highest for the FDW system, the ambient temperature of the RPV decreased at a low rate.

(3) With the start of water injection to the CS system, the ambient temperature of the RPV sharply dropped.

(4) With a decrease in the amount of injected water for the CS system, the temperatures of the water supply nozzle (N4B) and the lower RPV head rose.

With the characteristics shown in (1), (3), and (4) above, it is estimated that a heat source may exist inside the RPV.

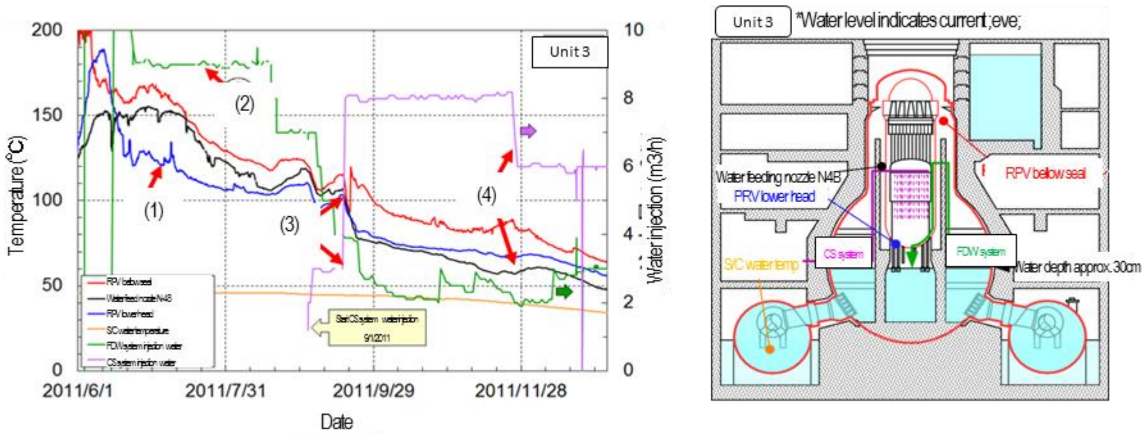


Figure A4.13-4: Changes in Plant Parameter and Measurement Locations for Unit

Appendix 4.14: Estimation of Fuel debris properties

Figure A4.14-1 Estimation of fuel debris properties (1/3) (Provided by IRID)

Based on the case of TMI-2, the internal conditions of the reactor vessel of Unit 2 was assumed using the results (FY2016) of analysis by the accident progression analysis code, SAMPSON.

The images shown are from TMI-2 photographs, and the actual condition of Unit 2 may be different. As the internal conditions of the reactor vessels become clear, reassessment may be required.

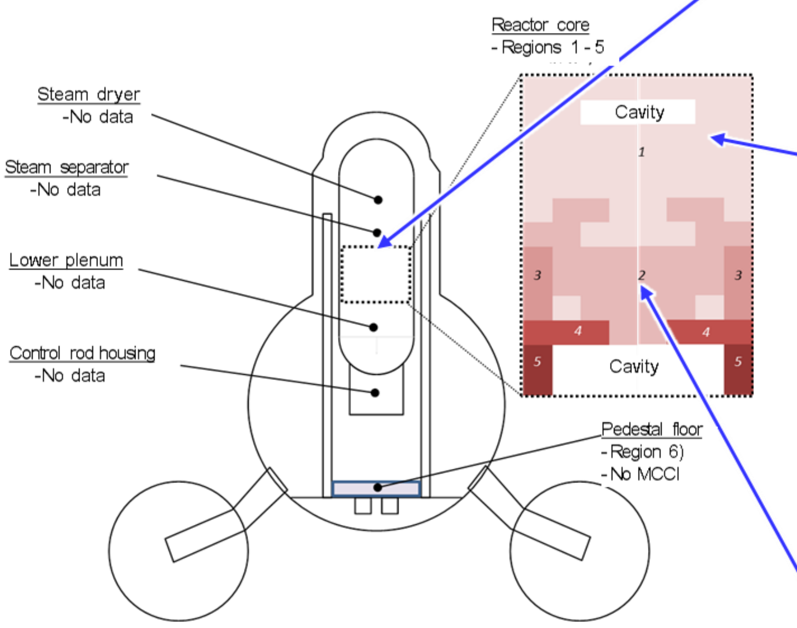
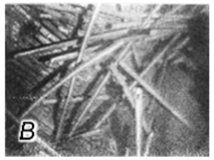
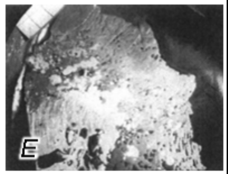
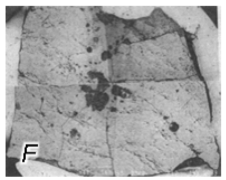
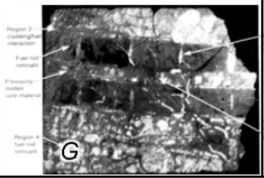


Image		Characteristics
Molten upper part Plenum	A No image	[Image A: Structures around the plenum] Molten or damaged upper plenum (debris attached) [Main composition: stainless steel, Zry-2, UO_2]
Loose debris	 B  D C No image	[Image B: Unmelted fragments] Unmelted cladding and fuel structural materials [Main composition: Zry-2, UO_2, stainless steel] [Image C: Unmelted fragments, small rock-shaped debris] Unmelted fragments, fragments of re-solidified fuels, melted objects solidified into granular debris after rapid cooling [Main composition: Zry-2, UO_2, stainless steel, $(U,Zr)O_2$] [Image D: Small rock-shaped debris] Small rock-shaped debris [Main composition: $(U,Zr)O_2$]
Upper crust	 E	[Image E: Upper curst] Molten fuels that cooled and solidified into fuel debris relatively quickly
Solidified molten material (Molten pool)	 F	[Image F: Solidified molten material] Molten fuels slowly solidified into fuel debris
Lower curst	 G	[Image G: Fuel pellets in the solidified molten material] Molten fuel assembly that cooled and solidified into fuel debris relatively quickly [Main composition: $(U,Zr)O_2$, (U- and Zr- rich phases), boride, metal-containing debris] ・In TMI-2, upper crust (surface crust of a few centimeters thickness), solidified molten materials (agglomerate), lower crust (crust of 0.1m thickness)

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B : Reprinted with permission from G. R. Eidam, "Core Damage" Chapter 5 of "The Three Mile Island Accident," 1986 American Chemical Society, Volume 293. Copyright 1986 American Chemical Society.

D ~ G : Reprinted from R. K. McCardell, M. L. Russell, D. W. Akers and C. S. Olsen, "Summary of TMI-2 Core Sample Examinations," Nuclear Engineering and Design 118 (1990) 441-449, Copyright 1990, with permission from Elsevier.

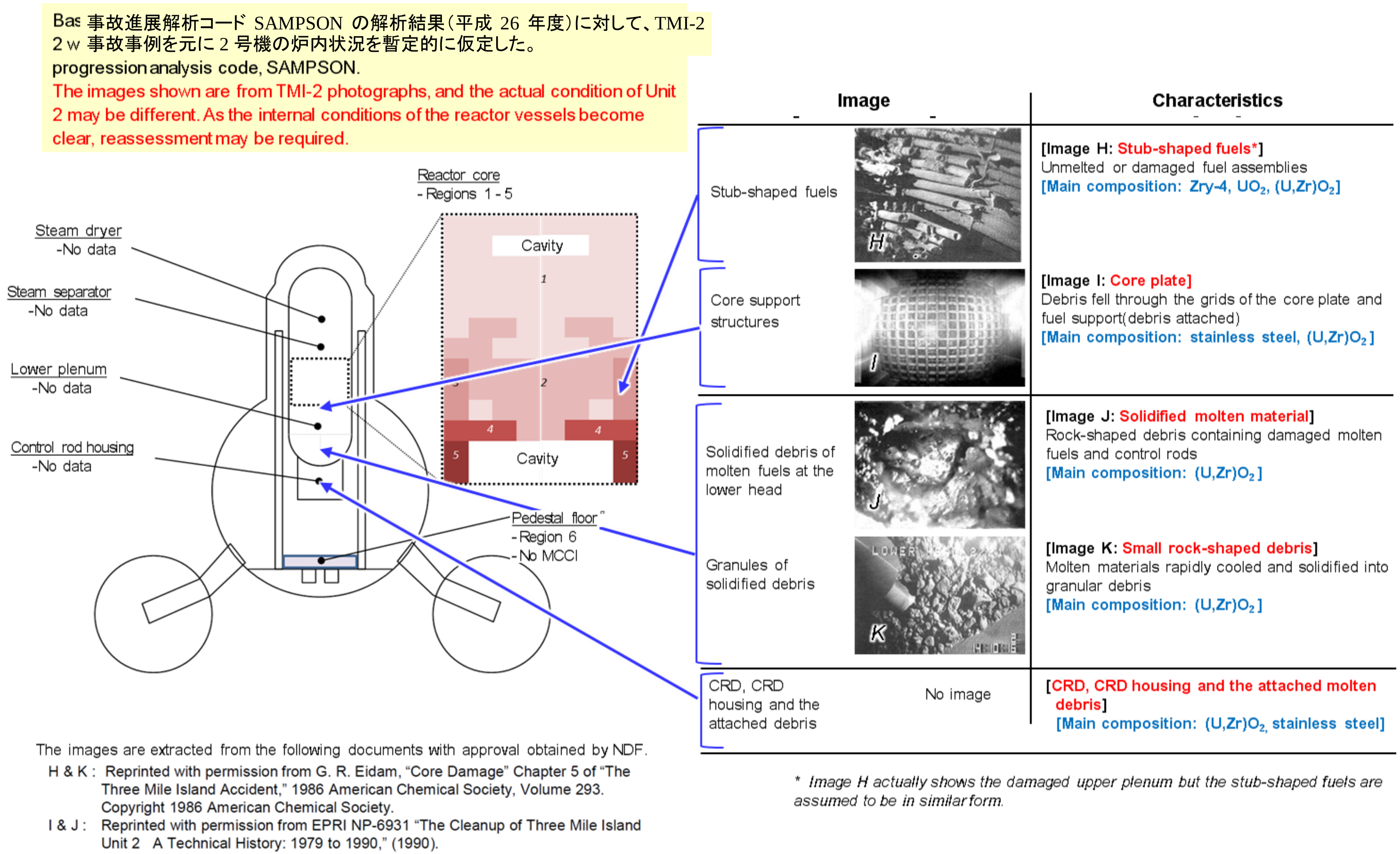


Figure A4.14-2 Estimation of fuel debris properties (2/3) (Provided by IRID)

事故進展解析コードMAAPの解析結果（平成26年度）に対して、海外の大型MCCI試験を元に2号機の炉内状況を暫定的に仮定した。
SAMPSON.

The images shown are from photographs of MCCI tests, and the actual conditions of Unit 2 may be different. As the internal condition of the reactor vessel become clear, reassessment may be required.

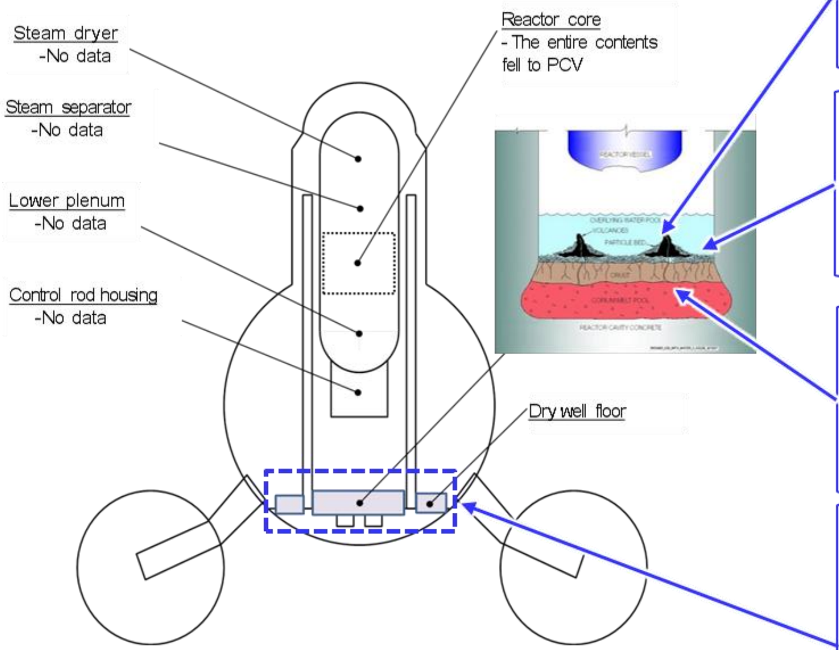
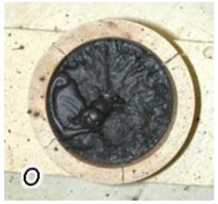


	Image	Characteristics
Volcano-like debris generated from MCCI	 L	**[Image L: Volcano-like debris generated from MCCI]** Debris formed when the molten corium inside streamed out after the solidification of crusts (CEA VULCANO test)
Powder-like debris generated from MCCI	No Image M	**[Image M: Powder-like debris generated from MCCI]** Corium that fell into the water during the initial stage and the molten corium inside streamed out after the solidification of crusts (ANL CCI test)
Agglomerates generated from MCCI	 N	**[Image N: Agglomerates generated from MCCI]** Molten corium inside has solidified (ANL CCI test)
Debris generated from MCCI (upper part)	 O	**[Image O: Debris generated from MCCI (upper part)]** Overall image after the test on silica-rich concrete (ANL CCI test)
Debris generated from MCCI (vertical cross-sectional view)	 P	**[Image P: Debris generated from MCCI (vertical cross-sectional view)]** Overall image after the test on silica-rich concrete (ANL CCI test) **[Main composition: $(U,Zr)SiO_4$, $(U,Zr)O_2$]**

The images are extracted from the following documents with approval obtained by NDF.

L : Reprinted with permission from C. Journeau, P. Piluso, J.-F. Haquet, S. Saretta, E. Boccaccio, J.-M. Bonnet, "Oxide-Metal Corium-Concrete Interaction Test in the VULCANO Facility," Proceedings of ICAPP 2007, Nice, France, May 13-18, 2007, Paper 7328.

N & P : Reprinted with permission from Argonne National Laboratory
Source: M. T. Farmer, S. Lomperski, D. J. Kilsdonk, and R. W. Aeschlimann, Argonne National Laboratory, S. Basu, U.S. Nuclear Regulatory Commission
Published in: "OECD MCCI Project, 2-D Core Concrete Interaction (CCI) Tests: Final Report," February 28, 2006, OECD/MCCI-2005-TR05, http://www.ipd.anl.gov/anlpubs/2011/05/69907.pdf.

O : Reprinted with permission from M. T. Farmer, "Thermite as a Validated Option to Melt Large Corium Masses," Plinius 2 International Seminar, Marseille, France, May 16, 2014.

Figure A4.14-1 Estimation of fuel debris properties (1/3) (Provided by IRID)

Appendix 4.15: Information for comprehensive analysis for the internal PCV conditions

Tables 4.15-1-4.15-3 show the result of the analysis using severe accident analysis code, estimation according to the trend of heat balance method and plant parameter, muon detection, PCV internal survey and BSAF Phase-1 and information that summarizes the amount and location of fuel debris and position.

Table A4.15-1 Comprehensive analysis/evaluation results of fuel debris amount and location and plant survey situation (Unit 1) (Unit: ton)

<table>
<tr><th colspan="2">Area</th><th>Severe accident analysis (incld. concrete)</th><th>Heat balance method</th><th>Estimated from plant parameter</th><th>Muon detection</th><th>BSAF Phase-1</th></tr>
<tr><td rowspan="2">In the RPV</td><td>Core region</td><td>0</td><td rowspan="2">Results of cooling state at the time of accident/accident progressions analysis indicate there is no heat source.</td><td rowspan="2">A few heat sources</td><td rowspan="2">High density materials (fuel) was not observed in the reactor core location</td><td>0-3</td></tr>
<tr><td>Bottom of the PCV</td><td>10-15</td><td>0-8</td></tr>
<tr><td rowspan="2">In the PCV</td><td>Inside the pedestal</td><td>187-209</td><td rowspan="2">With heat source</td><td rowspan="2">Possibility of heat source in the periphery of CDR piping (RPV bottom) heat source.</td><td rowspan="2">Out of scope of the measurement conducted so far</td><td rowspan="2">105-164</td></tr>
<tr><td>Outside the pedestal</td><td>52-85</td></tr>
</table>

*The result of the analysis performed by TEPCO indicates that the temperature change during the nitrogen injection can be explained by the temperature distribution after the accident focusing on the local heat source in the periphery of the CRD pipe.

Reference: IRID document

Table A.4.15-2 Comprehensive analysis/evaluation results of fuel debris amount and location and plant survey situation (Unit 2) (Unit: ton)

<table>
<tr><th colspan="2" rowspan="2">Area (Unit and No. of Institutions)</th><th rowspan="2">Severe accident analysis (incld. concrete)</th><th rowspan="2">Heat balance method</th><th rowspan="2">Estimated from plant parameter</th><th rowspan="2">Muon detection</th><th colspan="2">BSAF Phase-1*</th></tr>
<tr><th>6 organizations</th><th>3 organizations</th></tr>
<tr><td rowspan="2">In the RPV</td><td>Core region</td><td>0-13</td><td rowspan="2">Heat source rate: approx. 30-60%</td><td rowspan="2">With a certain extent of heat source</td><td rowspan="2">No large-scale high density material (fuel) is found in the reactor core region</td><td>0-14</td><td>0-32</td></tr>
<tr><td>Bottom of the RPV</td><td>25-58</td><td>0-91</td><td>0</td></tr>
<tr><td rowspan="2">In the PCV</td><td>Inside the pedestal</td><td>90-129</td><td rowspan="2">Heat source rate: approx. 40-70%</td><td rowspan="2">Heat removed is estimated to transfer to the S/C.</td><td rowspan="2">Out of scope of the measurement conducted so far</td><td rowspan="2">0</td><td rowspan="2">147-240</td></tr>
<tr><td>Outside the pedestal</td><td>5-106</td></tr>
</table>

*Analysis result of BSAF Phase-1 describes the results for the case where fuel debris had and had not fallen on the pedestal.

Reference: IRID document

Table A4.15-3 Comprehensive analysis/evaluation results of fuel debris amount and location and plant survey situation (Unit 3)

(Unit: ton)

Area (Unit and No. of institutions)		Severe accident analysis (include. concrete)	Heat balance method	Estimated from plant parameter	Muon detection	BSAF Phase-1*	
						4 orgs.	9 orgs.
In the RPV	Core region	0-29	Heat source rate: approx. 20-70%	With a certain extent of heat source	Under studies	0-21	0-36
	bottom of the RPV	25-79				8-81	0
In the PCV	Inside the pedestal	73-154	Heat source rate is about 30-80%	Difficult to estimate (Difficult to detect changes by parameter due to a large amount of stagnant water)		0	140-268
	Outside the pedestal	0-102					

*Analysis result of BSAF Phase-1 describes the results for the case where fuel debris had and had not fallen on the pedestal.

Reference: IRID document

Appendix 4.16: Feasibility studies on fuel debris retrieval from R/B bottom

This is the FS on the establishment of access routes by using bottom-access method, which is a presumption of fuel debris retrieval from the bottom of the R/B. It was considered whether it is possible to create a hole to establish a route to access the inside of the PCV, at the bottom of the R/B bases, holes through which equipment or robots can enter for removing the fuel debris that have dropped onto and are accumulated at the bottom of the PCV (near the RPV pedestal legs). During the consideration process, attention was focused on (1) prevention of risk for additional contamination expansion, (2) avoiding a negative impact on the foundation ground of the R/Bs and the soundness of the R/Bs, and (3) suppressing the costs and work period.

(1) Present Situation of the Fukushima Daiichi NPS and Investigation of the Existing Techniques including Construction Techniques

The existing techniques were investigated and examined against the technical requirements to identify the elements of technical development. As a result, as Figure A4.16-1 shows, the examination revealed that the shield method and pipe jacking method can be used to drill a shaft that connects the ground with underground, a horizontal tunnel for horizontal underground travelling, and an upward shaft that gives access from the underground of the R/B to the bottom of it.

Investigations and comparison for the existing technologies for opening a hole from the bottom of the R/B foundation base to the inside the PCV was conducted in order to introduce the equipment and robotics for fuel debris retrieval to access the inside the PCV. Although full-turning all casing method was selected as high applicability, however considering the current performance verification of applicability by the mockup test facility is considered necessary in advance.

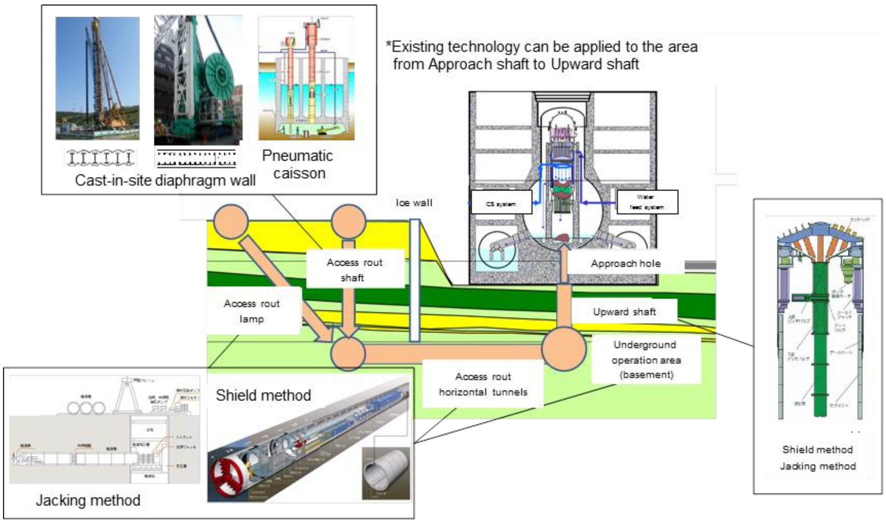
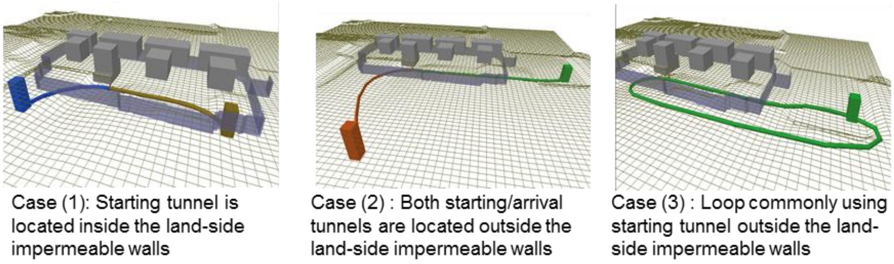


Figure A4.16-1: Examples of Applicable Existing Techniques

(2) Review of the Construction Concept

Several underground access routes were proposed for removal of fuel debris. First, the basic route configuration (an access shaft, horizontal tunnel, upward shaft, and approach hole) was reviewed. Then, comparative review was conducted of considerations about access route determination, including the location of the pit mouth and entry method, horizontal tunneling depth, tunnel connection and line shape, starting-point depth for drilling an approach hole, positioning transfer of drills machines, and handling of fuel debris when the approach hole is reached. Figure A4.16-2 shows the line shape by the installation locations of starting tunnel and arrival tunnel.



From left (1) Starting tunnel located inside land-side impermeable walls, (2) Both starting /arrival tunnels are located outside the land-side impermeable walls, (3) Loop commonly using starting tunnel outside the land-side impermeable walls

Figure A4.16-2: Examples of Line shape of installation locations of starting tunnel and arrival tunnel

A concrete method was reviewed for drilling an approach hole connecting the workroom at the bottom of the building with the fuel debris area as shown in Figure A4.16-3. In other words, it is a method for constructing an access route to the underground workroom at the bottom of the R/B. This method regards the PCV as the containment vessel, which penetrates the hole, as the boundary, and takes into consideration control of contamination spreading by using a double-pipe structure when penetrates PCV. It is, however, deemed to involve many issues concerning overall radiation measures such as handling of contaminated water and waste that are assumed to deliver high radiation environment and require further treatments. In addition, the method must allow for remote operation after the reached to PCV, and will be consequently required to address technical challenges such as robotization of the entire equipment and life extension of the drilling bits.

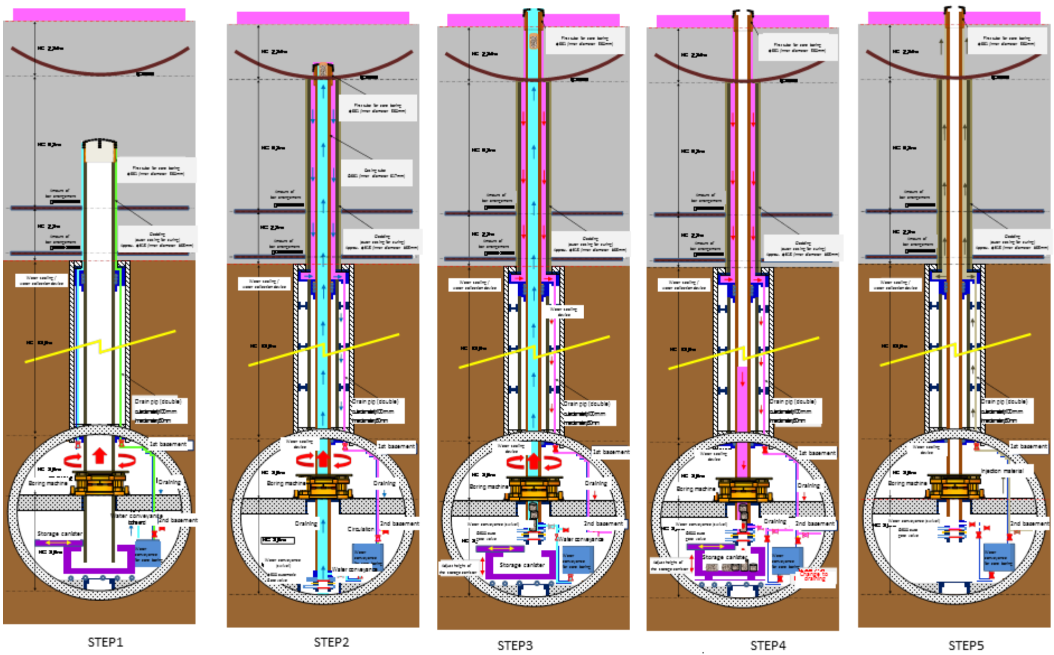


Figure A4.16-3: Procedures of boring from the lower part of the building to the fuel debris area

(3) Review of the Impact on the foundation ground of the Reactor Buildings and the Soundness of the Reactor Buildings

A review was conducted of how the foundation ground of the R/Bs and the soundness of the R/Bs would be affected by the access route constructed based on the drilling method by creating a construction equipment area immediately just under the PCV that seems to be highly applicable.

The impact of the settlement of earth surface based on a theoretical formula was determined by calculation, which revealed that for a horizontal tunnel with a diameter of 8.0 m, the settlement would be 1.4 mm at a point just above the tunnel, which in turn would be almost zero at a point 15 m away from the center of the tunnel. The decrease in area caused by an upward shaft would be as low as approx. 0.3%; which seems to hardly impair the bearing capacity of the foundation ground of the R/B. During the boring process inside the foundation base of the R/B, reinforcing steel would be cut, which would not probably weaken the foundation base. The complicated impacts caused by an earthquake, however, must be separately reviewed in details.

(4) Issues in the establishment of a access route to the lower part

For the access to the bottom of the building, access shaft, horizontal tunnel, and upward shaft, an existing method, such as the shield method, can be probably used. For the approach hole to be arranged at the base of the building, the full turning all casing method is assumed to be relatively applicable; however, as Figure A4.16-4 shows, it must address considerations such as how to stop water leakage from the bearing during the drilling process, remote operation and control of drilling, , installation of a water sealing equipment that can retain the water tightness of the containment vessel, boring method for the fuel duel debris area with uncertainties in the hardness, recovery of cores and contaminated water that contain debris, and how to secure long-term soundness of the watertight seal.

Judging comprehensively, establishment of an access route to the lower part involves many challenges that should be overcome. The achievement of it requires much technical consideration and development.

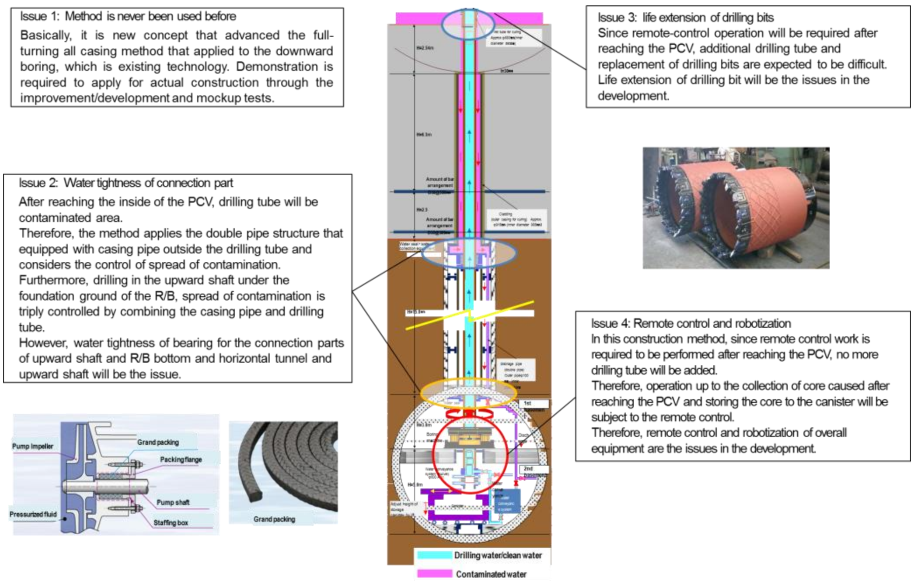


Figure A4.16-4: Issues in the technical development of approach hole boring work
Considerations Associated with Technical Development Required for Approach Hole Drilling

Appendix 4.17: Overview of analytical evaluation of air cooling inside the reactors

Objective

This analytical evaluation is aimed to estimate the temperatures of fuel debris and inside the PCVs at the target timing (2021) for fuel debris removed from Unit 1 of the Fukushima Daiichi NPS, given that the dry method will be used.

Analysis code

The evaluation used STAR-CCM+, a general-purpose, thermal-hydraulics analysis (CFD) program. STAR-CCM+ provides a variety of physical models capable of handling complicated engineering issues.

Analytic model

A three-dimension analytic model for air cooling inside the reactors was built based on the geometry data on the Fukushima Daiichi NPS as shown in Figure A4.17-1. Validated with the measured PCV temperatures, the analytic model can be used to analyze the distribution of the temperatures inside the reactors assuming that the dry method will be used.

For Unit 1, the analytic model was based on the assumption that the all fuel debris is to fall inside the RPV pedestal. Specifically, it is based on the assumption that the debris that has dropped first accumulates in the sump pits inside the pedestal and then overflows the sump pits, forming discoid accumulation inside the RPV pedestal (see Figure A4.17-2).

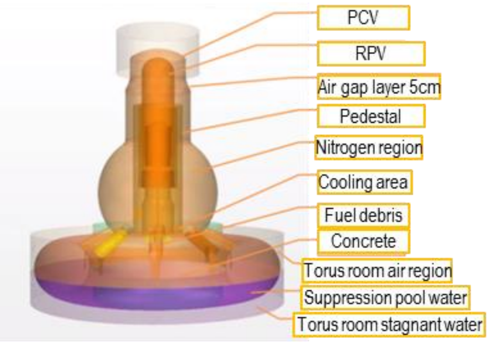


Figure A4.17-1: 3-D Temperature-distribution analytic model

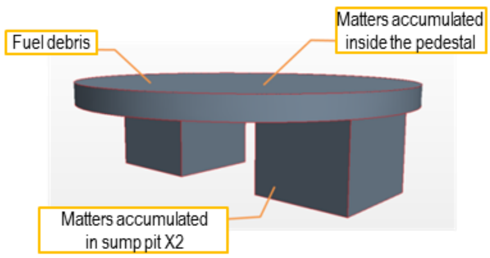


Figure A4.17-2: Shape of Debris inside the RPV Pedestal[3]

Boundary conditions

As boundary conditions, the model used ambient temperatures based on atmospheric-temperature data from the meteorological observatory of Namie City, Fukushima Prefecture. The heat-resistance value was set to $0.8m^2 \cdot k/W$ considering that the heat of the fuel debris is eliminated by natural convection of the N2 gas inside the PCVs.

The heat-resistance value was determined and used for analysis based on the heat transfer between the peripheral area of the analytic model and peripheral atmosphere when the concrete wall of the building is approximately 2 m.

Analytical input of decay heat

[3] For the debris inside the RPV pedestal, no consideration is given to the MCCI or leakage to the outside of the pedestal.

The amount of (decay) heat from fuel debris is referred from the published data from JAEA [JAEA-Data/Code2012-018[4]]. The heat from FPs (Cs-134 and Cs-137) takes into consideration the amount of heat released into the environment at the time of accident. It assumed that it adhered evenly to the inner wall of the PCVs because the distribution is not clear (see Table A4.17-1).

It was hypothesized that heat is evenly discharged from fuel debris according to the modeled fuel debris and that the heat is discharged from entire FPs considering the fact that FPs evenly adhered to the inside wall of the PCVs.

Table A4.17-1: Heat from Fuel Debris and FPs

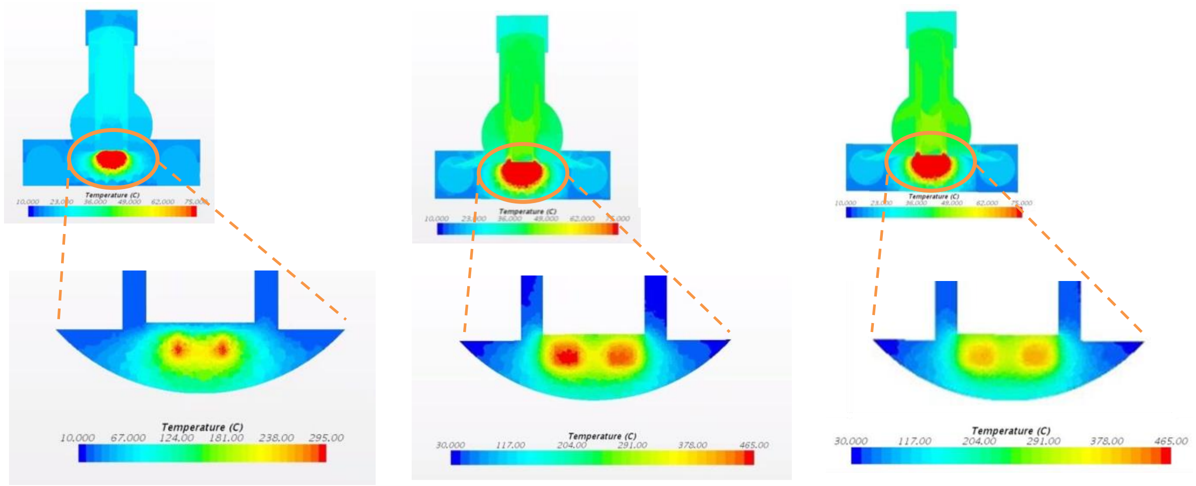
Period [year]	Fuel debris [kW]	FP [kW]
2015	61.5	9.37
2021	44.0	4.10
2031	32.9	2.40

Analysis results

Figure A4.17-3 shows the analytic evaluation results of air cooling inside the reactors. The analyses were conducted on the conditions in 2015 in which cooling is performed with injected water, in 2021, a target year for starting removal of fuel debris, and in 2031, 10 years after the removal target year, with the assumption that the dry method will used.

The analyses based on the dry method assume that no cooling water exists inside the PCVs or torus rooms and that all heat from fuel debris is removed through the coolant (nitrogen) inside the reactors.

The results of the Unit-1 analyses that assume the use of the dry method at the target timing for starting removal of fuel debris indicated that the maximum temperature of the fuel debris accumulated inside the RPV pedestals is approximately 480°C and maximum surface temperature of fuel debris, approximately 350°C.



Analysis Results inside a Reactor [Water injection cooling condition in 2015 simulated]

Analysis Results of Air Cooling inside a Reactor (2021) [Dry method assumed]

Analysis Results of Air Cooling inside a Reactor (2031) [Dry method assumed]

Figure A4.17-3: Analysis Results of Air Reactors

[4] Nishihara, Iwamoto, and Suyama, Estimation of Fuel Composition in Fukushima-Daiichi Nuclear Power Plant, 2012

Appendix 4.18: Applicability of fuel debris retrieval method for each locations (details)

Table A4.18-1 Applicability of the fuel debris retrieval method according to the locations (details)

Legend △ : Highly difficult ▲ : Extremely highly difficult

*1: Reactor well shield plug, PCV upper head, RPV head heat insulation materials, RPV head, steam dryer, steam separator, upper grid plate *2:Applicability is to be estimated considering the state of attachment of the fuel debris, relation with fuel debris retrieval method for other parts.

Fuel debris position / method	Periphery of RPV		Inside the PCV	
	Inside the RPV (core region and bottom of the reactor)	CRD housing	Inside the RPV pedestal	Outside the RPV pedestal
The Submersion-Top entry method Partial submersion-Top access method	△ -Removal of upper structure *1 or disassembly is estimated to be possible as a part of the existing technologies and technical development.	△*2 -Removal of upper structure *1 or disassembly is estimated to be possible as a part of the existing technologies and technical development. -Although a large size opening is required to be created in the bottom of the RPV, it is estimated to be possible as a part of the existing technologies or technical development. High degree of difficulty. Requires a long period of time for construction. -Assuming that the fuel debris in and outside the CDR housing is retrieved with its housing; it is estimated to be possible as a part of the existing technologies and technical development.	△ -Removal of upper structure *1 or disassembly is estimated to be possible as a part of the existing technologies and technical development. -Although a large size opening is required to be created in the bottom of the RPV, it is estimated to be possible as a part of the existing technologies or technical development. High degree of difficulty. Requires a long period of time for construction. -Since it is several 10 m away from the construction location, workability is required to be improved by lowering the stage for construction and it is possible to be achieved by using technical development.	▲ Direct access to the outside of the RPV pedestal cannot be achieved even an opening is installed in the bottom of the RPV. To access to the outside of the RPV pedestal, it is required to move further in a transverse direction after being after traveling to a long distance to the bottom of the RPV pedestal. It will take an extremely long time since the functions required to be equipped with the fuel debris are complicated.
Partial submersion-Side access method	▲ -Using X-6 penetration, access route to the inside of the RPV pedestal can be established by short-term technical development. - Approx.20m to the bottom of the RPV pedestal to the fuel debris inside the RPV. Disassembly of CRD housing, construction of opening in the bottom of the RPV and fuel debris retrieval inside the RPV requires high degree of stiffness and large scale fuel debris retrieval equipment. Estimated to be extremely difficult to perform installation and operation of large scale equipment with the limited opening and space.	△*2 -Using X-6 penetration, access route to the inside of the RPV pedestal can be established by short-term technical development. -CDR housing is located approx. 10m above from the bottom of the RPV pedestal and requires high stiffness and large scale fuel debris retrieval equipment. Estimated to be extremely difficult to perform installation and operation of large scale equipment with the limited opening and space.	△ -Using X-6 penetration, access route to the inside of the RPV pedestal can be established by short-term technical development. -Removal of the obstacles (e.g. Grating and replacement trolley) required to access the fuel debris on the floor is estimated to be possible by technical development.	△ -Establishment of access route to the outside of the RPV pedestal using the penetration to be installed inside the PCV is estimated to be possible by the implementation of technical development. -Many pieces of equipment are installed outside the RPV pedestal. Although it depends on the distribution of the fuel debris, Removal of equipment that impedes and fuel debris retrieval are estimated to be possible by the implementation of technical development.

Appendix 4.19: Studies on containment function (Boundary)

The accident on the first resulted in a long-term loss of power that deprived the power station of its cooling capability, resulting in a core meltdown. This led to losses of the RPV and PCV boundaries and the confinement capability of the R/Bs, resulting in severe environmental contamination. The accident took away all of the so-called quintuple walls. Under the situation where the confinement capability (boundaries) is incomplete, our critical challenge is to reduce the radiation dose and the impact of it, i.e., the risk level, through emergency measures, stabilization of the facilities, and cleaning after the accident.

This section considers the concept according to which the confinement capability of the power station was designed in the first place, the measures taken against the accident, and the present situation is as well as the idea on which the measures should be based in working on a new phase—removal of fuel debris. Table A4.19-1 shows an overview.

1. Confinement Capability (Boundaries) of a NPS under a Normal Condition

Speaking of confinement, it must be clarified what should be contained. A typical power station must contain roughly two types of substances: fission products (FPs) and corrosion products (CPs).

FPs are produced inside fuel pellets by nuclear fission, which partially go out of the pellets but are contained inside the fuel cladding. If this tube is damaged, FPs leak out into to the water, the coolant, in the nuclear reactor. In the design phase, a reactor is given an upper limit of FP concentration so that the reactor can continue to operate unless this limit is exceeded. In overseas, even if the fuel cladding was damaged, reactors, particularly in early stages, used to be operated continuously. In Japan, however, the nuclear reactors are stopped to prevent contamination if the fuel cladding has turned out to have a pinhole that allows even a slight amount of FPs to leak out. The FPs that require particular attention are rare gases and iodine because they are likely to leak out in a gaseous state. These FPs are released after they are decayed with an off-gas (OG) treatment system.

On the other hand, CPs exist in the coolant as products that are resulted from corroded structure materials, adhere to the fuel at the core, and then become liberated. The CPs that require particular attention are cobalt 60 and manganese 54. CPs are securely controlled with a treatment system for liquid waste as a matter of course to prevent them from having an impact on the outside of the power station. In addition, since they greatly expose workers to radiation, measures for reducing radiation exposure are being carried out such as use of low-cobalt materials as structural materials, improvement of reactor water chemistry, and removal of corrosion products.

During normal operation, the FPs and CPs mentioned above, including small amount of leaks, are kept within the control area and treated with a waste treating system to ensure that they satisfy the criteria before they are released.

During periodic inspection, fuel is replaced and the inside of the reactor is inspected with the RPV and PCV opened and even the nuclear reactor well filled with water. In this case, FPs and CPs are also kept within the control area and treated with a waste treating system to ensure that they satisfy the criteria before they are released. During construction, radioactive dust may be a problem. If this is the case, arrangements such as barrier placement and local air conditioning may be made to restrict the release of radioactive dust from the control area.

2. Confinement Capability (Boundaries) at the Time of Accident

Originally, the confinement capability is expected to prevent FPs from being released when an accident occurs. From the viewpoint of containment (boundary), FPs are contained by the so-called quintuple walls: (1) fuel pellets, (2) fuel cladding, (3) reactor coolant-pressure (RPV) boundary, (4) reactor containment vessel (PCV boundary), and (5) R/B (secondary containment). Among these confinement capabilities (boundaries), the RPV boundary is for containing liquid-phase FPs and the RCV boundary and R/B are for containing gas-phase FPs. When the reactor is in operation, the PCV is filled with nitrogen to keep the inside of it inactive for the purpose of preventing a hydrogen explosion when an accident occurs. According to the scale of the accident, the release of FPs is prevented by the fuel cladding tube, RPV boundary, or PCV boundary.

Actual accidents at BWR plants so far have not resulted in significant core damage caused by core re-submersion. This even applies to an accident that undergoes a guillotine break of the RPV boundary piping and results in a loss of the coolant. If an accident occurs, with the assumption that leakage of FPs from the PCV is 0.5% per day, leaked radioactive substances are to be retained inside the R/B and treated with an emergency gas-treatment system before they are released. For the liquid phase, a circulation loop is established as a boundary that allows cooling water that has flown out from the breakage to move to the S/C and then be poured into the reactor by the ECCS system. For a serious or virtual accident (siting evaluation), the performance of the containment by the PCV boundary and R/B is evaluated assuming that the reactor core is damaged. This evaluation is also based on the assumption that leakage of FPs from the PCV is 0.5% per day and leaked radioactive substances are to be retained inside the R/B and treated with an emergency gas-treatment system before they are released.

For a giant structure like a nuclear facility, it is impossible to completely eliminate leakage and therefore evaluation is made with a practically controllable leakage rate assumed and with an expectation that dynamic equipment has a suppression effect. The dose requirement at the time of accident occurrence is set to 5 mSv or less per accident.

The accident at the Fukushima Daiich Power Station, however, resulted in a long-term loss of power, leading to conditions severer than those assumed in the design phase. The loss of the cooling capability of the reactors and the damage to the cores resulted in losses of two walls: (1) fuel pellets and (2) fuel cladding tube. Furthermore, the molten cores damaged the reactor pressure vessels, resulting in a loss of the (3) RPV boundary. Subsequently, the temperature and pressure inside the reactor containment vessels became high, which damaged the (4) PCV boundary, causing leakage of steam containing radioactive substances and hydrogen produced by a water-zirconium reaction into the inside of the R/Bs. Since the emergency gas treatment system was not also operable due to the loss of power, hydrogen explosions and other factors damaged the (5) R/Bs.

All of the five walls were thus damaged, resulting in release of a large amount of radioactive substances. However, the existence of the PCVs somewhat suppressed the release of gases other than volatile gases such as rare gases and iodine. Although the reactor core melting was not avoided, it is notable that agile measures including fire engines helped the power station recover the cooling capability. This development was never seen for Chernobyl. As a result, the release of cesium to the outside air of the PCVs was controlled under 2%. For the other nuclides, the lease amounts were much lower.

On the other hand, water must be kept poured to cool fuel debris; the water leaks from the breakages of the PCVs, meaning that water is endlessly applied for cooling. This causes radioactive substances contained in leaked water to continuously flow out and mix with the accumulated water in the buildings, presenting a continuous problem—contaminated water—associated with the degraded confinement capability for the liquid phase. The water leak blockage capability is also degraded and therefore causes ground water to flow in, increasing the amount of contaminated water. This is further complicating the problems. To prevent contaminated water from increasing and radioactive substances from flowing out to the outside of the system, a circulating water injection cooling system is used that purifies contaminated water and then reuses it to cool reactor cores with the building used as a boundary. The level of the accumulated water inside the building is kept lower than that of ground water so that ground water will flow in to allow the building boundary to prevent radioactive substances from flowing out to the outside of the building. This continuous water injection successfully cooled fuel debris and decreased the temperature inside the containment vessels to a level well below 100°C. This remarkably suppressed the release of radioactive substances, achieving a so-called stable cold shutdown condition.

3. Present Confinement Capability (Boundaries) at the Fukushima Daiich Power Station

The present confinement capability (boundary) at the Fukushima Daiich Power Station is as follows.

The boundary for the liquid phase uses a circulating poured-water cooling system that purifies accumulated water inside the building using a water treatment system and then reuses the purified water to cool the reactor core with the PCV used as the primary boundary to retain cooling water poured into the reactor and the building used as the secondary boundary to retain water leaked from the PCV. The level of the accumulated water inside the building is kept lower than that of ground water so that ground water will flow in to allow the building boundary to prevent radioactive substances from flowing out to the outside of the building. This causes the ground water that has flowed into the reactor to be redundant water, which must be stored as water treated as the water treatment system. This requires a storage tank to be kept inside the premises, presenting a problem with contaminated water. To counter this problem, sub-drainage equipment is installed near the building to pump up ground water and release it into the harbor through a purification system, thus lowering the level of ground water to reduce the amount that flows into the building. In addition, multi-layer measures—a frozen wall—are taken to decrease the amount of ground water that comes close to the building. Equipment for transferring accumulated water inside the building is also installed so that the level of the accumulated water inside the building can be controlled. To reduce the amount of ground water that comes in and prevent contaminated water from flowing out, it is required to be able to securely control the level of water inside and outside the buildings under any condition.

At the boundary for the air phase, a nitrogen filler is used to inactivate the inside of the PCVs and a PCV gas controller is installed to extract gases inside the PCVs, filter them, and measure radiation from them before releasing them. This arrangement keeps the pressure in the air phase slightly positive to prevent a hydrogen explosion and minimizes the release of radioactive substances. With these measures, the estimated radiation dose at the boundary of the premises due to the radiation released from Units 1-4 is sufficiently low—approximately 0.00087 mSv/year.

As the goal to be achieved within fiscal 2015, it is required by the Secretariat of the Nuclear Regulation Authority that the additional dose at the boundary of the premises due to direct radiation from the contaminated-water holding tank and released liquids and gases should be less than 1 mSv/year.

4. Problems with the Confinement Capability (Boundaries) Associated with Removal of Fuel Debris and the Concept of the Measures against the Problems

Removal of fuel debris involves cutting of it, which probably causes α particulates to move to the liquid or gas phase inside the PCVs. In addition, changing the level of water inside the PCV is being considered, meaning that the requirements for the confinement capability (boundaries) may change.

For α particulates, the upper limit of concentrations associated with internal exposure through breathing, in particular, is stricter than for the other nuclides. For this reason, attention must be paid to particulates that may be released as dust from the gas phase and the gas-phase boundary must be arranged based on careful consideration. For this reason, in terms of control, it is easier to cut or treat otherwise fuel debris underwater or with water being applied, where possible, to let α particulates move to the liquid phase. In this case, raising the level of water inside the PCVs must be considered. This is one of the major aims of deliberately carrying out research and development activities about PCV repair (water leak blockage).

Basically, α particulates should be moved to the liquid phase, where possible, without dependence on the method for fuel debris removal, which inevitably raises the concentration of radioactive substances in the liquid phase (i.e., the risk level = hazard potential). It is questioned whether the present control of the water level difference between inside and outside the building sufficiently acts as the confinement capability (boundary) for the liquid phase. Measures for suppressing risks associated with fuel debris removal may involve the following:

(1) While fuel debris is cut, cutting particles nearby are sucked to control the concentration of α particulates that spread into the liquid phase and decrease the concentration of the radioactive substances in contaminated water so that the total hazard potential will be the same or lower.

(2) The PCV (blocking water leakage) is repaired to control leakage from the PCV and reduce the amount of radioactive substances that may mix into the accumulated water inside the building in order to reduce the risk level (it is difficult, however, to completely block (eliminate) water leakage).

(3) The methods in (1) and (2) may reduce the risk level during normal work. If an abnormally large amount of water leaks with the level of water raised inside the PCV, the level of water inside the torus rooms may increase and become higher than that of ground water. To counter this situation, a system is installed that emergently transfers water using a large-capacity pump to lower the water level. However, depending on the assumed leakage rate, it is difficult to avoid temporary water level reversal. It is also possible to use a method that does not raise the level of water inside the PCV.

(4) Temporary water-level reversal is evaluated for the effect in an emergency, where water-level reversal seldom occurs, to verify that it has no significant effect on the outside. To further lower the risk level, water leakage for the R/Bs is blocked. It is, however, expected to be difficult to completely eliminated water leakage from the buildings.

(5) To further reduce the risk level, an impermeable wall of, for example, clay is installed around the buildings as the final barrier.

It must be evaluated how much each level of measures can reduce the risk.

For the confinement capability (boundaries) for the gas phase, on the other hand, although an action is taken to transfer as many as α particulates to the liquid phase at the time of fuel debris removal, the concentration of α nuclides is expected to increase. As a solution to this, it may be possible to install a cell at the top of the PCV to arrange a system (primary boundary) that keeps the inside pressure negative as well as to install a container in the R/B to arrange an air conditioning system (secondary boundary) that controls the inside pressure to be negative. In this case, it is necessary to evaluate the impact on workers and the outside based on an estimated scattering ratio of α particulates and the actual filtering performance of the leakage rates at the primary and secondary boundaries. It is necessary to evaluate the impacts not only in normal conditions but also in an emergency.

5. Concept regarding Securing of the confinement Capability (Boundaries) Associated with Removal of Fuel Debris

At the Fukushima Daiichi NPS, rationally feasible measures have been implemented based on the precondition that the facilities were damaged after emergency measures taken after the accident and they are in a severe environment; these measures have resulted in the present stable state. At present, the Secretariat of the Nuclear Regulation Authority has established requirements (goals) that must be satisfied (in normal times) at present: i.e., the dose caused by additional exposure at the boundary of the premises (entire site) must be less than 1 mSv/year and workers must not be exposed to a dose of more than 100 mSv per five years or 50 mSv per year—the upper limits of normal-time doses defined by the Law Concerning Prevention of Radiation Injury due to Radioisotopes, Etc. These requirements are based on the control of releases and radiation in the control area for normal reactors. The evaluation made at the time of accident was based on an upper limit of 5 mSv per accident as a guideline from criteria for various accidents.

When fuel debris is removed, there is apprehension that the concentrations of FPs and α particulates—nuclear fuel materials—may rise inside the PCVs. Even if the concentrations increase in the gas and liquid phases, it presents no problems technically if a confinement capability (boundary) can be designed and evaluated that satisfies the goals for normal times and emergencies defied by the Secretariat of the Nuclear Regulation Authority.

However, the fact that a considerable amount of α particulates will be released during normal removal of fuel debris will not be probably accepted in the areas where efforts are being made to clear up damage caused by rumors. Under the circumstances, it may be appropriate to make the maximum efforts toward controlling the release to a value under the detection limit (ND) under normal conditions and use an upper premises-boundary limit of 5 mSv per accident as a guideline for very rare events such as accidents.

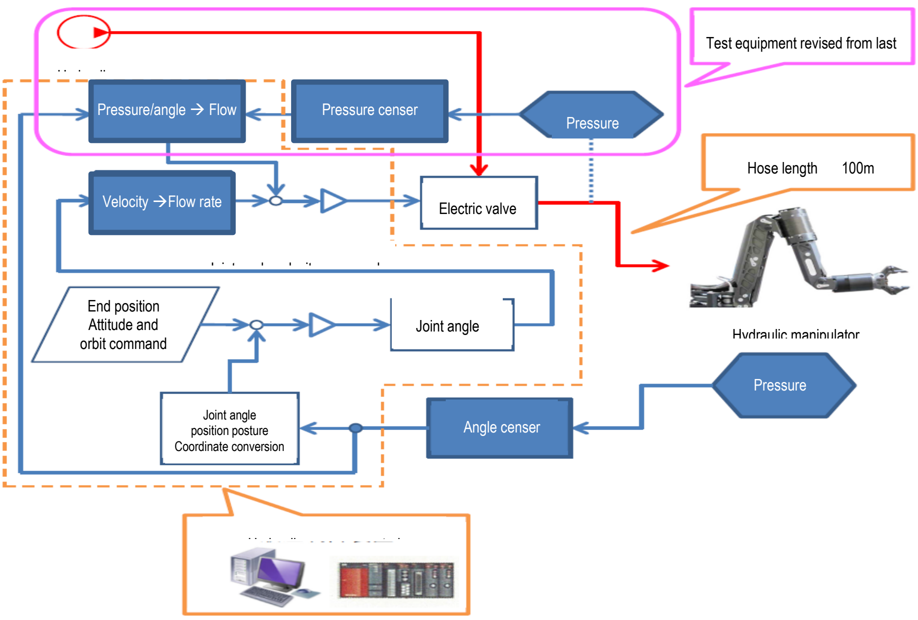
Table-A4.19-1 Containment functions at the Fukushima-Daiichi

	Before accident		At the time of accident	Current status	During fuel debris retrieval work	
	During operation period	During periodic inspection			Partial submersion method:	Submersion method:
Containment functions -FP confinement	(1) Fuel pellet +(2) Fuel cladding	(1) Fuel pellet +(2) Fuel cladding	(1) Fuel pellet + (2) Fuel cladding is melted into fuel debris with reactor internals. Volatility FP has already been released.	(1) Fuel pellet + (2) Fuel cladding is melted into fuel debris with reactor internals. Volatility FP has already been released.	(1) Fuel pellet + (2) Fuel cladding is melted into fuel debris with reactor internals. Volatility FP has already been released.	(1) Fuel pellet + (2) Fuel cladding is melted into fuel debris with reactor internals. Volatility FP has already been released.
-Liquid-phase part	(3) Containing cooling water by the RPV boundary. There is emergency cooling water in some part of SC.	Filling water to the reactor well, liquid phase is contained by (3) RPV boundary +reactor well	(3) Since leakage may be occurred from the (4) PCV boundary (DW/SC) due to the loss RPV boundary function, assuming building basement (e.g. R/B and T/B) as a liquid-phase boundary, the confinement (groundwater inleak) is to be ensured by the difference with groundwater level.	(3) Since leakage may be occurred from the (4) PCV boundary (DW/SC) due to the loss RPV boundary function, assuming building basement (e.g. R/B and T/B) as a liquid-phase boundary, the confinement (groundwater inleak) is to be ensured by the difference with groundwater level. Water level management liquid-phase boundary is composed of circulating injection cooling (large scale loop)	Liquid-phase leakage of debris cutting particles prevention is also important for the partial submersion method If a complete repair on the lower part of the PCV is achieved, PCV will be able to be functioned as a liquid-phase boundary. If complete repairs of the lower part of the PCV cannot be achieved, torus room can be functioned as liquid-phase boundary and the leakage can be prevented by the difference with groundwater level. If water is poured without repairing the lower part of the PCV, torus room can be functioned as liquid-phase boundary and the leakage can be prevented by the difference with groundwater level.	Important to prevent leakage of debris cutting particles. If complete repairs of the PCV is achieved, PCV can be functioned as liquid-phase boundary If complete repairs of the PCV cannot be achieved, torus room can be functioned as liquid-phase part boundary and the leakage can be prevented by the difference with groundwater level.
-Gas phase	(4) Gas phase boundary is composed of PCV boundary (N2 injection)+(5)R/B (maintain negative pressure by air-conditioning system)	(4) Releasing PCV boundary, gas phase boundary is composed of (5) R/B (maintain negative pressure by air-conditioning system).	(4) PCV boundary has been damaged and leakage is occurred in the gas phase. (5) R/B lost gas phase boundary function due to hydrogen explosion.	Using PCV gas management facilities and maintaining (4) PCV boundary within slightly negative, leakage from the gas phase is to be prevented.	Although (4) PCV boundary is released, installing cells and (5) R/B container as a measure to the debris cutting particles dust, gas phase boundary is composed of negative pressure/dust control system.	Although (4) PCV boundary is released, requirements on the gas phase boundary is low since measures to the debris cutting particles dust is performed by submersion.

Appendix 4.20: Development of equipment and device for fuel debris retrieval

Designation	(1) Test based on a fluid-pressure manipulator
Objective	Checking the basic feasibility of the control characteristics helpful in designing a manipulator for removing fuel debris, and identifying their challenging items
Test details	• Evaluating the effect of the hose length (10 m or 100 m) based on the assumption that a long distance is required between the fluid-pressure controller and access unit when the latter must be remotely controlled • Evaluating the effect of the load (15 kg) to be mounted at the end of the manipulator • Setting the moving speed of the laser end as the operating speed (2 mm/second or so)
Present situation	• Element tests are being conducted. • A joint research with a university is underway concerning improvement of the control performance of fluid-pressure manipulators
Evaluation and challenges	• In element tests conducted in fiscal 2015, the required target end positioning accuracy was not achieved. To improve the accuracy, a pressure feedback system was added.

Overview

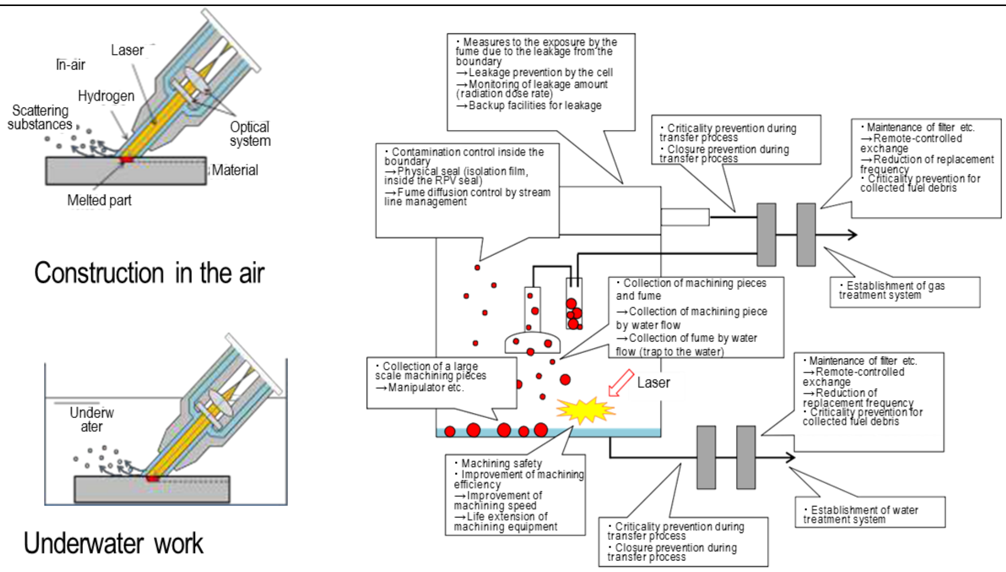


Configuration of the fluid-pressure manipulator

Photos and drawings provided by IRID

Designation	(2) Development of technologies for debris cutting, dust collection, and vision/measurement
Objective	• For easy removal, fuel debris must be processed into easy-to-handle sizes. Identifying the characteristics of cutting technologies that seem to be effective for fuel debris processing and developing technologies for collecting dust and aerosols produced during processing • Developing a radiation-proof camera to be used in the environment of fuel debris removal
Test details	• Improvement bit: Checking the stability in early stages of cutting and the amount of chips to be produced, and verifying the processibility of the mock fuel debris • Non-core bit: Checking the processibility of mock fuel debris • Laser: Measuring the processing efficiency and the weight and the particle size distribution of secondary products (fumes) moving into water and air Analytic evaluation to select a method for preventing fumes that move into the air from spreading Checking the effect of removing fume from gases • Camera: Irradiation test on the pickup camera tube and radiation-resistant camera
Present situation	Element tests are being conducted.
Evaluation and challenges	• In element tests conducted in fiscal 2015, there were problems on the amount of chips produced and the stability in early stages of cutting. • For the vision technologies, a pickup camera tube is expected to achieve a required radiation resistance.

Overview



Conceptual rendering of the laser cutting technique

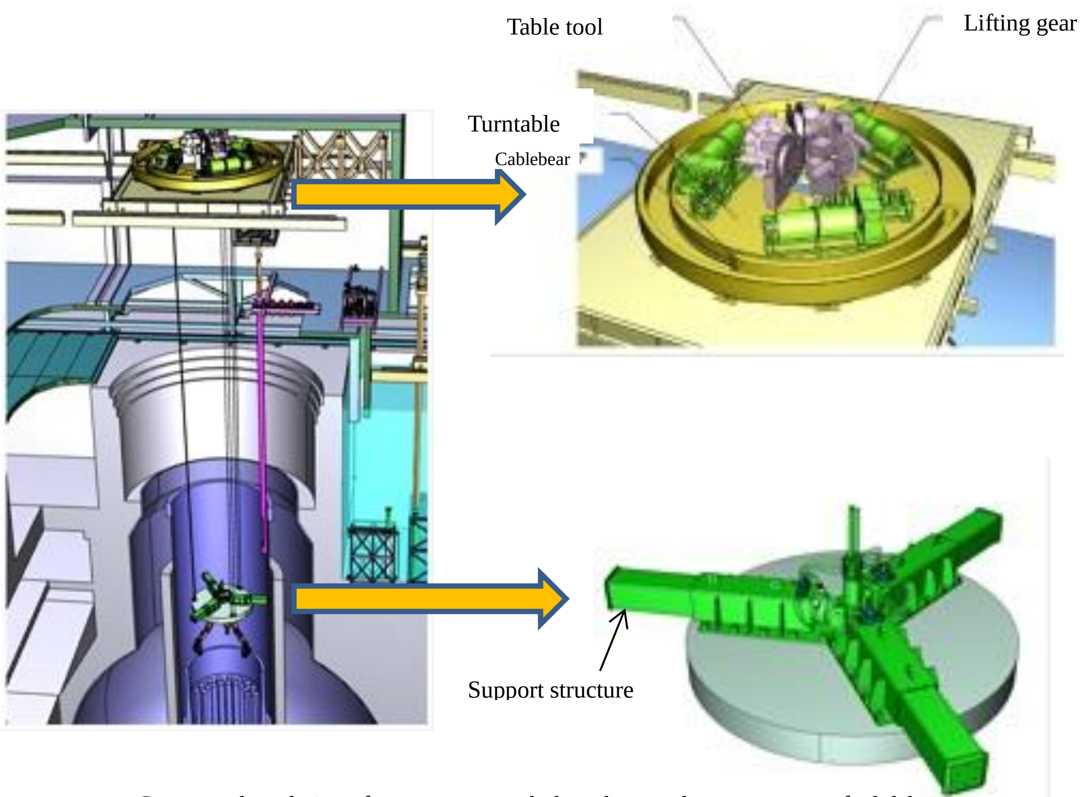
Conceptual rendering of bits

Development of visual systems

Photos and drawings provided by IRID

Designation	(3) Development of an access unit to be used in the RPV
Objective	Developing a device that brings the fuel debris handler handing device nearby fuel debris and supports the reaction force during fuel debris removal (processing and collection)
Test details	• Prototyping a 1/1 scale element of the lower table, which is a common platform for work inside the RPV • Checking the basic action of the support mechanism as well as the performance of remove at a (single) malfunction • Identifying the most appropriate supporting method (contact or pressing support) • Prototyping an upper table and conducting a test on it in or after fiscal 2017 reflecting the design conditions, such as the interfaces cell
Present situation	A prototype device is being designed.
Evaluation and challenges	• Maintaining or replacing contaminated wires and others • Synchronous control for keeping the lower table horizontal • Suppressing vibration of the lower table during slewing Executing function design, system design, and equipment design, and checking their feasibility in a mockup test.

Overview

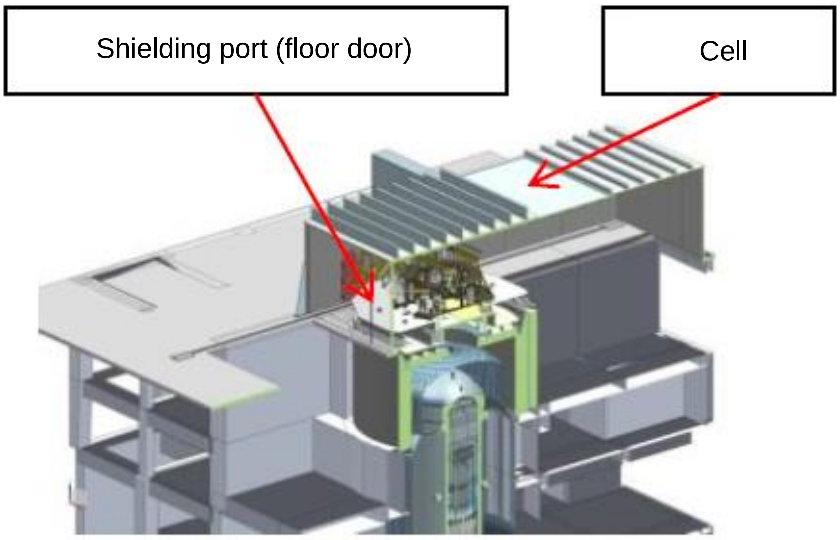


Conceptual rendering of a system to reach the submerged area to remove fuel debris

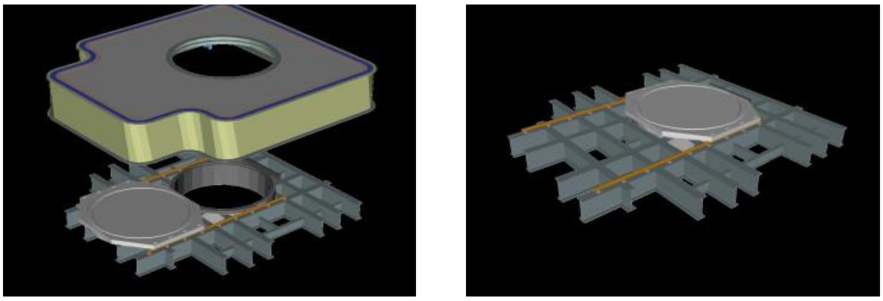
Photos and drawings provided by IRID

Designation	(4) Development of a platform and cell
Objective	Developing a system that contains the radioactive substances generated from the PCV and shielding from radiation by using a floor door installed between the cell installed on the refueling floor and the PCV
Test details	• Prototyping a 1/1 scale element of the shielding port • Checking the action of the opening/closing door • Checking the airtightness of the seal
Present situation	A test device is being prototyped.
Evaluation and challenges	A comparison and evaluation have been executed the concepts of the fixed and mobile cells from the viewpoints on safety, impact on the buildings and containers, and workability; as the result, a fixed-type cell has been selected because it is more advantageous in maintaining the boundary for controlling the release of radioactive substances and preventing contamination from spreading.

Overview



Conceptual rendering of a fixed-type cell

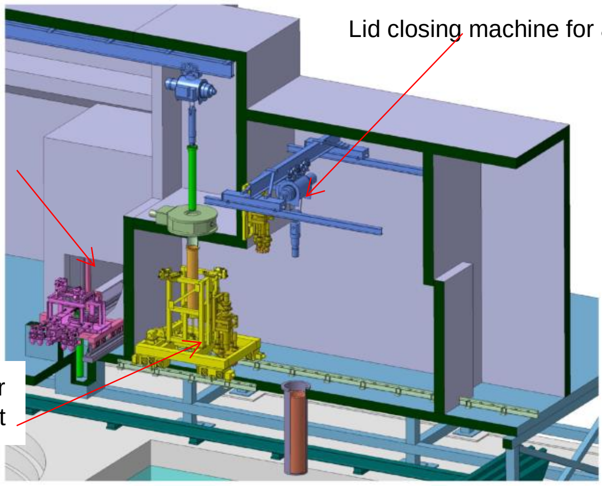


Conceptual renderings of a shielding port

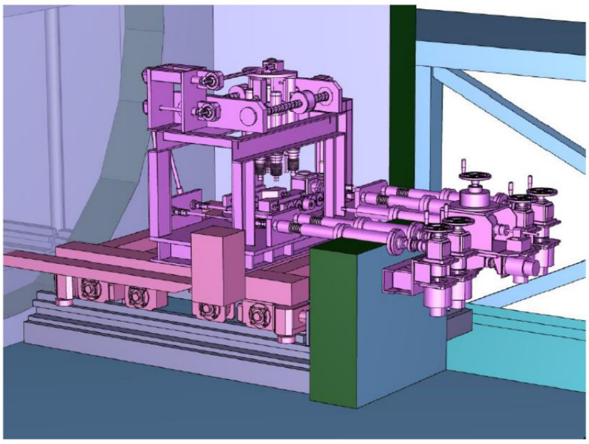
Photos and drawings provided by IRID

Designation	(5) Development of equipment for handling container cans containing fuel debris
Objective	Developing equipment that can be remotely controlled to transfer container cans to the RPV, close the lids, and clean the surfaces of the cans
Testing details	• Prototyping a 1/1-scale element of a device for closing the lids of container cans • Checking the basic action of the bolt tightening mechanism • Checking the appropriate tightening method and procedure • Checking the easiness of removing, disassembling and carrying out for maintenance of the remote device
Present situation	A testing device is being designed.
Evaluation and challenges	Design conditions have been determined for the equipment for handing container cans. But, the design must be executed inconsideration of the entire removal system.

Overview



Conceptual rendering of equipment for handling container cans

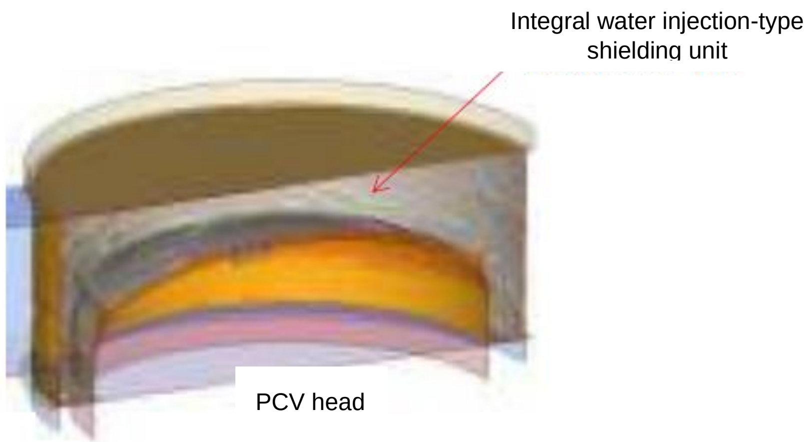


Conceptual rendering of equipment for closing the lids of container cans

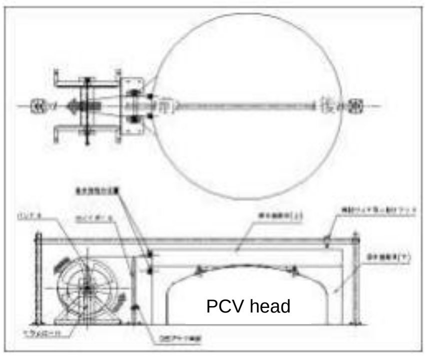
Photos and drawings provided by IRID

Designation	(6) Development of a lightweight shape-following shield
Objective	Developing equipment for facilitating installation and removal of a shield to be installed between the refueling floor and PCV in the upper-access method
Test details	• Reviewing the required strength through stress analysis based on distortion simulation • Reviewing the removing and recovering method a installed water-filled shield • Reviewing the shape of the drainage nozzle and drainage pressure conditions
Present situation	A lightweight shield is being prototyped
Evaluation and challenges	In element tests conducted in fiscal 2015, a simulation has shown that sections under relatively high stress are likely to break. The sheet must be enhanced in strength and also be split and the cover for a hanging device should be considered

Overview



Sample of distortion simulation in the case where shielding unit of water injection type is applied to the PCV head



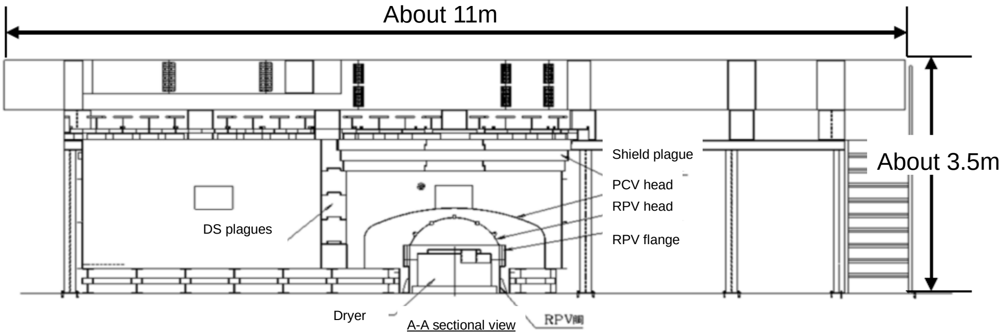
Collection procedures for water injection type shielding unit

Testing scene

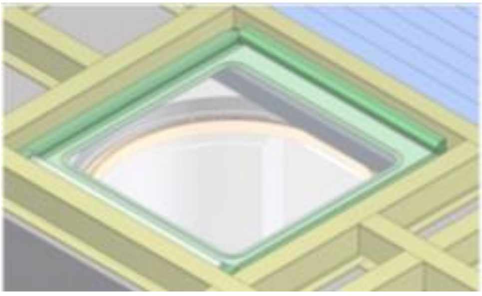
Photos and drawings provided by IRID

Designation	(7) Development of a method utilizing films and sheets for preventing contamination from spreading
Objective	Developing equipment for partitioning work areas and preventing radioactive dust from spreading removal of large structures
Test details	• Checking the mechanisms of equipment for preventing contamination from spreading, large opening/closing equipment, and remote equipment using models of an approximately 1/4 scale model of the actual equipment and checking the operating procedure for the equipment • Checking the airtightness of film and sheet to be used as partitions between areas • Checking the weldability and airtightness of contaminated equipment during cure
Present situation	1/4-scale models and test devices are being prototyped
Evaluation and challenges	In element tests conducted in fiscal 2015, it has been verified that equipment sealed with a polyurethane sheet can maintain an air pressure of 200 Pa. Air leakage has been observed in two of three tests. The repeatability of leakage test should be reconsidered

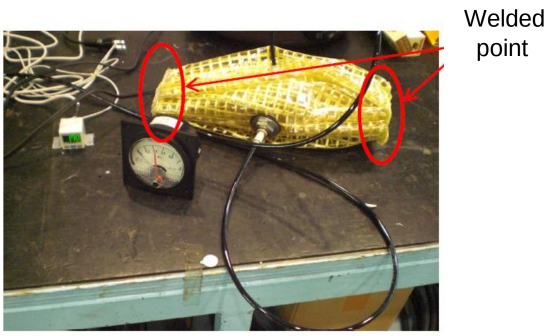
Overview



Scheme drawing of a 1/4-scale model



Conceptual rendering of an area separation sheet installed

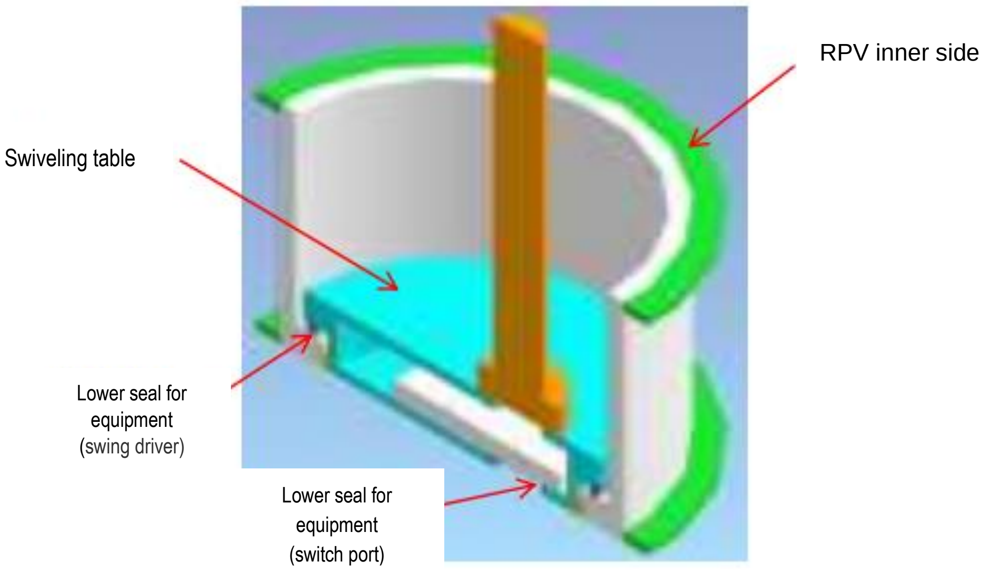


Airtightness test on a cylindrical sheet

Photos and drawings provided by IRID

Designation	(8) Development of a sealing technology for the access unit to be used in the RPV
Objective	Developing equipment for preventing radioactive dust spreading at access unit and also at the interface between the access unit and RVP, in the in-air upper access method
Test details	• Checking the sealing performance by partial simulation tests (including tests on slewing and opening and closing of the port) using a full-scale model • Checking the sealing mechanism inside the RPV • Checking the sealing mechanism at the bottom of the equipment
Present situation	A test device is being prototyped.
Evaluation and challenges	Reviewing a sealing method that keeps the seal performance under high frequent operations

Overview



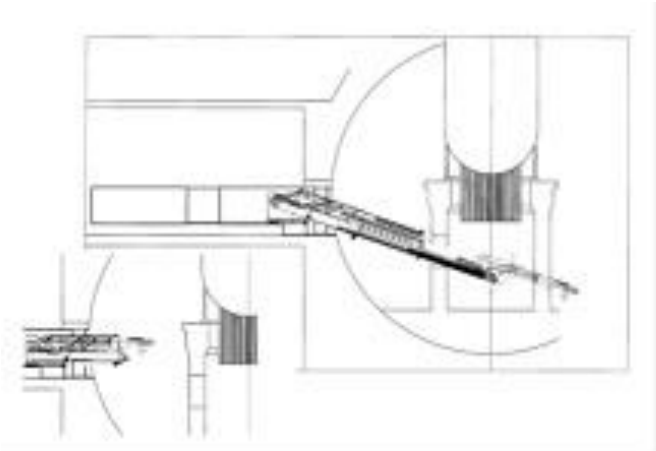
Conceptual rendering of the equipment inside the reactor

Element tests conducted on seal inside the RVP

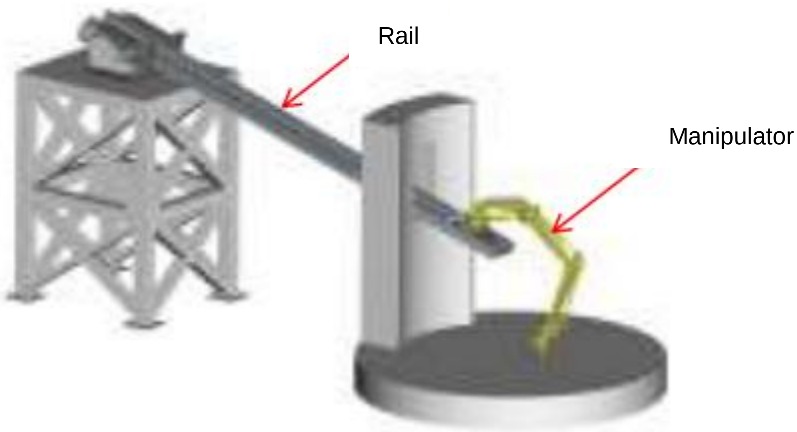
Photos and drawings provided by IRID

Designation	(9) Development of an access unit inside the pedestal
Objective	Checking the basic feasibility of a series of processes from installation of an access rail inside the pedestal to cutting of fuel debris using a robot arm the horizontal access method
Test details	• Checking the positioning accuracy of the robot arm (in with the conditions of end load of 2 t and an arm length of 6.5 m) • Checking whether a rail can be installed remotely • Checking the accessibility to the inside of the pedestal • Checking the cutting action inside the pedestal
Present situation	A test device is being prototyped.
Evaluation and challenges	A joint research with a university is underway concerning the design of and tuning procedure for the control system for the hydraulic equipment of large manipulator and the control performance of fluid-pressure manipulators. The results shall be reflected.

Overview



Conceptual rendering of fuel-debris removing equipment for in-air horizontal access

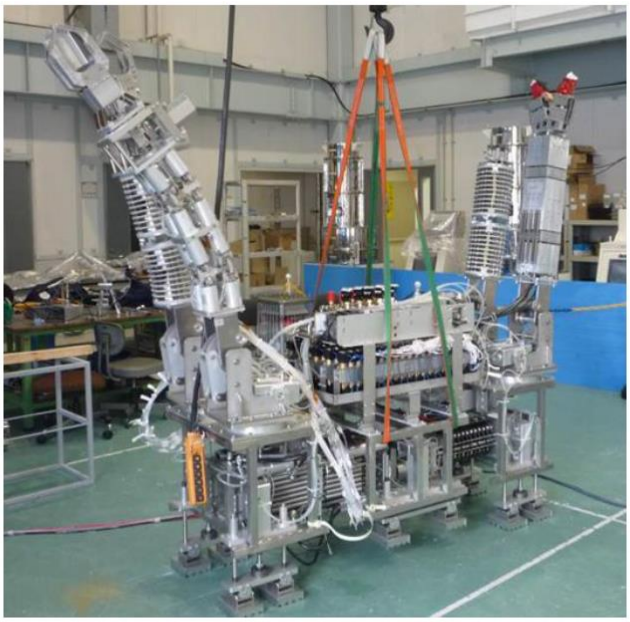


Outline drawings of the mockup test device for in-air horizontal access

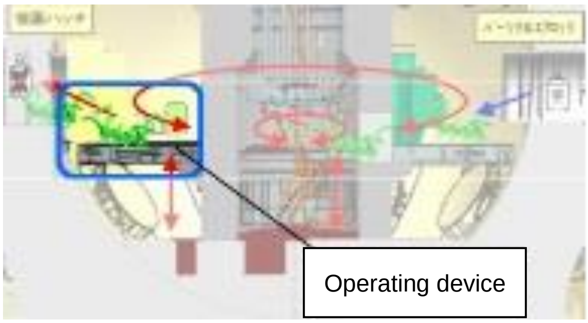
Photos and drawings provided by IRID

Designation	(10) Development of a flexible-structure arm for remote operation
Objective	Developing the arm and its ancillary devices for removing obstacles that block fuel debris removal work inside the PCVs for the horizontal access method
Test details	• Checking the accessibility, remote operability, and handleability by using mockup facility simulating the condition of the pedestal inside • Conducting element tests with the in-air horizontal removal method in mind to check the applicability of the method under review for removing fuel debris and devices
Present situation	Equipment based on the actual specifications is being fabricated
Evaluation and challenges	In element tests conducted in fiscal 2015, the first prototype equipment was used to check actions such as its mobility (self-propelled), retaining steel with two arms and cutting it with a grinder, and passing equipment and materials between arms.

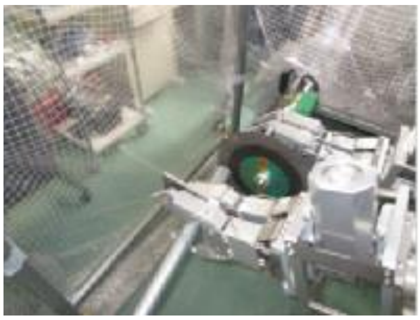
Overview



First prototype equipment



Conceptual rendering of access to the inside of the PCV

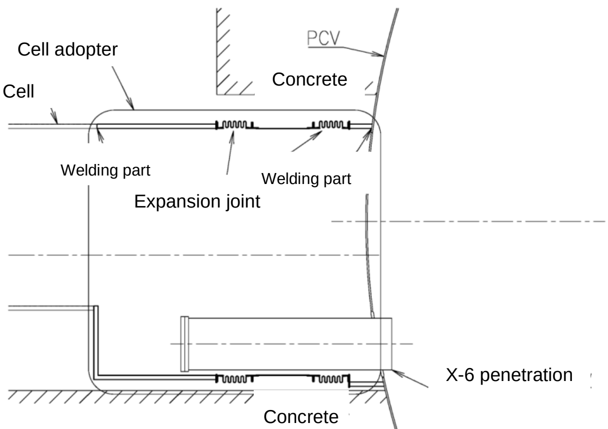


Steel being cut with a grinder

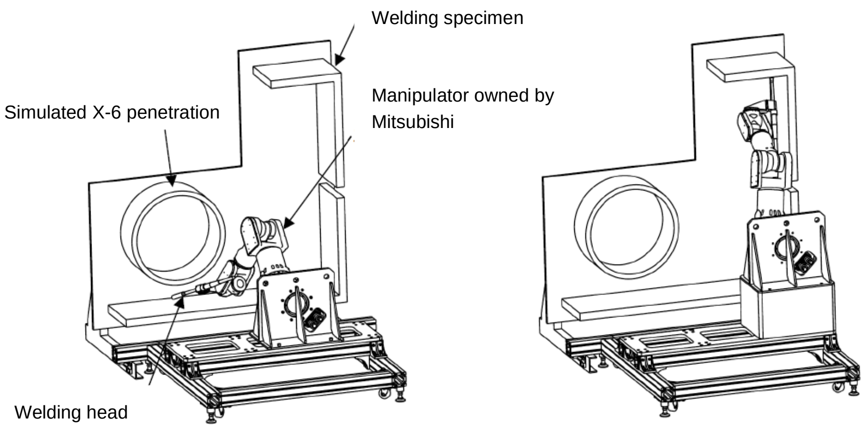
Photos and drawings provided by IRID

Designation	(11) Development of PCV welding equipment for remote seal welding the cell
Objective	Developing welding equipment for building boundaries between the cell being installed at a side of the PCV and the PCV connection in the horizontal access method
Test details	• Checking the feasibility of welding for the narrow section simulating the actual working conditions • Checking the quality of cross sections of welds • Checking the tensile strength of weld samples • Weld performance of 30-m hydraulic head pressure (0.3 MPa)
Present situation	A test device is being prototyped.
Evaluation and challenges	Under the Project for Sophistication of the Method and System for Removing Fuel Debris and PCV internal structures, the applicable method for checking the soundness after welding is being reviewed from the viewpoints of, for example, the checking procedure (methods), welding quality, and working environment. Considering the high radiation environment that the checking method shall be reviewed under many constraints.

Overview



Conceptual rendering of the cell adaptor



Conceptual rendering of the welding test device

Photos and drawings provided by IRID

Appendix 5.1: Management status of radioactive solid waste and storage plan

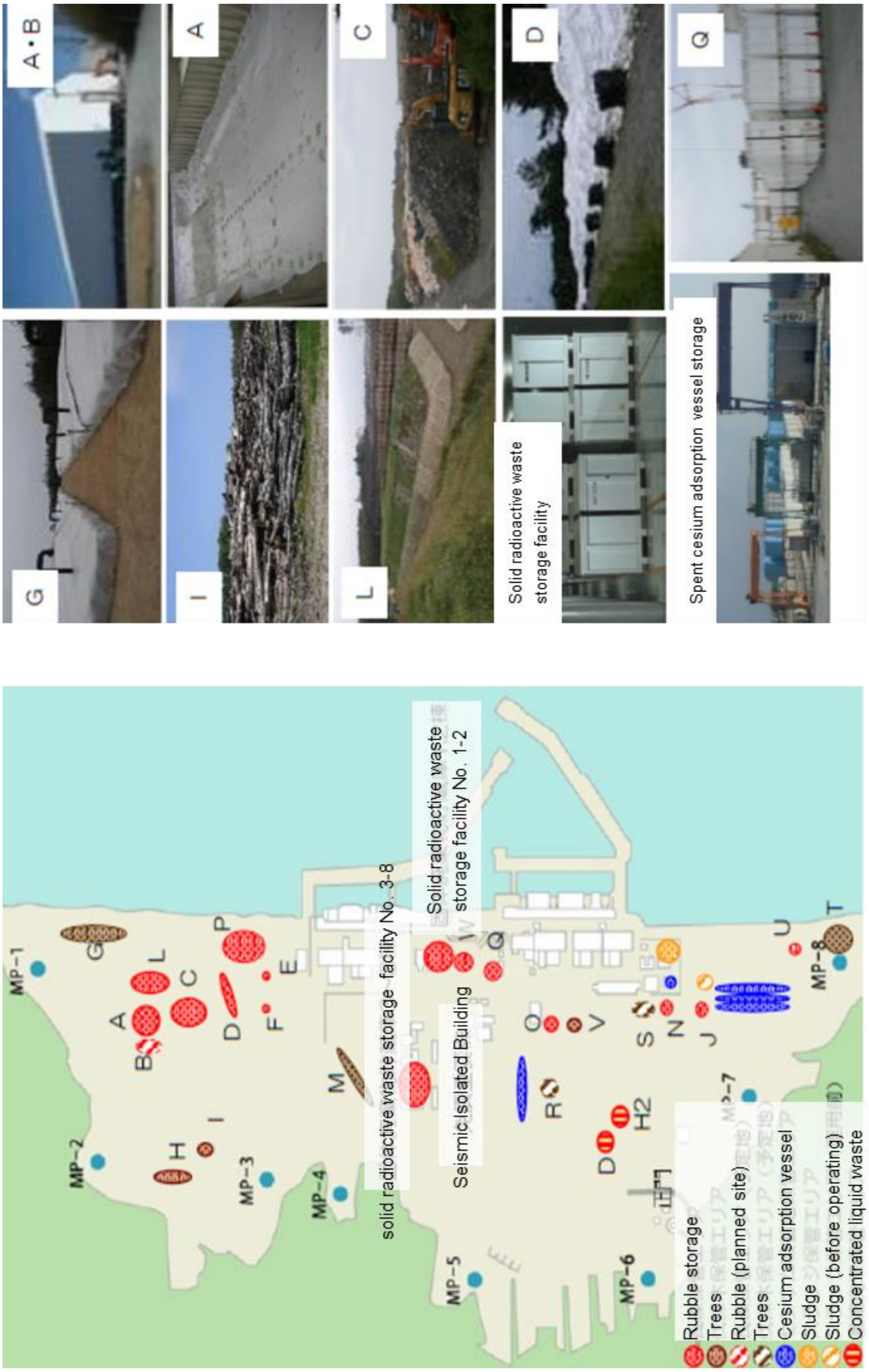


Figure A5.1-1: Storage of rubble and secondary wastes from water treatment

Source: TEPCO Document 3-4: Processing and Disposal of Radioactive Waste "Status of the Management of Rubble and Cut-down Trees (as of February 29, 2016)", Meeting of the Secretariat of the Team for Countermeasures for Decommissioning and Contaminated Water Treatment (28th Meeting), March 31, 2016.

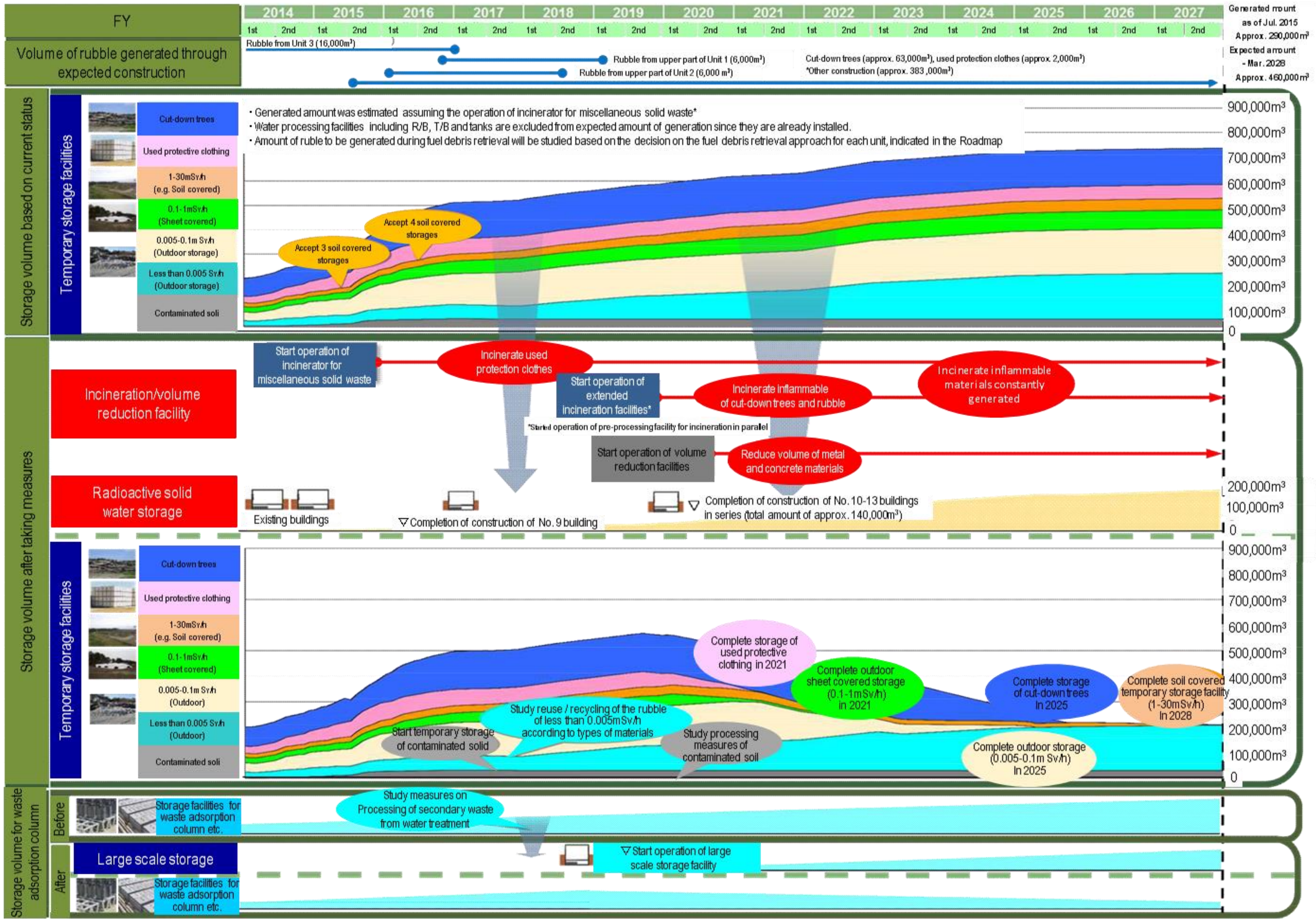


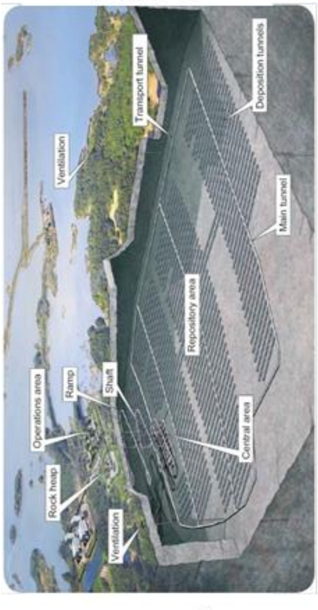
Figure A5.1-2: Conceptual Rendering of the Entire Storage Plan for Solid Wastes at the Fukushima Daiichi NPS

Source: TEPCO Document 3-4: Processing and Disposal of Radioactive Waste

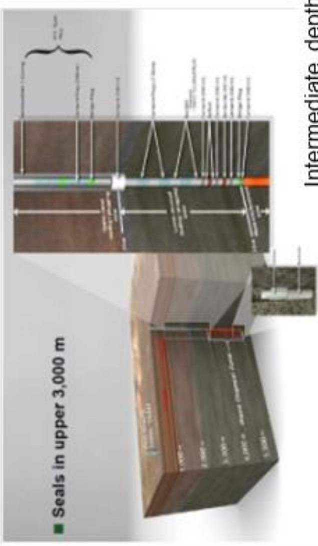
"Storage Plan for Solid Wastes at the Fukushima Daiichi NPS of Tokyo Electric Power Company, Incorporated (main body)", page 18, Meeting of the Secretariat of the Team for Countermeasures for Decommissioning and Contaminated Water Treatment (28th Meeting), March 31, 2016.

Appendix 5.2: Disposal facilities in Japan and abroad

Geological disposal:
Finland, Sweden,
U.S., Belgium, France,
U.K., Japan etc.
(Case of Sweden[1]) →



Pit disposal: France,
India, Spain, U.K.,
U.S., Japan etc.
(Case of France[5]) →



Deep borehole disposal:
IAEA, U.S.
←(Studies in the U.S.[2])

Trench disposal: France,
India, Spain, U.K., U.S.
South Africa etc.
← (Case of France[6])

Intermediate depth disposal :
Germany, Japan
←(Case of studies in Japan[3])

Monolith disposal: Belgium,
France (Case of France[7]) →

Rock Cavern disposal:
Finland, Sweden
(Case of Finland[4]) →

Tumulus disposal: France, Canada,
Sweden, U.K. and U.S.
← (Case of Sweden[8])

Figure A5.2-1: Examples of disposal facilities in Japan and abroad

Sources of the information shown in Figure A5.2-1

[1] Environmental Impact Statement, Interim storage, encapsulation and final disposal of spent nuclear fuel, (SKB, March 2011): p 206

[2] Timothy Gunter, U.S. D.O.E., “Deep Borehole Disposal Research and Development Program”, International Technical Workshop on Deep Borehole Disposal of Radioactive Waste, Washington, D.C. October 20-21, 2015.

[3] Overview of Marginal-depth Burying, Material 2 of January 19, 2011 of the working group on burial disposal techniques (6th), waste safety subcommittee, task force on nuclear safety and security of Advisory Committee for Natural Resources and Energy, Federation of Electric Power Companies of Japan

[4] Posiva Oy, “Nuclear waste management of the Olkiluoto and Loviisa nuclear power plants, Summary of operations in 2012.”

[5] J.L. Tison, “40 Years of operation of Near Surface Repositories. Andra Experience.”, 2009 CEG WORKSHOP FEB 24-26 BOMMERSVIK.

[6] Energy Solutions Website (http://www.energysolutions.com/waste-management/facilities/).

[7] Jean-Pierre VERVIALLE, “Historical Background of the Operation, Closure and Monitoring of Andra’s CSM Disposal Facility”, IAEA-Andra International Workshop, Cherbourg, 23 September 2009, (2009).

[8] Alvaro Rodríguez Beceiro, “Disposal solutions implemented for VLLW”, IAEA Scientific Forum, RADIOACTIVE WASTE: MEETING THE CHALLENGE, Science and Technology for Safe and Sustainable Solutions, 23-24, Sep., Vienna, Austria, (2014).

Abbreviations and short forms

Abbreviations and short forms	Definitions
AC	Atmospheric control System
CEA	Commissariat a l'energie atomique et aux energies alternatives: Commissariat a l'energie atomique et aux energies alternatives in France
CRD	Control Rod Drive
CS	Core Spray System
CST	Condensate storage tank
DHC	Drywell Humidity Control System
DOE	U.S. Department of Energy
D/W	Drywell
FDW	Feed Water System
FP	Fission Product
FS	Feasibility Study
HIC	High Integrity Container
HIC slurry	Slurry that contained in the secondary waste (high integrity container (HIC) from multi-nuclide removal system and expanded multi-nuclide removal system and advanced multi-nuclide removal system)
HVH	Heating and Ventilation Handling
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IRID	International Research Institute for Nuclear Decommissioning
JAEA	Japan Atomic Energy Agency
KEK	Kou Enerugii Butsurigaku Kenkyusho: High Energy Accelerator Research Organization
NDA	Nuclear Decommissioning Authority (U.K.)
NDF	The Nuclear Damage Compensation Facilitation Corporation
OECD/NEA	Organization for Economic Cooperation and Development/ Nuclear Energy Agency
PCV	Primary Containment Vessel
PCV internal structures	Structures and buildings including radiated materials, other than FPs as radioactive material
PLR	Primary Loop Recirculation system
RPV	Reactor Pressure Vessel
S/C	Suppression chamber
TMI-2	The Three Mile Island Unit 2: Three Mile Island Nuclear Power Plant in the U.S. Unit 2
Fuel in dry cask	Fuel assemblies stored in the Dry casks
Common pooled fuel	Fuel assemblies stored in the common pool
R&D Duties Execution Policy	Approaches to the R&D of the technologies required for decommissioning
Implementation Plan	Implementation plan on the Fukushima Daiichi NPS, which is Specified Nuclear Facility
Heavy nuclide	Actinide nuclides such as uranium and plutonium
The Strategic Plan	Technical Strategic Plan for Decommissioning of the Fukushima Daiichi Nuclear Power Station of Tokyo Electric Power Company Holdings, Inc.
Matters to be addressed	Matters which are requested to be taken by Tokyo Electric Power Company, Inc. when the Fukushima Daiichi NPS was designated as Specified Nuclear Facility.

Contaminated water in the buildings	Highly contaminated water accumulated in the R/B and seawater piping trenches
The Roadmap	Mid-and-Long-Term Roadmap towards the Decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Station Units 1-4
TEPCO	Tokyo Electric Power Company Holdings, Inc.
Solid waste in the storage	Solid radioactive waste generated by rubbles and fallen trees stored in the storage facilities.
The Fukushima Daiichi NPS	Fukushima Daiichi Nuclear Power Station of Tokyo Electric Power Company Holdings, Inc.
Waste sludge	Secondary waste in the sludge storage tanks of decontamination equipment
Waste adsorption column	Secondary waste generated from cesium and second cesium adsorption system.
Activated structure	Activated material inside the RPV/PCV

Terms

Terms	Definitions
PDCA cycle	A method to facilitate management task for production control and quality control in business activities. Repeating four steps which are Plan→Do (implementation) →Check (evaluation) →Action (improvement), improvements are made to the operation continuously.
Well shield plug	A concrete lid on top of PCV installed for radiation shielding (a part of the floor of the uppermost floor of the R/B during operation)
Submersion method:	A retrieval method by filling to the top of the PCV with water to submerge the fuel debris
Partial submersion method:	A method to retrieve the fuel debris exposed in the air and without filling the PCV with water
Technology Readiness Level	Index to indicate the level of technical development (TRL)
Clearance	Clearance system refers to a system under which the government confirms that the concentration of radioisotopes of materials used in a nuclear facility is below the "Clearance Levels" (the level at which the impact is negligible to human health). The materials confirmed by the government are removed from the regulations on nuclear reactors, and will be subject to regulations under laws on wastes and recycle as conventional industrial wastes or valuables
Actual debris	Actual fuel debris retrieved from the reactor vessels as opposed to simulated debris
Defense in depth	One of the fundamental safety principles which means that all safety activities are subject to multiple layers of overlapping provisions, so that if a failure should occur it would be detected and compensated for or corrected by appropriate measures
Fuel debris	Nuclear fuels molten and mixed with parts of reactor internals due to loss of reactor coolant and resulted in a re-solidified state
Hazard potential	Hazard potential Level of impact that the harmful materials may bring about
Fuel debris detection using muon system	A technology to identify the locations and shapes of fuels using the behavior of cosmic ray muons from space and the atmosphere to change the number of particles and trajectories when penetrating a material according to its density
Simulated debris	Artificial objects manufactured by estimating the chemical composition and forms based on the examples of TMI-2 accident
Robustness	The capability to maintain the robust function even when the condition is changed to a certain extent from what is expected