Technical Strategic Plan 2021 for Decommissioning of the Fukushima Daiichi Nuclear Power Station of Tokyo Electric Power Company Holdings, Inc.

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Nuclear Damage Compensation and

Decommissioning Facilitation Corporation

Looking back on NDF's efforts to the decommissioning of Fukushima Daiichi Nuclear Power Station

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March 2021 marked the 10th anniversary of the Fukushima Daiichi Nuclear Power Station accident. Among the ongoing reconstruction and decontamination efforts led by those in the affected areas and other stakeholders over the past ten years, action to bring the power station accident to an end and to decommission the reactors has been continuous. As a result, these measures have been successful at lowering radiation risk, and have allowed work intended to decommission the reactors over the mid- to long-term to proceed steadily.

Established in 2011, the Nuclear Damage Compensation Facilitation Corporation has conducted research and development, as well as provided advice, guidance, and recommendations, required for reactor decommissioning following legal reforms executed in 2014, and has served as the Nuclear Damage Compensation and Decommissioning Facilitation Corporation (NDF).

Nevertheless, we must take seriously the fact that many of those from the evacuation order area are still living in refuge. Under the fundamental principle of "Balancing between Reconstruction and Reactor Decommissioning," NDF believes that in order to proceed in tandem with these two efforts and to reassure the people in the areas affected by the disaster, we must work tirelessly while maintaining a constant awareness of the expectations and concerns of local communities and society.

NDF has published the Technical Strategic Plan annually since 2015 with the intent of broadly sharing a technical understanding of the complex reactor decommissioning process with stakeholders and the public. The Plan has also served to clarify the technical grounds for assisting the government in formulating its reactor decommissioning policies and the Tokyo Electric Power Company Holdings, Inc. (TEPCO) in making more solid engineering judgments. In addition to sharing the latest technical information, the Plan aims to help realize a safe, reliable reactor decommissioning process in line with the government's Mid-and-Long-term Roadmap by systematically compiling mid- to long-term issues and strategies. To this end, we drew up the Plan through exchanges of views among stakeholders, including opinions from the Decommissioning Strategy Committee and the expert committees of this Corporation.

We believe that the Technical Strategic Plan has thus far been successful in helping to progress reactor decommissioning and to realize future plans. Specifically, the Plan has achieved these by making recommendations for determining fuel debris retrieval policies and finalizing fuel debris retrieval methods for the first implementing unit, as well as for compiling the basic concepts concerning solid waste processing/disposal, on the basis that risks caused by radioactive materials are to be successfully reduced in the short-term, as well as over the mid- to long-term.

In addition to these technical efforts, we have provided advice on overall project management, thereby enabling TEPCO to properly and reliably proceed with decommissioning projects through the management of the Reserve Fund for Decommissioning established in 2017. Although these efforts have allowed TEPCO to begin enhancing its systems for managing and implementing projects to achieve its targets, we believe TEPCO must take further steps to enhance its overall capabilities going forward.

At the same time, in order to steadily promote the decommissioning process, which involves highly challenging issues, over the mid- to long-term, we are striving to promote an understanding among both the international and local communities. These efforts include holding the International Forum on the Decommissioning of the Fukushima Daiichi NPS to disseminate information to Japan and the world and to gather wisdom from around the globe, as well as holding dialog with local communities.

In the future, we will begin tackling new challenges such as making the shift to full-scale retrieval of fuel debris, an unprecedented undertaking anywhere in the world. NDF will provide technical assistance for the trial retrieval of fuel debris and to further expand the scale of retrieval. In addition, we are committed to contributing to the next Mid-and-Long-term Roadmap by advancing a review of solutions for issues in anticipation of what lies beyond the start of fuel debris retrieval. This review will be based on the prospects of processing/disposal method and technology related to its safety.

Finally, we would like to ask for your ongoing support as all stakeholders continue to work under the all-Japan framework to make steady progress in their efforts regarding decommissioning.

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1. Introduction

The overall approach to the decommissioning of the Fukushima Daiichi Nuclear Power Station of Tokyo Electric Power Company Holdings, Inc. (hereinafter referred to as the “Fukushima Daiichi NPS”) started under the Mid-and-Long-term Roadmap[1] towards the Decommissioning of TEPCO’s Fukushima Daiichi NPS Units 1 to 4, released by the Japanese Government in December 2011, and it has been 10 years in March 2021.

The most urgent issues, such as treating contaminated water and removing fuel from the spent fuel pools (hereinafter referred to as “fuel removal from SFP”), have been given top priority in this effort. However, to complete the decommissioning, long-term measures are required including fuel debris retrieval work, and so it is essential to prepare a mid- and long-term strategy. On August 18, 2014, the former Nuclear Damage Compensation Facilitation Corporation was reorganized into the Nuclear Damage Compensation and Decommissioning Facilitation Corporation (hereinafter referred to as “NDF”), a new organization responsible for technical studies needed to proceed with the decommissioning properly and steadily from the mid- to long-term perspective. NDF’s duties include, in addition to those assigned to its forerunner, conducting R&D of decommissioning technologies, and providing advice, guidance and recommendations for ensuring the appropriate and steady implementation of the decommissioning.

At present, contaminated water management, which required urgent measures immediately after the accident, have been stabilized, including the reduction of the amount of contaminated water generated and the completion of the treatment of stagnant water in the buildings (other than the reactor buildings of Units 1-3, the main process building, and the high-temperature incinerator building). Fuel removal from SFPs has been completed in Units 3 and 4, the decommissioning work is steadily in progress. In addition, countermeasures against disasters such as earthquakes and tsunamis are progressing. During the earthquake that struck off the coast of Fukushima Prefecture on February 13, 2021, the water level in the primary containment vessels of Units 1 and 3 dropped, and the medium- and low-concentration tanks and storage tanks for stagnant water in Units 5 and 6 slid (shifted), but As for the ALPS-treated water, the government has announced a policy of discharging it into the ocean, on the premise that safety will be ensured and that measures against reputational damages will be thoroughly taken.

On the other hand, due to the impact of the new coronavirus infection, the fuel debris retrieval from Unit 2, which was scheduled to start within 2021, is expected to be delayed by about one year, and efforts are currently being made to minimize this delay. The Tokyo Electric Power Company, Incorporated (hereinafter referred to as “TEPCO”) has updated its Mid-and-Long-term

[1] The first edition was formulated in December 2011 as a way for the government to define the basic concept and main target processes for promoting the decommissioning and the contaminated water management of Fukushima Daiichi NPS. It has been revised five times based on the progress of the decommissioning and the contaminated water management, and the latest 5th edition (revised on December 27, 2019) can be found at the following link.
https://www.kantei.go.jp/jp/singi/hairo_osensui/dai4/siryou2.pdf

Decommissioning Action Plan, which was announced in 2020, in March 2021 to specify the outlook for decommissioning work based on the above.

The "Technical Strategic Plan 2021 for Decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Station" (hereinafter referred to as the "Technical Strategic Plan 2021") describes the issues to be addressed for the trial retrieval of fuel debris to minimize the impact of the new coronavirus infection, the issues to be discussed for the selection of a method for further expansion of retrieval scale, and the efforts for the ALPS-treated water, while offering the prospects of processing/disposal method and technology related to its safety (hereinafter referred to as the ”Technical Prospects”), that was to be presented around FY2021 in the " Mid-and-Long-term Roadmap for Decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Station" (hereinafter referred to as the "Mid-and-Long-term Roadmap").

With regard to the impact of the new coronavirus infection, which has been ongoing since FY2020, TEPCO has established a task force to deal with the new coronavirus infection, and is implementing measures based on the business continuity plan and measures to prevent infection and spread[2]. Although there have been cases of infected employees and workers engaged in the work at the Fukushima Daiichi NPS, the safety of decommissioning work has not been affected. There have been delays in the start of the trial retrieval of fuel debris and the lack of direct communication regarding international cooperation and regional symbiosis, but efforts are being made to minimize these impacts.

1.1 Structures and systems toward the decommissioning of the Fukushima Daiichi Nuclear Power Station

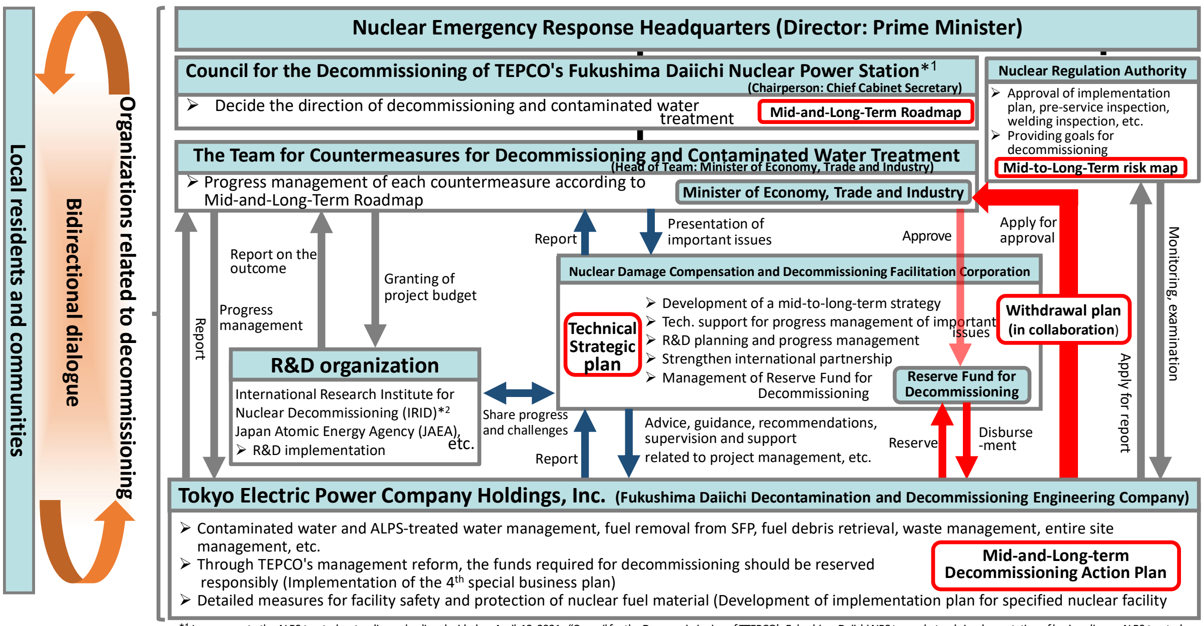
In order to make systematic and steady progress in addressing issues of decommissioning work over the mid-to-long-term, TEPCO has been working to build and strengthen the project management structure. In April 2020, though its organization was reorganized, and the management structure and scheme were established, after this, it is important to enhance and upgrade the management methods and make them effective and rooted in the field operations. From the financial perspective, the Nuclear Damage Compensation and Decommissioning Facilitation Corporation has been carrying out the management of a reserve fund for decommissioning since October 2017 to ensure immediate decommissioning work. The management task aims are, in every fiscal year, (1) TEPCO will deposit the amount at NDF that is specified by NDF to implement decommissioning appropriately and steadily, as well as that approved by the Minister of Economy, Trade and Industry, and (2) based on the “Withdrawal Plan for Reserve Fund for Decommissioning” (hereinafter referred to as “Withdrawal Plan”), that was jointly prepared by NDF and TEPCO and approved by the Minister of METI, TEPCO will withdraw the reserve fund and implement decommissioning. At present, three years have passed since the

[2] TEPCO, “Status of the outbreak of New Coronavirus infection in our group” https://www.tepco.co.jp/press/news/2021/1612325_8971.html

decommissioning reserve management was introduced, and it is contributing to the proper and steady implementation of decommissioning. (Fig.1)

Under this management task, NDF will (1) appropriately manage the fund for decommissioning, (2) manage the implementation structure for proper decommissioning and (3) manage the decommissioning work based on the Reserve Fund for Decommissioning appropriately, and NDF assumes responsibility as an organization to manage and oversee TEPCO's decommissioning activities. NDF prepared "the Policy for Preparation of Withdrawal Plan for Reserve Fund for Decommissioning" (hereinafter referred to as "The Policy for Preparation of Withdrawal Plan"), which was drawn up based on the "Technical Strategic Plan", and presented to TEPCO the work goals and main activities to be incorporated in the Withdrawal Plan, and evaluated the appropriateness of TEPCO's efforts in the process of jointly preparing the Withdrawal Plan from the perspective of symbiosis and communication with the community, etc. (Fig. 2).

In addition to the operation of these systems, the division of roles among the organizations directly involved in the decommissioning of the Fukushima Daiichi NPS, including the Japanese government, NDF and TEPCO, as well as organizations specializing in R&D, such as the International Research Institute for Nuclear Decommissioning [IRID] and the Japan Atomic Energy Agency [JAEA], is shown in Fig.1, which also indicates how the abovementioned systems are implemented. Among these roles, R&D are discussed in Chapter 5, and dialogue with local residents and communities is described in Chapter 6.



*1 In response to the ALPS-treated water disposal policy decided on April 13, 2021, "Council for the Decommissioning of TTEPCO's Fukushima Daiichi NPS toward steady implementation of basic policy on ALPS-treated water disposal" was founded.

*2 TEPCO, a decommissioning project operator, participates as a member of IRID and shares the needs, challenges, and results of research and development.

Fig.1 Division of roles of related organizations responsible for decommissioning of the Fukushima Daiichi NPS

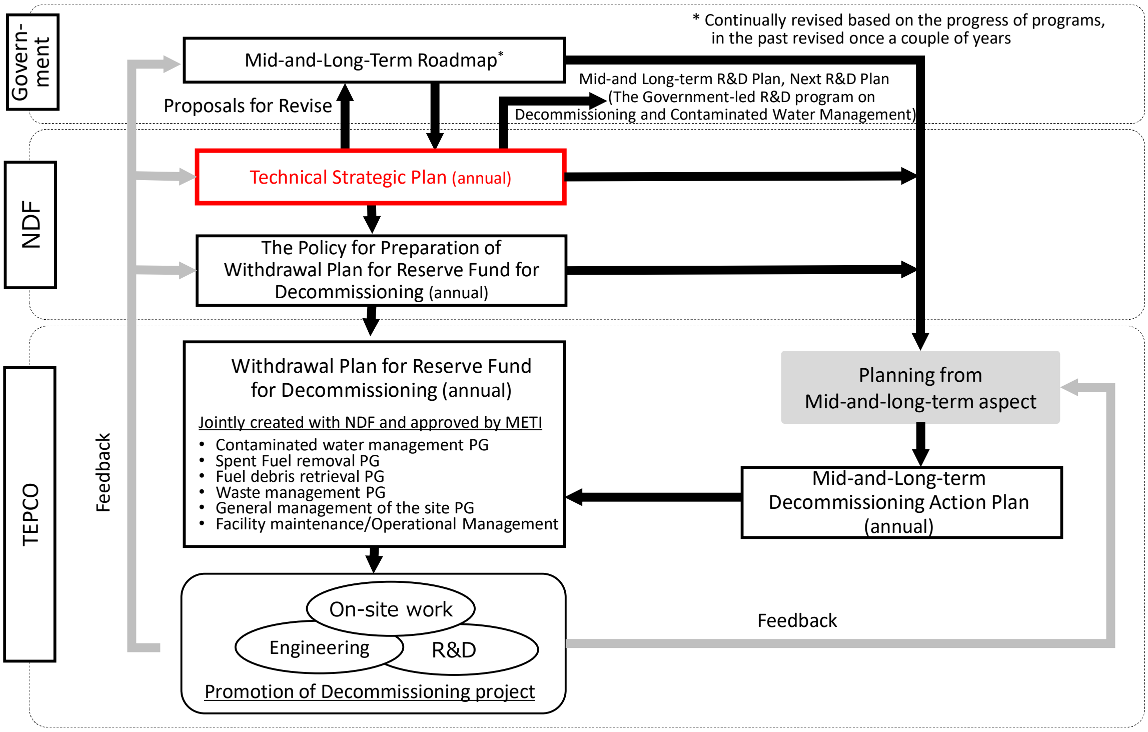


Fig. 2 Positioning of the Technical Strategic Plan based on the Reserve Fund

1.2 The Technical Strategic Plan

1.2.1 Positioning of the Technical Strategic Plan

NDF has been compiling the Technical Strategic Plan every year since 2015 for the purpose of providing a solid technical basis for the Mid-and-Long-term Roadmap, contributing to its smooth and steady implementation, consideration of revisions, achievement of the goals of the Nuclear Regulation Authority (NRA)'s "Target Map for Reducing Medium-Term Risk at TEPCO's Fukushima Daiichi NPS," and providing a basis for the Policy for Preparation of Withdrawal Plan (Attachment 1). The Technical Strategic Plan makes suggestions that contribute to the annual revision of the Mid-and-Long-term Decommissioning Action Plan from a technical perspective.

The Technical Strategic Plan 2021 presents a technical strategy from a medium-to-long-term perspective that overlooks the overall efforts at the Fukushima Daiichi NPS in order for project operators to steadily implement decommissioning work toward achieving the goals of the Mid-and-Long-term Roadmap that was revised in 2019. In particular, since fuel debris retrieval, which is a highly difficult task, is approaching, and the roles of the government, NDF, TEPCO, research institutes, etc. are becoming more significant to realize this task. The Technical Strategic Plan designed with these perspectives in mind as well.

1.2.2 Overall structure of the Technical Strategic Plan 2021

The Technical Strategic Plan 2021 consists of six chapters.

Chapter 1 (Introduction) describes the current situation 10 years after the accident and the progress of the efforts to realize the milestones of the Mid-and-Long-term Roadmap, as well as the main points of the Technical Strategic Plan 2021.

Chapter 2 (Concept on risk reduction and safety assurance for decommissioning of the Fukushima Daiichi NPS) presents the basic policy as the approach to risk reduction and ensuring safety. In implementing risk reduction strategies, it also describes the immediate targets, basic approach to risk reduction, and the approach on the order of priority, as well as the approach to ensuring safety, including the basic policy of ensuring safety based on the characteristics of the Fukushima Daiichi NPS, incorporating the safety perspective and the operator's perspective, and the preliminary implementation and utilization of the information obtained in the later stages.

In Chapter 3 (Technological strategy toward decommissioning of the Fukushima Daiichi NPS), sector-specific goals are set for each of the four areas of fuel debris retrieval, waste management, contaminated and treated water management, and fuel removal from SFP.

Section 3.1 of Chapter 3 (Fuel debris retrieval) describes the issues to be addressed for the trial retrieval to minimize the effects of the new coronavirus infection and the gradual expansion of the retrieval scale, as well as the efforts to study the retrieval method for further expansion of the retrieval scale.

In Section 3.2 of Chapter 3 (Waste management), in accordance with the basic concept of solid waste processing/disposal, in addition to the Technical Prospects presented in the Technical Strategic Plan 2021, issues based on the achieved status and matters to be implemented thereafter are described.

Section 3.3 of Chapter 3 (Contaminated water and treated-water management) describes the initiatives and direction of countermeasures for the newly detected relatively high total alpha nuclide, reinforcement and renewal of the entire water treatment system from the perspective of service life and aging, and issues for discharge of ALPS-treated water into the ocean.

Section 3.4 of Chapter 3 (Fuel removal from spent fuel pools) describes the new goals for fuel removal from SFPs and the appropriate and specific work plan depending on the situation of each unit. This section also describes the direction of efforts to determine the future processing and storage methods, such as securing the necessary capacity to properly store the removed fuel on site and evaluating the long-term integrity of the fuel in SFP.

In Chapter 4 (Analysis strategy for promoting decommissioning) shows, as the significance and system of analysis, the importance of building and maintaining analytical facilities and functions required for handling waste and fuel debris, and establishing a system including human resource development, as well as the characteristics of fuel debris, the policy for reducing uncertainty and the issues of sample analysis and the policy for handling them.

Chapter 5 (Efforts to facilitate research and development) describes the individual research and development described in Chapters 3 and 4 with a view to the medium and long term as a whole, and summarizes the approaches expected of the government, project operators, and related research organizations. This chapter also describes the importance of ensuring the continuity of research and development and accessibility to past results after the termination of IRID, and the efforts to strengthen the system for these purposes.

Chapter 6 (Activities to support our technology strategy) describes the significance, current status, and major issues and strategies in the following areas: further enhancement of project management, improvement of capabilities as a decommissioning operator, enhancement of international collaboration, and regional symbiosis.

Section 6.1 of Chapter 6 (Further strengthening of project management and improvement of capability required as a decommissioning executor) describes the need for constant awareness to comply with corporate ethics in response to the problem of non-conformity with the nuclear materials physical protection program of at the Kashiwazaki-Kariwa Nuclear Power Station and the importance of insourcing of technology through decommissioning work.

Section 6.2 of Chapter 6 (Strengthening of international cooperation) describes the necessity of strengthening international collaboration, such as strengthening partnerships with relevant organizations in each country that are engaged in decommissioning in overseas legacy sites, in order to concentrate wisdom in Japan and abroad, and the current status of efforts affected by the new coronavirus infection.

Section 6.3 of Chapter 6 (Local community engagement) describes the approach to be taken when related organizations, mainly TEPCO, work in cooperation with each other based on the principle of "Balancing between reconstruction and decommissioning" in order to continue decommissioning the Fukushima Daiichi NPS for a long time. This report describes the approach to be taken by TEPCO and other related organizations when they work together.

2. Concept on risk reduction and safety assurance for decommissioning of the Fukushima Daiichi NPS

2.1 Basic policy for the decommissioning of the Fukushima Daiichi NPS

<Basic policy for the decommissioning of the Fukushima Daiichi NPS>

Continuously and quickly reduce the risks arising from the radioactive materials caused by the accident that do not exist in normal nuclear power plants

The Fukushima Daiichi NPS has the necessary safety measures required by the NRA in place for the matters for which measures should be taken and it is being maintained in a state with a certain level of stability.

However, there are still enormous risks at the Fukushima Daiichi NPS because fuel debris and spent fuel still remain in the reactor buildings damaged by the accident, part of the status of the NPS has not yet been sufficiently ascertained, and the site has radioactive contaminated water and enormous amounts of extraordinary radioactive wastes. If left unaddressed, these risks may increase due to aging degradation of the facilities and other factors. Quickly and swiftly reducing these risks is an urgent matter for the NPS.

Accordingly, the basic policy for the decommissioning of the Fukushima Daiichi NPS is "to continuously and quickly reduce the risks arising from the radioactive materials caused by the accident that do not exist in normal nuclear power plants" by taking measures specifically designed to reduce risks. Generally, the following measures are effective for reducing risks at facilities where an accident has occurred; (1) Improving the containment functions of the damaged facilities; (2) Changing the properties and form of the contained radioactive materials to be more stable; and (3) Strengthening monitoring and control over the equipment to better prevent or mitigate the occurrence or propagation of abnormalities. In order to achieve these measures in an integrated way, in addition, (4) Collecting radioactive materials from the damaged facilities or insufficient containment conditions and placing them in more robust storage is effective.

Since the accident, these diverse measures for risk reduction have been taken with careful preparations aimed at preventing accidents and radioactive exposure of workers (Attachment 2).

2.2 Concept of reducing risks caused by radioactive materials

2.2.1 Quantitative identification of risks

The term "risk" has various meanings depending on the field or situation in which it is used. In general, in the context of appropriate risk management, "risk" can be understood as an expectation value of the negative impact of an event. In other words, the magnitude of a risk (risk level) posed by a subject (risk source) can be expressed as the product of the level of impact and the likelihood of occurrence of an event.

The Technical Strategic Plan uses a method based on the Safety and Environmental Detriment score (hereinafter referred to as "SED") developed by the Nuclear Decommissioning Authority (hereinafter referred to as “NDA”) to express the magnitude of risk (risk level) for radioactive materials. The risk level expressed by SED is given by the calculation formula below.

Risk Level expressed by SED = “Hazard Potential” × “Safety Management”

“Hazard Potential” here, is an index of the impact of the event, namely, the impact of internal exposure in the event of human intake of radioactive material contained in the risk source. It can be expressed as the product of Inventory, which is the amount of radioactive material contained in the risk source (taking account of toxicity of the radioactive material), and factors that depend on the form of the risk source and the time allowable until the manifestation of the risk. “Safety Management” is an index of the likelihood that an event will occur. It is determined by factors that depend on the integrity and other aspects of the facility and on the packaging/monitoring status of the risk source (Attachment 3).

The major risk sources of the Fukushima Daiichi NPS are summarized in Table 1, and Fig. 3 shows the risks of the Fukushima Daiichi NPS as the sum of these risk sources. The current risk levels assigned to the respective risk sources are expressed in Fig. 4 with “Hazard Potential” and “Safety Management” as the axes.

In the Mid-and-Long-term Roadmap, management of these risk sources is broadly classified into three major categories: (1) Relatively high risks given high priority (stagnant water in buildings and fuel in SFPs), (2) Immediate risk unlikely, but risk may grow when handling with haste (fuel debris), and (3) Increased risk unlikely in the future, but appropriate decommissioning efforts are required (solid waste such as sludge generated by the decontamination device). Their priorities are set, and appropriate measures are being taken. In Fig.4, above (1) is represented in pink, (2) in yellow, and (3) in green, with the risk sources in the “sufficiently stable management” region (in pale blue area) are shown in light blue.

Major risk sources identified at the Fukushima Daiichi NPS are shown in Table 1. In addition, the overall decommissioning work over the long term includes waste that existed before the accident and the risk sources that have low hazard potential but are not adequately controlled in a stable manner. Since the Technical Strategic Plan 2019, these issues have also been presented. In particular, regarding the facilities containing risk sources that were not expressly considered before, investigations and examinations are being conducted in consideration of external events such as earthquakes, tsunamis, and rainwater (Attachment 4).

It is also important to identify risks that have not been anticipated before. Although it is not easy to identify such risks, when an unexpected event occurs, analyzing the event to clarify causes that had not been anticipated before provides a clue for risk identification.

At the event of total-β contamination leakage in the rubble temporary storage area reported on March 25, 2021[3], leakage of radioactive materials occurred from a container whose contents were

[3] Tokyo Electric Power Company Holdings Inc., “Report on the accident event of the drainage at the shallow draft wharf and storage of rubble”, Study group on monitoring and assessment of specified nuclear facilities meeting (90th), Material 4, April 19, 2021

not identified. Before this event, it was assumed that solid content such as rubble would not immediately transfer radioactive materials to the environment due to container damage. In light of this event, however, it is important for risk identification to understand physicochemical state and its changes over time, in addition to the location of the risk sources and radioactivity. At the time of the earthquake on February 13, 2021, with its epicenter off the coast of Fukushima Prefecture[4], lowering of the PCV water levels at Units 1 and 3, and sliding of tanks on site exceeding the sliding amount evaluated at the time of tank installation were observed. For the PCVs for which the current state is not well-understood, understanding the damage condition by internal investigation and assessment of the situation at the accident, and estimation of aging by monitoring/evaluation are useful for risk identification. Regarding external events such as natural disasters, it is necessary to thoroughly evaluate in advance the consequences of and the necessity of countermeasures against beyond-design-basis events in existing/new systems.

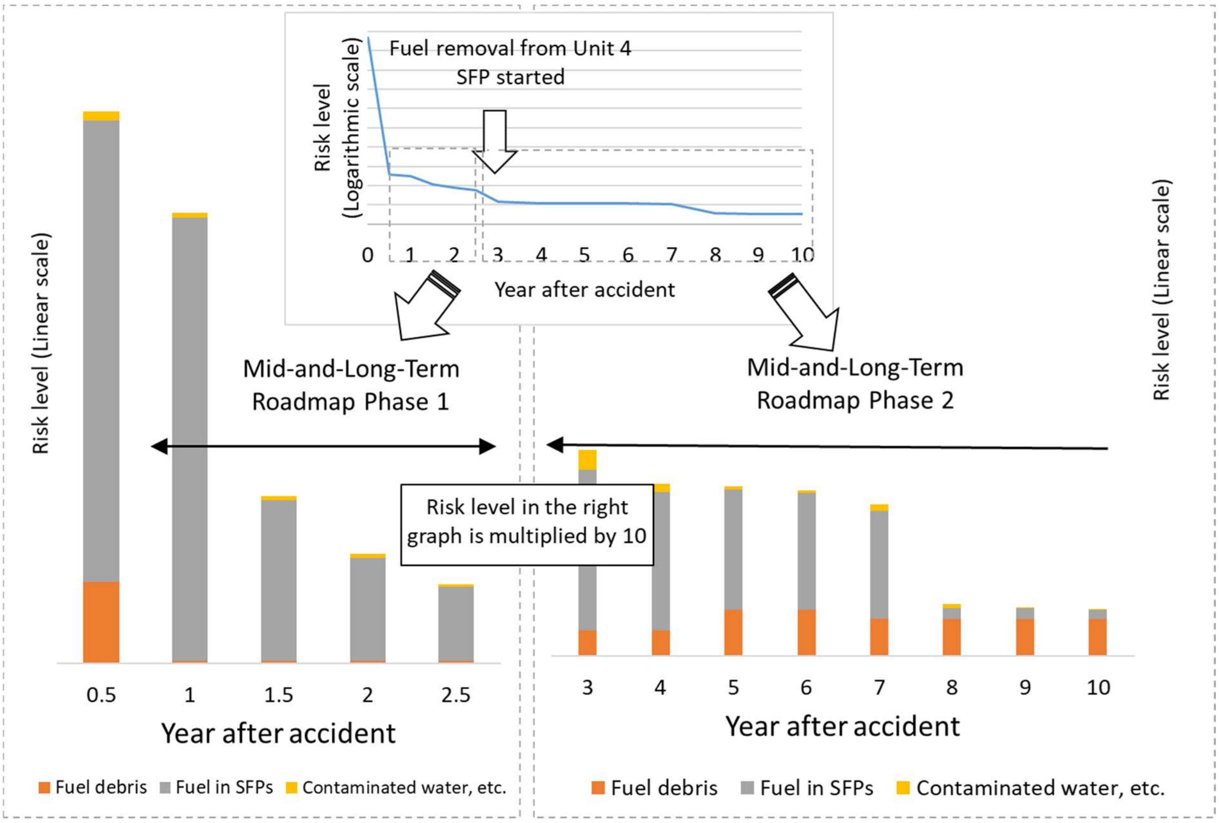
Although none of the above events resulted in significant consequences, it is important to carefully analyze the events using methods such as root cause analyses, and to identify risks that had not been anticipated in order to help prevent the occurrence of significant consequences. For this purpose, TEPCO needs to make efforts to learn from the unexpected events as described above.

Table 1 Major risk sources at the Fukushima Daiichi NPS

Fuel debris		Fuel debris in RPVs/PCVs in Units 1 to 3
Spent fuel	Fuel in SFPs	Fuel assemblies stored in the spent fuel pools (SFPs) in Units 1 and 2
	Fuel in the Common Spent Fuel Storage Pool	Fuel assemblies stored in the Common Spent Fuel Storage Pool
	Fuel in dry casks	Fuel assemblies stored in dry casks
Contaminated water, etc.	Stagnant water in buildings	Contaminated water accumulated in the reactor buildings of Units 1 to 3, process main building and high-temperature incinerator building, and sludge containing α-nuclides at the bottom of buildings of Units 1 to 3
	Zeolite sandbags	Sandbags containing zeolite placed on the basement floors of the process main building and high-temperature incinerator building
	Stored water in welded tanks	Strontium-treated water and ALPS-treated water, etc. (ALPS-treated water and water under treatment) stored in welded tanks
	Residual water in flanged tanks	Concentrated saltwater and sludge containing α-nuclides left at the bottom of flanged tanks
Secondary waste generated by water treatment	Waste sorption vessels, etc.	Spent sorbents used in a cesium sorption apparatus, a second cesium sorption apparatus, a third cesium sorption apparatus, advanced multi-nuclide removal equipment, mobile-type strontium removal equipment, a second mobile-type strontium removal equipment and mobile-type treatment equipment, etc.
	ALPS slurry	Slurry produced during treatment by the multi-nuclide removal equipment and added multi-nuclide removal equipment, and stored in high integrity containers (HIC)
	Sludge generated by decontamination device	Flocculated sludge generated during the operation of the decontamination system
	Concentrated waste liquid, etc.	Concentrated waste liquid generated by evaporative concentration of concentrated salt water with further volume reduction by concentration, and carbonate slurry collected from the concentrated waste liquid

4 Tokyo Electric Power Company Holdings Inc., “Additional system inspection and seismic evaluation in response to the earthquake on February 13 at the Fukushima Daiichi Nuclear Power Station”, Study group on monitoring and assessment of specified nuclear facilities meeting (90th), Material 5-1-3, April 19, 2021

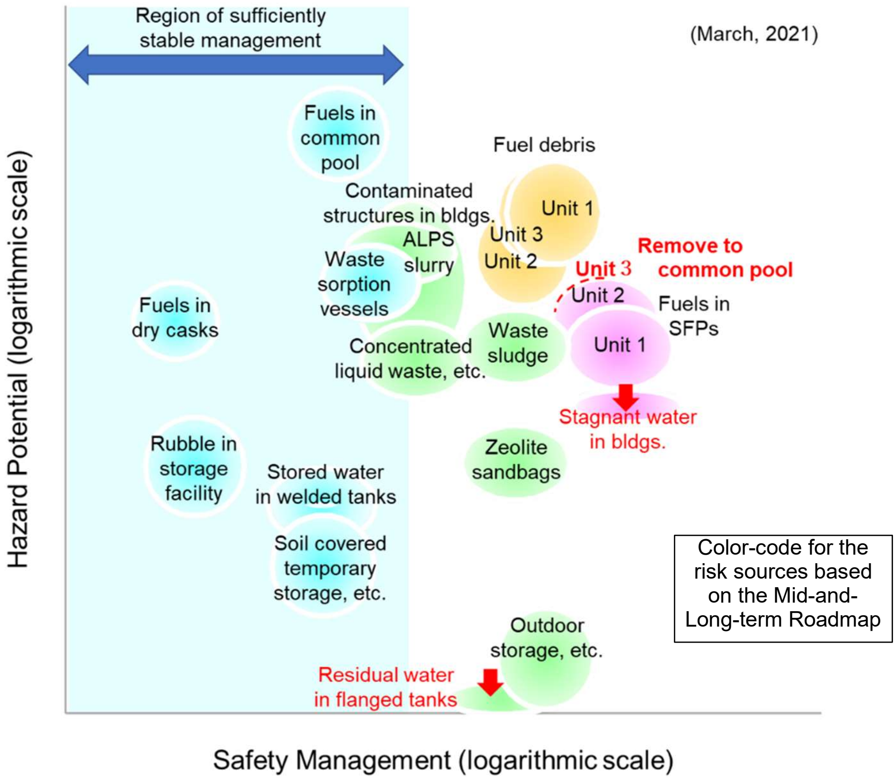
Rubble, etc.	Solid waste storage facility	Rubble (30 mSv/h and above) stored in the solid waste storage facility
	Soil-covered temporary storage, etc.	Rubble stored in the soil-covered temporary storage facility and containers (1-30 mSv/h), felled trees stored in the temporary storage pool
	Outdoor storage, etc.	Rubble stored in outdoor sheet-covered storage (0.1-1 mSv/h), rubble stored in outdoor storage (below 0.1 mSv/h), felled trees stored in outdoor storage
Contaminated structures, etc., in the buildings		Structures, pipes, components, and other items (shield plugs, standby gas treatment system pipes, etc.) inside the reactor buildings and PCVs/RPVs that are contaminated with radioactive materials dispersed due to the accident; and activated materials generated from operation before the accident



*1 The risk level was high due to fuel debris right after the accident, however, it became significantly lower because the hazard potential was decreased a lot by attenuation of the radioactive materials inside the fuel debris during the one year after the accident.

*2 In the evaluation eight years after the accident, as a result of incorporating the insight that the rise in the water temperature after cooling shutdown was slower than expected, the risk associated with fuel in SFPs is lower than previously estimated, because the time margin before the risk becomes apparent is increased.

Fig. 3 Reduction of risks present in the Fukushima Daiichi NPS



* Risk sources that are “relatively high risks given high priority” are shown in pink, those that are “immediate risk unlikely, but risk may grow when handling with haste” are shown in yellow, those that are “increased risk unlikely in the future, but appropriate decommissioning efforts are required” are shown in green, and those that are in the “sufficiently stable management” region are shown in light blue.

The red letters present risk sources that have changed significantly since last year, and the dotted lines or the starting points of the arrows indicate the locations in the previous year.

Fig. 4 Example of risk levels assigned to the major risk sources at the Fukushima Daiichi NPS

2.2.2 Risk reduction strategy

2.2.2.1 Interim targets of the risk reduction strategy

Measures for risk reduction include the reduction of the “Hazard Potential” and the reduction of the “Safety Management” level. Examples of reduction of the “Hazard Potential” include the decrease in inventory and decay heat associated with radioactive decay, and changing the form of liquid and gas into a less moveable form. Processing contaminated water to change it into secondary waste is an example of form change.

Examples of reduction of the “Safety Management” level include transferring fuel in SFPs to the Common Spent Fuel Storage Pool, and placing rubble stored outdoors into storage. Of the various risk reduction measures, reduction of the “Safety Management” level is generally considered to be easily realized. Consequently, the decommissioning of the Fukushima Daiichi NPS, which is implemented under the basic policy of “to continuously and quickly reduce the risks arising from the radioactive materials caused by the accident and that do not exist in normal nuclear power plants” (refer to Section 2.1), should first focus on steadily managing risk sources by keeping them in higher-integrity facilities to lower their Safety Management levels. The interim target of the risk reduction strategy is to bring the risk levels into the “Sufficiently stable management” region (the pale blue area) as shown in Fig. 4.

Regarding the progress from the Technical Strategic Plan 2020 towards this goal, Fig. 4 shows the completion of transfer of the fuel from Unit 3 SFP to the Common Spent Fuel Storage Pool in February 2021[5]; the completion of treatment of the remaining water (ALPS-treated water, etc.) at the bottom of the flanged tank in July 2020 (transfer to welded tanks)[6]; and the decrease of the stagnant water in buildings (transfer to sorption vessels, etc.). In addition, Table 1 specifies that the shield plugs and the piping of the standby gas treatment system, whose state have been verified through field investigations by the NRA[7], are included in the contaminated structures, etc., in the buildings.

In considering the station-wide risk reduction strategy for the Fukushima Daiichi NPS, the above-mentioned SED is a quantitative indicator of risks attributable to radioactive materials at a certain time, and is an effective method for prioritizing risk sources for risk reduction.

In response to the "Direction of risk consideration for external events" described in the Technical Strategic Plan 2020, TEPCO is proceeding with studies related to natural disasters, including measures against tsunami and large-scale rainfall, and building integrity evaluations, as stated in the Mid-and-Long-term Decommissioning Action Plan 2021[8]. TEPCO should continue verification on the system and facility integrity against external events including natural disasters, and develop their actions commensurate with the degree of risk.

2.2.2.2 Progress of risk reduction

Fig. 5 shows the process to bring major risk sources into the "Sufficiently stable management" region as the interim goal, and the decommissioning work progress along this process. Fig. 5(a) shows the outline flow of the decommissioning work to date and the future plans to represent the overall decommissioning process in a comprehensive way. Using the coloring in Fig. 4 to indicate the risk level of each risk source, Fig. 5(a) also shows the flow of risk reduction. Based on this flow, it is possible to visualize how the risk sources have changed compared with the time of the accident by applying it to fuel debris, spent fuel, and Cs-137 released during the accident. The number of spent fuel assemblies as an indicator to make the work progress easier to see in Fig. 5(b), and for Cs-137, the estimated radioactivity (Bq) common to various risk sources as an indicator in Fig. 5(c), both indicate the progress of the decommissioning work by representing the status of transition to the "Sufficiently stable management" region in a pie chart format.

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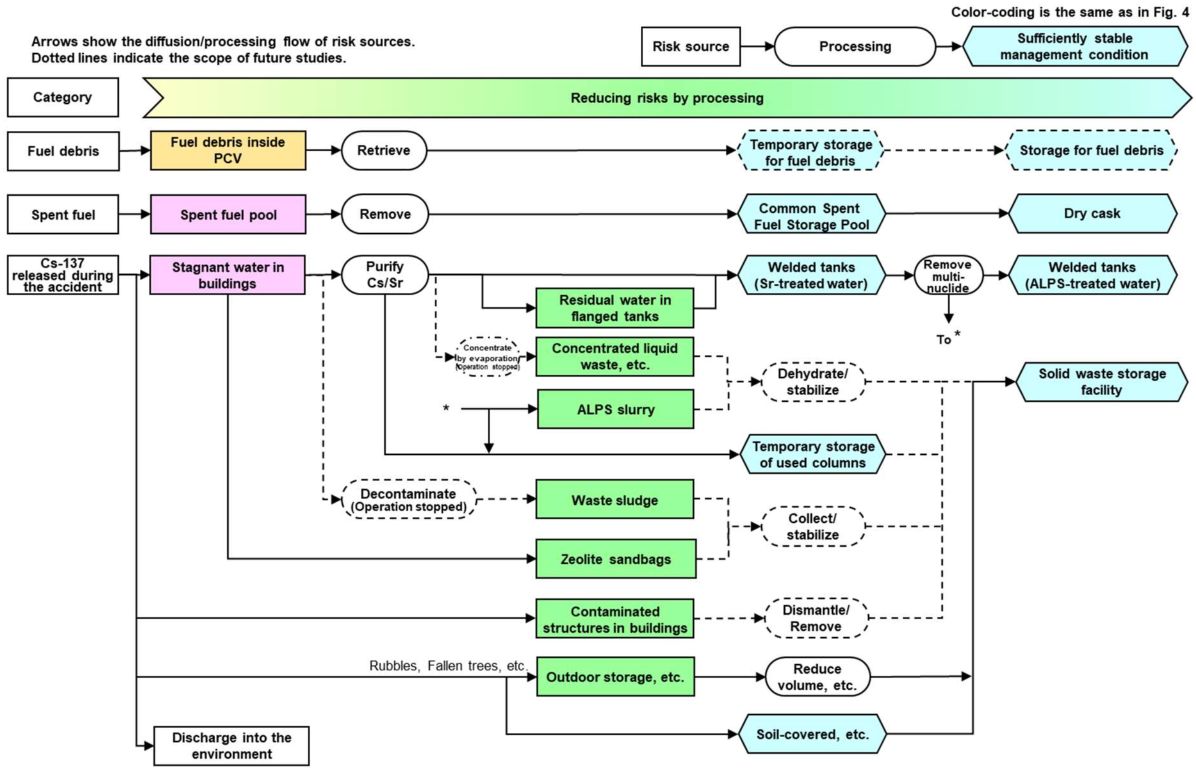
[5] Tokyo Electric Power Company Holdings Inc., "Completion of fuel removal at Unit 3", at the 88th Meeting of the Secretariat of the Team for Countermeasures for Decommissioning and Contaminated Water Treatment, March 25, 2021

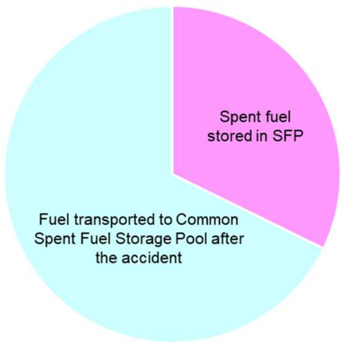
[6] Tokyo Electric Power Company Holdings Inc., "Status of storage/treatment of the accumulated water containing high-concentration radioactive materials at the Fukushima Daiichi Nuclear Power Station (459th report), July 13, 2020

[7] The 19th Study Committee on Accident Analysis of the Fukushima Daiichi Nuclear Power Station, Material 4, "Interim report on investigation/analysis of the accident at the TEPCO Fukushima Daiichi NPS (draft)", March 5, 2021

[8] Tokyo Electric Power Company Holdings Inc., "Mid-and-Long-term Decommissioning Action Plan 2021", March 25, 2021

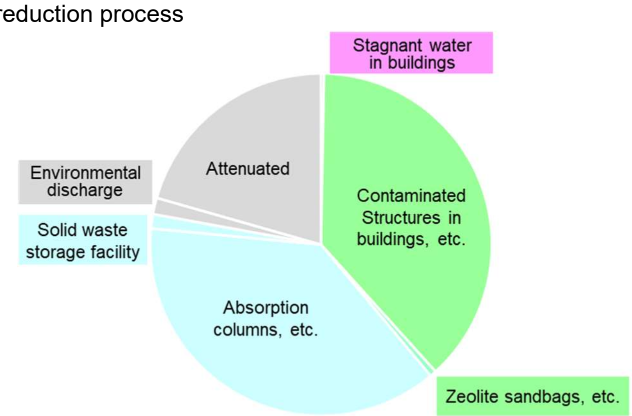


(a) Risk reduction process



*New fuel not included

(b) Number of fuel assemblies of spent fuel (Units 1 to 4)



(c) Radioactivity of Cs-137 released at the accident (Units 1 to 3)

Fig. 5 Risk reduction process for major risk sources and its progress (example as of March 2021)
(Attenuation in Fig. 5(c) takes into account radioactive decay of Cs-137 from the time of the accident to the end of March 2021)

2.2.2.3 Basic approach to risk reduction

The decommissioning of the Fukushima Daiichi NPS is a project that involves considerable uncertainties. To date, the internal status of the Primary Containment Vessels (hereinafter referred to as "PCVs") of Units 1 to 3 has been estimated to some extent through simulation of the accident development process, estimation of the places with fuel debris by muon-based fuel debris detection technology, placement of investigation equipment into the PCVs, radiation dose measurement and video photographing in the buildings, and other means. There are however still significant uncertainties. Eliminating these uncertainties requires many resources and, in particular, a considerable amount of time. In order to realize prompt reduction of risk, it is necessary to promote the decommissioning work through a flexible and prompt approach, based on the directions determined with previously obtained experience and knowledge, and with experiment and analysis-based simulation, making safety the top priority, even though uncertainties remain to a certain extent.

Regarding the perspective from which these comprehensive decisions will be made, NDF summarizes the following five guiding principles:

(Five guiding principles)

- Safe — Reduce the risks posed by radioactive materials and ensure work safety
  (Issues such as containment of radioactive materials [environmental impact], exposure of workers to radiation, assessment of the effect of risk reduction)
- Proven — Highly reliable and flexible technologies
  (Issues such as conformity to requirements, effectiveness and flexibility against uncertainty)
- Efficient — Use resources effectively (e.g., people, things, money and space)
  (Issues such as reduction of waste generation, cost, efficiency, securing necessary work area and site)
- Timely — Be conscious of time
  (Issues such as the early start of fuel debris retrieval and estimation of time required for fuel debris retrieval)
- Field-oriented — Comprehensive three-reality policy by checking actual site, actual things, and actual situation
  (Issues such as workability including environment-friendliness, accessibility, and operability, and maintainability including ease of maintenance and troubleshooting)

In applying the five guiding principles to the actual site, it is important to proceed with the decommissioning operation after greatly emphasizing safety assurance for the purpose of protecting human beings and the environment from the radioactive materials associated with the operations, thoroughly conducting radiological impact evaluations, and taking appropriate radioprotective measures. (“Safe” in the five guiding principles)

In the decommissioning of the Fukushima Daiichi NPS, because the public risk level is rising with time as the degradation of facilities damaged by the accident progresses, controlling this risk to be as low as reasonably achievable (“Efficient”) as promptly as possible (“Timely”) in light of the situation at the site, and proceeding with the decommissioning in a reliable manner (“Proven”) by feasible ways in the harshest on-site state (“Field-oriented”) will lead to ensuring safety in the medium-to-long-term.

As for the result judged based on these guiding principles, it is also important to work to disseminate information carefully so that the results will be widely accepted by society.

2.3 Approach to ensuring safety during decommissioning

2.3.1 Basic policy for ensuring safety based on the characteristics of Fukushima Daiichi NPS

Decommissioning of the Fukushima Daiichi NPS containing the reactors involved in the accident is an unprecedented activity that takes place in a peculiar environment different from that of a normal reactor, and therefore, to ensure safety, the following characteristics (peculiarities) regarding safety should be fully recognized:

- A large amount of radioactive material (including α-nuclides that have a significant impact in internal exposure) is in an unsealed state, as well as in unusual (atypical) and various forms
- Barriers for containing radioactive materials, such as reactor buildings and PCVs, are incomplete
- Significant uncertainties exist regarding the state of these radioactive materials and containment barriers, etc.
- Difficulty in accessing the site and installing instrumentation devices to obtain on-site information due to constraints such as high radiation levels on site
- Since the current level of radiation is high and further degradation of containment barriers is a concern, it is necessary to take measures in consideration of the time axis without prolonging the decommissioning activities

Consequently, TEPCO, as the operator of the decommissioning project, needs to pay special attention to the following points in proceeding with the decommissioning based on the five guiding principles.

Firstly, with regard to “safety”: There is great uncertainty about the state of radioactive materials and containment barriers, and on-site access and installation of instrumentation devices to reduce the uncertainty are also restricted. Under these circumstances, a large amount of atypical and unsealed radioactive material will be handled in an incomplete state of containment. Therefore, the starting point for all reviews should be confirmation of the feasibility of ensuring safety with a wide range of possibilities (cases) assumed. At the same time, with regard to “safety”, it is important not to prolong the work period in light of risk reduction over the entire work period. Therefore, it is necessary to avoid excessive safety measures and to take optimum safety measures (ALARP[9]). Such perspective on “safety” (the safety perspective) should be reflected in the decommissioning work review.

Secondly, with regard to “field-oriented”:

- The on-site environment is in a peculiar state that includes a high level of radiation, and therefore attention should be paid to the feasibility of construction/implementation of safety measures on site.
- An approach through design alone has limitations due to significant uncertainties.

[9]Abbreviation of As Low As Reasonably Practicable. This is the principle that the radiological impact must be as low as reasonably achievable.

From the above-mentioned reasons, it is essential to accurately apply the information gained on site into engineering. In order to ensure the implementation of unprecedented engineering such as fuel debris retrieval, the views and feelings of the individuals and organizations (operators) that are responsible for the on-site work (including operation, maintenance, radiation control, instrumentation, analysis, etc.) and very familiar with actual site should be highly respected. Moreover, it is important to respect their perspectives and judgements directly based on the site (the operator’s perspective). In promoting the prolonged decommissioning work, it is necessary to maintain and strengthen the operator's perspectives/feelings, and TEPCO themselves should inherit their perspectives. Therefore, TEPCO needs to take action that always accounts for the worksite in the overall decommissioning work process, such as by inviting outside experts and technicians with operator’s perspectives for coaching/educational training, including experienced workers in difficult operations and those who experienced on-site operations.

In the actual study of the decommissioning work, TEPCO, as the decommissioning project executor, should clearly define the “requirements” for the work in advance, and should consider specific safety measures to achieve them. In doing so, it is essential to apply the safety perspective and the operator’s perspective to handling the characteristics (peculiarities) of decommissioning the Fukushima Daiichi NPS. Specifically, requirements should be established in consideration of “the safety perspective” and “the operator’s perspective”, and specific safety measures selected for the work that satisfies the requirements, considering the two perspectives. In each stage of reviewing the decommissioning work, in this manner, sufficient attention should be paid to “the safety perspective” and “the operator’s perspective”.

In this decommissioning work with significant uncertainties, it is frequently difficult to clearly define requirements in advance. Even in such cases, the decommissioning work should be carried out flexibly and promptly with by verifying and improving the selected, specifical safety measures as the “preliminary implementation and utilization of the obtained information in the latter stages” and “iteration-based[10] engineering” as described later.

This section describes first the importance of the safety assurance measures in terms of the characteristics of Fukushima Daiichi NPS based on safety assessment which includes the operator’s perspective. Then, it describes the operator’s perspective-specific importance that should be incorporated at multiple levels in the safety assurance process while requesting spread of “Safety first” performed by project operator. Lastly, the section refers to the importance of ALARP judgment.

2.3.1.1 Promulgating the “Safety First” principle that safety perspective comes first

The use of any method or device is basically unacceptable unless the safety perspective is sufficiently reflected in them. Therefore, it is important that all who work in the processes (projects) leading up to the use of methods and devices on the site, keep the safety perspective first in mind as they engage in their work (safety first). The specific application of the general “safety first”

[10] Method that gradually increases the level of completion of engineering by obtaining the next result from one result and repeating this process.

principle in the projects means, “Conducting extensive assessments on safety matters associated with methods and devices when reviewing any project and, upon verifying that necessary and sufficient levels of safety have been ensured, taking into account factors such as technical reliability, reasonableness, speed, actual site applicability and project risks in a comprehensive manner to decide which methods or devices to use, and which safety measures to apply consequently”.

Since the accident at Fukushima Daiichi NPS, leaders of nuclear operations at TEPCO have stepped up to the plate and continue to work hard to raise awareness on the issue of nuclear safety, such as through dialogue amongst themselves, as well as through messages that they communicate to other TEPCO employees. In order to thoroughly disseminate the “safety first” principle to all persons involved in the projects including on-site workers, the attitude of top management (the approach to reiterating the special nature of nuclear safety and that special attention is needed accordingly) is important.

2.3.1.2 Optimization of judgement with a safety assessment as its basis and ensuring timeliness in decommissioning

With an aim of reducing risk through decommissioning, it is most important to take appropriate measures and ensure the safety of work in which a large amount of radioactive material is handled that is technically difficult and has significant uncertainties, such as fuel debris retrieval. Thus, decommissioning work should be carried out with such “safety perspective”.

Specifically, when designing safety measures for each decommissioning activity, it is essential to make decisions based on the five guiding principles after conducting a thorough safety evaluation and confirming that the required safety is ensured. As mentioned above, the decommissioning work of the Fukushima Daiichi NPS is unprecedented and has significant uncertainties. Using deliberated safety evaluation as the basis for making decisions regarding safety measures, the decisions will not be significantly unstable (that is, without devoting too little or excessive resources), and thus necessary, sufficient, and reasonably feasible safety measures can be realized (optimization of judgment based on safety assessment). In regard to reasonably feasible safety measures, it is particularly important in the safety assessment of the Fukushima Daiichi NPS to conduct safety assessment with incorporating operator’s perspective stated in the section 2.3.1.3.

In addition, the importance of making progress in the decommissioning work without delay (the importance of time-axis-conscious action) can be mentioned as “the safety perspective” unique to the decommissioning of Fukushima Daiichi NPS. Considering the high radiological impact that has already materialized, and the possibility of further degradation of containment barriers, etc., making progress in the decommissioning work without delay will have great significance for ensuring the safety of the entire decommissioning process from a medium-and-long term perspective. Therefore, it should be noted, for ensuring safety, that different perspectives from normal reactors are required, which have a certain margin in terms of human, physical, and financial resources and have low radiological impacts and high stability. On the condition that safety is secured, rational judgement should be made on resource allocation and the time-axis-conscious progress in the decommissioning work without delay based on the relationship with the overall balance (ensuring timeliness in decommissioning activities).

2.3.1.3 Ensuring safety by incorporating “the operator’s perspective”

To ensure that safety measures are truly effective, it is necessary to satisfy the needs from the standpoint of those who actually perform the operations and tasks on site, “the operator’s perspective” (perspectives and judgements from the standpoint of those who are familiar with the site and perform operations and tasks on site) is important. In addition to this standpoint, the Fukushima Daiichi NPS is a facility that has suffered from an accident, and unlike a normal reactor its decommissioning requires an unprecedented approach that can be carried out in a peculiar environment, such as one with high radiation levels. Therefore, when determining the feasibility of safety measures on site, the peculiarities of the on-site conditions, such as high radiation levels and the environment, shall be considered.

“The operator’s perspective” is also important for ensuring safety from the following perspectives, which are different from those of normal reactors.

- Complementation of design by operations, including operating controls:
  Due to significant uncertainties, there is a limit to addressing all situations by design alone. Therefore, it is effective to supplement the design with operator response and on-site operation, and improve safety collectively with operation.
- Utilization of information in design obtained through monitoring, analysis, etc.:
  To cope with significant uncertainties, it is important to utilize information obtained through on-site operations such as monitoring and analysis, etc., in designing safety measures. When utilizing such information, it should be linked with calculational evaluation, etc., to “make a comprehensive use of it”.
- Handling an abnormality:
  Although it is essential to take all possible measures to prevent progress of an abnormality, on-site response as a precaution to prevent the occurrence of an abnormality is effective considering the characteristics that the progress of abnormalities is moderate and there is sufficient time to respond[11].

2.3.1.4 ALARP judgment based on safety

For safety, there is a minimum level of safety standards that must be met before the relevant retrieval method or equipment can be used. At levels above the level that meets this minimum level, there is a range of choices and, within this range, retrieval methods and equipment to be adopted will be determined based on trade-off between the safety level to be achieved and project cost and duration (note that retrieval method and equipment involving long-term large safety measures are

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[11] Since a long time has passed since the accident at the Fukushima Daiichi NPS, the intrinsic energy (decay heat) that drives dispersion of radioactive materials is small. Therefore, in general, abnormalities have the characteristic of progressing slowly so there is a large time margin to respond.

not necessarily beneficial to safety, especially at Fukushima Daiichi NPS), a kind of ALARP. There is also an issue as to whether such retrieval methods and equipment are feasible in the field.

Based on the above, in the process of determination, it is important to decide retrieval methods and equipment to be employed eventually through the cycle of “defining the safety standards (safety perspective)”, “indicating the feasibility on-site (operators’ perspective)”, and “examining and discussing at projects (project management)” as shown in Fig.6. As shown in this figure, the safety perspective and the operator’s perspective are not independent from each other. The ALARP judgement made by the project based on the safety perspective will be linked to the decision of the retrieval method and equipment after going through the feasibility check based on the operator’s perspective. The operator's perspective is essential to actually incorporate the safety perspective into the site, and the judgment based on the safety perspective is needed to utilize the operator's perspective.

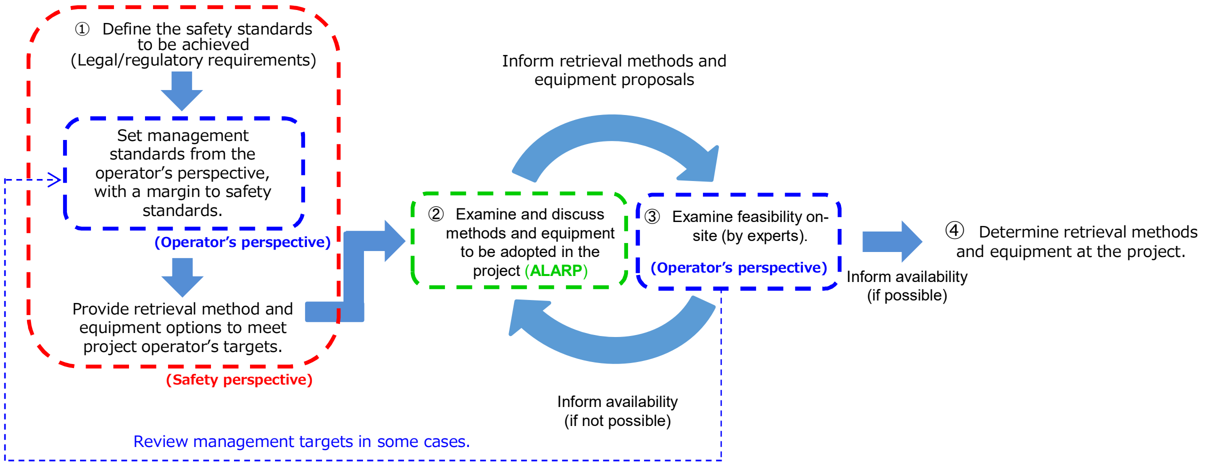

Fig.6 ALARP centered on safety (conceptual diagram)

2.3.2 Preliminary implementation and utilization of the obtained information in the latter stages

The on-site conditions at the Fukushima Daiichi NPS containing the reactors involved in the accident includes considerable uncertainties. If the whole operation of a large-scale project such as fuel debris retrieval is to be designed only with existing knowledge, assumptions of an extremely large safety margin and wide range of technical options will be needed. Thus, extension of the work period or the risk of rework will be unavoidable. As a result, the feasibility or predictability of the entire project may be reduced, leading to a delay in the entire decommissioning, a rise in the decommissioning cost, or increased radiation exposure of workers.

However, considering the current environment with an already high radiation level, further deterioration of containment barriers, and the possibility of future major natural events (such as earthquakes or tsunamis), it is necessary to immediately improve the state of such risks and reduce uncertainties. Therefore, a “sequential type approach” is important where the whole operation is divided into several stages, “operation at first stage” is implemented for which practical safety can be ensured, and then the information obtained there is utilized in the next stage. With this approach[12], operation proceeds with safety ensured through monitoring the condition inside reactor, restricting operational actions and flexible on-site responses[13] at each stage of the process. The information obtained at each stage of operation is utilized in the design of subsequent stages. This approach reduces uncertainties in the operations in subsequent stage as well as improve the reliability of safety assurance and rationalize design.

TEPCO should actively introduce an approach like this into actual engineering and project management[14].

In a case that uses a similar approach, TEPCO has been conducting reactor water injection shutdown tests since FY 2019. One of the purposes of these tests is to contribute to determining whether or not to terminate water injection in the future, which also takes into account maintaining flexibility in selecting fuel debris retrieval methods. Knowledge on whether or not to terminate water injection has been accumulated by identifying different risks (rising temperature of fuel debris and the RPV bottom, increased dust scattering outside the PCV, and re-criticality when resuming water injection), and by gradually extending the testing time while taking certain risks.

These reactor water injection shutdown tests have resulted in the clarification of the relationship between the lowering of the PCV water level and pressure in Unit 1. According to the information on the damaged piping that was obtained from the on-site investigation conducted before the tests, it was assumed that the PCV pressure might decrease when the water level reached the pertinent damaged area. In fact, the occurrence of the event, which developed as predicted, further enhanced the degree of confidence of the presumed cause that “the PCV pressure dropped as a result of exposing the damage area due to the lowering of the PCV water level”.

This example of the reactor water injection shutdown test yielded results that led to reducing the uncertainty through examination in combination with the information gained from on-site investigations, although this information is not directly related to the purpose of the test.

Hereafter, it is recommended to make it clear as a policy that the information to be gained through on-site operation should be fully incorporated and accumulated as knowledge in consecutive activities for ensuring safety. For example, the same applies to risk identification associated with hydrogen at the time of fuel debris retrieval. Testing to reduce nitrogen supply for an experimental purpose may help identify hydrogen risk, and determine requirements on the necessary amount of nitrogen supply and reliability of the exhaust systems to ensure safety. It is important to accumulate successful/unsuccessful experience gained in the process of these sequential approach as a track record, allowing gradual reduction in major uncertainties in the overall decommissioning work in the future. This will lead to steady progress in decommissioning and contribute to ensuring safety in decommissioning the Fukushima Daiichi NPS from the perspective of risk reduction in the medium-and-long term.

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[12]This is also used in UK, for example, for the decommissioned facilities in Sellafield, and is called Lead & Learn.

[13] Some example measures include installing nuclear instrumentation to the extent feasible; limiting the amount of debris retrieving; and setting the value for managing radioactive dust concentration and regulating operations.

[14] This is stated in the Decommissioning Implementation Plan (March 17, 2021, Tokyo Electric Power Company Holdings Inc.,), which summarizes the policies on implementing decommissioning at the Fukushima Daiichi Nuclear Power Station. https://www.tepco.co.jp/press/release/2021/1585525_8711.html

2.3.3 Approach to address a temporary increase in risk level associated with decommissioning operations

While the decommissioning work is striving for prompt risk reduction from a medium-and-long-term perspective, careful deliberation of the possibility that the performance of decommissioning work may temporarily change the risk levels and may increase the radiation exposure of workers is required. Executing the decommissioning work involves taking some action on the current situation of the NPS, which is maintained in a state with a certain level of stability despite some risks. Such risks may materialize, depending on the way action is taken. For example, accessing the inside of the reactor to retrieve fuel debris will affect the current containment status, and the special operations and maintenance performed in the retrieval work will increase the exposure of workers involved in these activities.

The possibility of a temporary increase in the risk level and a rise in workers' exposure arising from such decommissioning work must be addressed by taking measures to prevent and restrict them. In particular, as for the radiation safety of workers, it is imperative to limit the increase in the risk level during decommissioning as much as practicably possible by thorough preparations as achieved through application of the concept of ALARA (to suppress radiation exposure to As Low As Reasonably Achievable).

Note that the basic stance for promptly implementing the decommissioning must stand firm because if the decommissioning work is delayed excessively, it means that existing major risks will remain over the long term and their risk levels may gradually rise as the buildings and facilities deteriorate over time. Therefore, with regard to the selection of work methods, the design and manufacture of equipment and safety systems, and the development of work plans for the decommissioning work, cautious and comprehensive decision making is required for early implementation of decommissioning in consideration of many constraints such as time, cost, and worker's exposure needed for relevant preparations and work, while giving priority to limiting the risks involved in the decommissioning work (Attachment 5). The approach to risk reduction and ensuring safety in the decommissioning of the Fukushima Daiichi NPS, as described in this chapter, RA), NDF, TEPCO, and others to cooperate with each other to reduce risks based on the approach to ensuring safety based on their respective positions. In doing so, it is important to establish a system for on-going risk monitoring which enables a wide range of people to easily understand how the overall risks at the site have been continuously reduced through the decommissioning work, and to communicate such progress to the public. In addition to sharing the status of risks through the Technical Strategic Plan on a constant basis, NDF is considering providing the status of risk reduction along with the progress of the decommissioning work described in 2.2.2.2. TEPCO also needs to develop a mechanism to identify risks for the entire site and become aware of the need to take action to communicate the status of risk reduction to society in a proactive manner.

3. Technological strategies toward decommissioning of the Fukushima Daiichi NPS

3.1 Fuel debris retrieval

3.1.1 Targets and progress

(Targets)

(1) Retrieve fuel debris safely after thorough and careful preparations, and bring it to a state of stable storage that is fully managed.

(2) Trial retrieval in Unit 2 was scheduled to begin within 2021, but the process has been delayed due to the COVID-19 infection. In order to limit the delay to about one year, preparations will be made for starting retrieval. Continue a series of work including the gradual expansion of fuel debris retrieval to acquire knowledge and experience necessary for the further expansion of fuel debris retrieval (For the target of fuel debris retrieval, see Attachment 6).

(3) With regard to the further expansion of fuel debris retrieval, consideration will be given to the methods including those for containing, transferring, and storing of fuel debris, by assessing fuel debris retrieval at the first implementing unit, internal investigations, research and development, and the on-site environmental improvement, etc.

(Progress)

Fig. 5 shows the estimated fuel debris distribution, access route and surrounding structures of Units 1 to 3. The progress in each unit is also shown below.

(1) Unit 1

Within FY 2021, further detailed information inside PCV is planned to be gained by inserting a boat-type access investigation device with diving capabilities (hereinafter referred to as “underwater ROV”) into PCV and investigating the internal status such as distribution of deposits widely scattered at the bottom of outside the pedestal, presence or absence of fuel debris involved in deposits, and structures inside the pedestal. In preparation for the start of this investigations, removal of obstacles within the PCV is being promoted while taking measures to control dust dispersion and monitoring dust concentration by considering the change in dust concentration when opening the inner door at the penetration X-2. Also, at the beginning of 2021, during the preparation of the obstacle investigation in PCV, for the pressure decreasing that were expected to have occurred due to the addition of external force to the outer door of the penetration X-2, removal of obstacles within the PCV is being promoted while performing measures to suppress the occurrence. As the immediate response to the PCV water level drop caused by the earthquake off the coast of Fukushima Prefecture on February 13, 2021, a study is underway to implement continuous water level monitoring. Specifically, when conducting PCV internal investigation using underwater ROV, consideration is being given to increase the amount of injection water once to the level before the earthquake because the risks of interference with deposits increases when the water level is low, and to return to the current water level after PCV internal investigation.

As forward-looking measures, lowering of the water level is planned to improve the seismic resistance of the suppression chamber (hereinafter referred to as “S/C”), and preparations are currently being made to lower the water level, such as by water intake using existing piping and water quality surveying of the S/C water.

(2) Unit 2

The 2019 Med-and-Long-term Roadmap specified Unit 2 as the first implementing unit for fuel debris retrieval, and its trial retrieval was supposed to launch within 2021. However, the process has been delayed due to the new coronavirus infection. In order to limit the delay to about one year, preparations are being made for starting fuel debris retrieval. As for dust dispersion prevention, in consideration of the changes in dust concentration while drilling the inner door of the penetration X-2 for Unit 1 PCV internal investigation (June 2019), preparations are underway for low-pressure water cleaning of deposits, spray curtains on the penetration outlet and countermeasures to reduce pressure in the PCV. The arm-type access equipment (robot arm) has arrived in Japan and started testing. In addition, in order to allow the arm-type access equipment to be used to enter from the penetration X-6, removal of the deposits in the penetration X-6 is planned. In 2020, deposit contact investigation and 3D scanning investigation in the penetration X-6 showed that the deposits deform due to contact, the deposits are accumulated on the slope from the building side toward the pedestal, and the remaining cable is not stuck and can be lifted. Based on these results, removal of the deposits in the penetration X-6 is under consideration.

A plan for the gradual expansion of fuel debris retrieval is also underway, and the retrieval device will be improved by increasing the weight capacity and enhancing accessibility while complying with specifications for the devices for trial retrieval and PCV internal investigations. In this plan, the requirements related to the arm performance of the arm-type access equipment and the interface between the equipment and enclosures have been clarified and examined. The retrieved fuel debris will be stored in unit cans in the enclosure, and then transferred to the receiving/delivery cells on site and stored in temporary storage cells. In addition, some of the fuel debris will be collected in the receiving/delivery cells for analysis and transported to the facility for analysis. Designing of the retrieval device, receiving/delivery cells, and temporary storage cells is in progress (Fig. 8, Fig. 9).

In response to the unprecedented approach to retrieving fuel debris from the first implementing unit, NDF continues working while verifying the actual site applicability of the device and the results of the review on modifications to the safety system from the perspectives of safety, reliability, reasonability, timeliness, and a field-oriented stance, in accordance with the progress of engineering at TEPCO.

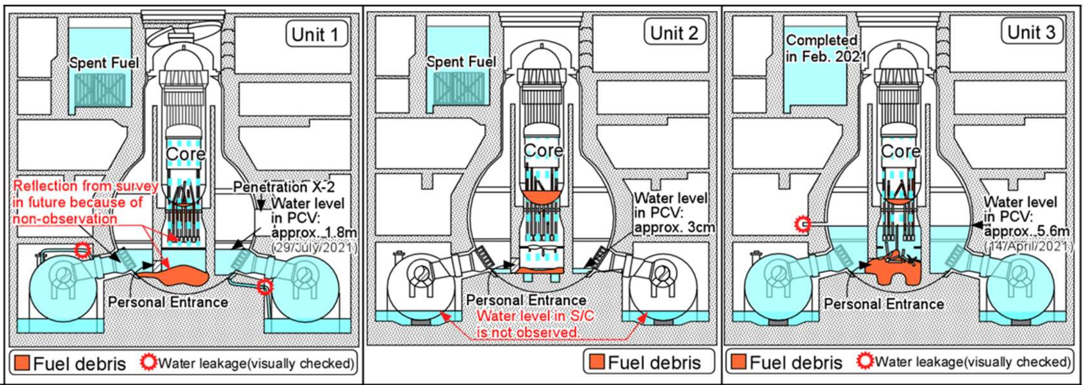
(3) Unit 3

As for Unit 3, due to high water levels in the PCV, it is planned to gradually lower the PCV water levels, taking into account the improved seismic resistance of the S/C and conducting PCV internal investigation. The result of sampling the S/C water (conducted in 2020) using the piping connected to the S/C shows that the concentration of radioactive materials such as Cs-137 is higher than that of the stagnant water in buildings. Therefore, its impact on the treatment of contaminated water is being considered, and the analysis results are being incorporated into the PCV water intake system. In addition, the conceptual study on further expansion of fuel debris retrieval is in progress.

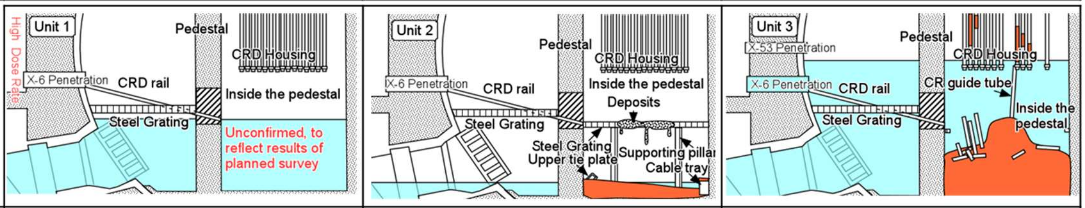
(4) Impact of and response to the earthquake on February 13, 2021, with its epicenter off the coast of Fukushima Prefecture

Due to the earthquake that occurred off the coast of Fukushima Prefecture on February 13, 2021, lowering of the PCV water level was observed in Units 1 and 3, however, water injection into the reactor is continuing. As there are no significant changes in plant parameters, the cooling condition of fuel debris is considered to be intact. It is assumed that this drop in water level may have been caused by an increase in leakage from inside of the PCV due to changes in the condition of the damaged areas previously identified in the PCV and its newly damaged areas. Going forward, the change in parameters such as water level will be checked by water injection shutdown testing, and expansion of knowledge be considered.

In light of the recent earthquake, it is necessary to enhance monitoring systems to observe changes in plant conditions, perform impact assessment to maintain/manage systems and buildings over the med-and-long-term, and promote technical development required to understand the situation. (Refer to 3.1.2.4.1.5: Issues in the structural integrity of PCVs and buildings)



	Unit 1	Unit 2	Unit 3
Core region	· Little fuel debris remains.	· Little fuel debris remains. (Partially intact fuel might exist in the peripheral region)	· Little fuel debris remains.
At RPV lower head	· A small amount of fuel debris is present · A small amount of fuel debris is present inside and on the outer surface of the CRD housing.	· Large amount of fuel debris is present · A small amount of fuel debris is present inside and on the outer surface of the CRD housing.	· Part of fuel debris is present · A small amount of fuel debris is present inside and on the outer surface of the CRD housing.
At the PCV bottom (Inside the pedestal)	· Most of the fuel debris is present.	· A certain amount of fuel debris is present.	· Amount of fuel debris in Unit 3 is more than that in Unit 2.
At the PCV bottom (Outside the pedestal)	· Fuel debris may have spread outside the pedestal through the personal entrance.	· The possibility of fuel debris spreading outside the pedestal through the personal entrance is low.	· Fuel debris may have spread outside the pedestal through the personal entrance.
Radiation dose in operation site*1	· Radiation dose around the penetration X-6 on the first floor of R/B is high (630 mSv/h).	· Radiation dose on the first floor of R/B had reduced to approx. 5 mSv/h as a whole.	· Radiation dose on the first floor of R/B is higher than tens of mSv/h.



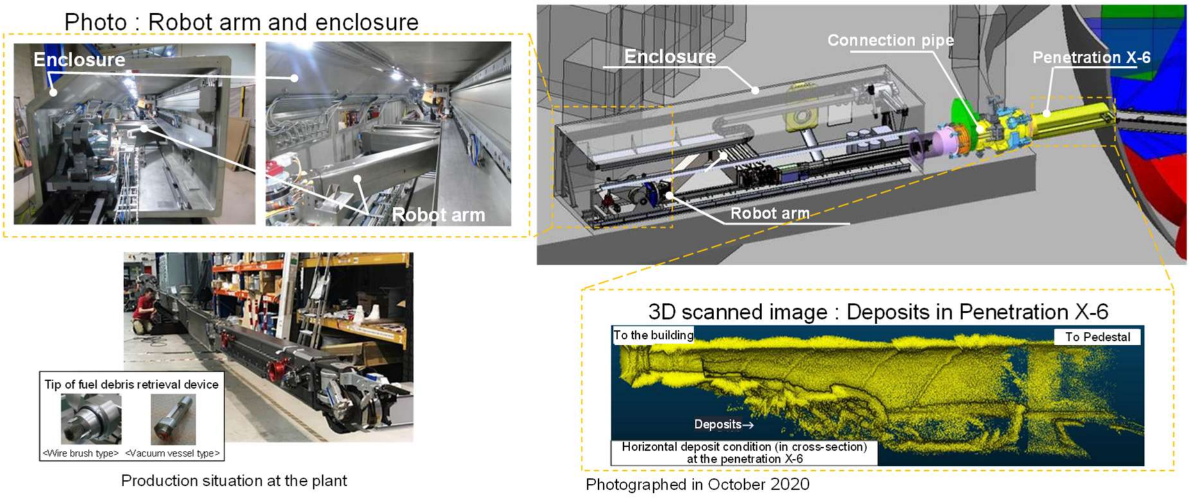
	Unit 1	Unit 2	Unit 3
Information on the access route to fuel debris*2	• The D/W bottom outside the pedestal is accessible from the upper side of the steel grating. • Condition around the CRD rail connecting into the pedestal from the penetration X-6 has not been observed.	• No large obstacles have been observed on the CRD rail and around the pedestal entrance. • The bottom inside the pedestal is accessible through the pedestal entrance.	• The bottom inside the pedestal is accessible through the pedestal entrance.
Information on the condition of structures around the access route	· No significant damage has been observed on the wall surface outside the pedestal on the steel grating upper side.	• While a part of fuel assemblies have fallen, no damage has been observed on the CRD housing support in the examined range . • No damage has been observed on the wall surface and the structures (CRD exchanger, etc.) inside the pedestal.	• Some damaged structures and fallen objects (which may include internal structures), and the fall and deformation of a part of the CRD housing support have been observed inside the pedestal. • No damage has been observed on the wall surface inside the pedestal.

*1 Data provided by TEPCO

*2 Results obtained through PCV internal investigation performed up to date were presented for judging whether any obstacles such as fallen objects may exist on the route to the inside of the pedestal from X-6 penetration, which is considered as a dominant access route for fuel debris retrieval by the side access method.
Other access routes through the equipment hatch and others have been investigated under the Governmental-led R&D program on Decommissioning and Contaminated Water Management.
Due to high dose rate around X-6 penetration of Unit 1, an access route through the equipment hatch may be used in case that it is difficult to improve the environmental condition around X-6 penetration.
PCV internal investigation of Unit 1 will be performed through X-2 penetration (equipment hatch) considering accessibility of devices for PCV internal investigation.

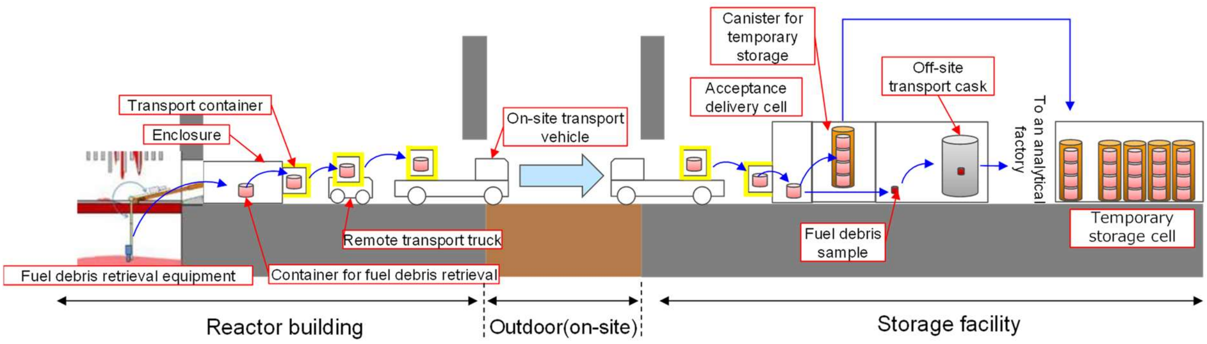
(Prepared in reference to "Material 4-1: Progress of treatment of stagnant water in buildings", the 81st meeting of the Study group on monitoring and assessment of specified nuclear facilities)

Fig. 7 Estimated fuel debris distribution, access route and surrounding structures of Units 1 to 3



(TEPCO material edited by NDF)

Fig. 8 Image of fuel debris retrieval system
(Trial retrieval and subsequent gradual expansion of fuel debris retrieval)



(TEPCO material edited by NDF)

Fig. 9 Image from retrieval to temporary storage of fuel debris
(Gradual expansion of retrieval scale)

3.1.2 Key issues and technical strategies to realize them

Since the understanding the situation inside the PCVs is still limited, the current design and the plan for on-site operations related to fuel debris retrieval should be continuously reviewed based on knowledge that will be obtained in the future, and it is also important to accurately incorporate the results of studies, research and development toward fuel debris retrieval.

In Unit 2, trial retrieval and PCV internal investigation will be performed, and implement the gradual expansion of fuel debris retrieval based on the findings. In addition, a conceptual study on the further expansion of fuel debris retrieval for Unit 3 is planned.

3.1.2.1 Trial retrieval and PCV internal investigation, and gradual expansion of fuel debris retrieval

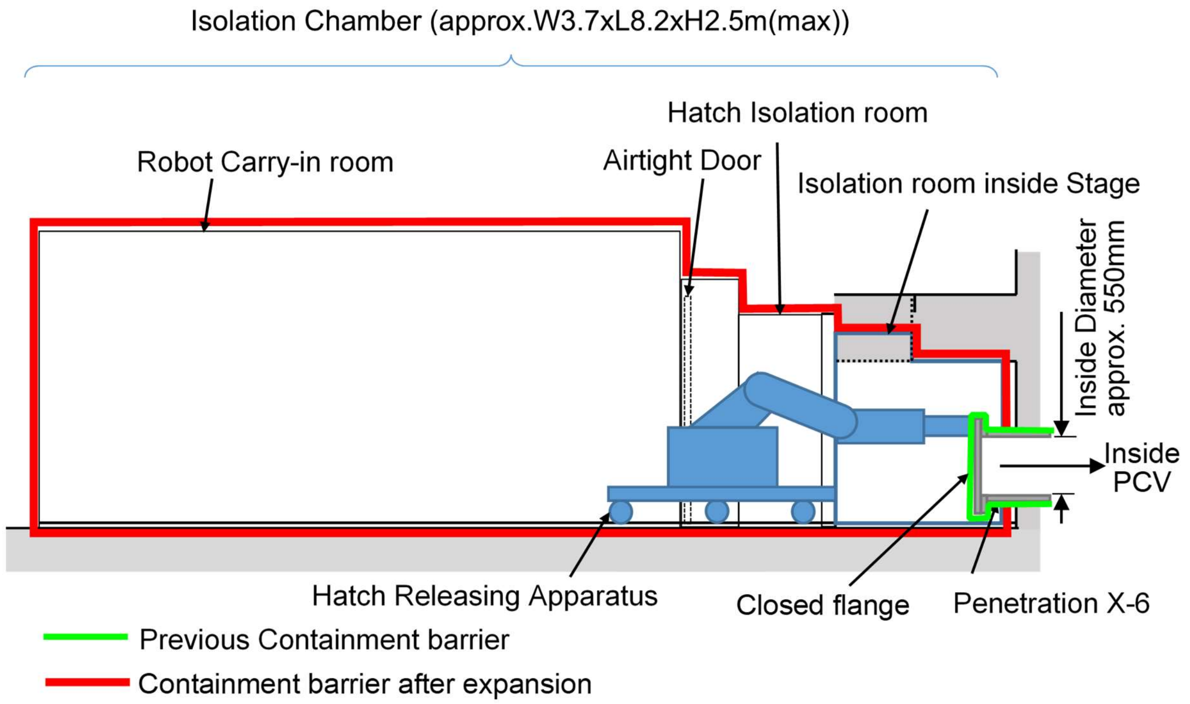
3.1.2.1.1 Development status and prospects of trial retrieval/internal investigation equipment

For the trial retrieval and PCV internal investigation in Unit 2, the operation will be performed by opening the flange of the penetration X-6 to make a larger opening than before, through which the arm-type access investigation equipment is moved in/out to retrieve fuel debris inside the PCV. In this operation, an expansion will be made to provide an isolation chamber (composed of a robot carrying-in room, etc.) to be built during opening the penetration X-6 (Fig. 10), and an enclosure to be newly provided (which will enclose an arm-type access investigation equipment, etc.) (Fig. 11), since the conventional containment barrier was located in the blank flange part of the penetration X-6. Although small in scale, this is a fundamental form of site construction for future retrieval work, in which an opening will be newly provided in the PCV to extend the containment barrier outside the PCV. This presents an approach that enters a new stage.

Thus, as the opening of penetration X-6 is a task related to maintaining to containment barrier function, careful consideration should be given to safety in particular. From the preparatory work stage, detail of the work should be thoroughly checked, potential risks should be identified, and countermeasures should be considered. For its implementation, it is important to conduct more careful and thorough preparation, verification and training.

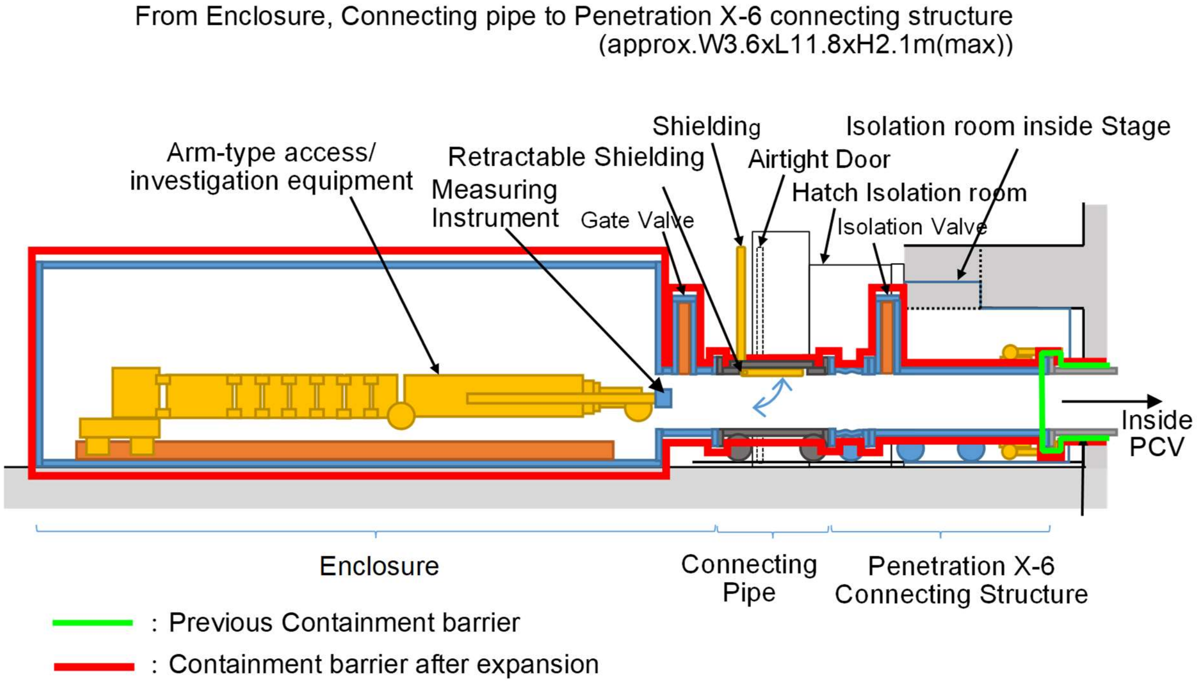
The fuel debris retrieval through the trial retrieval and PCV internal investigation was scheduled to begin within 2021, but the process has been delayed under the impact of the COVID-19 infection, considerations are being made to limit the delay to about one year. The performance confirmation test of the arm-type access investigation equipment in the UK has been delayed. As there was a possibility of further delay, reconsideration is given to cancel the performance confirmation test and mockup test simulating the on-site environment, transport the equipment to Japan, and conduct the performance confirmation test, mock-up test and training domestically. No changes due to this reconsideration have been made to the initial plan for testing and training menus.

Although it is important to make efforts to minimize delays caused by the COVID-19 infection, mockup testing that takes full account of uncertainties on site is also important in terms of actual site applicability and ensuring safety. In addition to simulating the severe environment on site, this mockup testing should clarify the parts that cannot be simulated and make enough preparations for practical application. NDF will also confirm the test plan and whether measures are sufficient.



(TEPCO material edited by NDF)

Fig. 10 Schematic drawing of isolation chamber to be installed at penetration X-6
(during opening the Penetration X-6)



(TEPCO material edited by NDF)

Fig. 11 Schematic drawing of enclosure to penetration X-6
(during trial retrieval and PCV internal investigation)

The key technical issues, countermeasures, and points to consider are described below.

- Dust dispersion prevention associated with removal of deposits in the penetration X-6.

In order to prevent dust dispersion due to removal of deposits inside the penetration X-6, measures such as low-pressure water cleaning of deposits, spray curtains to the penetration outlet and lowering pressure in the PCV are in preparation, in consideration of the changes in dust concentration during drilling the inner door of the penetration X-2 for Unit 1 PCV internal investigation (June, 2019). During AWJ cutting of obstacles from the penetration X-6 to the pedestal opening (ground floor), it is also planned to perform the work after lowering PCV inner pressure.

For the removal of deposits in the penetration X-6 and AWJ cutting of obstacles, it is essential to develop a detailed work plan in accordance with the preliminary implementation and use it in the later stages as mentioned in 2.3.2 and ensure safety accordingly (for example, break down work steps; proceed to the next step after confirming there is no problem with the dust dispersion monitoring results in each step; and take countermeasures and proceed to the next step if any sign of abnormality is observed).

NDF will confirm whether a sufficient level of safety is secured for performing work, including whether the work plan is well-developed by TEPCO, the plan is fully implemented without fail, and work is stopped when necessary.

・ Considerations for the risk of spreading impact of the COVID-19 infection

The performance confirmation test and mockup test simulating the on-site environment in the UK have been canceled, and tastings by bringing the equipment into Japan are being promoted. For the performance confirmation test in Japan, it is essential to secure the UK engineers, and it is necessary to maintain the backup system on the UK side in the event of a defect, while sharing information and communicating smoothly with the UK engineers.

It is also important to make all possible preparations for the risk of the COVID-19 pandemic expanding in Japan. NDF will confirm these responses.

・ Considerations in project management

It is important to proceed with the project while paying attention to the process progress management of the contractors including overseas enterprises and subcontractors. As part of their project management activities, TEPCO needs to make further efforts to perform prior-evaluation of risk of delays, and develop alternative plans and measures to prevent the occurrence of risks. NDF also participates in meetings with contractors and their subcontractors to closely check the status and support risk assessment.

・ Limitations in the scope of trial retrieval and internal investigation, and incorporation into gradual expansion of the retrieval scale

In the PCV internal investigation using a robot arm, it is planned to ascertain the state of existing structures, and the distribution of deposits inside the pedestal (3D data), the distribution of gamma rays and neutron counts at the bottom and on the platform, in as wide a range as possible. However, since more structures and platforms in the pedestal remained than the initial design plan, the range in which the arm can access the bottom of the pedestal is limited. Thus, the possible range of neutron measurement and trial retrieval from the bottom of the pedestal is limited. Assuming that debris at the bottom of the pedestal cannot be retrieved, it is also planned to retrieve the deposits on the platform which are highly likely to be fuel debris, as same as those at the bottom of the pedestal. Given the limited scope of investigation and trial retrieval, greater consideration is required in advance to determine what information is needed to gradually expand the retrieval scale as a next step for promoting the retrieval work in a reliable manner.

・ Human resource development and technology transfer for the next step (gradual expansion of the retrieval scale)

With regard to the trial retrieval, there are uncertainties and difficulties in the development of the robot arm and the removal of deposits and obstacles due to a limited understanding of the conditions inside the PCV. Therefore, when performing such work, it is necessary for TEPCO and related institutions to utilize human resources with a wealth of field experience, including those invited from outside as needed, to develop human resources to foster field-oriented perspectives/feelings, and to transfer techniques cultivated through these activities.

3.1.2.1.2 Development status of gradual expansion of retrieval scale and prospects

The retrieval equipment to be used for gradual expansion of the retrieval scale will be improved by increasing the weight capacity and enhancing accessibility while complying with specifications of the devices for trial retrieval and PCV internal investigation.

It is planned to expand the range of retrieval step by step while making achievements, starting with retrieval of fuel debris that can be gripped and sucked, and expanding it to fuel debris retrieval with cutting. Consideration will also be given to the possibility of cutting platform beams and the range of cutting. The enclosure containing the arm-type access equipment, etc. is connected to the PCV via penetration X-6 connection structures to secure containment functions. In order to bring fuel debris into the enclosure, it is necessary to consider shielding, measures against hydrogen and prevention of the spread of contamination, methods for transferring fuel debris from the enclosure, and methods for confirming the maintenance of boundary and dynamic equipment functions and for remote maintenance.

From the perspectives of research/development and engineering by TEPCO, and in terms of ensuring actual site applicability and safety, NDF continues to observe and check the status of technology development and preparations for application to the site in a timely manner.

The key technical issues and countermeasures are described below.

・ Ensuring containment performance of enclosures for fuel debris

In the retrieval operation, the process is repeated from carrying debris retrieved from the PCV into the enclosure, storing in unit cans and to carrying out to the outside of the enclosure for on-site transportation. As a result, the enclosure gradually becomes contaminated, and it is important to secure the containment performance of the enclosure.

This work is performed by controlling the pressure in the enclosure as the arm-type access equipment is moved in/out. Therefore, in order to confirm airtightness performance and operation reliability, through the duration, it is important to perform prior mockup testing, post-installation testing of the equipment, and subsequent abnormality monitoring.

・ Ensuring reliability of a dual arm manipulator (Fig. 12)

The dual arm manipulator to be installed in the enclosure plays an important role in performing various operations and maintenance in the enclosure (cleaning and decontamination of the arm, replacement and maintenance of cutting/recovery device, storage of fuel debris in containers, carry-out of storage containers, decontamination in enclosures, replacement of parts, etc.), and thus it is important to ensure its reliability. Therefore, it is necessary to improve the reproducibility of work through a wide range of operation/maintenance training in advance, and to train operators.

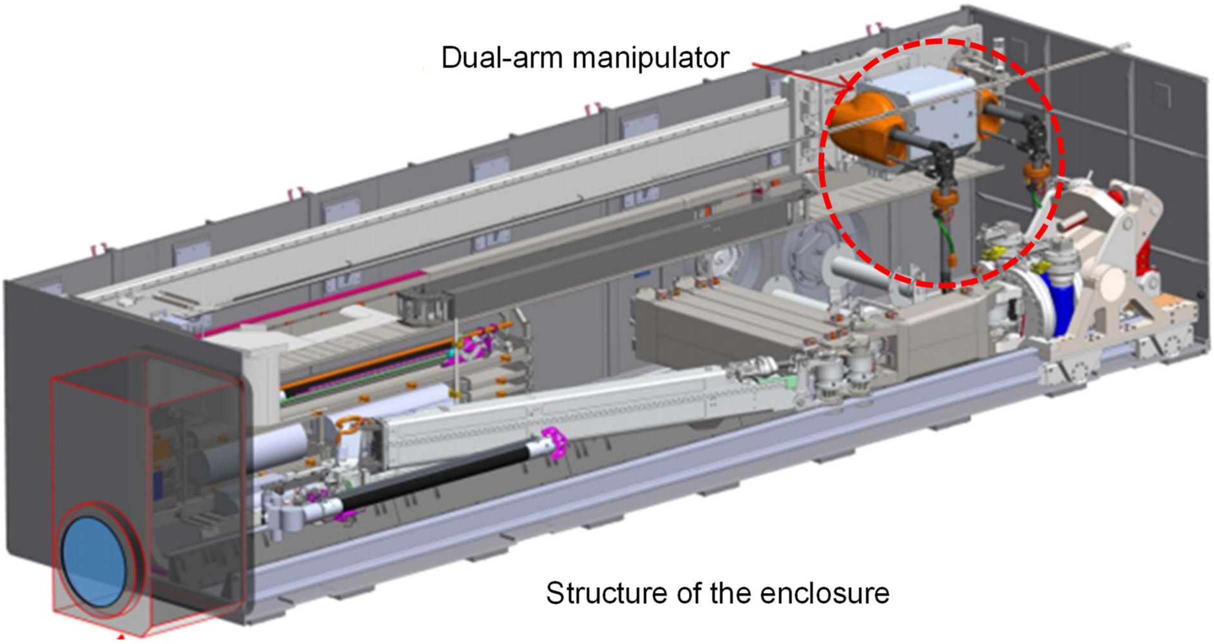


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Fig. 12 Enclosure and dual arm manipulator

・ Ensuring maintenance of devices and countermeasures during the in-service period

To expand the retrieval scale in a gradual manner, in addition to periodic maintenance, repair or replacement is required in case of failure. Since the radiation dose in the Unit 2 reactor building, where the enclosure will be installed, is high and it is difficult to perform maintenance in that place. Therefore, it is planned to construct a maintenance building outside the building, transfer the equipment or enclosure itself, and decontaminate, dismantle, repair or replace it inside the maintenance building.

In addition, since a dual arm manipulator that performs various operations may be repaired or replaced during the in-service period, the device to carry out the manipulator to the maintenance building is under development.

Since it is extremely important to ensure the maintenance of equipment/devices and their measures, including repairs, NDF will check the examination and preparation status for them in TEPCO. It is also important to leverage the experiences gained through the in-service maintenance of equipment/devices for further expansion of the retrieval scale. Therefore, a system that can reliably preserve maintenance records, including failure histories and their measures, should be established.

3.1.2.1.3 Further expansion of fuel debris retrieval

Toward further expansion of the retrieval scale, methods should be selected based on the viewpoint that “fuel debris retrieval is an important process in decommissioning, and its retrieval in a reliable manner affects the success/failure of the decommissioning project”, and from a comprehensive standpoint (in anticipation of technical feasibility as well as business continuity). In addition, TEPCO should take responsibility for selecting the method. Therefore, this section describes in detail how to select methods.

At the Fukushima Daiichi NPS, where uncertainty still exists, the uncertainty of the condition inside the PCV hinders examination, which forces preconditions to be set to perform examination. In judging technical feasibility in the future, it is important to clarify and examine the requirements (boundary conditions) and constraints (site use area, existing system interface, etc.) for methods and systems, including criticality control, dust containment, shielding, and heat removal.

TEPCO has been engaged in the conceptual study for further expansion of the retrieval scale. As part of this conceptual study, TEPCO will consider scenarios for fuel debris retrieval and, at the end of FY 2021, identify promising methods (top/sub candidates). Subsequently, based on the information obtained from the survey results, TEPCO will finally narrow down the possible methods (top/sub candidates), and decide the method with its design in mind.

The following are the points to be considered for examining retrieval scenarios and methods.

・ How to select retrieval methods

In selecting the method, based on the five principles mentioned in Section 2.2.2.3 (Safe, Proven, Efficient, Timely and Field-oriented), a determination should be made not only to satisfy the target of safety level (radiation exposure dose of the general public and workers, robustness against natural events such as earthquakes and tsunamis) but also to use the attributes (evaluation items) such as cost and schedule as indexes for selection. It is necessary to quantify these evaluation items as much as possible by using a multi-attribute utility analysis method[15], etc. It is believed that the most important factors in the process of selecting methods are the evaluation items used as indexes for determination and how to weight these indexes. In setting these indexes, a decision should be made not only based on the approach of TEPCO as an operator, but also from a comprehensive viewpoint developed through discussions with experts. Normally, the selection of methods should be based on the results of internal investigation. However, in a situation with many uncertainties like at the Fukushima Daiichi NPS, it is considered necessary to proceed with examination based on the currently available information and then to feed back the results gained from the investigation. As for the results of method selection, it is also important to make efforts to disseminate information in a careful manner so that the evaluation results will be widely accepted by society.

[15]Method for determining the relative merits and demerits for decision-making based not only on one attribute (evaluation item) but also multiple attributes (evaluation items). This methodology is applied to the process of selecting methods, and those with a high score calculated from “Σ (evaluation of each attribute (evaluation item)) x (weight of each attribute (evaluation item) = importance)” will remain as promising methods.

・ Development of retrieval scenarios

Given the limited understanding of the situation in the PCV, it is important to examine several scenarios of fuel debris retrieval by each unit and to clarify several paths from start to completion. In considering fuel debris retrieval scenarios, different cases are assumed, where fuel debris will be retrieved by the side-access or top-access methods, or combining both. Then, examinations including internal investigation required for each case will be made to consider several paths. This study intends to assume in advance different results obtained from PCV/RPV internal investigations or technical studies in the future, and then conduct examination based on the preconditions of using such results.

After reviewing these numerous paths, it is important to narrow down promising candidates for retrieval methods at a certain point of the path (e.g., narrow down promising candidates for the side-access method), and then further narrow down the path to take according to the information obtained afterward (e.g., perform internal investigation, and narrow down promising candidates for the top-access method).

It is also important to formulate a specific schedule with considering these fuel debris retrieval scenarios.

・ Clarification of requirements

With regard to further expansion of the retrieval scale, the methods will be considered, including those for containment, transfer, and storage of fuel debris, based on the findings from fuel debris retrieval in Unit 2 (trial retrieval, gradual expansion of the retrieval scale), PCV/RPV internal investigation, research and development and the on-site environment improvement, etc. In doing so, operations, devices and equipment, and facilities will be larger than, and the scope of construction will be wider than in the case of retrieving fuel debris from Unit 2. Therefore, much more attention should be paid in overviewing the entire Fukushima Daiichi NPS, including other work. In addition, because of the high radiation dose on site and the limited understanding of the situation inside the PCVs, the scope of work may also be extensive. Therefore, it is important to specify the requirements (containment, criticality, operability, maintainability, throughput[16], etc.) required for operations and devices more clearly and proceed with the work. Attention should also be paid to the interaction between the requirements.

・ Process for narrowing down promising retrieval methods

Based on the development results of the Project of Decommissioning and Contaminated Water Management, the latest findings at home and abroad are taken into consideration to derive ideas. Then, primary/secondary screening will be performed to gradually narrow down the methods. The primary screening verifies compliance with requirements and constraints, while the secondary screening quantifies and weighs each evaluation item. A multi-attribute utility analysis method, etc., is utilized for such screening process (Fig. 13). Diverse ideas are expected to be derived, but

[16] It represents the ability to retrieve fuel debris and indicates the processing time and work efficiency of the retrieval operation.

it is important to conduct objective evaluation based on this process and to narrow down the methods.

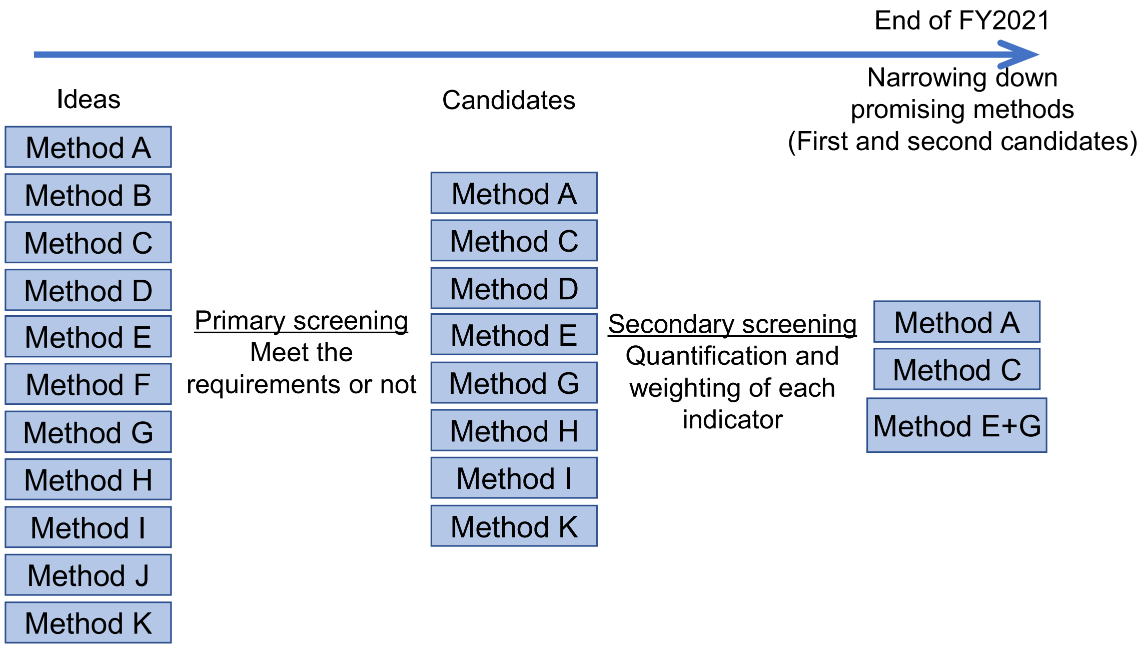

Fig. 13 Image diagram of the process for narrowing down retrieval methods

Based on the above considerations, NDF will evaluate the validity of the results of TEPCO's conceptual studies (Fig. 14).

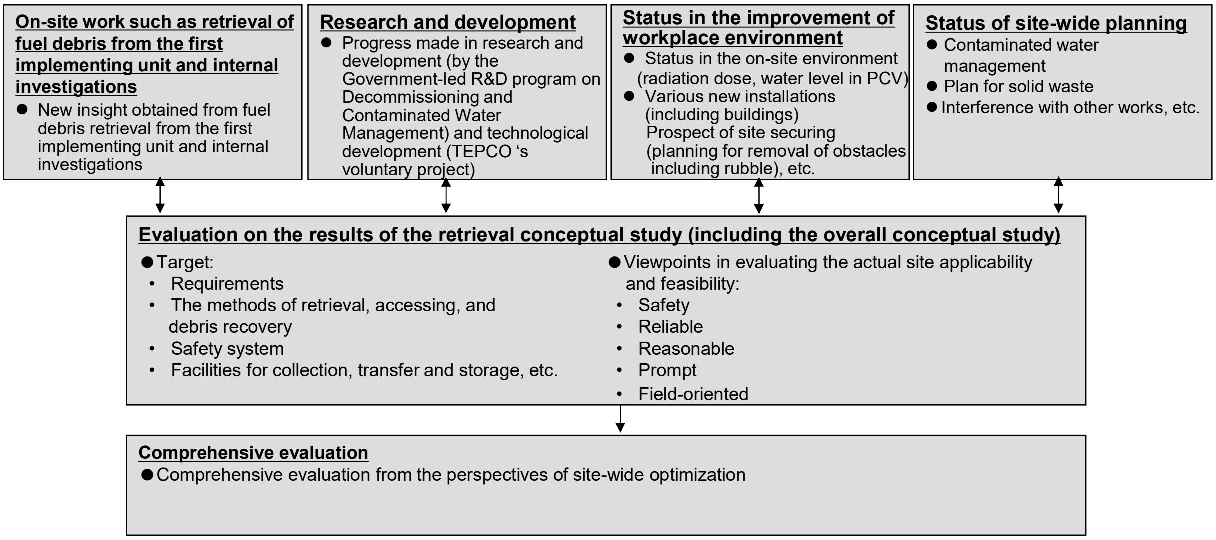

Fig. 14 Flow of retrieval method study (conceptual diagram)

Issues in each technical field for further expansion of the retrieval scale are described in Section 3.1.2.4.

3.1.2.2 Continuation of accident analysis activities (clarification of events that occurred at the accident and the process of accident progression)

Analysis of deposit samples collected by the previous internal investigation for fuel debris retrieval is in progress[17]. The information obtained by such investigation and analysis is directly incorporated into fuel debris retrieval methods and storage management. Moreover, examination and study in light of the accident history will promote understanding of phenomena, contribute to determining the cause of the accident and decommissioning, and indirectly to the enhancement of nuclear safety.

TEPCO and JAEA are cooperating in implementing activities for estimating and verifying individual events that occurred at the accident, including overheating, melting, chemical reactions and hydrogen explosions, the process of their progression over time, and the operation status of emergency cooling and depressurization equipment, by comparing the results of sample analyses with mock-up testing on accident progression and past scientific knowledge[18]. Moreover, TEPCO is independently investigating the operating floor [19],[20],[21] and standby gas treatment system[22] (hereinafter referred to as “SGTS”) and so forth in each unit.

The NRA responsible for continuing accident investigations/analyses[23] established “The Study Committee on Accident Analysis of the Fukushima Daiichi Nuclear Power Station” and investigated the operating floor in Unit 2, the inside of the reactor building in Unit 3, and SGTS filter lines, etc., with cooperation from TEPCO, and compiled an interim report[24]. Based on these investigations, the NRA has assessed that a large amount of Cs exists between the first and second layers of the shield plug installed on the operating floor in Units 2 and 3. Since the information related to such accident analysis may be affected by the dismantling of the facility due to the progress of decommissioning work, “Liaison Council on the accident investigation of Fukushima Daiichi Nuclear Power Station”[25,26] has been established, and communication and coordination for accident analysis and decommissioning has been conducted among NRA, Agency for Natural

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17 The 84th Secretariat Meeting of the Team for Countermeasures for Decommissioning and Contaminated Water Treatment, “Material 3-3: Analysis results of samples related to the internal investigation of the Unit 1 to 3 PCVs”

18 PowerPoint report on the development of analysis and estimation technologies for fuel debris characterization in the national project (To be updated as the material will be posted online around September 2021)

19 The 69th Secretariat Meeting of the Team for Countermeasures for Decommissioning and Contaminated Water Treatment, “Material 3-2: Investigation on obstacles and well plugs in the SFP of the Unit 1 reactor building”

20 The 88th Secretariat Meeting of the Team for Countermeasures for Decommissioning and Contaminated Water Treatment, “Material 3-3: The 24th meeting of the Team for Countermeasures for Decommissioning and Contaminated Water Treatment/Secretariat, “Material 3-2: Preliminary report on investigations of the operating floor of the Unit 2 reactor building”

21 The 24th Secretariat Meeting of the Team for Countermeasures for Decommissioning and Contaminated Water Treatment, “Material 3-2: γ-ray spectrum measurement results of the operating floor of the Unit 3 reactor building”

22 The 84th meeting of the Study group on monitoring and assessment of specified nuclear facilities, “Material 4-3: Surveillance results for removal of SGTS piping at Units 1 and 2”

23 Article 4, Paragraph 1, Item 11 of the Act for Establishment of the Nuclear Regulation Authority stipulates the scope of authority as “Affairs concerning investigations of causes of accidents that have resulted from the operation, etc. of reactors and causes of damage that has arisen from nuclear accidents”.

24 The 19th Study Committee on Accident Analysis of the Fukushima Daiichi Nuclear Power Station, “Material 4: Interim report on investigation/analysis of the accident at the TEPCO Fukushima Daiichi NPS”

25 NRA, " Material 3, Continuous survey and analysis on accident of Fukushima Daiichi Nuclear Power Station” (27th meeting)

26 NRA, The 7th Committee on the accident analysis of the Fukushima Daiichi Nuclear Power Station “Material 2, Continuous survey and analysis on accident of Fukushima Daiichi Nuclear Power Station”」

Resources and Energy, TEPCO and NDF. As an example of approaches to balance decommissioning work and accident investigation, in the partial removal of SGTS piping for Units 1 and 2[27], TEPCO is planning to take measurements of the radiation dose rates and with a gamma camera[28], and to collect samples of the piping for analysis after cutting. As a countermeasure against radioactive dust emission during cutting, injection of urethane form and local exhaust ventilation are being considered to ensure that the decommissioning and accident investigation will not affect the surrounding residents and the environment.

TEPCO is also planning to systematically conduct on-site investigation while estimating the state of the reactor core and PCV, and unsolved issues[29]. After a lapse of 10 years since the accident, the radiation dose has decreased due to the decay of fission products (hereinafter referred to as "FP”) and environmental improvements on site, and accessibility to the reactor buildings has improved. However, there are still many places with high radiation doses. In order to locate FP released at the accident with a low radiation exposure dose, it is important to collaborate with each organization, and continue these activities to a reasonable extent to clarify the events that occurred at the accident, the process of their progression and the equipment operation status. When new facts about the accident are revealed through further investigations and other activities, it is also important to deepen and incorporate knowledge by performing severe accident progression analysis evaluations, etc.

Since the shield plugs are installed on the operating floor, , PCV internal investigation and gradual expansion of the retrieval scale from the first floor of the reactor building are not directly affected by the trial retrieval. In order to further expand the retrieval scale, however, it is important to fully recognize high radiation doses in the shield plugs, taking into account the possibility that access from the operating floor (top-access) is required, and examination of retrieval methods with decontamination, shielding, and containment in mind.

3.1.2.3 Technical issues for technical requirements and future plans

3.1.2.3.1 Technical issues for ensuring safety of fuel debris retrieval work

In general, when considering ensuring safety at a nuclear facility, a series of evaluations are conducted by assuming accident scenarios in which the potential hazards of the facility become materialized, evaluating that these scenarios fall within the safety standards, and confirming that the safety measures are appropriate. In an ordinary nuclear power plant, such a series of safety assessment procedures are established and standardized by national regulations and guides. However, since there are no established and standardized regulations and guides for the decommissioning work at the Fukushima Daiichi NPS, it is necessary to organize an approach to

[27] NRA, The 21st Committee on the accident analysis of the Fukushima Daiichi Nuclear Power Station “Material5-1, Partial removal of SGTS piping of Units 1 and 2 of Fukushima Daiichi Nuclear Power Station

[28] A camera that visualizes gamma rays by superimposing the gamma-ray detection results from the radiation detector and the images from the camera

[29] Tokyo Electric Power Company Holdings Inc., “Estimation of the state of the reactor core and PCV in Units 1 to 3 of the Fukushima Daiichi NPS, and discussion on outstanding issues, the 5th progress report”

ensuring safety based on the safety-related features of the Fukushima Daiichi NPS and to share it with the parties concerned.

Decommissioning work of the Fukushima Daiichi NPS containing the reactors involved in the accident is an unprecedented activity that takes place in a peculiar environment different from that of a normal reactor, and therefore, to ensure safety, the following characteristics (peculiarities) regarding safety should be fully recognized:

- A large amount of radioactive material (including α-nuclides that have a significant impact in internal exposure) is in an unsealed state, as well as in unusual (atypical) and various forms
- Barriers for containing radioactive materials, such as reactor buildings and PCVs, are incomplete
- Significant uncertainties exist regarding the state of these radioactive materials and containment barriers, etc.
- Difficulty in accessing the site and installing instrumentation devices to obtain on-site information due to constraints such as high radiation levels on site
- Since the current level of radiation is high and further degradation of containment barriers is a concern, it is necessary to take measures in consideration of the time axis without prolonging the decommissioning activities

Based on these characteristics, NDF is organizing an approach to ensuring safety with the following as the basis:

- Optimization of judgement with safety assessment as its basis:

  When making decisions by comprehensively considering technical reliability, reasonableness, promptness, etc., safety assessments should be fully used to avoid a great variance in decisions for safety measures (to avoid too excessive or too insufficient resource input).

- Ensuring timeliness in decommissioning activities:

  While paying attention to the prevention of accidents and the mitigation of their impacts, measures should be taken with concentration on the time axis so as not to prolong the decommissioning period, in consideration of high radiological impacts that have already become apparent, as well as further deterioration of containment barriers, etc.

- Complementing design by operating controls, monitoring, analysis, and on-site operation in the event of an abnormality:

  Due to significant uncertainties, there is a limit to addressing all situations by design alone. For this reason, the information obtained at the operation stage, including that obtained through monitoring and analysis, should be utilized in design, and design should be complemented by operators' efforts and on-site operation to enhance safety in total with operations. In preparations for abnormalities, consideration should be given to on-site response considering the characteristics that the progress of abnormalities is moderate and there is sufficient time to respond.

In addition, along with organizing the concept for ensuring safety, technical requirements have been established for ensuring safety of fuel debris retrieval and intensive studies are being conducted as shown in the following Sub-sections 3.1.2.4.2 to 3.1.2.4.7.

3.1.2.3.1.1 Establishing the containment functions (gas-phase)

Dispersion of radioactive material in ordinary operating nuclear power plant is prevented by keeping interior of a reactor building under negative pressure against the ambient air (active containment function by negative pressure control) and pressure in a PCV is maintained to be equal to that inside of a reactor building (passive containment function). However, the reactor buildings, PCVs, etc. of the Fukushima Daiichi NPS were partially damaged by the hydrogen explosion and their containment function is deteriorated. Due to this, establishment of an active containment function by negative pressure control is being considered during fuel debris retrieval work. Moreover, from the perspective of prevention of hydrogen explosions due to steadily generated hydrogen by the process of radiolysis of water and of corrosion (inactivation) of structural materials due to the presence of oxygen, nitrogen is injected into the PCV to maintain it in a nitrogen atmosphere. As for the exhaust from inside the reactor buildings, the release of radioactive materials has been prevented by the PCV gas control system, which is furnished with filters to remove radioactive materials and measure radioactivity[30].

We expect that existing safety systems will be able to cope with the retrieval of fuel debris, such as gripping and sucking, in the case of a trial retrieval or gradual expansion of fuel debris retrieval. In the subsequent work such as fuel debris cutting, it is necessary to construct the containment function of the gas phase system in consideration of re-scattering of Cs, etc., that adhere to the equipment and structures in the PCV, aerosolization of water containing radioactive materials, and generation of short-lived iodine and noble gases if criticality should occur.

In addition to re-scattering of Cs, etc., the fact that dispersed fine particles (α-dust) containing α-nuclides may be generated and the radioactivity concentration in the PCV gas-phase may increase is a concern. Therefore, dispersion of α-dust from inside the PCV must be suppressed as much as possible, and a function for containing the gas-phases should be provided to make the radiation dose impact on workers and the public fall within the allowable value.

Accordingly, it is reasonable to expand the retrieval scale while understanding the tendency of α-dust dispersion at each stage of expanding the fuel debris retrieval scale, and verifying the appropriateness of the containment function built in the subsequent stage. In TEPCO’s engineering work, the improvement of dust monitoring installations inside the reactor buildings and the study of lowering pressure or reducing pressure in the PCVs using existing equipment are in progress based on the outcome of the Project of Decommissioning and Contaminated Water Management. In the future, the effect on the surroundings will be assessed based on the monitoring results of the

[30] TEPCO, evaluation results of the additional release amount from the Units 1 - 4 reactor buildings (June, 2020), the Team for Countermeasures for Decommissioning and Contaminated Water Treatment/Secretariat (80th) material 3-6, May 28, 2020
https://www.tepco.co.jp/decommission/information/committee/roadmap_progress/pdf/2020/d200528_11-j.pdf

changes in condition such as the dispersion of α-dust associated with the work, and the retrieval scale of fuel debris will be gradually expanded.

In the process, TEPCO is considering establishing a secondary containment function and studying its necessity through their engineering work, while assuming the possibility of an increasing impact on the surroundings.

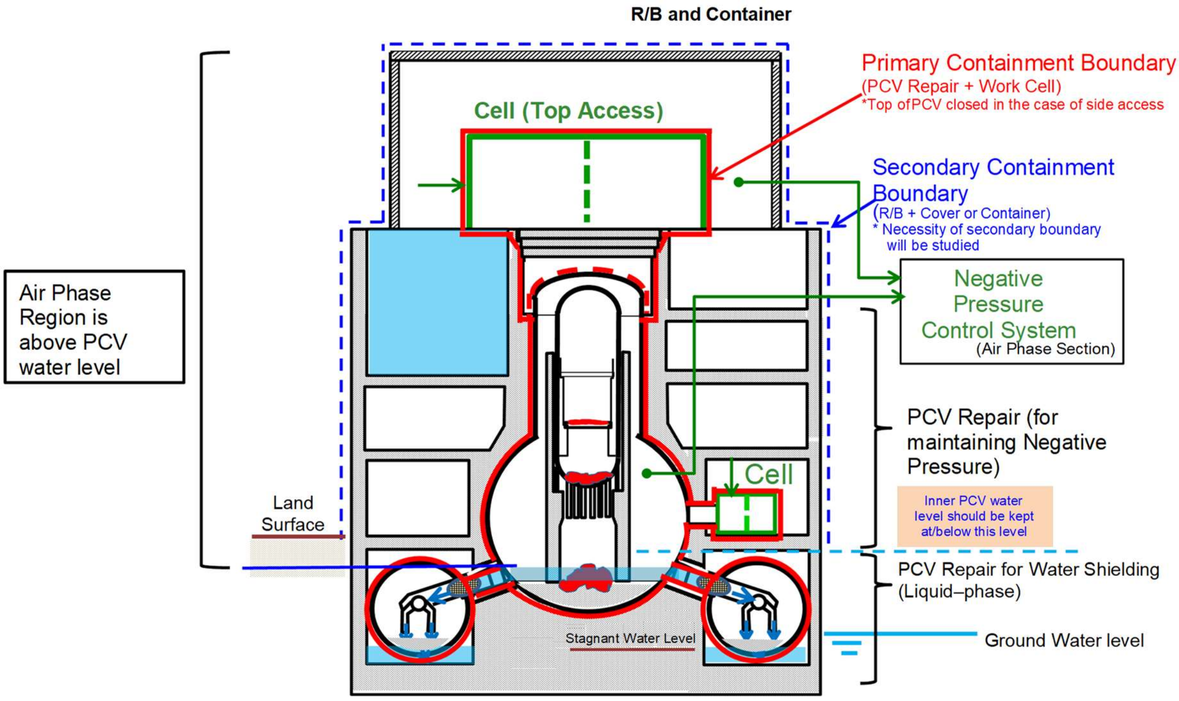

Fig. 15 Example of containment function (gas-phase) by negative pressure control

In establishing the containment function (gas-phases), technical issues to be addressed immediately for further expansion of fuel debris retrieval are as follows:

(1) Understanding of the dispersion rate of α-dust

As described above, for the purpose of fuel debris retrieval work, it is necessary to collect data such as α-dust dispersion rate and to establish measures to suppress the transition of α-dust to the gas phase as much as possible based on the collected data.

To collect data such as the dispersion rate of α-dust, it is necessary to plan demonstration and confirmation of the dispersion rate measurement during a trial retrieval and the gradual expansion of fuel debris retrieval. Moreover, in order to proceed with the review of technologies and R&D activities for the fuel debris retrieval method/system under conditions where such demonstration data have not been obtained, the general behavior related to α-dust dispersion should be roughly understood. For this purpose, verifications are currently underway by using the mock-up debris for developing analysis and estimation techniques to characterize the fuel debris.

In order to suppress the transition of α-dust to the gas phase, it is desirable to submerge the fuel debris and to fabricate it underwater as much as possible. However, the water level in the PCV is to be adjusted with other technical requirements such as the building of the containment function of the liquid phase described in the next section. Therefore, not all fuel debris can be fabricated underwater, and transition of α-dust to the gas phase is considered to be mitigated by splashing water on the fuel debris that is not submerged.

(2) Ascertaining the feasibility of negative pressure control in the PCV

A. Technical feasibility of negative pressure control based on the site conditions

In order to maintain negative pressure in the PCV, sufficient gas exhaust system capacity that considers the condition of PCV damage is required. Although damaged parts have not been identified yet, the exhaust capacity is currently established based on the relationship between actual nitrogen supply volume and actual PCV internal pressure. At this time, it is necessary to maintain sufficient differential pressure to respond to the internal pressure rise of the PCV due to abnormal events such as an internal temperature rise or shutdown of the gas exhaust system. In order to achieve this, repair of the damaged parts of the PCVs will be considered as necessary, but some difficulties such as remote work or exposure of workers are assumed due to work under high radiation dose conditions.

In this way, it is necessary to ascertain the technical feasibility of maintaining the negative pressure in the PCV based on the site conditions and the information obtained during trial retrieval or the gradual expansion of fuel debris retrieval.

B. Effect of air flow into the PCV during negative pressure control

Since air flows into the PCV in case of negative pressure control, measures of maintaining inactivated condition in the PCV by increasing nitrogen gas supply into the PCV will be examined as necessary based on evaluation on occurrence of accidents such as fire and hydrogen explosion using accumulated information regarding volume of hydrogen generated by radiolysis of water in the PCV and inflow volume of air (oxygen) into the PCV.

C. Study on the necessity of a secondary containment function

As illustrated in Fig. 15, for fuel debris retrieval, it is assumed that a working cell is newly installed as connected to the PCV under negative pressure control, and the work from retrieving the fuel debris to collecting the container for fuel debris retrieval into a transfer cask is performed in this cell. The PCV and this working cell have a primary containment function to prevent α-dust to the exterior (out-leakage).

In addition to this, in order to respond to an event in which radioactive materials are dispersed from the containment boundary caused by loss of primary containment function through negative pressure control, the necessity of the secondary containment function has been investigated by installing building covers and containers in the existing reactor buildings and maintaining slightly negative pressure inside of the reactor buildings to recover and treat radioactive materials. However, a large capacity gas exhaust system is considered to be required to maintain negative pressure in the secondary containment boundary since the reactor building has a large volume and its leak tightness may have deteriorated due to the accident. Therefore, based on the accumulated results of the tendency of dust dispersion obtained hereafter, it will be necessary to ascertain the required functions to establish a secondary containment function and to proceed with research and development accordingly.

D. Deterioration control of the containment function of the PCV

In order to maintain negative pressure in the PCV during the fuel debris retrieval work, it is necessary to manage risks of earthquakes and aging after considering how to handle deterioration of the primary containment function consisting of the PCV and the attached cell. This is outlined in Section 3.1.2.3.1.5.

(3) Study on exhaust gas management

In control of exhaust gas associated with negative pressure control, it should be confirmed that radioactive materials in gases that may contain nuclear fuel materials derived from fuel debris are maintained below the radiation dose standard for the public in the vicinity of the facility by measuring and controlling the release concentration and the release amount. In addition, α-nuclides derived from fuel debris should be included in the assessment and constantly monitored/measured during fuel debris handling so as to evaluate their normal fluctuation range in advance. By using such data, a system for early detection of abnormal events such as leakage and implementing appropriate impact mitigation measures should be established for preventing any impact on environment and workers.

The reliability and accuracy of the mechanical properties and chemical composition of the fuel debris needs to be improved because these are essential information for designing the decontamination equipment for efficient collection of radioactive dust.

As described above, regarding establishing a containment function (gas-phase), TEPCO has been ascertaining the feasibility of negative pressure control in the PCV through their engineering work, and in parallel, has started to examine the necessity of a secondary containment function, and since 2020 consideration is being given to establishing a secondary containment function. In the future, TEPCO will incorporate the results gained from the Project of Decommissioning and Contaminated Water Management, trial retrieval and gradual expansion of the retrieval scale, and give shape to the system specification necessary for establishing a containment function through their engineering work. (See Chapter 3.1.2.3.2.3 for the progress status of installing covers on the side and top of the Unit 3 reactor building to improve its containment performance.)

3.1.2.3.1.2 Establishing the containment functions (liquid-phase)

To mitigate the dispersion rate of generated α-dust and to minimize the transition to the gas phase, fuel debris cutting, etc., would be performed by pouring water over the fuel debris for fuel debris retrieval. Existing safety systems are expected to be capable of fuel debris retrieval by gripping and suction. For the subsequent work such as fuel debris fabrication and removal of obstacles, a large amount of α-particles will flow into cooling water (liquid phase). To prevent the cooling water containing α-particles from leaking to the environment, it may be of great importance to establish a cooling water circulation/purification system, and a liquid phase containment function in consideration of prevention of spread of contamination (Fig. 16).

For this reason, it is necessary to examine technologies for removing soluble nuclides that may be leached from fuel debris to the circulating cooling water as well as treatment technologies for solid matter trapped by the filter equipped in the circulating cooling water system. Accordingly, the c Decommissioning and Contaminated Water Management[31] has been promoting research and development. In parallel with this, the establishment of a PCV circulating cooling system that takes water from the PCV and injects it into the reactor for cooling, which is beneficial in terms of preventing the spread of cooling water containing α particles, was considered in research and development[32] by the Project.

To establish a reasonable containment function of liquid-phase in each stage of the scale expansion of fuel debris retrieval, it is rational to monitor the radioactive concentration of cooling water by stage and verify the validity of the containment function to be built in the subsequent stage based on the results (information on debris properties, etc.) obtained from research and development by the Project of Decommissioning and Contaminated Water Management. As with the containment function (gas phase), from the viewpoint of verifying/investigating the impact of the retrieval work on the liquid phase, TEPCO, through engineering work, has been discussing system addition/installation, etc., for the purpose of monitoring the circulating water system according to the results of the Project[33]. With regard to the effects on the liquid phase during fuel debris retrieval operations, the scale of fuel debris retrieval will be expanded gradually, based on the results of monitoring changes in the state of waste liquid containing α-nuclides. The water level in the reactor building is required to be maintained lower than the groundwater level to prevent the outflow of cooling water to groundwater and to appropriately control the water level in the PCV. Safety systems are to be established taking this into consideration.

In establishing the containment function (liquid-phases), the technical issues to be addressed immediately for further expansion of fuel debris retrieval are as follows:

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31 IRID, additional subsidies in FY 2018 for the Project of Decommissioning and Contaminated Water Management, "Technology development for further expansion of technologies to retrieve fuel debris/in-core structures", actual results in FY 2019, August 2020.
https://irid.or.jp/wp-content/uploads/2020/09/2019008kibonosaranarukakudai.pdf

32 IRID, additional subsidies in FY 2017 for the Project of Decommissioning and Contaminated Water Management, "Development of technology to establish primary containment vessel water circulation system (full-scale test), final report for FY 2019, August 2020.
https://irid.or.jp/wp-content/uploads/2020/09/2019006mizujyunkan.pdf
IRID, additional subsidies in FY 2017 for the Project of Decommissioning and Contaminated Water Management, "Development of technology to establish primary containment vessel water circulation system (full-scale test), final report for FY 2019, August 2020.
https://irid.or.jp/wp-content/uploads/2020/09/2019007mizujyunkanjitukibo.pdf

33 IRID, supplementary budget in FY 2016, "subsidies for the Project of Decommissioning and Contaminated Water Management", sophistication of retrieval method and system of fuel debris and internal structures, final report in FY 2018, July 2019.
http://irid.or.jp/_pdf/20180000_13.pdf

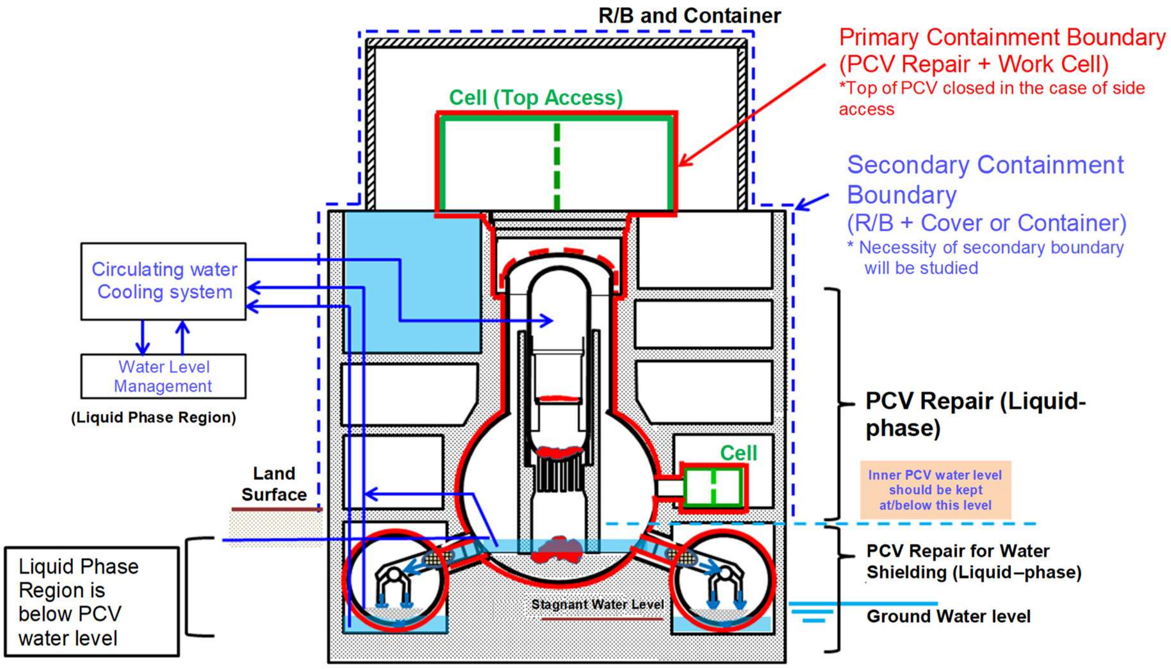

Fig. 16 Build example of containment function (liquid phase)

(1) Suppression of radioactive concentration increase in cooling water due to fuel debris retrieval work

As an approach to assess the radioactive concentration in the cooling water, it is necessary to plan to investigate the impact on waste liquid when gradually expanding the scale of retrieval.

To suppress an increase of radioactive concentration in cooling water in the PCV, cutting particles generated by fabrications such as cutting is planned to be collected through the PCV circulating cooling system to mitigate dispersion of dust. It is recommended that the results of monitoring by the water circulation system are taken into account and incorporate them into modifications of the PCV circulating cooling system as necessary in the stage of gradually expanding the scale of retrieval.

(2) Setting of water level in the PCV

Due to a low seismic margin of S/C support columns, it is recommended that the water level in the S/C is lowered, and thus, considerations are being given to lower the water level. In this instance, it is necessary to appropriately set and manage the water level in the PCV with consideration for the damaged condition of the PCV in each unit and prevention of the outflow of cooling water to groundwater (maintaining the water level in the reactor building below the groundwater level), etc., and to confirm that safety is ensured in terms of fuel debris cooling and control of dust dispersion.

As described above, regarding establishing a containment function (liquid-phase), based on the results of the Project of Decommissioning and Contaminated Water Management (technology for removing soluble nuclides that are considered to have leached from fuel debris into circulating cooling water, technology for treating solids collected by filters of the circulating cooling water system, etc.), since 2020 construction of a PCV circulating cooling system has been under investigation through TEPCO’s engineering. Going forward, based on the monitoring results obtained from trial retrieval and gradual expansion of the retrieval scale, consideration will be given to modifications required for the PCV circulating system while verifying the validity of the containment function to be established in the next phase.

3.1.2.3.1.3 Issues in cooling functions

Fuel debris generates heat due to the decay of radioactive materials. In Unit 2 after a lapse of 10 years since the accident, for example, the amount of heat generation has reduced to 1/1000 of the level at the accident. However, the maximum heat generation is still estimated to be 69 kW[34]. Therefore, unless cooling is continued, the surrounding materials gradually absorb the generated heat and there is concern that the following events may occur:

- Oxidation caused by increasing the temperature of the uranium oxide in fuel debris (increased O/U ratio) proceeds resulting in volume expansion, crack generation, and thereby progression of pulverization.
- Moisture in concrete structures is also dissipated and dried by heat, and then cracks occur to decrease the concrete strength.
- The inside of the PCV becomes dry, causing radioactive dust to be easily scattered and sent airborne.
- When water is injected after the PCV becomes dry, the water that contacts fuel debris turns into steam and raises the PCV internal pressure, which leaks from damaged areas with radioactive dust.

In order to prevent the above four events and excessive temperature rise of the fuel debris, the temperature is being maintained below 100°C (cold shutdown state) currently by conducting circulating cooling with water injection into the reactor. In FY 2019, water injection into the reactor was temporarily suspended with the aim of optimizing operation/maintenance management of cooling systems and emergency response procedures, etc. From FY 2020 onward, based on the results of the water injection shutdown test conducted in FY 2019, testing will be planned and performed depending on the purpose for the condition of each unit. Based on this policy, water injection shutdown tests are being performed for each unit (Unit 1: 5 days from November to December 2020; Unit 2: 3 days in August 2020; Unit 3: 7 days in April 2021), and based on the condition of lowering PCV water level during termination of water injection, it is planned to study water injection methods for the future such as further reduction of the amount of water injection.

In addition, during the fuel debris retrieval work, it is necessary to keep the temperature below the level at which the fuel debris retrieval device can continue to work without any problems for a long period of time.

However, it should be considered that the injection of cooling water may become redundant in the future due to further decrease in the amount of decay heat along with reduction in the remaining amount of fuel debris, as the fuel debris retrieval progresses.

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[34] Kenji Nishihara et al., "Evaluation of fuel properties at the Fukushima Daiichi Nuclear Power Station”, Japan Atomic Energy Agency (JAEA), JAEA-DATA/Code 2012-018(2012).

In maintaining this cooling function, the technical issues to be addressed for the time being include setting of the target temperature inside the PCV to make each task feasible, as well as the countermeasures to be taken under the assumption of cooling function abnormalities when each task is performed. While essential countermeasures would be to continue cooling by early recovery of the cooling water circulation system or by mobile equipment, etc., it is necessary to evaluate changes in the PCV internal condition based on the time margin in an emergency and to consider emergency response measures and procedures, etc., including collection of devices.

As temperature monitoring in the PCV is also evaluated by the temperature of devices and water around the fuel debris, the fuel debris temperature is not directly measured. In preparation for reducing the amount of cooling water, a temperature measurement of fuel debris must be made or, if direct measurement is difficult, examining techniques to estimate it from the temperature of the devices and water used for the current assessment. Therefore, examinations are being made through research and development by the Project of Decommissioning and Contaminated Water Management.

In addition, during fuel debris retrieval operation, the processing of cutting fuel debris while spraying water is conceivable from the perspective of dust dispersion control, and attention should also be paid to water level control inside the PCV, as well as controlling of the contaminated water generated.

From the above, monitoring parameters and their criterion need to be studied and prepared through TEPCO's engineering work in order to carefully promote fuel debris retrieval work while observing how this work will affect the existing circulating water cooling and purification system, as well as its cooling function.

3.1.2.3.1.4 Issues in criticality control

At present, monitoring of the concentration of Xe-135, which are short-half-life fission products, has shown no sign of criticality as the concentration remains lower than the criticality criterion of 1 Bq/cm3. In addition, the possibility of re-criticality of the fuel debris at the Fukushima Daiichi NPS is presumed to be low based on the expected condition of the existing fuel debris in engineering, because the alternation of molten fuel assemblies is not likely to reach criticality due to the abundance ratio with water, and the mixture of impurities, such as internal structures, can be expected in the course of core meltdown. Furthermore, the fuel debris is presumed to be scattered in a wide area beyond the core as a result of the accident progression. Even assuming the possibility that control rods might have melted down before the fuel elements in the course of core meltdown and that the optimum mixing of incidentally crushed fuel debris with water occurs, the possibility of criticality is considered to be small.

Though the possibility of criticality is low, fuel debris retrieval alters the shape, etc., of fuel debris. Therefore, it is essential to ensure reliable criticality management during retrieval without fail by investigating what conditions would lead to criticality if shapes, etc., of fuel debris change, and to establish an appropriate control method for ensuring prompt detection and shutdown in case of an unexpected criticality.

In the initial stage of fuel debris retrieval work, fuel debris should be retrieved by limiting the processing amount based on methods that will not significantly change the fuel debris shape, such as by gripping and sucking, as well as the estimated fluctuation of reactivity. Also, in the process of expanding the retrieval scale and cutting, the retrieved volume of fuel debris will likely be increased, while taking measures such as measurement of pre-work subcriticality and preparation for insertion of neutron absorbers. In addition, in the retrieval operations overall, unless criticality is unlikely to occur in consideration of the retrieval conditions, it is necessary to ensure the safety of fuel debris retrieval while evaluating for criticality by checking the amount of neutron flux fluctuation in the vicinity of the fuel debris, as well as reliable criticality prevention (criticality monitoring) that combines measures in design with operator monitoring and judgment to stop/resume work.

To maintain the subcritical condition more reliably for fuel debris storage after retrieval, it is important to store it stably while controlling the shape and size, for instance, by storing the debris in containers. In TEPCO's engineering work, the concept design is in progress for circulating water system configurations and system specifications for further expansion of fuel debris retrieval. Criticality assessment of equipment needs to be performed early in order to avoid rework at a later stage, and criticality prevention measures that greatly affect the equipment specifications have been extracted in advance.

For this criticality control, technical issues to be addressed for the time being are as follows.

(1) Establishment of criticality evaluation methods

An evaluation method has been developed to obtain the information on the conditions for reaching re-criticality of fuel debris based on the information to be obtained from each stage of fuel debris retrieval, including internal investigations, and to estimate the conditions for subcriticality and the degree of influence of criticality if it should happen. In conducting these evaluations, a plan should be made so that information on the critical parameters with high impact on the criticality evaluation can be obtained in the course of internal investigation and retrieval work, and be revised appropriately through information updates as needed for incorporating on-site information during preliminary work into safety assessment in later stages.

(2) Local neutron measurement around retrieval point

There are various kinds of existing neutron detectors according to application (fission chamber, B-10 proportional counter tube, semiconductor detector, etc.). Taking advantage of each feature, selection of neutron detectors for each stage has been considered. The required specifications for the neutron detectors for criticality monitoring are; (1) ability to survive the accumulated radiation dose (Gy) for the operation period and (2) installability of the assumed equipment (size/weight, cable diameter) or installability at a work site (size/weight, cable routing); and (3) the guaranteed detection efficiency at the required level (time, accuracy). Accordingly, optimal detector selection will be made according to the information on PCV radiation dose rate obtained by internal investigation and the progress of the equipment development of each unit.

In addition to subcriticality measurement, a small detector is highly likely to be used as a standalone detector for constant monitoring of local neutron measurement. Toward practical application of local and constant monitoring, studies are underway in the Project of Decommissioning and Contaminated Water Management, including specifications of neutron detectors and their actual location and quantity, and approach-to-criticality monitoring technologies in combination with evaluations based on the obtained data. In FY 2020, confirmation testing plan was formulated to verify the applicability of neutron detectors to the on-site environment. As specific operation methods, it is necessary to set up determination criteria on operation suspension/resuming if fluctuation of the neutron flux is detected, and on boron injection as a neutron absorber.

The possibility of criticality needs to be examined in places other than retrieval locations. For example, criticality may occur in places where fuel debris cutting particles accumulate and cannot be collected in the circulating water cooling system (i.e., outside the PCV bottom pedestal, in piping, water filters, waste water receiving tanks, etc.). Although criticality can be detected by the PCV gas management system, countermeasures will be considered in accordance with the feasibility of approach-to-criticality monitoring, criticality risk scenarios and evaluation results.

(3) Feasibility study of measuring the degree of subcriticality

When measuring the degree of subcriticality, in addition to the required specification of (2), it is necessary to select a highly sensitive detector in order to measure a weak neutron signal, capturing a neutron fluctuation in a very short time under a gamma ray environment. From examination performed to date, considering the equipment mountability (size/weight/electromagnetic noise) and operation methods (measurement time and duration, installation location, etc.) by sensitivity are key issues due to the necessity of lead shielding in a high γ-ray environment (assuming 1000 Gy/h). In the future, it will be necessary to consider neutron detector selection and optimization, applying the constraints (weight, size, cable handling, interference with arms, balance between measurement and fabrication time, etc.) introduced by the fuel debris retrieval methods and systems. Meanwhile, improving the actual site applicability of the detectors is under review in light of an approach to assess the γ-ray radiation dose rate and neutron count rate in the vicinity of the fuel debris as well as constant monitoring, and by downsizing with the combined use of several detectors, etc.

In addition, in order to verify the applicability to fuel debris, where various mixes of compositions and properties of the fuel debris are expected, it is necessary to assess the technical feasibility, including applicability, by planning and demonstration.

(4) Feasibility study of neutron absorber

Based on the information obtained in each stage of scale expansion, in preparation for cases where the level of fuel debris criticality is high or the applicability of measuring the degree of subcriticality has limitations, assessments on the required boron concentration and feasibility engineering studies on installations are in progress by TEPCO for cases of filling with sodium pentaborate during normal fuel debris retrieval. As a result, environmental impact in the event of leakage and compatibility with concrete as structural materials has been evaluated[35].

While examining operation details to maintain the boron concentration in consideration of the impact on the PCV circulation cooling system as well as the impact on systems and waste during segregation, collection, re-use and processing of boric acid, it is necessary to verify the applicability on site for the case of sodium pentaborate injection, together with the approach-to-criticality monitoring technologies described in (2) Local neutron measurement around retrieval point.

In addition, in the case that criticality should occur and emergency sodium pentaborate injection would be used to achieve a subcritical state, the methods for maintaining subcriticality after transition (lowering the water level, maintaining the boric acid concentration, etc.) as well as recovery methods need to be examined.

Moreover, to ensure maintenance of subcriticality when the margin for criticality of fuel debris is small, development is also underway for a non-soluble neutron absorber that can locally limit the impact on the PCV circulating cooling system. Until now, fundamental property testing and radiation resistance testing have been conducted to list B_4C metal sintered materials, glass containing B/Gd, Gd_2O_3 particles, and sodium silicate/Gd_2O_3 granulated powder as candidates for non-soluble neutron absorbers. For these candidates, the impact of long-term irradiation on the integrity of containers during storage of fuel debris, the method of spraying on fuel debris for debris crushing, and the effect after spraying have been verified. In FY 2020, the feasibility of a boric acid control system for a circulating water system was examined. In the future, technical development for on-site operation in combination with neutron detectors will be promoted.

To introduce non-soluble neutron absorbers, it is necessary to study the impact on PCV corrosion and environmental impact in the event of an environmental release.

(5) Detection of criticality by PCV gas management installations

Immediate criticality monitoring and sophistication of detectors in the PCV gas monitor is required to detect when approaching criticality and criticality in the vicinity of retrieval, and to detect criticality due to the fall of fuel debris and/or accumulation of powder debris in locations other than fuel debris retrieval. By measuring Kr-87/88 with high trackability for reactivity changes, in addition to Xe-135 that has already been measured, criticality detection can be accelerated and the level of subcriticality of the entire PCV can be presumed. Therefore, a method of practical application on site in combination with detection by neutron detector needs to be examined in the future[36].

3.1.2.3.1.5 Issues in the structural integrity of PCVs and buildings

As for the main equipment in the PCV/RPV pedestal, etc., and reactor buildings, their structural integrity has been evaluated in post-accident studies by TEPCO and the Project of

[35] IRID, supplementary budget in FY 2017, subsidies for the Project of Decommissioning and Contaminated Water Management, sophistication of retrieval method and system of fuel debris and internal structures (Technology development for establishing criticality control methods), final report, July 2019. http://irid.or.jp/_pdf/20180000_04.pdf

[36] IRID, supplementary budget in FY 2017, subsidies for the Project of Decommissioning and Contaminated Water Management, "Sophistication of retrieval method and system of fuel debris and internal structures (Technology development for establishing criticality control methods), final report, July 2019. https://irid.or.jp/_pdf/20180000_04.pdf

Decommissioning and Contaminated Water Management. As a result, it has been confirmed that the main equipment and reactor buildings have a certain level of seismic margin.

Hereafter, the existing main equipment and reactor buildings, as well as equipment/systems and buildings (including modified areas of the existing equipment/systems and buildings) to be newly installed for fuel debris retrieval over a relatively long period, should satisfy the functional requirements and (1) be capable of performing operations safely and (2) ensure the required level of safety against external events such as earthquakes and tsunamis. Assuming (3) long-term maintenance management, in addition, it is important to (4) feedback new knowledge to be gained from planned PCV internal investigations and debris analysis results, etc., to the design of fuel debris retrieval systems and the study of retrieval methods. The following shows the key functional requirements as examples.

○ Existing equipment and buildings (including modified areas. The impact of aging is also considered as necessary)
- Control deterioration of containment functions of PCV, RPV and reactor buildings, etc., and control/prevent large release of radioactive materials (maintaining containment functions).
- Reactor buildings, etc., safely support equipment/systems to be newly installed in the reactor buildings for fuel debris retrieval in addition to the existing main equipment (maintaining support functions).

○ Equipment/systems and buildings to be newly installed for fuel debris retrieval (including connections to the existing equipment/systems)
- Have functions according to design requirements and control/prevent large release of radioactive materials (ensuring containment functions).
- Safely support equipment/systems to be installed for fuel debris retrieval (ensuring support functions).
- New buildings, etc., provide a safe work environment as required (ensuring shielding performance, etc.).

In FY 2020, TEPCO has formulated a long-term maintenance management plan for existing on-site systems/equipment and buildings in consideration of the progress of time-related deterioration, and started implementation of the plan. When new facts about the accident are revealed through further investigations and other activities, it is also necessary to clarify the impact of the accident, especially damage, by performing severe accident progression analysis evaluations, etc. and to secure functions throughout the decommissioning period in consideration of the progress of time-related deterioration. Moreover, regarding the existing equipment/systems and buildings, the cooling function was maintained during the earthquake with its epicenter off the coast of Fukushima Prefecture that occurred on February 13, 2021,[37]

[37] Earthquake with its epicenter off the coast of Fukushima Prefecture. The maximum intensity of upper 6 was observed in Miyagi and Fukushima Prefectures. At the Fukushima Daiichi NPS, a seismometer installed on the

although lowering of the water level in the PCVs of Units 1 and 3 was confirmed. In light of this earthquake, in order to maintain and manage the equipment/systems and buildings with the above functions over the medium-and-long term, it is necessary to conduct impact assessments on the accident impact, aging and external events (earthquakes and tsunamis, etc.) anticipated during the decommissioning period. In view of the fact that the past assessment of these effects was limited, it is necessary to make maximum use of existing techniques and evaluation results for planning and implementing an investigation plan, in which remote control under a high radiation environment is a challenging issue, and to develop underlying technologies to understand the situation. In so doing, while giving priority to safety, it is useful to actively introduce the latest knowledge and achievements not only in the nuclear field but also in other fields.

As for existing and new equipment/systems and buildings, the loading conditions (layout, size, weight of the new equipment/systems, new openings on PCV/biological shielding walls, etc.) during fuel debris retrieval will be specified with further progress in designing. In order to ensure the structural integrity of equipment/systems and buildings, while considering the state of the site, examination will be promoted steadily based on the latest design information.

In the specific designing of new equipment/systems and buildings, it is important to define seismic classes and perform seismic evaluation accordingly. However, it is still challenging to repair and reinforce buildings and main equipment damaged by the accident in a high radiation dose environment. Because of that, earthquake ground motion and design criteria used in the design will be appropriately defined, taking into account the perspective of risk assessments.

3.1.2.3.1.6 Issues in reduction of radiation exposure during work

In accordance with the Mid-and-Long-term Roadmap and TEPCO's Mid-and-Long-term Decommissioning Action Plan, removal of obstacles and radiation dose reduction in the reactor buildings are in progress as improvement of the work environment in work areas/access routes. In the future, as work related to fuel debris retrieval, reduction of exposure during work such as removal of high-radiation dose equipment, etc. is an issue, and research and development has been promoted by the Project of Decommissioning and Contaminated Water Management to support TEPCO's engineering.

The main work areas related to fuel debris retrieval are high radiation dose areas such as inside the reactor buildings. Also, there comes the need to handle nuclear fuel materials containing α-nuclides from fuel debris with a large dose contribution in the case of internal exposure. Accordingly, enhanced control of not only for external exposure but also for internal exposure is essential for reduction of exposure.

Specifically, it is important to prevent excessive exposure to workers through appropriate radiation protection schemes depending on the working environment and working style. Regarding protection from external radiation exposure, the radiation exposure dose is evaluated considering the radiation sources and the radiation dose rate in the work area. Then, based on the three

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2nd basement floor (on the foundation plate) of the Unit 6 reactor building recorded the quake with its maximum acceleration of 235 gal. This is equivalent to the response level of about half of the seismic response analysis results of the buildings against the design basis earthquake ground motion (Ss).

principles, namely “time, distance, and shielding”, taking measures to reduce radiation exposure to as low as reasonably achievable will be needed.

Therefore, an appropriate combination of exposure reduction measures such as decontamination, shielding, remote technology etc. is to be selected, with the following ideas in mind.

- Priority should be given to reducing radiation exposure through a combination of remote technologies and decontamination. Then, plan on-site radiation exposure management for site workers by the “time, distance and shielding” approach.
- In the extremely high-radiation dose areas such as inside the PCV and torus rooms, work should be pursued by remotely controlled machines, etc. to avoid engaging personnel inside.
- With regard to the inside of the reactor buildings, except for the areas mentioned above, consideration should be given to the optimal combination of decontamination, shielding, removal of unnecessary objects, remote technology, and reduction of working time in order to keep the accumulative radiation dose for the entire project at a low level.
- Where remote technologies are employed, additional work will be required, such as the installation of systems, maintenance and technical troubleshooting, which must be taken into consideration in the above evaluation and planning.
- As for the decontamination tasks, the judgment between remote technologies and personnel employment must be made based on factors such as the radiation dose rate in the target areas, type of contamination, space for work, frequency of use, applicability and development situations of remote technologies, schedule and cost, etc.
- Priority must be placed upon areas where work requirements are clearly identified. Considerations must not be pursued if task requirements are unclear, or in a non-specific “betterment-oriented” manner such as to aim for an overall reduction of radiation dose.

Regarding protection from internal exposure, measures such as suppressing dispersion of radioactive dust and prevention of the spread of contamination are being taken and appropriate protective measures are to be selected depending on the target nuclides, airborne concentration and surface contamination density in the work area, to prevent inhalation ingestion and body contamination. In the event of intake, the effective radiation dose should be assessed using external counting (lung monitor) and bioassays. For this reason, it is important to select α-nuclides that are important for exposure assessment in advance, and to incorporate them into the control of airborne concentrations, standards for wearing protective equipment, and equipment calibration management. Controlling the surface contamination density in the work environment and the body of workers entering/leaving contaminated areas is also important to early detect the spread of contamination beyond the area division and to prevent an intake of re-suspended dust from loose contamination.

With the objective of dose reduction in long-term decommissioning, it is important to accumulate knowledge such as on-site operation experiences and lessons learned and to hand down knowhow.

For further expanding fuel debris retrieval, it is necessary to develop a database that enables sharing of information and prompt feedback for the next work plan.

In particular, in fuel debris retrieval operation, access to the PCV should be made from the penetration X-6, etc., after the work environment in the reactor building is sufficiently secured. To reduce the radiation exposure of workers in the reactor building, it is important to conduct sufficient investigations on the radiation dose distribution and state of contamination, including the contribution from the surroundings of the subject areas, to identify the source locations and intensity as much as possible and to build the radiation dose reduction plan. Upon adequate verification on the operation feasibility, the target dose rate in the work areas and access routes shall be set in consideration of the margin for the radiation exposure dose limit (50 mSv/year and 100 mSv/5 years) for workers specified by laws and regulations. In the radiation dose reduction plan for high radiation dose areas, it is important to take management measures to reduce the total radiation exposure dose to as low as reasonably achievable and accomplish operations with respect to work hours in accordance with dose limits and required work hours to accomplish operations. Based on these, and as R&D tasks by the Project of Decommissioning and Contaminated Water Management, the development of technologies to identify radiation sources using environmental survey data and to digitize the environment and radiation source distribution visualized by digital technology, for the formulation of safe and efficient work plan has been in progress since FY 2021. Moreover, in developing remote technologies for environmental improvement and removal of obstacles under high radiation dose, the obstacles to be removed have been selected and the elemental technologies have been extracted in accordance with the required functions, and technical investigation is underway since FY 2020.

In addition, adequate radiation exposure control should be conducted after formulated a long-term work plan that includes not to concentrate workers' radiation exposure on individual workers and to help reduce whole workers' radiation exposure. Support should be provided to develop a database that can improve work plan efficiency and radiation exposure control, and to establish a system that manages and operates various information on the entire Fukushima Daiichi NPS in an integrated, step-by-step manner.

3.1.2.3.2 Technical issues related to fuel debris retrieval methods

3.1.2.3.2.1 Issues in securing access routes

For carrying in, installing, and carrying out devices and equipment used for fuel debris retrieval work, and transporting fuel debris and waste, access routes should be established by removing obstacles on the access routes and reducing the radiation dose in the R/B to the level at which such tasks can be performed. When establishing new openings in the PCV or the like to construct access routes to fuel debris, suppression of the release of radioactive materials from the PCV and RPV and maintaining the integrity of existing structures should be kept in mind taking into account of the gas phase containment function described in Sub-section 3.1.2.3.1.1.

The Mid-and-Long-term Roadmap indicates that the first implementing unit would be Unit 2 and trial retrieval begins toward gradual expansion of fuel debris retrieval. Accordingly, TEPCO is currently proceeding with specific engineering studies to conduct an access route from penetration X-6 in Unit 2.

On the other hand, toward a further expansion of fuel debris retrieval, studies are underway on the construction of access routes from the side opening of the PCV to fuel debris (the side access method), based on the results of research and development conducted to date by the Project of Decommissioning and Contaminated Water Management. In the side-access method, the issue is to address containment, shielding for connecting structures between newly installed heavy structures and the side-opening of the PCV, and seismic displacement, and technical development of the method is underway using lightweight cells and fixed rails as well as access tunnel systems.

As for the construction of access routes including top access (top-access method), in addition to side access, the technology for removing obstacles and transportation methods that can shorten the preparation processes for retrieval are under study in order to enhance throughput. Since FY 2020, the Project of Decommissioning and Contaminated Water Management has been examining the feasibility of a method to cut, retrieve and transport interfering structures as a single or large unit while ensuring containment and shielding. The NRA pointed out the possibility that a large amount of Cs might exist on the undersurface of the shield plugs of Units 2 and 3 from their evaluation of TEPCO's radiation dose measurements on the operating floor. Therefore, the top-access method requires consideration of constructing access routes with this in mind.

In the future, based on the above-mentioned issues, it is necessary to clearly define the access route to be built at the next stage from the data obtained at each phase of scale expansion. In particular, when cutting the inner door of the penetration X-2 in unit 1, the dust concentration in the PCV increased more than expected before the start of the work, and the pressure in the PCV dropped while installing a camera chamber for investigating obstacles. Therefore, not only countermeasures to prevent dust dispersion, but also the extra time required for responding to when faced with such an unexpected situation need to be considered and planned.

Since the fuel debris retrieval policy stipulates that the optimum combination of retrieval methods should be selected depending on the location where fuel debris exists for each reactor Unit, it is important to proceed with research and development toward a planned scale expansion.

3.1.2.3.2.2 Issues in development of devices and equipment

In each phase of trial retrieval, gradual and further expansion of fuel debris retrieval, devices and equipment for fuel debris retrieval need to be developed with emphasis on safety, reliability, and efficiency. To flexibly respond to the situation inside the RPV and the PCV bottom where the fuel debris is predominantly present, the specifications of devices/equipment to be developed in these phases should be established in consideration of radiation resistance, dust resistance, waterproofness, range of temperature, remote inspection/maintainability, remote operability, securing visual field, seismic resistance, protection mechanism for collision avoidance or automatic shutdown in case of abnormality, high reliability, appropriate redundancy, a rescue mechanism that does not disturb the subsequent work when a problem occurs, and efficiency of fuel debris retrieval.

Equipment development for trial retrieval and gradual expansion of fuel debris retrieval has progressed as part of research and development of the Project of Decommissioning and Contaminated Water Management. After the gradual expansion of fuel debris retrieval, TEPCO needs to take over and substantiate the development results. TEPCO is proceeding with the engineering of the robot arms, etc., to be applied to Unit 2, while preparing education/training scheduled to be started from 2021 for the fuel debris retrieval operations using these remote devices. Prior to installing remote equipment such as robot arms on site, adequate performance verification and operation training are essential by using mockups simulating the expected PCV internal environment. For this purpose, development of mockup systems is required.

As for devices and equipment for further expansion of fuel debris retrieval, development is underway for retrieval methods for improving efficiency, retrieval/handling systems according to diverse conditions of fuel debris, and dust collection systems for dust generated during fuel debris fabrication. Specific examples of ongoing development activities include: research and development for removal of large-scale structures by the top access and removal of obstacles by top and side access methods; development of fuel debris cutting/fabrication systems (mechanical, thermal) and dust collection/dispersion control systems; development of retrieval/collection systems according to diverse conditions of fuel debris (fragments, sludge, fine powders, etc.); remote operation support for robot arms, etc.; on-site transportation of unit cans; removal of soluble nuclides in circulating cooling water; processing of deposits collected from the PCV; and prediction of dust behavior in the PCV. Furthermore, techniques for installing the devices and equipment used for fuel debris retrieval are required. Furthermore, techniques for installing the devices and equipment to be used for fuel debris retrieval are required. On the assumption of remote operation, research and development has been in progress since FY 2020 for installing work cells to establish radiation shielding and gas phase containment functions and for connection methods with the existing structures. Since the stage of further expansion of the retrieval scale is prolonged, and in order to ensure safe, efficient and continuous retrieval work, it is also important to develop integrated support technologies for decommissioning of the Fukushima Daiichi NPS, such as development of rational maintenance technologies for a wide variety of remote devices and equipment, and system development to continuously monitor environmental changes in the PCV along with work progress. These development activities have been promoted since FY 2021. TEPCO is currently conducting a concept study for selecting a method for further expanding the scale of retrieval, and plans to narrow down the number of methods for the subsequent design study by the end of FY 2021. Further development of devices and equipment should be planned and promoted in light of the selected method.

As for how to proceed with development, it is necessary to flexibly promote operations in the subsequent phase based on the information gradually obtained from preceding investigations and retrieval work, and to continue development for emerging important issues. The developed devices and equipment need to be combined as a system and undergo a series of mockup tests to demonstrate that they can realize their performance safely and reliably at the actual site. These mockup tests need to be implemented in a facility simulating the on-site environment in order to verify the applicability of remote equipment and operability/maintainability of the entire remote system under severe environmental conditions containing significant uncertainties. Therefore, in cooperation with the organizations concerned, NDF and TEPCO have been engaged in examining how to proceed with the remote mockup test plan, the test plan review, the scope of the mockup facility to be maintained, the time required, and operation management, etc. From 2021, it is expected that TEPCO will take the initiative in promoting examination and materialization.

3.1.2.3.2.3 Issues in system installations and working areas

Assuming that safety functions are ensured, and considering avoidance of excessive system specifications, it is necessary to examine the establishment of system installations, etc., take necessary measures such as system additions based on the results of such examination, and then to operate them properly. In carrying out examination, sufficient areas should be secured to satisfy the required environmental conditions while considering installing shields for reducing radiation exposure for workers in addition to system installation, operation/maintenance management.

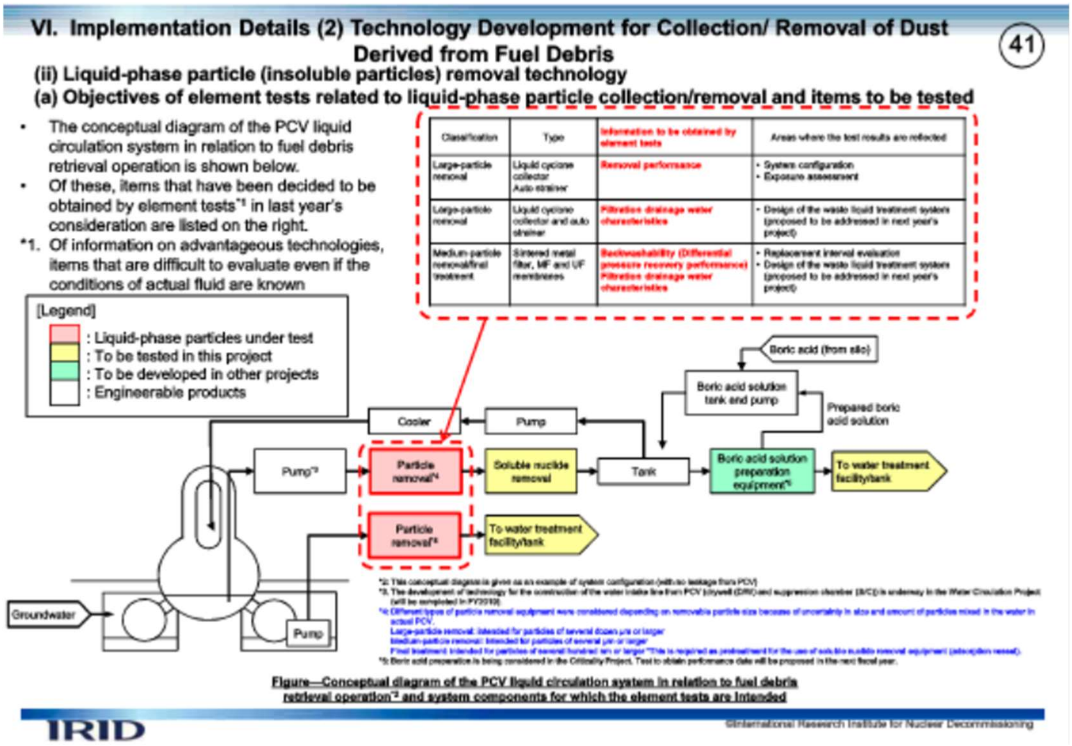
The system installations include a negative pressure control system required for establishing containment functions (gas phase), a circulating water cooling and purification system required for maintaining the containment functions (liquid phase) and cooling functions, and a criticality control system required for controlling criticality. Moreover, realization of measurement systems (for pressure, temperature, water level, radiation, etc.) to monitor the PCV internal state is a significant issue, which is essential for fuel debris retrieval. In order to build safety systems incorporating the above, TEPCO, through its engineering work, has tentatively set assumptions (system design conditions) based on research and development by the Project of Decommissioning and Contaminated Water Management, and has been examining system design and layout.

Fig. 14 shows an example case of research and development for the liquid-phase system by the Project[38]. In this project, the system configuration and the outline of its specifications for particle collection/removal are under consideration. In engineering work by TEPCO, a basic plan for system configuration has been developed in reference to the results of the Project. A case study is also conducted on the feasibility of the system layout (installing devices in the reactor building or installing only some devices in the reactor building if there are restrictions on arrangement) under the reactor building environment (ease of securing equipment installation space, radiation dose in the subject area, etc.). Based on the result of this examination, further investigation will be given on feasibility as a safety system. As described above, in establishing safety systems, steady efforts

[38] IRID, supplementary budget in FY 2016, subsidies for the Project of Decommissioning and Contaminated Water Management, sophistication of retrieval method and system of fuel debris and internal structures, final report, July 2019.http://irid.or.jp/_pdf/20180000_13.pdf

are required to improve the quality of design by repeatedly adjusting system layout design, and reconsidering device specifications as the situation demands.

It is important to ascertain the feasibility of systems designed to secure the safety functions necessary for proper operation of equipment and devices for safe retrieval and storage of fuel debris, and to steadily proceed with studies on the fuel debris retrieval method through a reliable approach.



(Source : IRID)

Fig. 17 Example of examining the liquid-phase particle (collection/removal) in the Project of Decommissioning and Contaminated Water Management

To further expand the retrieval scale, the working area required for installing fuel debris retrieval equipment/related devices and system installations is now being investigated in TEPCO's engineering work. In considering the handling of high radiation dose areas in the reactor buildings and interference with other tasks, installing systems outside of the existing buildings is also being considered.

In order to improve the containment performance of the Unit 3 reactor building, constructing covers on the side and upper part of the building is under investigation. In this process, consideration is given to the surrounding environment in addition to securing the space necessary for cover construction. As the radiation dose around the reactor building is currently high, construction of the cover by remote operation is under discussion. Verification that simulates the actual condition is required to achieve accurate construction of the cover. However, it is expected that there will be a limit to improvement of airtightness. Therefore, in parallel with the investigation of the structure and workability, the environment around the reactor building should be improved from a comprehensive viewpoint so that workers can be engaged in key operations and airtightness can be enhanced.

3.1.2.3.3 Technical issues related to safe and stable storage of fuel debris

3.1.2.3.3.1 Issues in handling fuel debris (containing, transferring, and storing)

Before initiating fuel debris retrieval work, a comprehensive system should be established that consists of a series of steps from containing and transferring to storing of retrieved fuel debris furnished with safety functions such as maintaining subcriticality, containment functions, countermeasures against hydrogen generation, and cooling. Accordingly, examination of the following is being progressed until the end of FY 2020:[39]

- Development of basic specifications for the container, such as overall length in consideration of handling, internal diameter, quality of materials and lid structure in light of work efficiency and maintaining subcriticality, etc., and demonstration of the structural integrity of the container by testing.
- Examination of a practical and rational prediction method of hydrogen generation from fuel debris stored in containers; determination of a vent mechanism for hydrogen gas release on the container lid by using the said prediction method; and establishment of safe transferring conditions with consideration for accumulation of hydrogen gas in transferring casks.
- Development of efficient drying technology applicable to fuel debris in unit cans, and consideration on a drying system using this technology.

Moreover, in reference to the results of these studies, TEPCO continues their activities to materialize systems and devices/installations used for the process from containing to storing fuel debris, which are required for gradually expanding the scale of retrieval, in coordination with other associated projects. In addition, specific transferring routes, storing technologies/forms and its locations in light of the usage plan for the entire site are taking shape.

Since information and knowledge on the fuel debris properties is limited, the systems and devices/installations will be designed by conservatively assuming the fuel debris properties. Therefore, for rationalization it is important to incorporate to the extent possible the information and data on fuel debris properties (amount of hydrogen generated, data on fuel debris properties), as well as knowledge and experience on handling fuel debris during the operations from receiving the fuel debris by containers for on-site transferring to temporary storage that can be collected/accumulated during trial retrieval and gradually expanding the scale of fuel debris retrieval work into the design of systems and facilities for containing, transferring, and storing fuel debris at further expansion of fuel debris retrieval.

Unit cans and containers in which fuel debris is stored should be handled and maneuvered by a remote device in a safe and reliable manner. Therefore, the assumed mockup operations need

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[39] IRID, supplementary budget in FY 2018, “Subsidies for the Project of Decommissioning and Contaminated Water Management (Development of Technology for Collection, Transfer and Storage of Fuel Debris)”, FY 2020 final report, June 2021

https://irid.or.jp/wp-content/uploads/2021/06/2020008syuunouisouhokanFIX_20210615r2.pdf to be executed (e.g., handling of unit cans and containers as well as fuel debris sampling for analysis by using actual or similar equipment/device including a remote device) in the initial stage of detailed design. Moreover, in terms of preventing design change/modification, the approach through the mockup operations, which determines the specifications/sizes of these remote devices and the devices/equipment required for containing, transferring, and storing fuel debris, their layout, and the flow of fuel debris handling, is considered to be useful. In developing the specific installations and systems for handling and storing retrieved fuel debris, it is also necessary to give consideration to the installations with respect to applying safeguards requirements.

The Mid-and-Long-term Roadmap stipulates that the processing/disposal method of the retrieved fuel debris shall be investigated and fixed during the third phase after starting the fuel debris retrieval work.

3.1.2.3.3.2 Issues in sorting out fuel debris and radioactive waste during fuel debris retrieval

In the fuel debris retrieval work, obstacles and structures to which molten fuel are partially adhered will also be retrieved from the PCV in addition to fuel debris in which molten core fuel are mixed with metals and solidified, and compounds (MCCI product) produced by mixing molten core fuel with concrete at the PCV bottom. Of these, if substances on which a small amount of molten fuel is adhered are all deemed as fuel debris, the amount would be enormous. It may become an obstructive factor in advancing decommissioning because scales of facilities and sites for fuel debris storage become larger. This requires the development of sorting techniques for fuel debris and radioactive waste, i.e., sorting scenarios (sorting is performed in which process from retrieval to storage), sorting criteria and necessary measurement technique/devices.

It is determined that those requiring special attention and facilities/systems for handling and storing to maintain subcriticality can be considered as fuel debris. For that purpose, it is recommended to aim for sorting out fuel debris based on the measurement results of the amount and concentration of nuclear fuel materials. The following studies were conducted in response to this[40].

- Consideration of which steps in the operation processes, from retrieval to storage, sorting is feasible to separate retrieved materials from the PCV (fuel debris, structures, etc.) into fuel debris and radioactive waste (consideration of sorting scenarios)
- Investigation of techniques/equipment that may be capable of measuring the content of nuclear fuel materials contained in the materials retrieved from the PCV (investigation of possible measurement techniques)

Based on these studies, it is currently considered as incredibly difficult challenge to measure or estimate the amount and concentration of nuclear materials in the retrieved materials from the PCV, which required innovative technology development.

[40] IRID, supplementary budget for FY 2018, subsidies for the Project of Decommissioning and Contaminated Water Management, “Technology development for further expanding the retrieval scale of fuel debris/in-core structures”, actual results in FY 2019, August 2020.To be revised once the 2021 report is released. https://irid.or.jp/wp-content/uploads/2020/09/2019008kibonosaranarukakudai.pdf

In developing measurement techniques and devices, it is important to understand the measurement errors first. Major factors affecting measurement errors include the fuel debris properties and the condition and location of nuclear fuel material in fuel debris storage condition in unit cans and containers, and there are many factors other than errors depending on the measurement device itself. However, the extent of such impact is highly uncertain at present due to a lack of knowledge on the fuel debris properties and the storage condition of fuel debris. Therefore, in parallel with the development of measurement technology and actual measurement device by iterating actual measurement using mockup fuel debris, etc., it is considered beneficial for R&D in terms of cost and time saving to identify factors influencing measurement errors and its strength accumulated by a number of numerical experiments by computer and incorporate findings such as the extent of influence into the development of actual measurement technologies/devices. Thus, the impact of fuel debris with different properties and storage conditions on the measurement errors as well as modifications/improvements (e.g., specifications of shielding materials and their installation location) of measurement techniques/devices to reduce measurement errors can also be examined through numerical experiments by computer. Such research and development activities have just started since FY 2020 in the Project of Decommissioning and Contaminated Water Management, and this R&D is greatly desired to be accelerated in the future.

Knowledge on the properties and actual storage conditions of fuel debris is anticipated to be accumulated through trial retrieval and gradually expanded-scale of fuel debris retrieval. By comparing the results of numerical experiments using the improved analytical model and expanded analysis condition based on these accumulated knowledge with the results of actual measurements using mock fuel debris manufactured based on the knowledge on the properties of actual fuel debris, it is possible to improve the accuracy of measurement technologies/devices and to further accelerate its development.

It is desirable that development of measurement techniques/devices required for sorting fuel debris and radioactive waste is to be continued in this manner. It is also important to continue activities to enhance the effectiveness and practical applicability of sorting methods (sorting criteria, scenarios, measuring techniques/devices) by leveraging knowledge and information on the fuel debris properties obtained by planned PCV internal investigation and analysis results of fuel debris samples taken during trial retrieval, and the gradual expansion of fuel debris retrieval and so on.

3.1.2.3.3.3 Issues in examining safeguards strategies

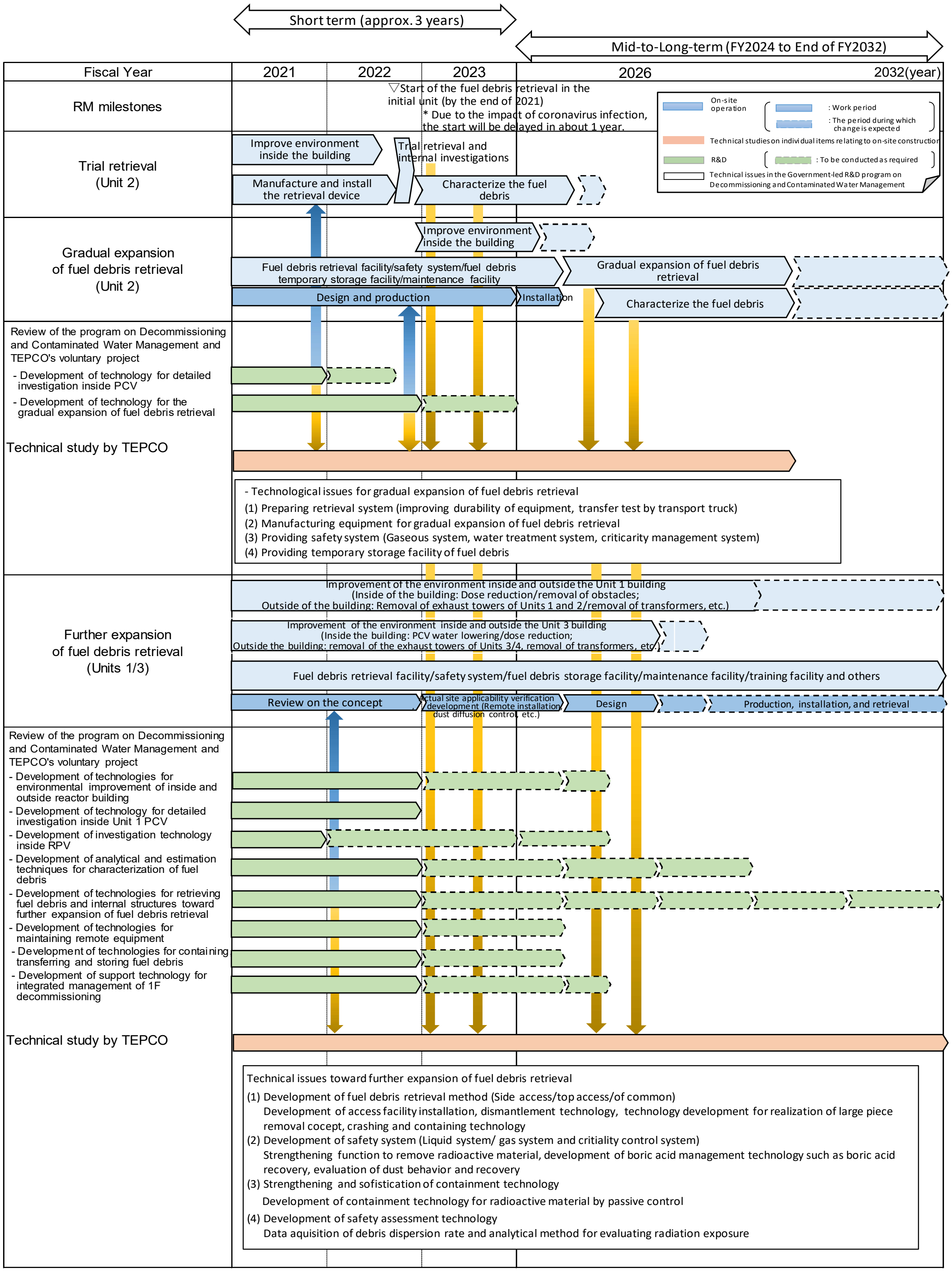
At the Fukushima Daiichi NPS, the safeguards implemented before the accident are not applicable due to the damage on fuel assemblies and destruction of reactor cores and reactor buildings. The appropriate safeguards have been applied along with the progress of the decommissioning process through cooperation and information sharing between Japan Safeguards Office (“JSGO”) of the NRA, International Atomic Energy Agency (“IAEA”) and TEPCO. As a result, the IAEA and JSGO confirmed that there had been no diversion of nuclear materials and undeclared nuclear materials or nuclear activities (For the concept of the safeguards, refer to Attachment).

New material accountancy and safeguards are expected to be applied to the retrieved fuel debris depending on its properties and state. As it is unprecedented, TEPCO may face technical issues in examining and applying them to the site.

In response to that, NDF will conduct wide-ranging surveys on existing technologies related to material accountancy and safeguards for all nuclear materials handled at the Fukushima Daiichi NPS to prepare in case that TEPCO needs technical assistance. NDF will also check the progress of the project from an engineering perspective to confirm that the application of safeguards to systems has not affected the decommissioning process.

3.1.2.4 Summary of key technical issues

The main technical issues and plans described in this section are summarized as shown in Fig. 18.



* It is assumed that Unit 3 will be examined in advance and expanded to Unit 1.

Fig. 18 Technical issues and future plans on fuel debris retrieval (progress schedule)

3.2 Waste management

3.2.1 Targets and progress

(Targets)

(1) The Solid Waste Management Plan (hereinafter referred to as the “Storage Management Plan”) is appropriately developed, revised and implemented including waste prevention, volume reduction and monitoring, with updating the estimated amount of solid waste to be generated in the next ten years periodically.

(2) Countermeasures integrated from characterization to processing/disposal of solid waste are studied from the expert point of view, and the prospects of processing/disposal method and technology related to its safety should be made clear by around FY 2021.

<Key points of "Basic Policies on Solid Waste”>

(1) Thorough containment and isolation
Thoroughly containment and isolation of radioactive materials to prevent human access to them, in order not to cause harmful radiation exposure.

(2) Reduction of solid waste volume
To reduce the amount of solid waste generated by decommissioning as much as possible.

(3) Promotion of characterization
Proper characterization addressing an increase in the number of analysis samples to proceed with studies on processing/disposal method of solid waste.

(4) Thorough storage
Generated solid waste should be stored safely and reasonably according to its characteristics. Storage capacity should be secured to ensure that the waste can be stored within the site of the Fukushima Daiichi NPS.

(5) Establishment of selection system of preceding processing methods in consideration of disposal
To establish selecting methods of processing for stabilization and immobilization (preceding processing) and then select preceding processing methods before technical requirements of disposal are established.

(6) Promotion of effective R&D with an overview of overall solid waste management
To confirm required R&D tasks after cooperating with each R&D field in characterization and processing/disposal and overviewing the overall management of solid waste.

(7) Development of continuous operational framework
To establish the continuous operational framework including development of relevant facilities and human resources in order to continue safe and steady solid waste management.

(8) Measures to reduce radiation exposure of workers
Thorough implementation of radiation exposure control, health and safety management based on the relevant laws/regulations.

(Progress)

Waste management is a long-term effort that needs to attain the prospect of implementing final disposal, while reducing risks in every stage from generation, storage, processing to disposal.

Since a large amount of solid waste with various characteristics is generated in association with decommissioning of the Fukushima Daiichi NPS, the efforts are being made based on the Basic

Policies on Solid Waste summarized in the Mid-and-Long-term Roadmap. TEPCO is required to ensure safe and reasonable storage of the solid waste generated. Led by NDF, the organizations concerned are promoting efforts based on each role to advance technical examination of integrated measures from characterization to processing/disposal of solid waste. The development for establishing a flexible and reasonable waste stream (the flow of the integrated measures from characterization to processing/disposal) is underway in addition to improvements in characterization analysis abilities. The goal is to provide the Technical Prospects by around FY2021.

3.2.1.1 Current status of Storage in Fukushima Daiichi NPS

Table 2 shows the current state of solid waste storage. To store the solid waste properly, TEPCO releases its Storage Management Plan, and estimates the volume of solid waste that will be generated in the next ten years, and shows their policy such as on installing waste management facilities to be required based on the volume.

According to this Plan, temporary outdoor storage of the solid waste will be eliminated completely by FY 2028, except for secondary waste generated by water treatment, metal and concrete rubble, and scrapped flanged tanks with extremely low surface radiation dose rates for reuse/recycling. Facilities needed to achieve this goal are under development (Attachment 10).

Among the targets of reuse/recycling, concrete rubble is crushed and recycled as roadbed material after confirming that the surface dose rate is equivalent to the background radiation dose. In addition, such as by melting is under consideration as a decontamination method for recycling metal.

Secondary waste generated by water treatment is planned to be transferred to store in a building, and a large waste storage building is being constructed as a storage facility for sorption vessels. Moreover, the slurry generated at ALPS (hereinafter referred to as “ALPS slurry”) generated by the multi-nuclide removal equipment, etc., and the waste sludge generated at the water purification system (hereinafter referred to as “waste sludge”) sludge have high water content and mobility. For safer storage, the slurry will undergo stabilization (dehydration) treatment (Scheduled start of operation in FY 2022), while the sludge will be transferred from the current storage area, an underground storage tank in a building, to higher ground (to be completed in FY 2023).

Such solid waste will continue to be generated with some exceptions, and additional solid waste will be generated from fuel debris retrieval.

Table 2 Status of solid waste storage
(a) Storage of rubble, felled trees, used protective clothing, etc. (as of June 30, 2021)

Storage method	Stored volume (m^3) / Storage capacity (m^3) (Percentage)
Outdoor storage (surface radiation dose rate ≤ 0.1 mSv/h)	226,400 / 270,200 (84%)
Outdoor sheet covered storage (surface radiation dose rate 0.1 - 1 mSv/h)	40,900 / 71,000 (58%)
Soil-covered temporary storage facilities, outdoor container storage (surface radiation dose rate 1 - 30 mSv/h)	17,900 / 24,600 (73%)

Containers* (in solid waste storage building)	25,600/ 39,600 (65%)
Total ----	310,700 / 405,400 (77%)

Felled trees

Classification	Storage method	Stored volume (m^3) / Storage capacity (m^3) (Percentage)
Trunks, roots, branches, leaves	Outdoor storage	99,500 / 134,000 (74%)
Branches, leaves	Temporary storage pool	37,300 / 41,600 (90%)
Total	----	136,800 / 175,600 (78%)

Used protective clothing

Storage method	Stored volume (m^3) / Storage capacity (m^3) (Percentage)
Container	33,700 / 68,300 (49%)

* Including secondary waste generated by water treatment (e.g. small filter)
Note that the storage volume is rounded to the nearest 100m^3, so the total and the breakdown may not be consistent.

(b) Management status of secondary waste generated by water treatment (As of July 1, 2021)
Sorption vessels, etc.

<table>
<tr><th colspan="3">Storage place</th><th colspan="2">Storage volume</th><th>Stored volume/capacity
(Percentage)</th></tr>
<tr><td rowspan="8">Outdoor temporary storage area of used sorption vessels</td><td colspan="2">Cesium sorption apparatus</td><td>779</td><td>Number of vessels and filters</td><td rowspan="8">5,158 / 6,372
(81%)</td></tr>
<tr><td colspan="2">2nd Cesium sorption apparatus</td><td>244</td><td>Number of vessels and filters</td></tr>
<tr><td colspan="2">3rd Cesium sorption apparatus</td><td>9</td><td>Number of vessels and filters</td></tr>
<tr><td rowspan="2">HICs from multi-nuclide removal system</td><td>Existing</td><td>1,923</td><td>Number of containers</td></tr>
<tr><td>Expansion</td><td>1,888</td><td>Number of containers</td></tr>
<tr><td>Used vessels from high-performance multi-nuclide removal system</td><td>High performance</td><td>83</td><td>Number of vessels</td></tr>
<tr><td>Used columns from multi-nuclide removal system</td><td>Existing</td><td>17</td><td>Number of Columns</td></tr>
<tr><td colspan="2">Used vessels and filters from mobile type strontium system</td><td>215</td><td>215</td></tr>
</table>

Waste sludge

Storage place	Stored volume (m^3) / storage capacity (m^3) (Percentage)
Sludge storage facility (indoor)	454 / 700 (65%)

Concentrated waste liquid

Storage method	Stored volume (m^3) / storage capacity (m^3) (Percentage)
Concentrated waste liquid storage tanks (outdoor)	9,380 / 10,300 (91%)

3.2.1.2 Prospects of processing/disposal method and technology related to its safety

The Mid-and-Long-term Roadmap states that Technical Prospects will be provided in the Technical Strategic Plan by around FY 2021. Specifically, it will “present measures toward reducing the volume of solid waste”, “develop analytical and evaluation methods for efficient characterization”, and “develop methods to reasonably select safe processing and disposal methods at the time when the necessary information such as solid wastes’ properties are proven”.

On “present measures toward reducing the volume of solid waste”, the Basic Policy for Solid Waste specifies that TEPCO should reduce the burden of overall waste management (for the entire process, from generation, storage, and processing to disposal) to the extent possible. The Technical Strategic Plan 2020 takes into consideration the possibility of further efforts based on precedents in other countries.

“Develop analytical and evaluation methods for efficient characterization” and “develop methods to reasonably select safe processing and disposal methods at the time when the necessary information such as solid wastes’ properties are proven” mean to develop and establish the methods necessary for disposing of materials that become waste even after undergoing volume reduction. NDF has organized and examined the following specific goals in the Technical Strategic Plan 2018. Of the wide variety of types of solid waste, the secondary waste generated by water treatment, which is highly mobile and for which there is no precedent for processing or disposing of it in Japan, has been selected as the main object for examination.

(1) Establish a safe and reasonable disposal concept based on characteristics and volume of the solid waste generated in the Fukushima Daiichi NPS with its applicable processing technology, and develop safety assessment methods that apply the features of the disposal concept, with considering examples of foreign countries.
(2) Clarify radiological analysis and evaluation methods for characterization.
(3) Clarify processing technology which practical application could be expected for stabilization and immobilization, considering disposal for important waste streams such as secondary waste generated by water treatment.
(4) Establish methods of reasonably selecting processing technology to stabilize and immobilize waste based on the above methodology before the technical requirements for disposal are determined (i.e., preceding processing).
(5) Have the prospect of setting processing/disposal methods for solid waste for which the processing technology considering disposal is not clarified, using a series of methods to be developed by around FY 2021.
(6) Clarify issues and measures concerning storage of solid waste until it is conditioned

The following shows the Technical Prospect based on the results of these studies.

3.2.1.2.1 Approach for volume reduction

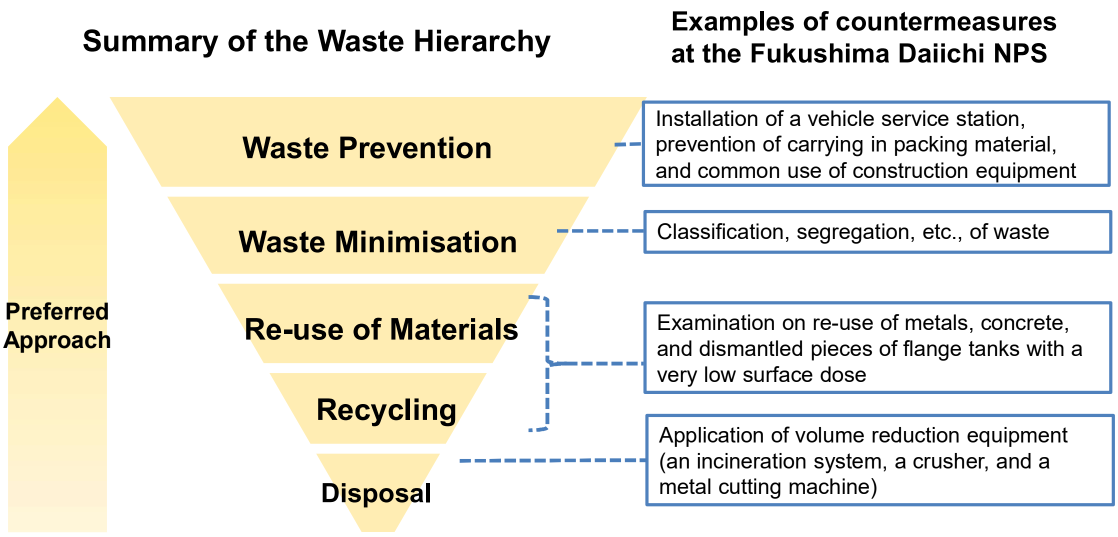
If a large volume of solid waste exists, not only does segregation and analysis take time, but the number of storage containers and the scale of storage facilities increase, and so does the load of waste management. Therefore, it is extremely important to reduce the physical volume of solid waste as much as possible.

At the Fukushima Daiichi NPS, it is important to instill initiatives on volume reduction, aimed at decreasing the burden of the overall solid waste management, into the overall decommissioning activities by referring to the examples of overseas countries that have implemented the waste hierarchy concept (Attachment 11).

Specifically, the priorities for measures to be taken as waste management are (1) prevention of waste generation, (2) minimization of waste volume, (3) reuse, (4) recycling, and (5) disposal. In waste management, it is important to prioritize (1) as much as possible, and consider (5) disposal as the last option (Fig.19) for volume reduction of waste to be stored, processed, and disposed of.

In terms of reducing waste generation, it is important to consider in the design and construction plan to reduce the volume of materials to be used. It is also important not to bring in substances that affect processing/disposal as much as possible. To minimize physical volume, consideration on preventing contamination through strict segregation, maintenance/management of manufactures, extension of product life, and waste volume reduction is important. Reuse should be promoted after contamination checks, decontamination, repair and parts replacement, and it is useful to consider ease of reuse from the design stage. Considering alternative uses is also beneficial. In recycling, it is important to consider the contamination condition of contaminated valuable sources, separate and process recyclable materials, and use them as new materials and products.

As shown in Fig.19, TEPCO has also been implementing initiatives corresponding to this concept. As new measures to be implemented as described in 3.2.1.1, among rubble accumulated outdoors (surface radiation dose rate ≦ 0.1 mSv/h) (Table 2), reuse/recycling of metals, concrete, and scrapped flanged tanks with extremely low surface radiation dose rates are under consideration. As part of this activity, decontamination methods for recycling metals are being examined. In promoting safe and reasonable waste management, it is important to examine further possibilities based on the characteristics of solid waste at the Fukushima Daiichi NPS in reference to precedent cases overseas.



Source: Strategy Effective from April 2011 (print friendly version), arranged by NDF

Fig.19 Summary of waste hierarchy at the NDA, UK, and countermeasures at the Fukushima Daiichi NPS

3.2.1.2.2 Development of analytical/evaluation methods for implementing efficient characterization

Since the solid waste at the Fukushima Daiichi NPS is characterized by diverse nuclide compositions and activity concentrations, and a large physical volume, efficient characterization is required. With the aim of developing the analytical/evaluation methods necessary for this purpose, the Project of Decommissioning and Contaminated Water Management and other initiatives have developed analytical methods for simplified and speed-up data acquisition. Furthermore, as a method to perform characterization with small analysis data, the R&D program has established a method for quantifying uncertainties in evaluation values (to identify variable distribution and the width) using statistical method with a method for efficiently identifying the inventory in combination with analytical data and contamination mechanism (Attachment 11).

As an analytical method for simplified and speed-up data acquisition, automation of pretreatment for analytical samples and a method using the triple quadrupole inductively coupled plasma mass spectrometry (ICP-MS/MS) (simplified method compared to the conventional radioactivity measuring method) have been developed. These results will be incorporated into the Building #1 of the Radioactive Materials Analysis and Research Facility under construction.

As characterization using statistical methods, in addition to using analytical data, statistical methods have been applied to the method for efficient characterization by combining with transition models[41], resulting in establishing the method to quantify uncertainties in evaluated values. A method that combines the Data Quality Objectives (hereinafter referred to as “DQO”) process[42]

[41] The process of contamination in reactor buildings and stagnant water by nuclides released from damaged fuel after the accident is modeled to calculate the radioactivity of each nuclide in solid waste.

[42] DQO process: A method developed by the US Environmental Protection Agency (US EPA) to plan sampling of analysis for decision-making. A sampling plan will be developed in seven steps. Planning and analysis are repeated as necessary to achieve the objectives. This process can be applied to various problems that can be solved by analysis.

with statistical methods has also been examined/tested to develop a medium-to-long-term analytical plan, and its effectiveness has been confirmed. In addition, a database called FRAnDLi (Fukushima Daiichi Radwaste Analytical Data Library) has been created, which contains information related to analytical data (sample information (type, sampling location, date, etc.) and analytical values of activity concentration, etc.), allowing data accumulation on a constant basis.

As described above, analytical/evaluation methods have been developed for efficient characterization and will be applied to characterization of solid waste.

3.2.1.2.3 Establishment of methods to reasonably select processing/disposal methods

In selecting processing/disposal methods in a reasonable manner, based on the waste properties, an appropriate combination of processing (waste form) and disposal (disposal facility) methods should be clarified so that the risk of buried solid waste to the public and environment can be maintained sufficiently low in the future.

In the case of solid waste from a normal reactor, its properties can be estimated to some extent by the previous findings (data) or analytical methods. Accordingly, the appropriate combination of processing (waste form) and disposal (disposal facility) methods can sufficiently reduce the risk to avoid a significant impact on the public and surrounding environment.

Even in the case of solid waste from the Fukushima Daiichi NPS, molten nuclear fuel is a major source of contamination, and the radioactivity concentration does not exceed that of spent fuel. Therefore, the risk can be sufficiently reduced by understanding the overall picture of the target solid waste (properties such as nuclide composition, activity concentration by waste, waste volume), and selecting a proper combination of processing (waste form) and disposal (disposal facility) methods, while utilizing the experience and knowledge on radioactive waste processing/disposal accumulated in domestic and overseas.

However, the overall picture of solid waste to be disposed of, including that which will be generated, will gradually become clear as the progress and plans for fuel debris retrieval, contaminated water management, and other decommissioning work are clarified. Therefore, it is necessary to repeatedly examine processing/disposal methods and safety assessments, starting from the waste for which properties have been clarified; to give consideration to making processing/disposal methods more appropriate; and to accumulate knowledge to consider safe and reasonable processing/disposal methods for diverse solid waste collectively. Aiming for safer and not extremely conservative storage of waste with high mobility such as slurry waste, processing (preceding processing) for stabilization/immobilization may be required before determining the disposal method (disposal facility). Reprocessing would be necessary if the specifications of the waste form even after preceding processing did not conform to those required by the disposal method (disposal facility) to be determined. Therefore, in order to minimize such possibility, a selection method for the preceding processing method with disposal in mind is needed.

As mentioned below, study on an appropriate combination of processing and disposal methods, or preceding processing methods, is considered for the waste for which properties have been identified to some extent.

- Establish several feasible disposal methods suitable for waste characteristics (without specifying the feature of facilities such as their locations and sizes)
- In parallel, establish several processing methods suitable for waste characteristics to be considered, and set the specifications of waste form after applying each processing method.
- Evaluate the safety of several selected disposal methods based on the specifications of waste form after processing to verify whether risk to the public and environment can be sufficiently low, and to consider more effective processing/disposal methods based on the evaluation results.

The above examination steps are repeated to narrow down disposal methods and specifications for waste form after processing. Clarifying the overall characteristics of solid waste concurrently with characterization helps identifying an appropriate combination of processing/disposal. When preceding processing becomes necessary, candidate processing methods will be selected in consideration of the status of examination and open issues at that point.

It is also important to consider the period during which pre-disposal management is to be implemented, taking into account the risk reduction during that period, and examining necessary and feasible technologies. Since storage is important to provide flexibility to respond to the progress of processing/disposal, and to reduce radiation exposure of workers due to the decay of the radioactive materials, it is important to consider storage strategies as part of this examination process.

A series of these studies is represented as a flowchart shown in Fig.20. Technical knowledge and evaluation methods necessary for these studies (establishment of processing technology and waste form appropriate for waste, safe, reasonable and feasible disposal methods, disposal safety assessment) have been developed through research/development (verification of applicability of processing methods using engineering-scale test equipment conducted mainly on the secondary waste generated by water treatment, establishment of reasonable and feasible disposal methods based on the waste properties and applicable processing technologies, and development of safety assessment methods) by the Project of Decommissioning and Contaminated Water Management, and has been established as a series of methods to select processing/disposal methods in a safe and reasonable manner (Attachment 11).

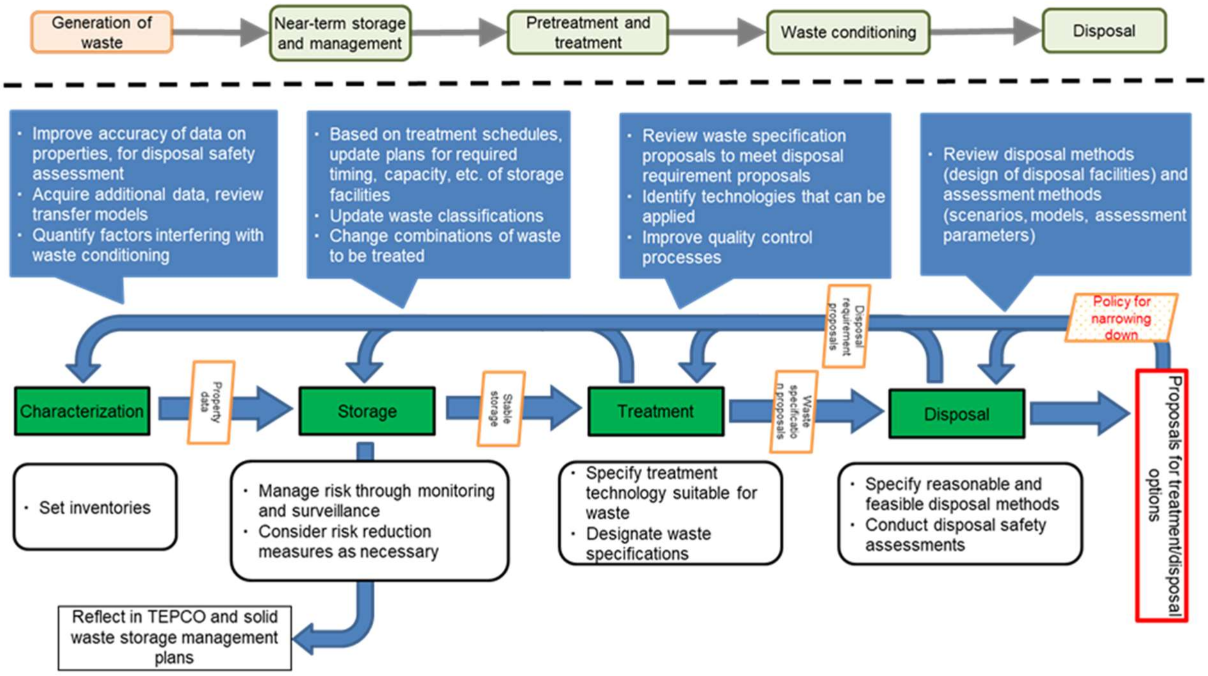


Fig.20 Development of methods to reasonably select safe processing/disposal methods of solid waste

3.2.2 Key issues and technical strategies to realize them

After providing for future issues based on the prospects of a processing/disposal method and technology related to its safety, the technical strategies by category to realize them are show below.

3.2.2.1 Issues based on the Technical Prospects

As volume reduction is extremely important for the safe and reasonable management of solid waste according to the progress of decommissioning work in the future, the measures in progress should be continued steadily. Since solid waste continues to be generated, it is important to continuously examine further possibilities by referring to advanced cases of overseas' for more volume reduction. It is recommended that volume reduction is realized in consideration of the expected outcome and feasibility.

For the development of analytical/evaluation methods for efficient characterization, it is necessary to improve evaluation methods and continuously incorporate them into solid waste management, including processing/disposal, while accumulating analytical data using efficient analytical methods established through achievements in research/development. In this case, efforts should be made for low-activity waste such as rubble, as well as high-activity waste such as secondary waste generated by water treatment and waste generated from fuel debris retrieval, according to the characteristics of each type of waste.

To establish methods to select safe processing/disposal methods in a reasonable manner, the methods shown in Fig.20 developed in 3.2.1.2.3 should be used to proceed with the examination toward the determination of waste form specifications and manufacturing methods for Phase 3, as specified in the Mid-and-Long-term Roadmap. Specifically, through these methods, the trial examples of optimization/rationalization of processing/disposal methods will be accumulated by waste stream according to the progress of characterization and with the assumption of ensuring safety to widely acquire findings on optimization by waste stream. Moreover, consideration will be given to specify strategies for optimization/rationalization of the overall picture covering the entire waste stream, allowing clarification of approaches toward such purposes. In doing so, it is important to flexibly examine appropriate measures in consideration of actual use and economic feasibility by reflecting the latest findings and applying the concept of Best Available Techniques[43]. As the examination progresses, and the processing/disposal methods for the overall picture of waste are finalized, it will be important to share the examination process for optimization, such as by sharing the awareness of problems with local communities and society.

3.2.2.2 Technical strategy by sectors

3.2.2.2.1 Characterization

For low-activity waste such as rubble, it is not so challenging to perform the analysis work itself. However, it takes an immense amount of time to measure entire quantity because of the enormous volume of waste, and therefore, volume reduction and a corresponding efficient analysis strategy are needed. For that purpose, it is important to take an approach that efficiently ensures the required accuracy. In order to achieve this, efficient analyses should be promoted by making them simplified/speed-up, and inventory evaluation methods that combine the DQO process with statistical methods be established.

For high activity waste, sampling and analysis themselves are difficult, and the amount of analysis data to be obtained is limited. Thus, inventory assessment based on the transition model becomes more important. It is necessary to obtain actual sample data such as by the ongoing efforts for sampling from Cs sorption vessels and its analysis, which are currently in progress. The application of inventory evaluation methods, which combine the DQO process with statistical methods, and the priority of data to be collected should also be considered to enhance the accuracy of the transition model.

Following the phase of analyzing samples that are easy to collect, characterization is now in the phase of collecting/analyzing samples that are important for waste management. Going forward, it is important to develop a medium-to-long-term analysis strategy that defines the solid waste to be analyzed, its priority, and quantitative targets for analysis, etc., and to proceed with analysis/evaluation accordingly. It is useful to accumulate trial results and verify their validity in order to establish a flow from the development of a medium-to-long-term analysis plan using

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[43] The term is defined as follows, as the concepts proposed in environmental protection and internal discussions.
・For the purpose of preventing environmental pollution, it should be defined as “a process, facility or method of state-of-the -art operation”, “with the principle of limiting emissions, discharge and disposal”, “taking into account technical and economic feasibility”.
・It can change in accordance with technological, economic and social factors, as well as with changes in scientific knowledge and understandings.
(Daisuke Sugimoto, Hiroshi Hasegawa, “The concept of ‘Best Available Techniques (BAT)’ for disposing radioactive waste – Review of overseas cases and suggestions for Japan-“, L06001 Report by Central Research Institute of Electric Power Industry (November 2006))

statistical methods; analysis and data acquisition; the incorporation of the acquired data into examination of processing/disposal methods and evaluation of the outcome; to the development of the next medium-to-long-term analysis plan based on the evaluation results.

As for facilities for analysis, in addition to the existing facilities in the JAEA's Ibaraki area, etc., it is planned to establish the Radioactive Material Analysis and Research Facilities under construction, as well as facilities for analysis by TEPCO, allowing characterization of a variety of solid waste in parallel. Since the target nuclides, analysis items, accuracy, and the number of samples for analysis depend on the target solid waste, a structure should be established based on the appropriate division of roles and according to the characteristics of facilities.

3.2.2.2.2 Storage

For storage of all waste, it is important to reconsider measurement items and timing, etc., in terms of diverse information for characterization, while acquiring necessary information through monitoring and surveillance of the storage status commensurate with the risks involved.

With regard to high-activity waste, such as waste generated from fuel debris retrieval, the issues and countermeasures assuming the further expansion of the fuel debris retrieval scale have been clarified according to the results of research/development as of FY 2021. Going forward, reviews should be performed along with the examination of the fuel debris retrieval methods. Measures should be taken to ensure storage of solid waste that is expected to be generated during fuel debris retrieval (trial retrieval, gradual expansion of the retrieval scale) before full-scale retrieval.

The site also has solid waste stored before the accident, and a large volume of dismantled waste is expected to be generated after the completion of fuel debris retrieval. Only increasing storage capacity for solid waste will eventually reach the limit, so efforts should be made to reduce the volume of solid waste to be generated as much as possible.

In considering further possibilities of volume reduction, with an aim to reuse/recycle metals with extremely low surface radiation dose, chemical decontamination (decontamination by phosphoric acid), physical (mechanical) decontamination (steel blasting), and decontamination by melting (decontamination by melting slag), are under consideration as metal decontamination methods for recycling.

As metal recycling with decontamination by melting slag has already been used in many Western countries, it is considered a promising candidate technology. Thus, it is important to focus on the areas where the conditions are different between Western countries and the Fukushima Daiichi NPS (target nuclides, etc.), and to evaluate the applicability of the method.

3.2.2.2.3 Processing/disposal

The objective is to establish safe and reasonable processing/disposal methods for all solid waste in which diverse waste streams exist, and to widely obtain knowledge for optimizing each individual stream. Therefore, it is necessary to continue development/research of processing and disposal technologies required for the series of studies as shown in Fig.20.

Regarding the processing technology, outstanding issues in low and high-temperature processing technology, for which research/development is promoted, should be addressed. Waste streams, for which the application of low and high-temperature processing technologies has not been investigated, will be evaluated as necessary, and performance such as leachability for solidified substances to be produced will be evaluated. As for low-temperature processing technology, consideration is given to transformation of solidified substances as well as inspection methods to verify the possibility of solidification. In the case of high-temperature processing technology, the feasibility of the whole processing system, including supply and exhaust systems, is an issue in addition to the solidification process, and therefore it is necessary to carry out examination in a timely manner according to the start time of processing. Furthermore, in order to expand technological options, it is important to examine the possibility of low-temperature solidification after interim treatment such as steam reforming.

Regarding disposal technologies, in order to establish reliable safety assessment techniques, important issues specific to solid waste at the Fukushima Daiichi NPS will be explored and identified based on the understanding of the sensitivity structure of parameters to radiation dose and the long-term transition behavior of disposal facilities. Then, priorities will be examined and incorporated into research plans. Development and improvement of the proposed disposal options consisting of proposed disposal concepts and waste to be disposed will also be promoted in reference to practices at home and abroad. Furthermore, while improving its reliability, after expanding the target of waste streams, on which trial assessments will be performed by applying safety assessment technology, group of disposal options will be examined with a bird-eye-view of all solid waste at the Fukushima Daiichi NPS. Then, contributions will be made to considering appropriate measures for overall waste management, in coordination with areas other than disposal such as presenting targets for waste form performance and the accuracy required for characterization.

3.2.2.3 Summary of key technical issues

The main technical issues and plans in the future described in this section are summarized as shown in Fig. 21.

The Mid-and-Long-term Roadmap stated that the properties of solid waste will be analyzed and the specifications of waste form and their production methods will be determined in the Phase 3,. Therefore, in Phase 3-1, as a systematic approach toward this goal, studies will be conducted to present appropriate measures for overall management of solid waste. Specifically, the first step is to create processing/disposal options of solid waste by examining unimplemented issues related to processing technology, interim treatment, and disposal options. Then, the options will be compared and evaluated using the property data that are becoming clear, and examinations will be conducted to identify waste stream that are suitable for characteristics of solid waste.

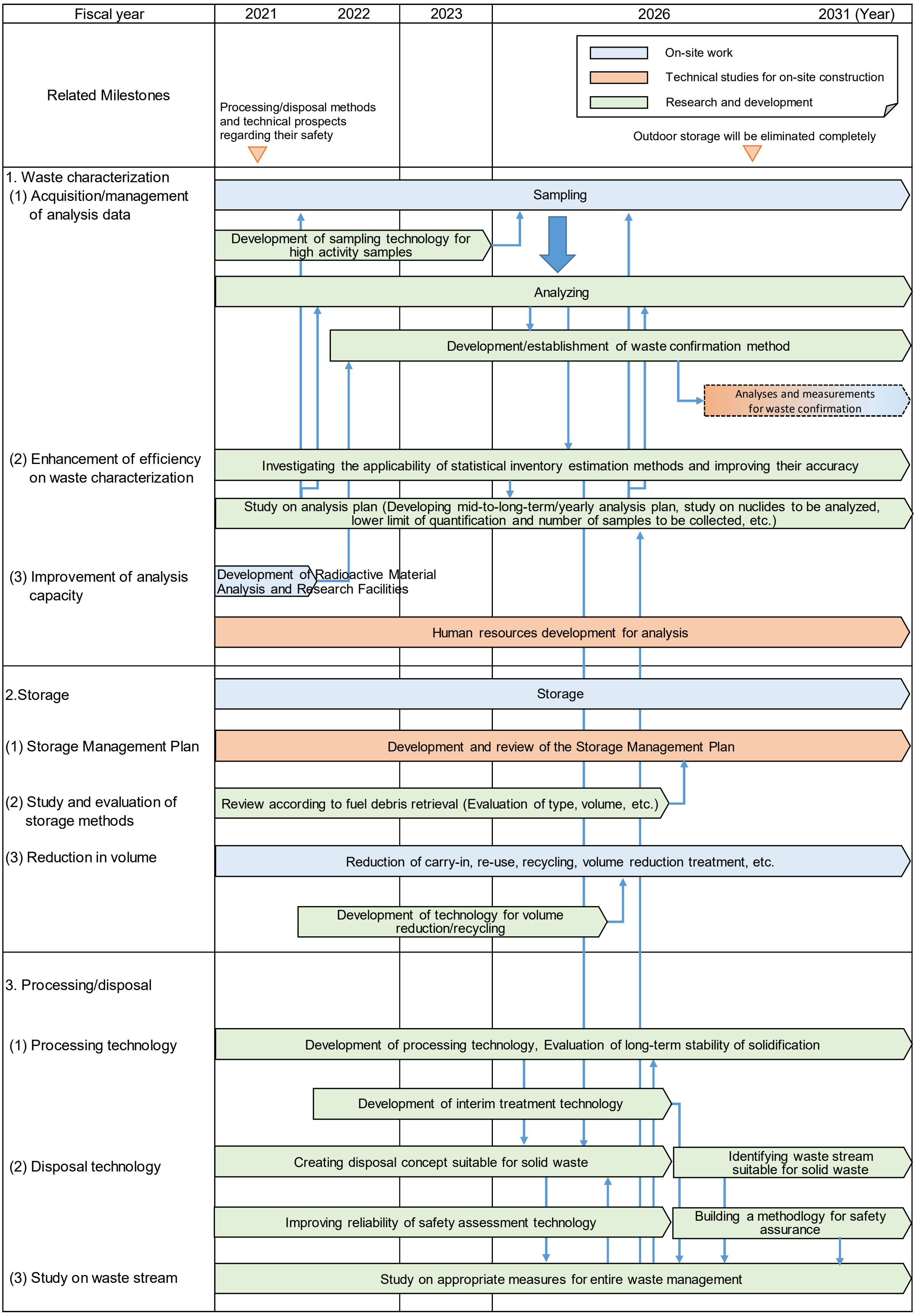


Fig. 21 Main technical issues and future plans on waste management (progress schedule)

3.3 Contaminated and treated water management

3.3.1 Targets and progress

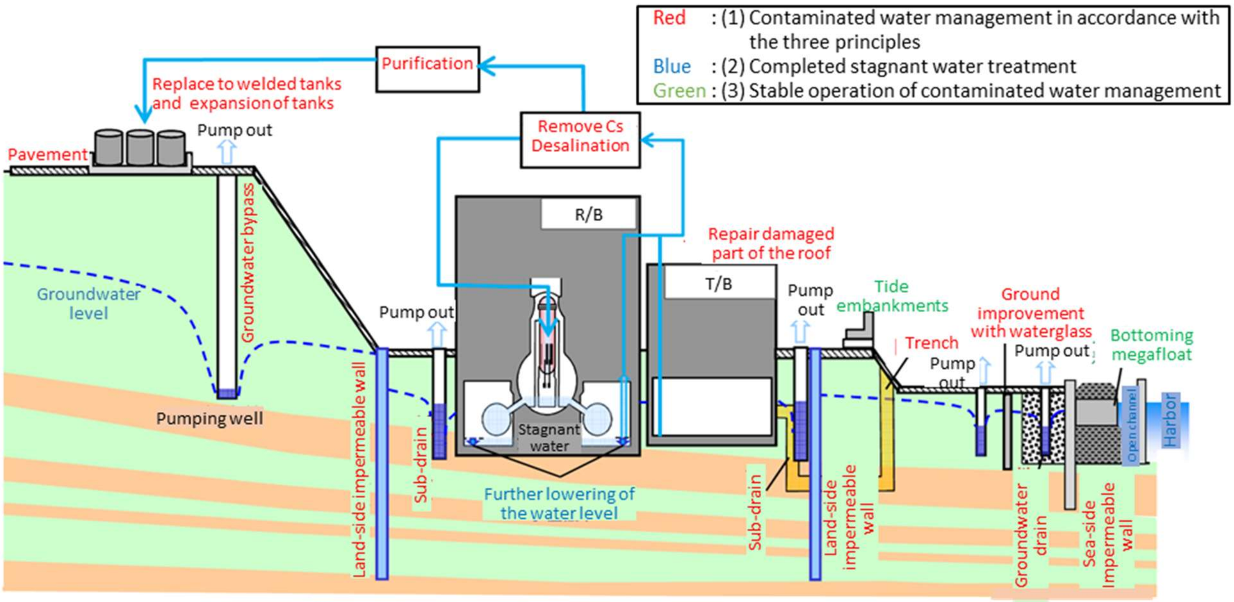
(Targets)

(1) Under the three principles concerning the contaminated water issues ("Removing" contamination sources, "Redirecting" fresh water from contamination sources, and "Retaining" contaminated water from leakage), to reduce the stagnant water in the reactor buildings in FY 2022 to FY 2024 to about the half of the amount of the end of 2020 while continuing the operation of the constructed water-level management system and controlling the generation amount of the contaminated water to 100 m^3/day or less in 2025. Moreover, to ensure stable implementation of contaminated water management, measures against large-scale natural disaster risks, such as tsunamis and storm rainfall, will be implemented in a planned manner.
(2) To arrange the relationship with a decommissioning process including full-scale fuel debris retrieval beginning in the near future, and to promote examination of the measures of the contaminated water management for medium-and-long term prospects.
(3) ALPS-treated water currently being stored in tanks will be handled in accordance with the government's basic policy announced in April 2021.

(Progress)

Fig. 22 shows the outline of the contaminated water management. Stagnant water in buildings, that is, the contaminated water with a mixture of cooling water contacted with the fuel debris and groundwater/rainwater flowed into the buildings is liquid containing a considerable amount of the dissolved radioactive materials (inventory) from the perspective of measures to reduce the risk from radioactive materials. Therefore, its hazard potential is high and so is the Safety Management level, as the storage condition of such stagnant water deviates from what is originally intended (refer to Section 2.2). Of this stagnant water in buildings, excluding the reactor buildings of Units 1 to 3, where circulating water injection is ongoing, the process main building and high-temperature incinerator building storing contaminated water temporarily for purification treatment, the treatment of stagnant water in buildings was completed in 2020, and the inventory was significantly reduced. However, the hazard potential is still high.

Currently, the following three measures are being implemented as contaminated water management:



(Source : TEPCO)

Fig. 22 Outline of contaminated water management[44]

(1) Efforts to promote contaminated water management in accordance with the three principles ("Removing" contaminant sources, "Redirecting" fresh water from containment sources, and "Retaining" contaminated water from leakage")

The groundwater level in the vicinity of the reactor buildings was stably controlled at low levels through multilayered contaminated water management such as land-side impermeable walls and sub-drains. The increase in the amount of contaminated water generated during rainfall also tended to be controlled by repair of damaged roofs and facings on site. As a result, the amount of contaminated water generated decreased from approx. 490 m3/day (FY 2015) before the measures were taken to approx. 140 m3/day (2020). In order to reduce the amount of contaminated water to 100m3/day or less, roof repair and expansion of facing range are being addressed while adjusting interference with other decommissioning work. [45] shows the results of the amount of contaminated water generated by factor, and the countermeasures to achieve the target by factor. Along with the completion of treating the remaining water, including ALPS-treated water, at the bottom of the flanged tanks in July 2020, measures and observation for risk reduction are being implemented through monitoring of groundwater and the harbor.

[44] Secretariat Meeting of the Team for Countermeasures for Decommissioning and Contaminated Water/Treated Water Treatment (92nd), Material 2, "Outline of decommissioning, contaminated water and treated water", July 29, 2021

[45] Committee on Countermeasures for Contaminated Water Treatment (23rd meeting), Material 4, "Issues and countermeasures concerning contaminated water management at the Fukushima Daiichi NPS", June 25, 2021

Factors in the contaminated water generation (Category)		2020 [FY2020*1] Contaminated water generation	Measures to achieve the amount of 100m^3/day	Targets for FY2025 (m^3/day)
①	Inflow into the buildings (Rainwater/groundwater inflow)	**Approx. 100m^3/day [about 90m^3/day]**	- Repairing wrecked parts of the rooftop - Facing around the buildings - Lowering water level in subdrain	**Approx. 50m^3/day**
②	Amount trandsferred from the El. 2.5m ground to the buildings	**Approx. 10m^3/day [about 10m^3/day]**	- Decrease in operational water level Reducing the amount of water transferred to the turbine buildings by securing a margin for water level rise during heavy rain	**Approx. 10m^3/day**
③	Chemical injection volume during purification with ALPS *2	**Approx. 10m^3/day [less than about 10m^3/day]**	- Reliable maintenance	**Approx. 10m^3/day**
④-1	Amount of transfer generated byto decommissioning work	**Approx. 10m^3/day [about 10m^3/day]**	- Scheduled treatment of stagnant water in trenches and so forth	**Approx. 10m^3/day**
④-2	Amount of generation transferred (Drainage at the shallow draft wharf)	**Approx. 10m^3/day [about 30m^7/day]**	- Reliable operation and management of facilities - Scheduled treatment of stagnant water	
Amount of contaminated water generated		**Approx. 140m^3/day [about 140m^3/day]**		**Less than 100m^3/day**
Reference	Rainfall (mm)	1,339m (3.7mm/day) [1,329mm(3.7mm/day)]	Average rainfall (3.9mm/day)	

*1 Data obtained through March 31, 2021

*2 Chemicals being injected into preprocessing facility for multi-nuclide removal system

*3 Invludes sprinkling of water on the operating floor, inflow into the buiding outer the frozen soil and transfer of stagnant water in trench

(TEPCO material editted by NDF)

Fig. 23 Results of contaminated water generation by factor and main measures to reduce the amount[45]

(2) Efforts to complete stagnant water treatment

In 2020, the treatment of stagnant water in buildings, excluding the reactor buildings of Units 1 to 3, the process building and high-temperature incinerator building, was completed.

It is planned to lower the water level in the reactor building while continuously lowering the sub-drain water level in order to reduce the amount of stagnant water in the reactor building by half. In association with this, the importance of issues in handling sludge containing α-nuclides (α-sludge) is increasing. As the particle size distribution and chemical composition of the α sludge at the bottom of the reactor building have been analyzed, it is expected that most of the sludge can be removed by a filter with an appropriate pore diameter. In order to complete the treatment of stagnant water in the process building and the high-temperature incinerator building, moreover, methods for radiation dose rate surveillance or recovery are under consideration for high-dose zeolite sandbags located on the lowest floor. In the case of a building where the treatment of stagnant water has been completed and the floor surface has been exposed, a method for recovering sludge located on the floor is being studied.

(3) Efforts for stable operation of contaminated water management

The measures against tsunami have been implemented, including installation of tsunami tide walls in the Japan Trench; closure of building openings; reinforcement of land-side impermeable walls (suppression of brine[46] leakage by electrifying closing valves of the brine supply piping); and relocation of water collection systems such as sub-drains from the revetment side to higher ground. As a countermeasure for heavy rain, reinforcement of discharge functions of the existing drainage, etc. is underway.

[46] Cooling liquid that circulates through underground frozen ducts for ground freezing

(4) Treated water management

For handling water treated with the multi-nuclide removal equipment, so-called ALPS-treated water[47], discussions by experts, including technical aspects and the social impact, such as reputational damage, were held at a national subcommittee, and a report was published in February 2020. In April 2021, based on opinions from a wide range of people through subsequent opinion exchanges with local governments and agriculture, forestry, and fisheries industries, the government announced the basic policy of discharging the ALPS-treated water into the ocean[48] on the premise that safety will be ensured and that measures against reputational damage will be thoroughly taken, while TEPCO published "Actions by TEPCO in response to the Government's basic policy on disposal of the treated water by multi-nuclide removal equipment"[49]. Currently, the equipment needed and actions against reputation are under consideration, TEPCO reported the status of these examinations and efforts in August 2021[50], [51].

3.3.2 Key issues and technical strategies to realize them

3.3.2.1 Issues in the future treatment of stagnant water in buildings

The following three points are the key issues for the future treatment of stagnant water in buildings.

(1) Prevention of spreading α-nuclides

At the bottom of the torus room of the reactor building, stagnant water containing α sludge and ionized α-nuclides exists, and relatively high concentration of α-nuclides has been detected (Fig. 24). These α-nuclides has significantly high effective dose factors when ingested by inhalation, and it is necessary to suppress this spread to a limited extent. At present, the properties of α-sludge and ionized α-nuclides are being analyzed, and removal methods for preventing the spread of α-nuclides are being studied. However, the chemical form of α-nuclides may change depending on water quality and coexisting substances. In order to ensure removal of α-nuclides, it is necessary to collect samples from as many places as possible and to understand the variation in their properties. Since α-sludge contains a high concentration of Cs -137, in addition, it is important to consider measures to reduce radiation exposure dose of workers, maintainability, and secondary waste.

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[47] Water that is treated with multi-nuclide removal equipment until the level of radioactive materials other than tritium fall below the regulatory standard on safety, without fail.

[48] Inter-Ministerial Council for Contaminated Water, Treated Water and Decommissioning Issues (5th meeting), Material 1, "Basic policy for disposing of treated water by multi-nuclide removal equipment at the TEPCO Fukushima Daiichi Nuclear Power Station (draft)", April 13, 2021

[49] TEPCO, Attachment 1, "Actions by TEPCO in response to the Government's basic policy on disposal of the treated water by multi-nuclide removal equipment", press release, April 16, 2021

[50] TEPCO, Status of examining dealing with water treated with multi-nuclide removal equipment at the Fukushima Daiichi Nuclear Power Station, press release, Report by the subcommittee dealing with water treated with multi-nuclide removal equipment, press release, August 25, 2021

[51] TEPCO, Attachment 3, Handling of compensation in the event reputational damage caused by discharging treated water by multi-nuclide removal equipment, press release, August 25, 2021

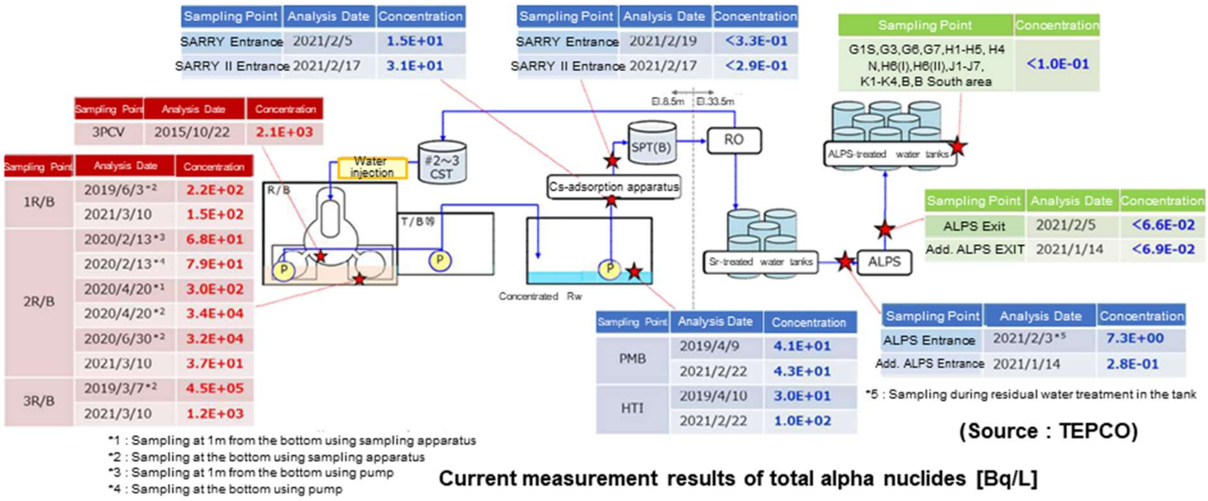


Fig. 24 Water treatment systems for stagnant water in buildings and measurement results of total α-nuclides[52]

(2) Treatment of high-concentration stagnant water

In treating stagnant water in buildings, water at the bottom of the buildings has not been pumped. Therefore, not only α-nuclides but also highly concentrated stagnant water containing radioactive materials and salt, which is close to the state immediately after the accident, exists at the bottom of the torus rooms of the reactor buildings. The S/C also contains contaminated water with higher concentration of radioactive materials and salt than usual stagnant water.

It is necessary to dispose of contaminated water containing such high concentration of radioactive materials and salt as soon as possible in terms of risk reduction. In order to suppress the fluctuation range of the contaminated water concentration for the stable operation of the water treatment system, however, contaminated water of high concentration and that of low concentration are mixed for treatment. For this reason, in order to accelerate the treatment of highly contaminated water, it is necessary to proceed steadily with careful planning that takes into account the water balance with low-concentration contaminated water.

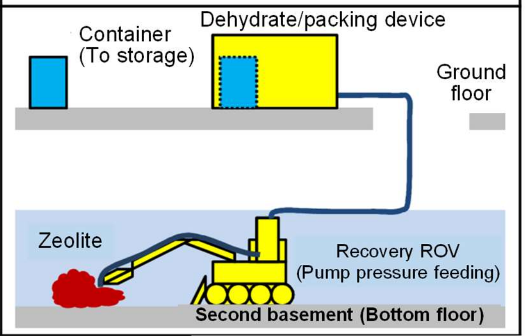
(3) Recovery of zeolite sandbags

As for the zeolite sandbags in a high radiation dose state[53] placed on the lowest floor of the process building and the high-temperature incinerator building, we are considering a method of underwater recovery that directly transfers them to the first floor by remote heavy machine or ROV, dehydrates them and places them in containers, prior to stagnant water treatment (Fig. 15). In this method, securing of visibility in turbid water and improvement of recovery rate of zeolite are issues, and prior examination using mockups is important. An impact assessment must also be performed for the case where part of the zeolite could not be recovered and to consider alternative measures for radiation dose reduction. The basement floors of these buildings have functions for equalizing the concentration of stagnant water, settle fine particles and temporarily store the water during

[52] The 89th meeting of the study group on monitoring and assessment of specified nuclear facilities, Material 4-3, Progress of treatment of stagnant water in the buildings, March 22, 2021

[53] Total weight of approx. 20 t. The maximum radiation dose rate on the sandbag surface is approx. 3,000 mSv/h in the process building and approx. 4,400 mSv/h in the high-temperature incinerator building.

heavy rain. Stagnant water will be transferred to the temporary storage tanks to proceed with stagnant water treatment. It is also necessary to consider the functions and operation method of temporary stagnant water storage tanks upon estimation of changes in water quality and volume.

Procedure	Set up drawing
【Underwater recovery】 1. The zeolite is pumped directly from the second basement to the container installed on the ground floor and dehydrated. 2. Complete zeolite measurement 3. Stagnant water treatment (To expose the floor of the second basement)	

(TECO material edited by NDF)

Fig. 25 Outline of the underwater recovery method for zeolite sandbags[54]

3.3.2.2 Issues of contaminated water management considering the decommissioning process such as fuel debris retrieval

The following three points are the key issues of contaminated water management considering the decommissioning process such as fuel debris retrieval.

(1) Handling of high-concentration α-nuclides

The possibility that fine particles containing high-concentration α-nuclides are produced and mixed into the water treatment system in association with cutting, fabrication, etc., of fuel debris cannot be denied. Therefore, it is necessary to take measures, such as strengthening the monitoring of water treatment systems, installing collection systems for fine particles containing α-nuclides, and monitoring for criticality. Further, in handling such α-nuclides, the chemical characteristics of actinide elements constituting the α-nuclides needs to be sufficiently understood. Actinide elements change into various chemical forms depending on the water quality (pH, dissolved oxygen concentration, coexisting materials, etc.), and the dissolution rate tends to increase significantly especially in the oxidizing environment. Therefore, it is important to understand and control the water quality. When examining the method of removing α-nuclides, moreover, more effective designing becomes possible by sampling even a small amount of actual liquid at an early stage and conducting performance verification using samples. It is necessary to evaluate the safety of recovered high-concentration α-nuclides from the perspective of criticality, heat generation, radiation dose, hydrogen generation, and waste management.

(2) System configuration for sustainable stable operation

The water treatment system at the time of fuel debris retrieval needs to secure reactor cooling water and purify a large amount of radioactive materials containing α-nuclides. Therefore, as a

[54] The 87th meeting of the study group on monitoring and assessment of specified nuclear facilities, “Material 3- 3, Status of study on progress for zeolite sandbags, January 25, 2021”

whole system, it is recommended that the cooling water circulation system has a simple system configuration that focuses on reliability. For the purification system which has a complex configuration, it is recommended to have a system configuration that takes and purifies a portion of contaminated cooling water, enabling both purification and supply of reactor cooling water. As for the purification system equipment, it is necessary to consider the evaluation of the volume of secondary waste generation and maintainability under a high radiation dose, as well as the evaluation of purification performance, and to examine purification methods in the medium-and-long term. In particular, system planning for the entire purification system equipment should be promoted with the view to improving and updating existing systems, based on the operation experience of existing systems, changes in the water quality environment up to the present, and predictions of changes.

(3) Medium-and-long term measures for contaminated water management systems

It is necessary to ensure that periodical inspection and updating of equipment, including land-side impermeable walls and sub-drain systems, is implemented in order to maintain the effectiveness of contaminated water management over the medium-to-long term. For this purpose, it is important to anticipate various risks, such as deterioration of system functions caused by aging, metallic fatigue due to traffic loads, damage of piping caused by natural disasters; to procure/arrange backup and alternative items for the enhanced structure for monitoring and early recovery, and for stable operation; and to promote maintenance/management and system updates in a planned manner.

While the current contaminated water management is shifting to a certain stable state, in addition, it takes a long time to complete fuel debris retrieval. Along with the selection of methods for further expanding the scale of debris retrieval currently in progress, it is important to see a medium-and-long term, overlook the current contaminated water management anew, and examine the principles of more stable contaminated water management and more appropriate maintenance/management.

3.3.2.3 Issues for discharging ALPS-treated water into the ocean

As described in 3.3.1 (4), Treated Water Management, in April 2021 the government announced to discharge ALPS-treated water into the ocean[48] on the premise that safety will be ensured and that measures against reputational damage will be thoroughly taken, and TEPCO is making preparations for such implementation[49]. For the peaceful uses of nuclear energy, the discharge of waste liquid into the ocean with a sufficiently small radiological impact on the human population and the natural environment is in accordance with the internationally recognized concept of "Dilution and Dispersion"[55], which is a method widely adopted in Japan and abroad[56].

[55] ICRP, 1998. Radiation protection recommendations as applied to the disposal of long-lived solid radioactive waste. ICRP Publication 81. Ann. ICRP 28 (4).

[56] Ministerial Ordinance for Commercial Nuclear Power Reactors concerning the Installation, Operation, etc., Article 90, Ordinance of the Ministry of International Trade and Industry 77 of 1978

At the Fukushima Daiichi NPS, efforts are being made to reduce risks for fuel debris retrieval in environments where the radiation level is still high. Under these circumstances, it is extremely difficult to reduce the amount of contaminated water generation to zero. The contaminated water generated is purified by multi-nuclide removal equipment, etc., and stored in tanks located on higher ground as ALPS-treated water containing large-volume tritium (ALPS-treated water/water under treatment). As most of the contaminated water is stored in welded tanks, the possibility of leakage is very low, but the risk is not zero. There is also concern that the tanks occupy a large portion of the site, making it impossible to secure enough space for stable, temporary storage of spent fuel as well as safe fuel debris retrieval.

On the premise that safety will be ensured and that measures against reputational damage will be thoroughly taken, the announced government policy on the discharge of ALPS-treated water into the ocean aligns with the international concept mentioned above, and is an important decision in terms of ensuring the sustainability of decommissioning work. In particular, discharging ALPS-treated water into the ocean enables reduction of risks associated with tank storage and allocation of limited resources to other high-risk operations, contributing to the steady progress of decommissioning work.

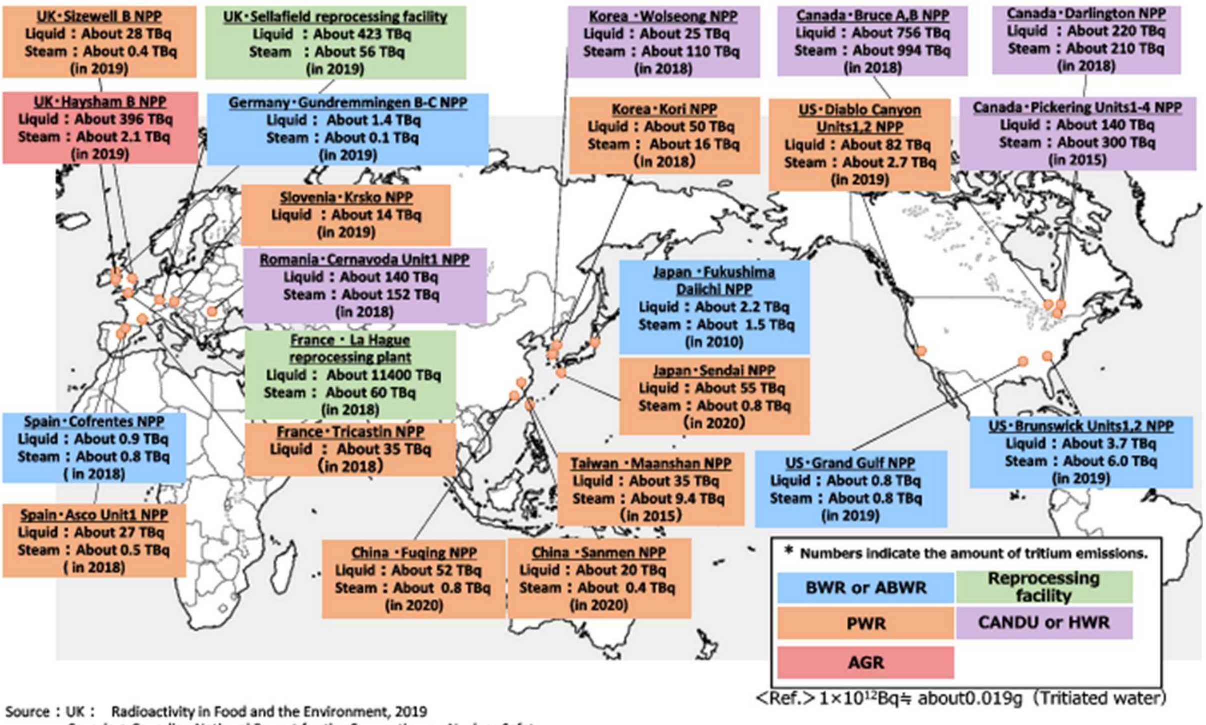
On the other hand, it is also a fact that there have been concerns about reputational damage due to the discharge of ALPS-treated water into the ocean. Therefore, efforts should be continued to deepen understanding to eliminate such concerns. The reliability of TEPCO has declined due to inappropriate incidents in terms of physical protection of nuclear materials at the Kashiwazaki-Kariwa Nuclear Power Station and insufficient provision of information during earthquakes at the Fukushima Daiichi Nuclear Power Station. TEPCO needs to take this reality seriously and respond more carefully than before. One of the reasons for this incident might be insufficient communication with local communities and commercial distributors. Therefore, greater transparency is required, for example, by repeatedly providing explanations in an easy-to-understand and careful manner, mainly by TEPCO, in order to increase understanding of (1) an operation plan for offshore discharge; (2) the effects of tritium contained in the water to be discharged to the ocean on the human body; and (3) the method for verifying the operation status as the basics for implementing safe discharge, and by verifying these through reliable third parties such as IAEA in cooperation with organizations concerned, and by delivering accurate information.

(1) Operation plans for offshore discharge (for details, see the Attachment 12)

Fig. 26 shows the annual discharge of tritium in nuclear facilities in Japan and abroad[57]. These nuclear facilities discharge tritium by means that can minimize the radiological impact on the surrounding environment as much as possible. Fig. 27 shows the treatment flow up to the discharge of waste liquid containing tritium in nuclear power plants in Japan. Waste liquid generated in a power plant is [purified] to reduce the concentration of radioactive materials in the waste water as much as possible by filtration, evaporation, adsorption by the ion-exchange resin method, etc., and

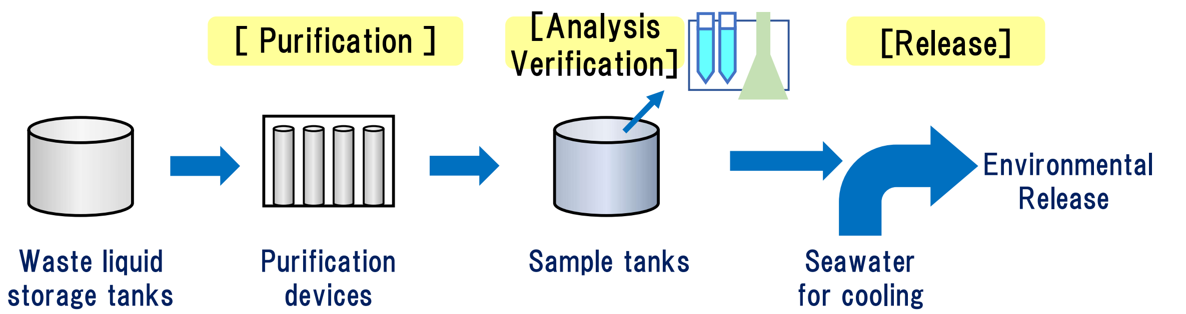
[57] Report by the subcommittee on handling of ALPS-treated water, February 10, 2020

decay of radioactivity with time. The purified liquid is collected in a sample tank where the concentration of the radioactive material is [analyzed and verified] by monitoring and sampling. Subsequently, waste liquid containing tritium is mixed with seawater used for cooling and then discharged to the environment. As such, [purification] → [analysis/verification] → [discharge] is the basic process, and the discharge is in compliance with the regulatory standards on the concentration of radioactive materials. In nuclear power plants in Japan, radiation monitors are installed to measure γ-rays before radioactive materials are discharged into the environment to monitor concentration.



Source : UK : Radioactivity in Food and the Environment, 2019
Canada : Canadian National Report for the Convention on Nuclear Safety
France : Tritium White paper
Other countries and regions : Prepared from reports published by electricity providers in various countries and regions.

Fig. 26 Annual discharge of tritium in nuclear facilities in Japan and abroad[58]



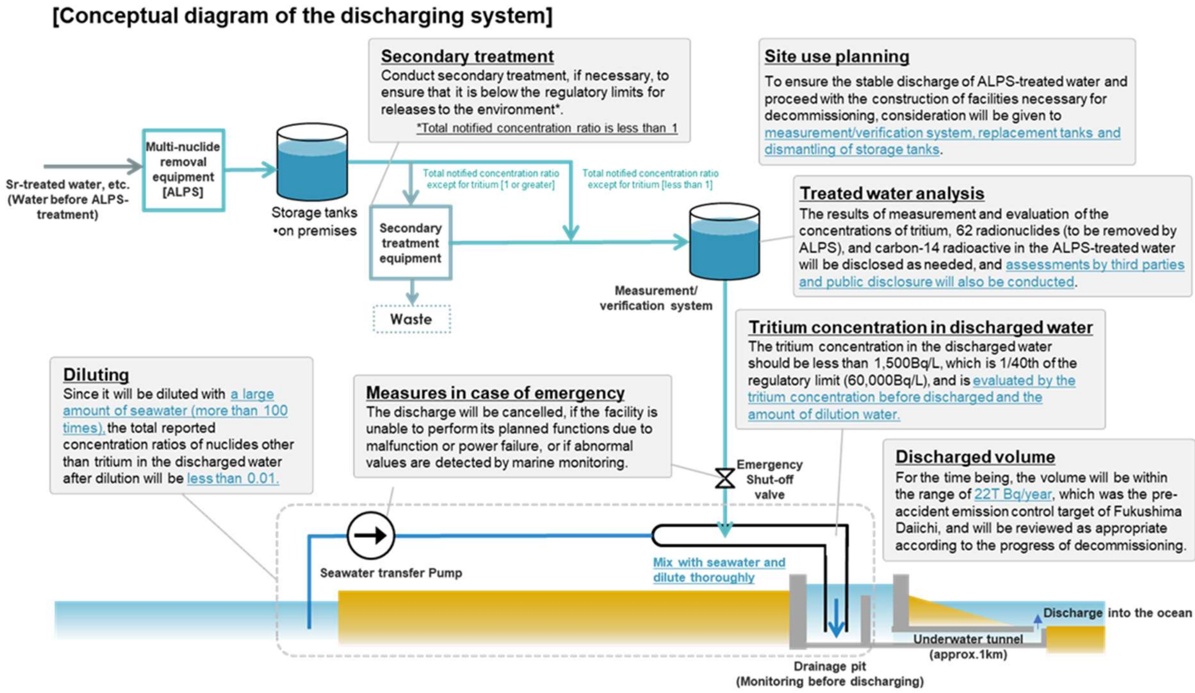
Source : Prepared by NDF with reference to materials on domestic nuclear power plants

Fig. 27 Treatment flow up to discharging waste liquid containing tritium in nuclear facilities

[58]The 1st Expert Meeting for marine monitoring on ALPS-treated water, Reference 2, June 18, 2021

The conceptual diagram of the discharge system planned by TEPCO is shown in Fig.28. In this discharge facility, the concentration of radioactive materials other than tritium in the ALPS-treated water is confirmed to be below the regulatory standard value for the discharge to the environment in the measurement and verification facilities. Then, the discharged water is mixed and diluted with seawater so that the tritium concentration is less than 1,500 Bq/L, which is 1/40 of the regulatory standard value (60,000 Bq/L).

TEPCO will evaluate the tritium concentration after the dilution from the tritium concentration obtained by the analysis at the measurement and verification facilities and the flow rate ratio of the ALPS-treated water and seawater. In addition, in order to confirm that the tritium concentration in the ALPS-treated water after seawater dilution is below 1,500 Bq/L, the tritium concentration will be confirmed by sampling every day during the discharge and will be announced immediately. (For details, refer to Section 3.4(2) c Dilution and Discharge in Attachment 12). In addition, monitoring in other areas of the ocean will be enhanced by adding measurement points and increasing the frequency of measurement[50].



(TEPCO material edited by NDF)

Fig. 23 Conceptual diagram of the discharge system planned by TEPCO[59]

A set of discharge systems covers from [purification], [analysis/verification], [dilution], [discharge] to [continuous monitoring in the sea area]. In addition, an emergency shut-off valve is planned to be installed in the process between [analysis/verification] and [dilution] so that the discharge can be suspended in the event of system failure or detection of any abnormal values during marine

[59]TEPCO.: Attachment 2 [Summary version] Actions by TEPCO in response to the Government's basic policy for disposing of treated water by multi-nuclide removal equipment", April 16, 2021

monitoring. The annual discharge of tritium is planned to be within 22 trillion Bq, a target value of controlling discharge set by the Fukushima Daiichi NPS before the accident.

This discharge system planned by TEPCO is based on the past results in Japan and overseas, and safe offshore discharge will be possible by ensuring that these results are reflected in the system, developing operational procedures and manuals for equipment operation and analysis, thoroughly educating, training, and observing operators, and strictly adhering to the implementation plan. By strictly following the implementation plan, safe offshore discharge will be possible.

(2) The effects of tritium contained in the water to be discharged to the ocean on the human body

Tritium is a radioisotope of hydrogen, and is contained in any water, including water vapor in the atmosphere, rainwater, seawater, tap water, and even in the human body. In addition, its properties are similar to those of ordinary water molecules. As its characteristic, therefore, it is difficult to separate tritium from water. The impact of tritium on the human body due to the disposal of ALPS-treated water has also been evaluated using the UNSCEAR[60] method, and it has been confirmed that the impact is extremely small compared with that of natural radiation (2.1 mSv/yr)[57]. It is also planned to conduct breeding tests of marine organisms in the ALPS-treated water diluted with seawater, with the aim of fostering understanding of the discharge of ALPS-treated water into the ocean and reducing the impact of rumors. It is also planned to conduct webcasting of the breeding situation[50]. TEPCO will review the annual discharge of tritium according to the progress of decommissioning, etc.

(3) The method for verifying the operation status

In the operation phase, it is necessary to confirm that the planned facilities are installed and operated reliably, that the analysis is conducted reliably in accordance with the established procedures, etc., and to publish the results. Moreover, it is necessary to increase the transparency of the implementation status of the plan by involving third parties other than TEPCO in analysis/verification before the discharge, and marine monitoring after the discharge. For third-party involvement, it is important to develop a quality assurance system because the results may vary depending on the institution and facility for analysis due to the fluctuation of the lower detection limit caused by the detection accuracy of measuring instruments or analysis techniques. For example, it is effective to analyze common samples in an integrated way before the discharge to see the fluctuation range of the result, and verify and disclose the quality of each analysis institution beforehand. It is also necessary to consider a structure for verifying the operation status of the developed system.

(4) Further initiatives

TEPCO's planned discharging system, if operated reliably in accordance with the implementation plan approved in the review by the NRA, will have no adverse effects on humans and the

[60]UNSCEAR (United Nations Scientific Committee on the Effects of Atomic Radiation): The Committee reports to the U.N. General Assembly on the effects of radiation among the human population, and on the natural environment from a scientific and neutral standpoint. The 27 member countries include Japan, the UK, the US, France, Russia, Germany, China, and South Korea.

environment, including other radionuclides, and therefore it is an important issue to operate the system "reliably" “as planned”. Going forward, TEPCO will need to proceed with the following preparations. During actual operation, it is necessary to ensure the implementation of the plan (system, operation, information distribution, etc.) established in the preparation stage, to perform check and review, and to review and expand, the plan as needed, as well as to ensure its transparency.

- In the operational phase, develop a series of operation plans including system operation, analysis of ALPS-treated water, flow control of the treated/diluted water, marine monitoring, maintenance, and troubleshooting, and then develop a system plan that minimizes risks and eliminates social concerns
- Perform a radiological impact assessment on the human population and the natural environment, and disclose evaluation results based on the specific discharge plan
- Verify safety by experts from the International Atomic Energy Agency (IAEA) and other agencies
- Develop a plan to strengthen marine monitoring, and perform marine monitoring before the discharge
- Education and training on system operation and analysis, etc., for parties concerned including contractors (TEPCO)
- Development of strategies to provide accurate and understandable information domestically and internationally without causing anxiety from a social perspective, and timely dissemination of the status of preparations
- Ensuring implementation of measures against reputational damage as set forth in the Government’s basic policy announced in April 2021

NDF will provide technical and professional support for TEPCO in designing the discharge system and considering discharging methods, while promoting distribution of accurate information and increasing understanding through various opportunities in Japan and abroad in line with changes in the interests of those who will receive the information. We will also make sure that TEPCO implements measures to minimize reputational damage, and that TEPCO takes appropriate and sufficient actions with compensation in the event of reputational damage.

3.3.2.4 Summary of key technical issues

The main technical issues and plans described in this section are summarized in Fig. , and Fig. 18 shows the future plans in regard to water treatment system for fuel debris retrieval.

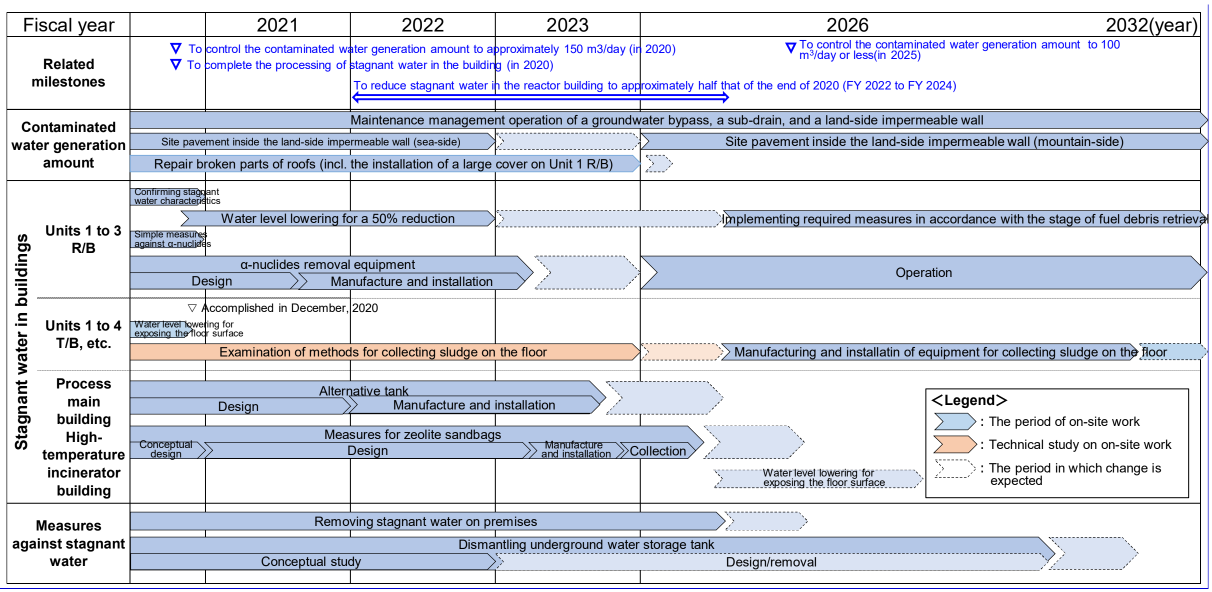


Fig. 29 Key technical issues and future plans on contaminated/treated water management (progress schedule)

3.4 Fuel removal from spent fuel pools

3.4.1 Targets and progress

(Targets)

(1) While the return of residents and reconstruction in the surrounding area is gradually advanced, to carry out a risk assessment and ensure safety including preventing dispersion of radioactive materials and to start removal of fuel in spent fuel pools (SFPs) in FY 2027 to FY 2028 for Unit 1 and FY 2024 to FY 2026 for Unit 2. To complete fuel removal from SFP in FY 2020 for Unit 3. (Completed in February 2021)

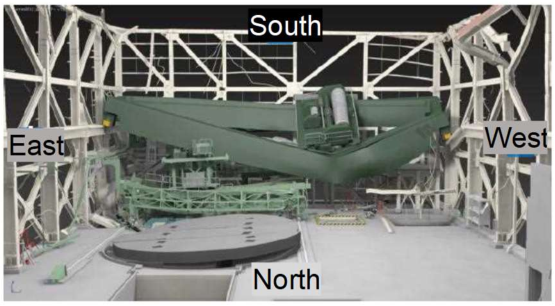
(2) The fuel in Units 1 to 4 that were affected by the accident are removed from the SFPs and transferred to the Common Spent Fuel Storage Pool, etc., where they are appropriately stored so that they are in a stable management state. In order to secure the Common Spent Fuel Storage Pool capacity, the fuel stored there is transferred to and stored in Dry Cask Temporary Custody Facility.

(3) To perform the evaluation of long-term integrity and the examination for processing for the removed fuel and to decide the future processing and storage method.

(Progress)

TEPCO is working on the work plan indicated in the Mid-and-Long-term Roadmap and the Mid-and-Long-term Decommissioning Action Plan.

In Unit 1, due to the hydrogen explosion, roof stabs, building materials, such as steel frames, which constituted the upper part of the building, an overhead crane, etc., have collapsed as rubble on the operating floor as shown in Fig. . While the residents were returning, from the perspective of further reduction of radioactive dust dispersion risk, the whole operating floor was covered with a large cover for fuel removal in Unit 1 SFP. The removal method was changed to one in which rubble removal and fuel removal from SFP are carried out inside the cover. Fig. 24 shows a conceptual drawing of this method. Currently, removal of the building cover (remaining parts) that interferes with installation of the large covers has been completed, and installation of the large covers is in progress within the area where the cleanup around the reactor building has been completed. In addition, in order not to affect fuel in SFP, measures to prevent and mitigate the dropping of rubble, such as installation of supports for overhead cranes and fuel handling machines, and curing of SFP, were completed in November 2020.

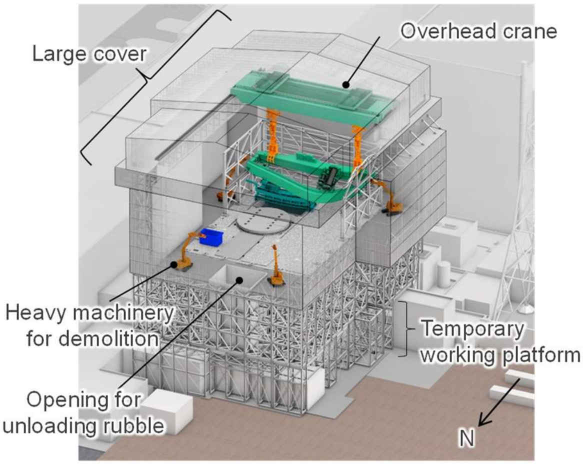


Condition of the existing installations under the collapsed roof (conceptual drawing)

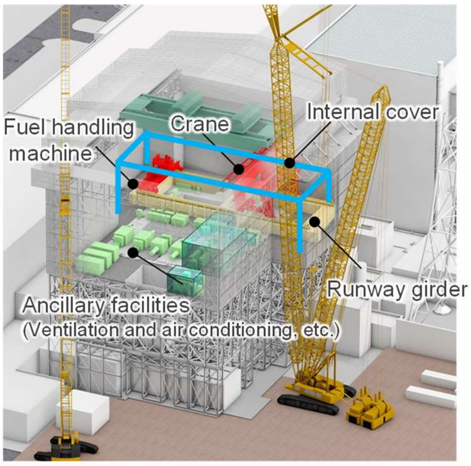
Condition of the collapsed south-side roof

(TEPCO material edited by NDF)

Fig. 30 State of the collapsed rubble on the Unit 1 operating floor



During rubble removal (Conceptual drawing)

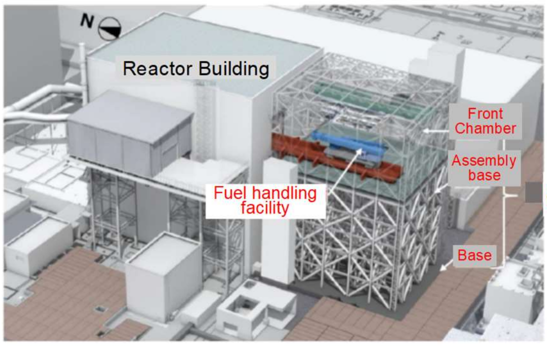


During fuel removal (Conceptual drawing)

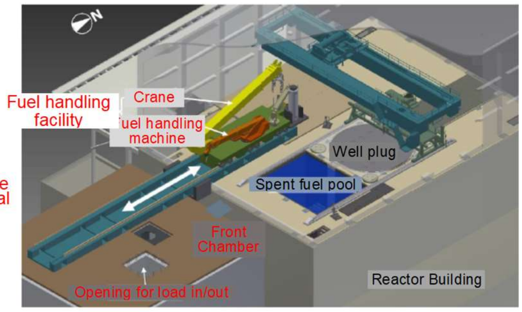
(TEPCO material edited by NDF)

Fig. 24 Fuel removal method from Unit 1 SFP

For Unit 2, a new method in which the upper part of the operating floor will not be dismantled and with access from the south side of the reactor building was adopted from the perspective of further reduction of radioactive dust dispersion risk similar to Unit 1. Fig. shows a conceptual drawing of this method. At present, yard cleanup and soil improvement work are underway as preparatory works for installing the fuel removal platform. In addition, based on the measurement results of the air dose rate and surface contamination of the operating floor conducted in March and April 2021, decontamination of the operating floor is ongoing in order to further reduce radiation dose.



Fuel removal method (conceptual drawing)



Fuel handling facility (conceptual drawing)

(TEPCO material edited by NDF)

Fig. 32 Fuel removal method from Unit 2 SFP

Removal of all fuel from Unit 3 was completed in February 2021. During that time, various problems occurred in the fuel handling equipment and operation, such as a problem in the procurement stage including crane voltage setting error; failure of mast rotation during fuel removal; and irregular winding of mast wire ropes. It is important to roll out the knowledge and lessons learned from these experiences concerning procurement and remote control to planning and implementation of future decommissioning work, including fuel removal from Units 1 and 2 and fuel debris retrieval from Units 1 to 3. Therefore, in addition to keeping records of all operations, it is important to organize the results, including efforts for troubleshooting and recurrence prevention. In particular, measures for strengthening quality control, improving device development, plans and results of preliminary tests, and consideration of contingency plans were compiled as a document for a reference. Furthermore, as part of its approaches to strengthen quality control, TEPCO has revised “The Basic Design Management Manual”, which is common to all Fukushima Daiichi NPS to improve its design and procurement process.

For Units 5 and 6, fuel will be appropriately stored in the SFPs of the units for the time being. Then, they will be removed in a range so as not to affect the work in Units 1 and 2.

Securing the available capacity of the Common Spent Fuel Storage Pool and transfer of some fuel in the Common Spent Fuel Storage Pool to Dry Cask Temporary Custody Facility are required to remove all the fuel in SFPs, including Units 5 and 6, and store them in the Common Spent Fuel Storage Pool. For this purpose, TEPCO is working on expanding storage capacity of Dry Cask Temporary Custody Facility and off-site transportation of new fuel. The available storage capacity in the Common Spent Fuel Storage Pool and Dry Cask Temporary Custody Facility is shown in Fig. .

Such efforts will be made to complete fuel removal in all Units in 2031.

As described above, fuel removal from Unit 3 was completed but other high-radiation dose equipment such as control rods, channel boxes and filters are also stored in SFP. Although cooling is not necessary for them, shielding is required, and there is a risk that the source in the pool will be exposed if the pool water leaks. Therefore, in terms of risk reduction, removal of such high-radiation dose equipment is needed following the fuel in SFP. In this case, it is efficient to utilize the device used for fuel removal and rubble removal. Therefore, as soon as preparations for removal are completed, including securing storage facility, removal work should be implemented immediately. Thereafter, pool water can be excluded from management by draining the pool. Prior to draining the pool, however, the radiation dose and dust dispersion from the pool after drainage should be evaluated to confirm safety. This leads to smooth fuel debris retrieval in the later stages because of the increased flexibility of use of the operating floor, etc.

As with Unit 3, for Units 1, 2 and 4, high radiation dose equipment is stored in each SFP. Giving priority to fuel removal from SFP (removal is completed in Unit 4), fuel removal and drainage of the pool should be performed in a planned manner. In designing fuel handling facility, it is necessary to consider not only the fuel in SFP but also the plan and process for removing the high-radiation dose equipment.

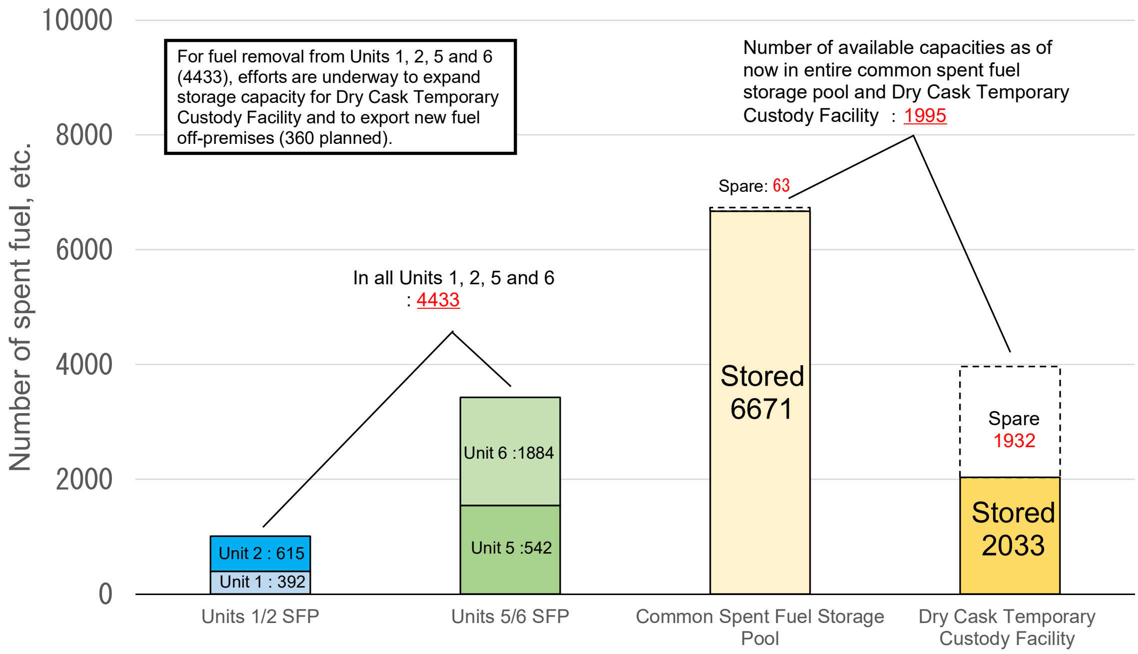


Fig. 33 Storage status of spent fuel (As of March 2021)

3.4.2 Key themes and technical strategies to realize them

3.4.2.1 Fuel removal from SFPs

For Units 1 and 2, it is necessary to advance the work steadily to realize the determined new removal method.

In promoting the project, it is important to make assessment of safety in association with work and confirming that necessary and sufficient safety is ensured. Moreover, it is essential to comprehensively consider technical reliability, rationality, promptness in the work schedule, actual site applicability and project risk, etc.

(1) Unit 1

For Unit 1, the installation of the large cover, and the removal of leftover objects such as rubble on the operating floor will be promoted. Although overhead crane support is installed on the upper part of the operating floor for fall prevention, it is still in an unstable state. Therefore, removing the overhead crane in a safe and reliable way is one of the main issues to prevent it from collapsing onto the fuel handling machine and falling into SFP. Therefore, in ongoing examination of how to remove the overhead crane, it is necessary to perform all safety assessments as an assumption, and it is important to carry out a comprehensive examination based on the perspectives of (i) formulating specific work procedures and work plans enabling identification of risk items, (ii) the risk scenario assumed from (i) and the measures, (iii) identification of points to consider such as exposure of workers, from an operator's perspective, and (iv) rationality and impact on other work.[61]

Regarding the contamination state of the well-plugs of Unit 1 to 3, the Study Committee on Accident Analysis of the Fukushima Daiichi NPS pointed out that the well- plugs have important implications in safety and decommissioning work due to the high level of their contamination[62]. Although the well-plugs of Unit 1 has been evaluated by the above mentioned Study Committee to be about two orders of magnitude less contaminated than several tens of PBq of Units 2 and 3, those in Unit 1 become deformed and unstable due to the impact of the explosion at the accident. For this reason, TEPCO is studying the impact of falling well-plugs on the PCV in the event of an earthquake. It is necessary to make a comprehensive decision on how to handle these well-plugs, based on the study results, and by taking into consideration the impact on the removal of fuel from SFP and fuel debris retrieval in the later stage, and by performing thorough safety assessments.

While applying overseas findings, a detailed handling plan for 67 fuel assemblies with damaged cladding tubes, which have been stored in Unit 1 SFP since before the accident, is under development toward the completion of fuel removal in 2031. In particular, efforts should be made to ensure verification of the post-accident condition, examination/development of handling methods, and risk study associated with handling.

(2) Unit 2

In Unit 2, fuel in SFP will be removed from the opening on the south side of the operating floor using a fuel handling machine composed of a boom-type crane-system, which has not yet been used for nuclear facilities in Japan. Since it is a new system, it is important to do the following: (i) to set up a design schedule with appropriate margins, (ii) to perform mockup tests fully simulating on-site situations and operation methods and ensure feedback on the results to design and production, and (iii) to be sufficiently familiar with the operation and functionality of systems beforehand in preparation for removal by remote operation[63].

As mentioned above, fuel removal from SFP basically assumes unmanned operation by remote control, however, manned operation is also assumed for system installation or system

[61]NDF, "Evaluation on the selection of fuel removal methods (plan) at the Fukushima Daiichi Nuclear Power Station Unit 1", Secretariat Meeting of the Team for Countermeasures for Decommissioning and Contaminated Water/Treated Water (73rd), Material 3-2, December 19, 2019.

[62]NRA, "Draft revision of the interim report (draft) based on the results of public comments", The Study Committee on Accident Analysis of the Fukushima Daiichi Nuclear Power Station (19th meeting), Material 3 (pages 81 to 83), March 5, 2021

[63] NDF, "Evaluation on the selection of fuel removal methods (plan) at the Fukushima Daiichi Nuclear Power Station Unit 2", Secretariat Meeting of the Team for Countermeasures for Decommissioning and Contaminated Water/Treated Water (71st), Material 3-2, October 31, 2019

troubleshooting. In addition, it is recommended that the radiation dose on the operating floor is reduced as much as possible, taking into account of high contamination of well-plugs indicated by the above-mentioned the Study Committee on Accident Analysis of Fukushima Daiichi Nuclear Power Station.[62] In the air dose rate measurement and surface contamination measurement of the operating floor in March and April 2021, it was confirmed that the radiation dose rate was reduced by about 20% compared with the result in FY 2018 due to the results of previous cleanup work. It is important to incorporate these survey results in decontamination and shielding installation methods for further dose reduction of the operating floor.

3.4.2.2 Decision of future processing and storage methods

The future processing and storage methods for the fuel in SFPs need to be decided after considering the impact of seawater and rubble exerted during the accident. The impact of seawater and rubble has been evaluated for the fuel removed from Unit 4, and it is expected that the impact is small. However, based on the situation of the fuel to be removed, it is necessary to advance the evaluation of long-term integrity and the examination for processing and to decide the future processing and storage methods.

It is planned to transfer fuel in SFPs of all Units to the Common Spent Fuel Storage Pool by 2031. Considering tsunami risk, however, it is recommended that fuel in the Common Spent Fuel Storage Pool be transferred to higher ground. Thus, TEPCO is considering fuel storage on higher ground. It is necessary to prepare this storage facility from the perspective of safety and operator. In the case of storage on higher ground, it is recommended that dry storage is used, in which natural convection (ventilation) of the air is used for cooling, and system and maintenance/ management can be simplified. However, consideration should be given in reference to overseas findings, including handling of fuel with damaged cladding tubes.

3.4.2.3 Summary of key technical issues

The main technical issues and plans described in this section are summarized as shown in Fig..

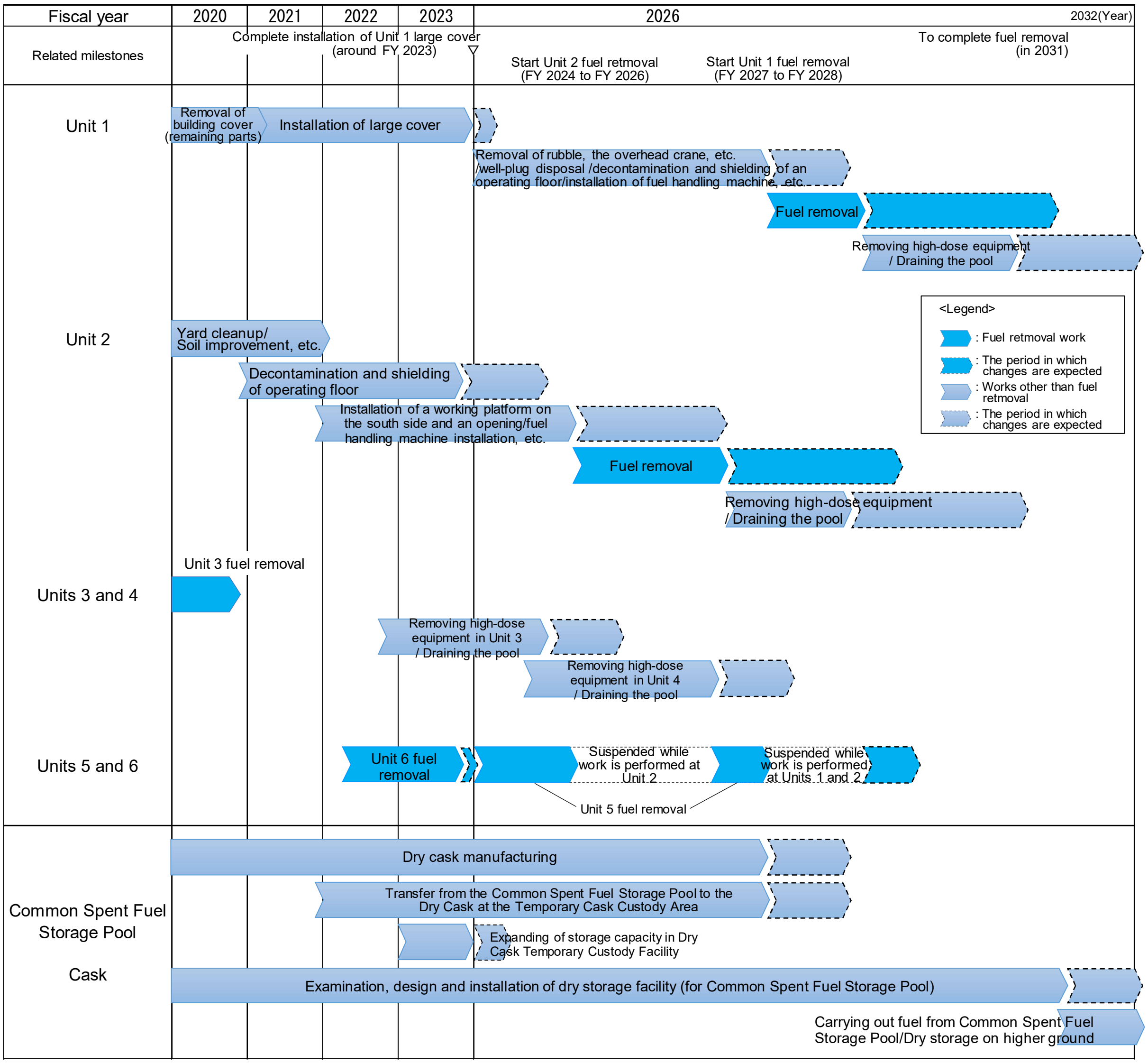


Fig.34 Key technical issues and future plans for fuel removal from SFPs (Progress schedule)

4. Analysis strategy for promoting decommissioning

4.1 Uncertainty of fuel debris, etc. and importance of analytical results

The accident at the Fukushima Daiichi NPS was the first core meltdown accident at a BWR in the world, and there are no records of temperature and other plant parameters due to loss of power at the accident. In addition, many uncertainties remain regarding the state inside the reactors, the state of the fuel debris, and FP release paths, etc., due to the unclear operational status of the safety equipment and the injection of seawater to bring the accident under control. Since there is uncertainty in the formation process of fuel debris and it is not under human control, fuel debris is considered to have heterogeneity in various physical properties such as chemical composition, microstructure and density.

If the uranium content is unknown and 97 - 98% of the uranium content of the fuel assembly before the accident should be set as the value for criticality control or off-site transportation, and safety evaluations/measures should be implemented accordingly. According to the calculation results of the severe accident (hereinafter referred to as "SA") codes and the videos and photos of PCV internal investigations, it is clear that the uranium content decreases by melting and mixing with the surrounding structural materials. However, since there is no value to be used for the evaluation, an excessive margin is included in the safety measures. If the range of such uncertainty can be reduced, there is no need to include excessive safety margins in safety assessments and safety measures, and thus, the promptness and rationality of decommissioning can be improved. In addition to conventional sample analyses, studies on reducing uncertainty of fuel debris properties by other measurement methods have already started by the Project of Decommissioning and Contaminated Water Management.

The analysis results of solid waste are important basic information for the study on processing/disposal methods for various kinds of waste generated by the accident. The analysis results of fuel debris are applied in a number of areas, including retrieval methods, storage management, necessity of processing, investigation to determine the cause of the accident, and improvement of nuclear safety. As shown in Fig. , their relationship changes with the progress made in decommissioning of the Fukushima Daiichi NPS. It is important to correctly recognize that the analytical results are one of the important criteria for decisions for reducing the range of uncertainty in the above examination for facilitating decommissioning. TEPCO, incorporating analysis results, should take the lead in establishing and developing analysis systems, facilities, and functions that can efficiently collect and evaluate analysis results.

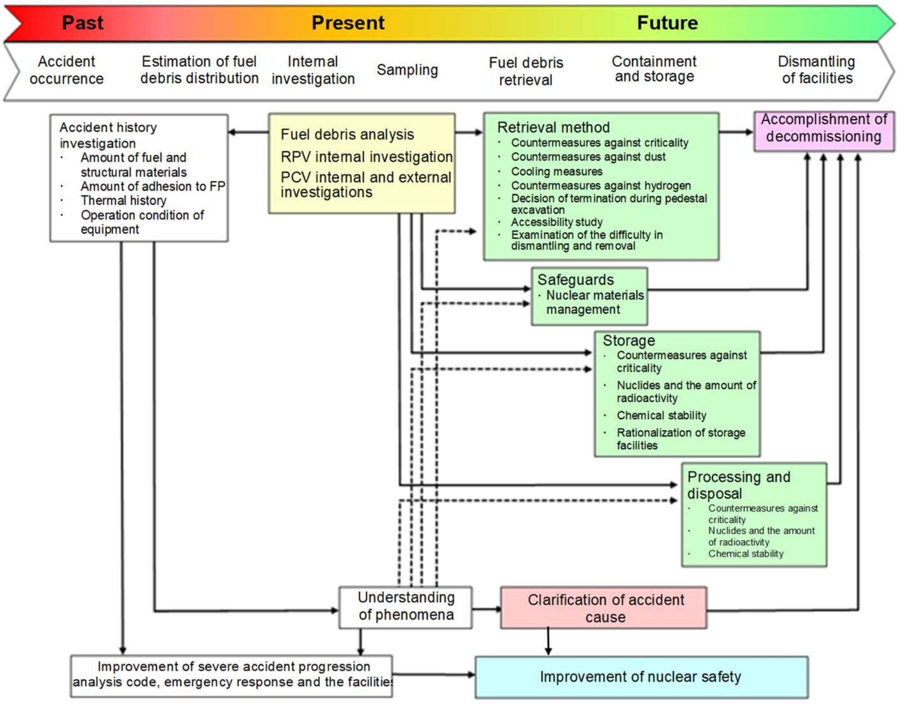


Fig. 35 Incorporation of analyses and investigation results, and their relationships

4.2 Three elements of analysis strategy

To safely and steadily proceed with decommissioning of the Fukushima Daiichi NPS, it is necessary for TEPCO to establish and develop facilities for analysis and the functions required for handling of solid waste or fuel debris. In addition, it is important to build a system that effectively utilizes analyzed results for each decommissioning operation.

In order to obtain good analysis results, as shown in Fig. , it is effective to properly maintain (i) the methods and systems for analysis, (ii) the quality of the analysis results, and (iii) the size and quantity of sample.

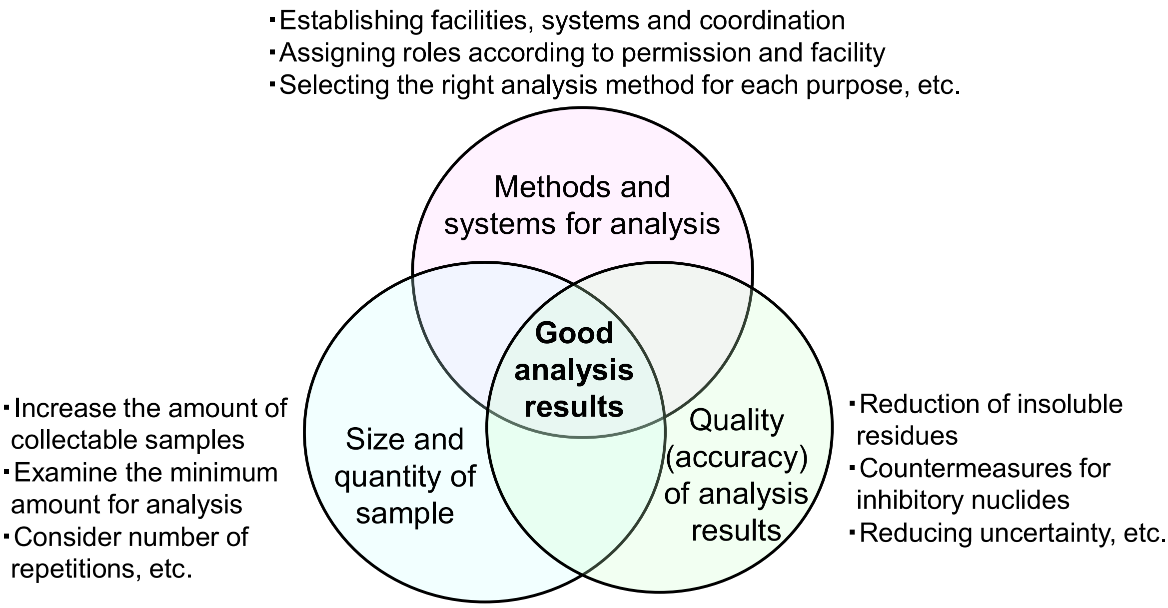


Fig. 36 Three elements of the fuel debris analysis strategy

4.3 Current status of establishing an analysis system and strategy

TEPCO currently utilizes three facilities, which are the laboratory for Units 5 and 6, the chemical analysis building, and the environmental management building (for pretreatment only), and performs the analyses required for facility administration and progressing with decommissioning. They are also planning to establish a facility for the analysis necessary for smooth performance of routine analyses related to processing/disposal of solid waste and fuel debris retrieval in the future. Moreover, as fuel debris retrieval proceeds, the risk of α-nuclides intake is assumed to gradually increase, and establishment of a bioassay function that contributes to assessment of internal exposure is planned.

Some deposit samples and solid waste samples are currently transported to JAEA's facility for analysis in Ibaraki prefecture, private facilities for analysis (Nippon Nuclear Fuel Development Co., Ltd. (hereinafter referred to as "NFD"), and Nuclear Development Co., Ltd. (hereinafter referred to as "NDC"), and analyzed there. It takes a lot of time and is costly to transport samples from the Fukushima Daiichi NPS to these facilities for analysis, i.e., off-site transportation using public roads.

As an essential facility for decommissioning of the Fukushima Daiichi NPS, the JAEA is proceeding with the construction of Radioactive Material Analysis and Research Facilities (facility management building, building #1, building #2) adjacent to the Fukushima Daiichi NPS[64] under the supplementary budget of the Government (FY2012)[65]. At commencing operations in buildings #1 and #2, they will be designated as controlled areas of the Fukushima Daiichi NPS, which has the advantage that off-site transportation is not required. Leveraging this, it is effective to promptly identify basic physical properties such as nuclide inventory, chemical composition, chemical form, shape and density, and incorporate them into safety assessment and work procedures[66].The purpose of building #1 is to analyze solid waste, and building #2 is to analyze fuel debris. The facility management building began its operation from 2018, building #1 commenced comprehensive functional tests from February 2021, and building #2 is undergoing safety review. However, the operation start of building #1 scheduled for June 2021 has been delayed due to a malfunction of the air supply/exhaust system.[67] Therefore, it is necessary to examine further utilization of facilities for analysis in the Ibaraki area and to organize division of roles according to the characteristics of the facilities for analysis in and near the Fukushima Daiichi NPS and those in the Ibaraki area. However, since all the facilities for analysis in the Ibaraki area have been in operation for more than 30 years, considerations on measures are required for aging facilities that will be used continuously.

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[64] The 52nd Secretariat Meeting of the Team for Countermeasures for Decommissioning and Contaminated Water Treatment, "Material 3-4: Opening of the Okuma Analysis and Research Center facility management building"

[65] The 24th meeting of the study group on monitoring and assessment of specified nuclear facilities, "Material 3-1: Development of R&D hub facilities for decommissioning"

[66] The 68th meeting of the study group on monitoring and assessment of specified nuclear facilities, "Material 1-4: Okuma Analysis and Research Center"

[67] The 87th Secretariat Meeting of the Team for Countermeasures for Decommissioning and Contaminated Water Treatment, "Material 4: Shortage of the air volume of the air intake/exhaust system in the building #1 of the Radioactive Material Analysis and Research Facility, and reconsideration of the time of operation start"

Not only the facilities for analysis in the Ibaraki area, but also the Radioactive Material Analysis and Research Facilities to be operated in the area adjacent to the Fukushima Daiichi NPS are short of the human resources required for stable facility operation, and the securing and maintaining of analytical engineers needs to be considered. In this respect, it is necessary to consider in advance the qualities expected of analytical engineers in various analytical works, and to develop a plan so that the required roles are appropriately achieved.

In a normal nuclear power plant, fuel is sealed in fuel cladding, and before the accident, the unsealed α-nuclides were not directly handled in the Fukushima Daiichi NPS. The fuel debris generated by the accident contains unsealed fuel, FP and activation products, and there is a risk of internal/external exposure or spread of contamination in performing analyses. Therefore, in addition to knowledge on general chemical analysis and electronic equipment analysis, a wide range of knowledge is required, including radiation protection, laws and regulations related to nuclear power and radiation, chemical reactivity of fuel and materials, physical, chemical and biological properties of radioisotopes, properties of various types of radiation and measurement methods. Although such knowledge is acquired over time, TEPCO must develop human resources for fields where there is little experience in as short a time as possible. Until now, JAEA and Japan Nuclear Fuel Limited have accumulated sufficient knowledge and experience on the handling of α-nuclides and fuel analysis techniques. Therefore, with the cooperation of JAEA and Japan Nuclear Fuel Limited, it is necessary for TEPCO to promptly work on developing analytical technicians. Personnel exchanges have already started between TEPCO and JAEA for the purpose of developing human resources engaged in handling and analysis of α-nuclides. In FY 2020, two TEPCO employees were transferred to JAEA, and eleven JAEA employees were transferred to TEPCO, and such transfers will continue.

However, since there are few talented personnel (analysis evaluators) who can design the analytical range and items in anticipation of how to use the analysis results in advance, it is also important to make efforts in increasing such personnel. TEPCO, which is responsible for decommissioning the Fukushima Daiichi NPS and who needs to evaluate the analysis results and incorporate them into each process, should take the lead in increasing the number of analytical evaluators.

Blind tests[68] by JAEA, NFD and NDC have been performed on simulated samples produced by Tohoku University in order to improve not only analytical accuracy but also a wide range of analytical techniques.[69] As a measure for increasing the number of analytical evaluators, it is effective to expand the scope of knowledge by performing this blind test and the comprehensive evaluation on fuel debris properties described in 4.4. Through these nationwide initiatives, it is expected that cooperation among organizations will be promoted for analysis, effective use of

[68] A blind test is a test conducted to acquire an objective evaluation of a product without notifying the subject of the product information.

[69] IRID (2021) Supplementary budget in FY 2020, “Subsidies for the Project of Decommissioning and Contaminated Water Management”, (Development of technologies for improving analytical accuracy and estimation of thermal behavior of fuel debris), final report in FY 2020, August 2021

facilities, development of analytical human resources and technical improvement, and expansion of knowledge.

4.4 Improvement of the quality of analysis results, diversification and expansion of analysis methods

4.4.1 Improvement of the quality of analysis results

Fuel debris contains difficult-to-measure nuclides, interfering elements, insoluble materials, etc., and it is considered difficult to conduct complete composition analysis. It is also an important perspective to question the analytical result of the samples in consideration of the impact of the error factor. Monitoring data, sampling analyses, PCV internal/on-site investigation, analyses using SA codes, and past knowledge and experimental results have been accumulated. As part of verification of sample analysis results, deriving consistent property evaluations in reference to analysis, surveillance and test results leads to improving reliability of analysis results, and thereby quality of the analysis results.

Fuel debris caused by the accident is a mixture of fuel and materials in the core. It is important to comprehensively review/evaluate at what stage of the accident progression the substance was produced, what elements were mainly contained, and what properties they have. This also enables provision of feedback on instructions for necessary sampling to cascade analysis results, instead of unnecessarily increasing the number of samples to be analyzed. As shown in the conceptual diagram shown in Fig. , it is expected that a flow will be established from [(1) Fuel debris sampling], [(2) Sample analysis], [(3) Evaluation of the characteristics] to [(4) Safety Assessment] and the cycle from [(1) Fuel debris sampling], [(2) Sample analysis], [(3) Evaluation of the characteristics] to [(6) Instructions for the next sampling]. TEPCO and JAEA are already cooperating in implementing forensic activities that estimate accident behavior and causes by comparing the results of sample analyses with mock-up testing on progression of meltdown and past scientific knowledge, and it is recommended to further expand these activities[70].

[70] IRID (2021), Supplementary budget in FY 2018, "Subsidies for the Project of Decommissioning and Contaminated Water Management", (Development of analytical and estimation techniques for characterization of fuel debris), Results for FY2020 implementation

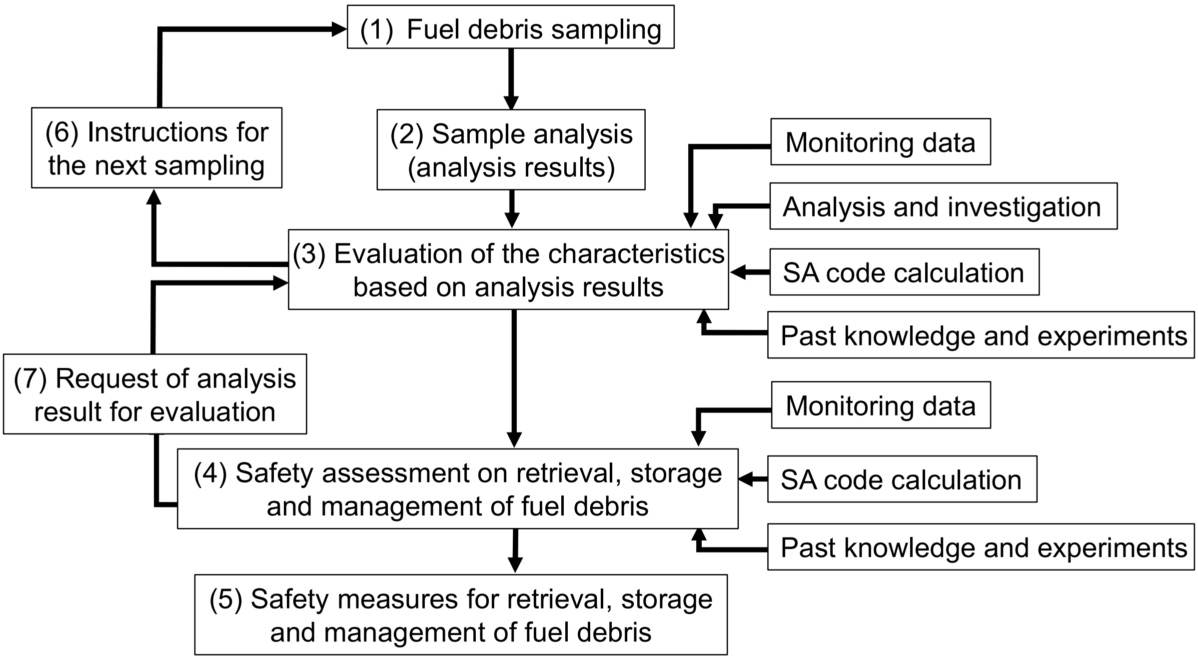

Fig. 37 Conceptual diagram from fuel debris analysis to evaluation/countermeasures

4.4.2 Diversification and expansion of analysis methods in light of sample size and quantity constraint

The current sample analysis is mainly performed using an electronic microscope after transporting smear samples[71] to a facility for analysis in Ibaraki area[72]. Since the density, hardness, and other items cannot be measured for micro or small samples, it is necessary to increase the size and quantity of the samples in accordance with the progress of the fuel debris retrieval process. It is difficult to analyze a sufficient number of samples because the analysis is performed by using a manipulator in each process in a hot cell[73], and by about 0.5 to 1 samples per month in one facility. Fig. shows the estimation results of the accumulated mass of fuel debris in Unit 3 and samples to be analyzed. In the figure on the left, the blue area indicating the cumulative mass of the sample is not visible, but in the figure on the right with the vertical axis enlarged, the blue area becomes visible. The fuel debris in Unit 3 is estimated to be approx. 364 tons[74], while the total amount that can be analyzed in the facility for analysis is estimated to be approx. 45 kg/reactor. This means that only about 0.01% of the fuel debris to be retrieved is analyzed, and there is a large gap between the amount to be retrieved/stored and the amount to be analyzed. As one of the criticality measures

[71] Smearing is a method of examining surface contamination by measuring the amount of free radioactive materials by wiping a surface where contamination is detected with filter paper, etc. This is called the smear method.

[72] The 84th meeting of the Team for Countermeasures for Decommissioning and Contaminated Water Treatment/Secretariat, "Material 3-3: Analysis results of samples related to the internal investigation of the Unit 1 - 3 PCVs"

[73] A hot cell is a compartment surrounded by shielding materials such as concrete and iron to analyze and test highly radioactive materials. In order to prevent leakage, the inside of the compartment is maintained below atmospheric pressure, and radioactive materials are handled by a manipulator and tongs.

[74] Supplementary budget in FY 2014, the subsidies for the Project of Decommissioning and Contaminated Water Management (Sophistication of internal PCV state analysis by severe accident progression analysis and actual equipment data, etc.), completion report (March 2016), Collaborative Innovation Partnership, International Research Institute for Nuclear Decommissioning (IRID), The Institute of Applied Energy (IAE)

in the hot cell, the upper limit is set for the amount of fuel handled and stored, and this calculation also assumes to set the upper limit.

*The right figure shows an enlarged

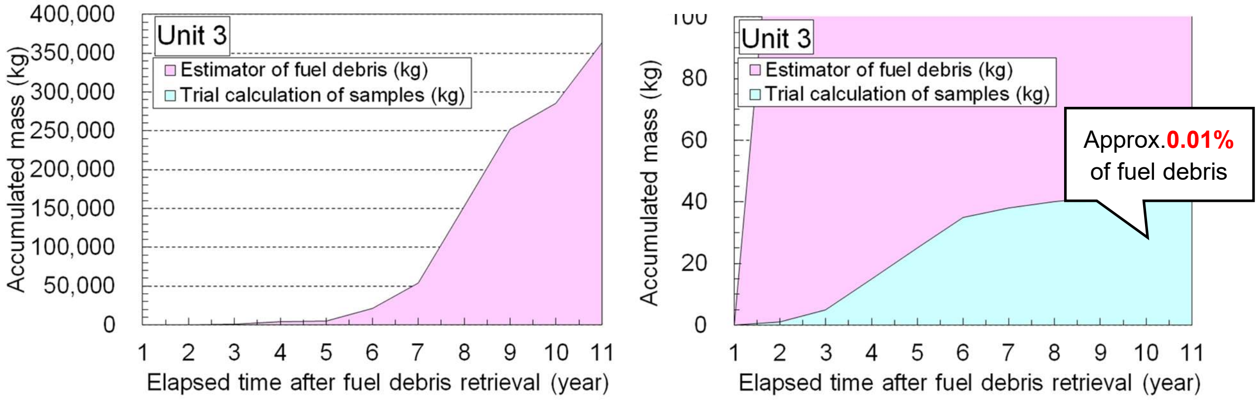

Fig. 38 Estimation results of the cumulative mass of fuel debris to be retrieved from Unit 3 and samples to be analyzed

Since fuel debris has heterogeneity, the analytical values vary depending on the sampled parts, and the situation is such that a sufficient amount of fuel debris cannot be analyzed, resulting in a wide range of uncertainty in evaluation. Applying this situation to Fig. 36, it is effective to diversify and expand the analysis methods and to perform comprehensive evaluation in order to expand the area of good analytical results, where three circles overlap, regardless of the restrictions on the improvement of analysis quality and sample quantity. For example, in addition to performing unsealed sample analysis in a facility for analysis, it is recommended that in-situ analysis[75] (simplified analysis) and non-destructive measurement are performed to increase the amount of information on the sample, including coordinating information during collection, and to consolidate the results to perform evaluations for controlling the range of uncertainty within a certain level. However, there are no practical examples of in-situ analysis or non-destructive measurement of fuel debris, and remote operation is required because of the high radiation dose of the subject. Therefore, it is necessary to promote research and development focusing on specific items such as uranium content.

Fig. illustrates an example timing of in-situ analysis and non-destructive measurement in the process from fuel debris retrieval to storage. In-situ analysis should focus on specific items at or in the vicinity of the retrieval worksite before transporting samples to the facility for analysis. For example, the content of uranium entering and fusing into concrete or steel materials becomes zero at the end of the retrieval work. Samples must be transported to a facility for analysis to confirm by compositional analysis that fuel debris retrieval at the worksite has been completed. This is the final stage of fuel debris retrieval and, by that time, the properties of fuel debris could be identified through sample analyses. Therefore, rather than investing a lot of time and resources in transportation, it is effective for the whole process to perform in-situ analysis at many parts and

[75] An analysis performed by bringing analytical equipment to the location where the sample can be obtained, as opposed to the analysis that is performed by taking the sample for analysis back to the location where the analysis equipment is located.

quickly verify the existence of uranium. This is important in terms of the compatibility between shortening of transportation process to the facility for analysis, cutting process of structures, and increase in analytical information.

In the case of non-destructive measurement, moreover, it is appropriate to provide non-destructive measurement after retrieving fuel debris, adding neutron absorber materials, and drying it, but before transition from the fuel debris retrieval stage to the storage stage. If the timing of non-destructive measurement is before sealing the storage container, fuel debris and solid waste can be sorted out. If it is after the storage container is sealed, pre-operation inspection of the long-term storage phase can be conducted leveraging the condition with zero risk of the spread of α-nuclides contamination. Since the significance, purpose, and difficulty of nondestructive measurement of fuel debris differ before and after sealing them in the storage container, thorough examination of the applicability to fuel debris containing neutron absorbers as well as the feasibility of combining various nondestructive measurement methods is required.

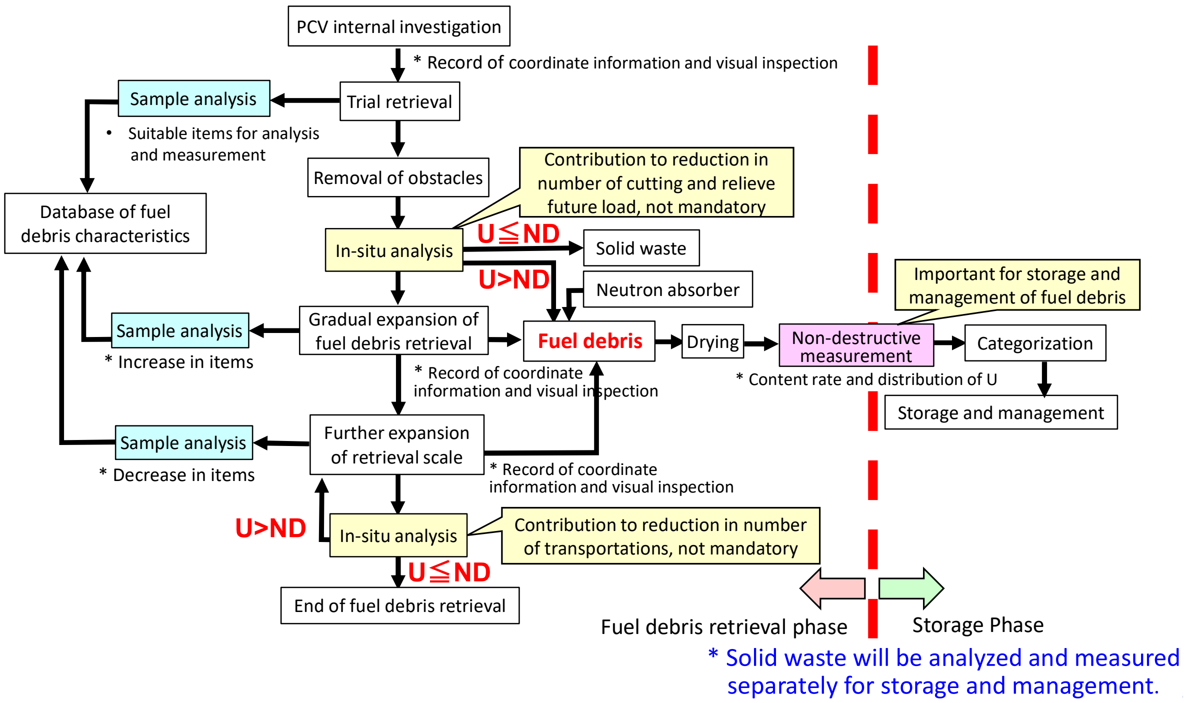

Fig. 39 Example timing of in-situ analysis and non-destructive measurement in the process from fuel debris retrieval to storage stage

5. Efforts to facilitate research and development

5.1 Significance and the current status of research and development

There are many difficult technical issues requiring research and development to promote the decommissioning of the Fukushima Daiichi NPS from the perspectives of safe, proven, efficient, timely, and field-oriented. At present, when trial retrieval of fuel debris is imminent, it is necessary to accelerate research and development in consideration of the practical application for a gradual expansion of retrieval scale and further expansion of fuel debris retrieval.

In order to solve these technical issues, basic/fundamental research and application research by universities in and outside of Japan and researching institutions such as JAEA, practical application research and field demonstrations by IRID, manufacturers including overseas enterprises and TEPCO, etc., are being performed by various industrial-academic-governmental institutions (Fig. 40).

The Government is supporting highly-difficult ones among application research, practical application research and field demonstrations by “The Project of Decommissioning and Contaminated Water Management”, and ones regarding basic/fundamental research by “The Nuclear Energy Science & Technology and Human Resource Development Project” (hereinafter referred to as "World Intelligence Project").

TEPCO is engaged in technical development that directly leads to practical application. NDF considers R&D medium-and-long term plans, next-term R&D plans, and promotes these plans and supports the World Intelligence Project. With institutes involved as its members, moreover, NDF has established the Decommissioning R&D Partnership Council, which considers information sharing on needs and seeds for R&D, adjustment of R&D based on the needs of decommissioning work, and issues on promoting cooperation in R&D and human resource development. Moreover, coordination between the Project of Decommissioning and Contaminated Water Management and the World Intelligence Project has been promoted through the Decommissioning R&D Partnership Council. Based on the discussions at this Council, NDF set up “a task force on research collaboration”[76], and identified six Essential R&D Themes mainly in the area of basic/fundamental research to be addressed with priority, in light of investigational issues and problem awareness on the side of the needs (Attachment 13). The R&D implementation system is shown in Fig. 41. Also, in promoting research and development, it is important to make use of the Naraha Remote Technology Development Center, the Okuma Analysis and Research Center, and Collaborative Laboratories for Advanced Decommissioning Science of JAEA (hereinafter referred to as “JAEA/CLADS”) and to arrange the decommissioning R&D base, including international perspectives.

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[76] It was established in NDF in 2016 and held 6 times in total. It consists of NDF, intellectuals from universities/researching institutions, and TEPCO. Six Essential R&D Themes were identified, which should strategically and preferentially ascertain the principle in promoting decommissioning at the Fukushima Daiichi NPS, and the interim report on the task force on research collaboration was developed (November 30, 2016).

It is important for the promotion of effective/efficient R&D to create a database of knowledge on R&D results, etc., and thus it is necessary for decommissioning operators and R&D institutions to address it. It is also important for these institutions to collaborate and cooperate to compile and archive information/research results obtained during the decommissioning work of the accident reactors, and to make them available to those who will be engaged in the associated research and development. NDF is currently examining the method.

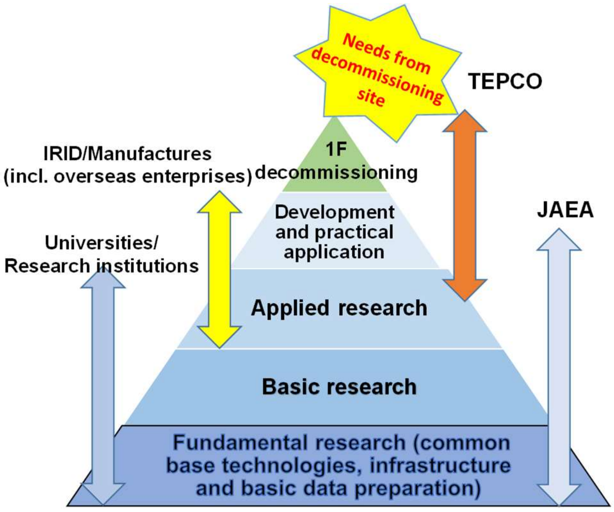
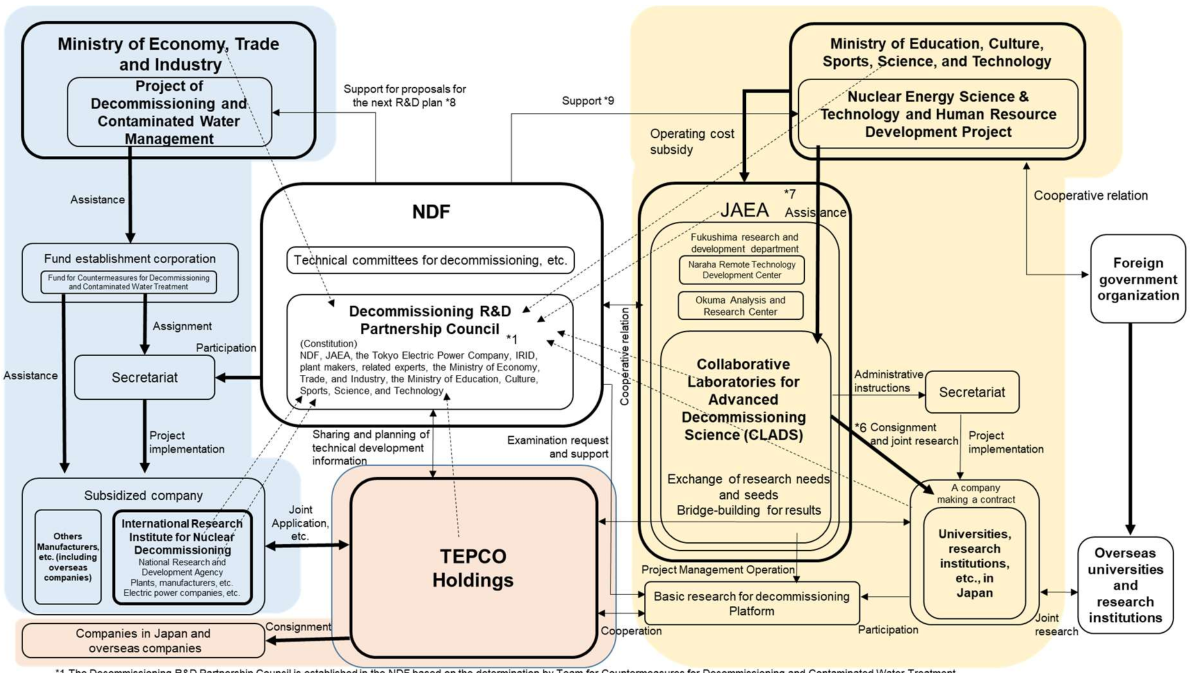


Fig. 40 Scope of R&D for decommissioning and implementation entities



*1 The Decommissioning R&D Partnership Council is established in the NDF based on the determination by Team for Countermeasures for Decommissioning and Contaminated Water Treatment.
*2 The arrow drawn in a thick solid line indicates expenditures such as research expenses and operating expenses (excluding facilities expenses), the arrow drawn in a thin solid line indicates a cooperative relation, etc., and the arrow drawn in a dotted line indicates the participation in the Decommissioning R&D Partnership Council.
*3 Some organizations such as JAEA are located at two or more places.
*4 Each organization has cooperative relations with foreign organizations based on MOU, etc., respectively.
*5 Research and development uniquely conducted by the Central Research Institute of Electric Power Industry, etc., are omitted in this figure.
*6 Although among the World Intelligence Project for Nuclear S&T and Human Resource Development the part adopted up to FY 2017 is the consignment from the Ministry of Education, Culture, Sports, Science, and Technology to a company making a contract, it is omitted in this figure.
*7 Although the subsidy for the World Intelligence Project for Nuclear S&T and Human Resource Development is delivered to JAEA, it is expressed as what is delivered to CLADS, to make it easier to understand.
*8 With regard to the Government-led Project of Decommissioning and Contaminated Water Management, based on the policy in the Mid-and-Long-term Roadmap or Strategic Plan and the progress situation of research and development, the NDF formulates a draft plan of the next research and development, and the Ministry of Economy, Trade, and Industry determines it.
*9 The NDF participates as a member in a steering committee of the World Intelligence Project for Nuclear S&T and Human Resource Development.

Fig. 41 Overview of the R&D structure of the decommissioning of Fukushima Daiichi NPS

5.2 Key issues and strategies

5.2.1 Updating the R&D medium-and-long term plan

In October 2020, NDF and TEPCO developed the R&D medium-and-long-term plan overlooking the overall decommissioning research and development for about ten years in order to appropriately manage the extraction and execution of the required research and development for fuel debris retrieval and waste management after the Government's Medium-and-Long-term Roadmap indicated the course of action for about ten years and the promotion policy of research and development to support it, and based on the Medium-and-Long-term Decommissioning Action Plan 2020 of TEPCO. Attachment 14 shows the R&D medium-and-long-term plan updated based on the Medium-and-Long-term Decommissioning Action Plan 2021 revised in March 2021, progress in R&D and Prospects of processing/disposal methods and technology for waste related to its safety.

In reviewing the R&D medium-and-long-term plan, TEPCO, the Agency for Natural Resources and Energy, and NDF have held R&D planning meetings, and shared the plan with the Ministry of Education, Culture, Sports, Science and Technology (MEXT) to discuss the required results, timing, implementation structure, etc.

The plan will be updated and expanded continuously based on the information made clear by the progress of PCV internal investigation and fuel debris analyses.

5.2.2 Initiatives for the Project of Decommissioning and Contaminated Water Management

5.2.2.1 The Project of Decommissioning and Contaminated Water Management

The Ministry of Economy, Trade and Industry is providing support through the Project of Decommissioning and Contaminated Water Management for research and development to solve technical issues that are difficult to solve in decommissioning. As shown in Fig. 40, the Project selects project operators, such as IRID, and other companies conducting R&D, and provide necessary assistance through corporations with funds and the secretariat. Through the Project to date, significant results, such as the progress in understanding the internal state by PCV investigations, have been obtained for promoting decommissioning work. Attachment 15 outlines the implementation of the Project up to now.

In the Project of Decommissioning and Contaminated Water Management, since last fiscal year, NDF has been participating in the secretariat to strengthen the functions of project planning and progress management, while TEPCO jointly applies for issuance in cooperation with research leadership and provides project management to incorporate requirements in terms of actual site applicability.

As a result of strengthening such functional enhancement, project proposals have been made, which incorporate TEPCO's on-site needs and actual site applicability in more detail, and the projects have been executed so that the results of such proposals can be effectively applied to the engineering work by TEPCO. As the secretariat for the Project of Decommissioning and

Contaminated Water Management, NDF will provide progress management, including verification as to whether the expected results are achieved in time of need, to meet the intended project objectives.

While actively participating in this system, it is important for TEPCO to increase their focus on their own R&D activities, and strengthen their structure. In August 2021, TEPCO has enhanced the system to promote the examination and implementation of technology development in the future.

5.2.2.2 Next-term R&D Plan

In order to support the Project of Decommissioning and Contaminated Water Management, and based on the R&D medium-and-long-term plan, every fiscal year NDF discusses the R&D to be carried out in the next two years with parties concerned at the R&D planning meeting, and then develops the Next-term R&D Plan. At that time, the plan is first deliberated on by the Fuel Debris Retrieval Expert Committee and Waste Management Expert Committee, and then by the Decommissioning Strategy Committee. After this, it is summarized as an NDF proposal. This plan was reported by the Ministry of Economy, Trade and Industry (METI) to the Secretariat Meeting of the Team for Countermeasures for Decommissioning and Contaminated Water/Treated Water Treatment, and The Project of Decommissioning and Contaminated Water Management has been implemented accordingly.

In considering the next-term R&D plan, it is necessary to evaluate the R&D results to date, moreover, to identify issues that should improve its achievement and issues to be newly addressed, as well as to identify emerging issues and organize technical issues with a view of the R&D medium-and-long-term plan. When identifying the issues, it is also important to identify them exhaustively, to confirm whether each issue is in line with the needs of TEPCO, as the entity responsible for decommissioning, and to aim for R&D results to be utilized for TEPCO's engineering.

Based on the next-term R&D plan, there are ongoing projects for gradual expansion of retrieval scale, further expansion of fuel debris retrieval, and waste management. In connection with the projects for “development of maintenance technology for remote operation devices” and “development of assistive technologies for integration management of decommissioning of 1F (development of continuous monitoring system in PCV)” started this fiscal year, it is particularly important to determine the direction for development, and therefore, opinions from the Fuel Debris Retrieval Expert Committee have been solicited to promote development.

5.2.2.3 Research and development implementation system for the Project of Decommissioning and Contaminated Water Management

The IRID has played a major role in research and development for decommissioning for about ten years since the time that the post-accident situation inside the reactors was unknown. In particular, the IRID has established a good track record in internal PCV condition analysis through its internal investigations, and in developing fuel debris retrieval equipment and storage containers.

Meanwhile, as the engineering work by TEPCO progresses, the situation in the reactors and the needs are gradually becoming clear. In addition, development is currently being promoted with the engineering work by TEPCO, which is a shift from joint activities through the Collaborative Innovation Partnership. In light of this changing situation, consideration is being given to a structure for after summer 2023, which is its deadline set by IRID.

In the Project of Decommissioning and Contaminated Water Management, it is important to establish a proper R&D implementation structure. The continuity of R&D activities that have been conducted mainly by the IRID should be ensured, and researchers/developers should cooperate more closely with TEPCO.

The results of the IRID through the Project of Decommissioning and Contaminated Water Management have been achieved with the national budget, and decommissioning of the Fukushima Daiichi NPS is a national/social issue. Therefore, it is important to establish an easy to access system that enables organizations involved in the research and development for decommissioning to make effective use of the R&D results.

5.2.3 Promotion of cooperation between decommissioning sites and universities/ researching institutions

5.2.3.1 The Nuclear Energy Science & Technology and Human Resource Development Project

Universities/researching institutions tasked with basic/fundamental research are expected to maintain and develop human resources, knowledge and infrastructure to make a quick response when technical issues requiring scientific knowledge occur. It is important that universities/ researching institutions share awareness of issues in the field of decommissioning. In order to facilitate the long-term decommissioning project of the Fukushima Daiichi NPS, it is important to conduct scientific and technological investigations based on understanding of the principles and theories from the medium-and-long term perspectives.

In this background, MEXT has been promoting fundamental/basic research and human resource development activities, which contribute to solving issues such as the decommissioning of the Fukushima Daiichi NPS, by bringing together domestic and overseas intelligence from universities/researching institutions as the World Intelligence Project since FY 2015, crossing barriers of the nuclear field, and through close coordination and alignment including international joint research. From new subjects adopted in 2018, the leadership was transferred from the MEXT to JAEA/CLADS to strengthen cooperation between JAEA/CLADS and universities, and establish a system to implement medium-and-long term R&D and human resource development, contributing to decommissioning more stably and continuously.

The six Essential R&D Themes mentioned above have been discussed and further deepened through the Platform of Basic Research for Decommissioning[77], a council to promote basic/ fundamental research, and utilized in the public invitation to the World Intelligence Project. In soliciting applications for the current World Intelligence Project, JAEA/CLADS is using the “overall

[77] Council for promotion of basic/fundamental research through joint management of JAEA/CLADS and the Ministry of Education, Culture, Sports, Science and Technology's program for strengthening research and human resource development in the World Intelligence Project. Refer to: https://clads.jaea.go.jp/jp/about/platform.html

map of the basic/fundamental research", which provides an overview of the entire decommissioning process from contaminated water management to waste processing/disposal, including these six Essential R&D Themes, and identifies the R&D needs and seeds required. Furthermore, as described in 5.2.1, MEXT has participated in R&D planning meetings to share information on the important basic/fundamental R&D, and used such information for project planning in the Project of Decommissioning and Contaminated Water Management.

5.2.3.2 Collaboration between the Project of Decommissioning and Contaminated Water Management and the World Intelligence Project, and initiatives for business-academia collaboration by TEPCO

Some basic/fundamental research contributing to problem-solving in on-site decommissioning have recently obtained outstanding research results mainly in the World Intelligence Project. It is an important issue to directly apply the results in on-site of decommissioning. In order to achieve this, it is essential to match the needs of the decommissioning site with the seeds of universities and researching institutions, build bridges for excellent results, while making use of the overall map of the basic/fundamental research. Based on such awareness, JAEA/CLADS has been tackling assessment of actual site applicability for its adopted subjects in the World Intelligence Project to date, and on-site application as the result of its assessment in cooperation with TEPCO. Moreover, the "steering committee[78]" that presents the basic policy for the operation of the World Intelligence Project has been established since July 2020 by the Ministry of Economy, Trade and Industry in charge of the Project of Decommissioning and Contaminated Water Management and domestic decommissioning related makers. In FY 2020, TEPCO also joined as Program Officer (PO) that is responsible for research management as well as the screening member for selection in the public invitation process, leading to strengthening and accelerating the efforts to apply the perspective of the needs of decommissioning in the Project.

At the 9th Decommissioning R&D Partnership Council held in February 2021, it was proposed that the MEXT and METI work together to resolve decommissioning issues to further deepen needs-driven R&D. In response to this, it is important to further strengthen ties by sharing information on R&D planning meetings and the results of both projects, and to deepen alignment between the World Intelligence Project and the Project of Decommissioning and Contaminated Water Management. In this manner, the Decommissioning R&D Partnership Council is required to continuously serve as a general coordinator, such as by promoting effective alignment between the R&D seeds and the needs of the decommissioning work as the initial task of the Council. For this purpose, it should be noted that TEPCO's decommissioning needs are becoming clearer along with the progress in decommissioning compared to the time when the Council was established in 2015. TEPCO is required to make more effective use of this Council according to the progress of decommissioning, for example, by communicating the needs of decommissioning work more proactively at Council meetings and strengthening interactive information sharing with universities/researching institutions, manufacturers, and academic societies.

[78] Consisting of program directors, intellectuals from universities/researching institutions, NDF, TEPCO, and manufacturers, and established in the Ministry of Education, Culture, Sports, Science and Technology.

Since FY 2019, TEPCO has also started joint research with universities based on the results of the World Intelligence Project to discover the technological seeds owned by universities for decommissioning[79]. The Government, JAEA/CLADS, NDF, TEPCO, and other organizations involved need to further strengthen their cooperation for better matching needs with seeds and serve as a bridge to share outcomes.

5.2.3.3 Establishment of the centers of basic research/research infrastructure

In order to make the long-term decommissioning of the Fukushima Daiichi NPS proceed steadier in technical aspects, it is essential to work on developing R&D infrastructure and accumulate technological knowledge, develop generic technologies and collect basic data, including the essential R&D themes, building up research centers, facilities and equipment, and human resource development. Decommissioning of the Fukushima Daiichi NPS is an opportunity for trialing state-of-art science and technology and accumulation of such activities is expected to become a source of innovation.

In the building for International Research Collaboration of JAEA/CLADS (Tomioka-machi, Fukushima Prefecture), universities, researching institutions, industries, etc., inside and outside Japan form a network and promote research and development and human resource development in an integrated manner. It is expected that a network will be formed for the exchange of diverse human resources from Japan and abroad, such as universities, researching institutions, and industries, and that JAEA/CLADS will become a central organization serving as a hub for such activities. For this reason, the World Intelligence Project has been implemented as a JAEA/CLADS project since FY 2018, transferred from the Ministry of Education, Culture, Sports, Science and Technology. The “Program for Decommissioning Research by Human Resource Development” was newly established in the World Intelligence Project in FY 2019[80]. It created a hub of research and human resource development (Collaboration laboratory) in both educational researching institutions and JAEA/CLADS, and has started R&D and a human resource development project that connects these organizations by cross-appoint system[81]. It contributes to functional enhancement of JAEA/CLADS bases.

It is also important to build research and development infrastructures as hardware. The Naraha Remote Technology Development Center of JAEA, which began full-scale operation in Naraha Town, Fukushima Prefecture, in April 2016, is a facility with various systems such as mock-up facilities for the development and demonstration of remote-control devices and equipment. In particular, it is essential to conduct full-scale mock-up tests, prior to the introduction of equipment into a severe environment that cannot be accessed by humans, not only for performance verification but also for training and establishment of operating procedures, etc. Therefore, active

[79] Tohoku University, Fukushima University, the University of Tokyo, and Tokyo Institute of Technology (Joined in FY 2020 except the University of Tokyo)

[80] For JAEA, the bases in Tokai and Oarai can also be used in addition to the one located in Fukushima Prefecture.

[81] A system that enables universities, public researching institutions, and private companies to enter into employment contracts with each other and operate under the responsibility of each institution.

use of these tests by operators is recommended. Hereafter, a mock-up test of the arm-type access equipment for trial retrieval of fuel debris is scheduled. Use of this facility will become even more important for facilitating decommissioning at the Fukushima Daiichi NPS.

Furthermore, in Okuma Town, Fukushima Prefecture, construction of the JAEA Okuma Analysis and Research Center (Radioactive Material Analysis and Research Facility) is ongoing. So far, starting with the opening of the facility management building, the construction of the building #1, which mainly handles low-radiation dose samples such as rubble, is well underway. The detailed design of the building #2, which handles high-radiation dose samples such as fuel debris, is in progress.

In this way, research facilities related to decommissioning projects are located in Fukushima Prefecture, where a global center for R&D for decommissioning has been established, and R&D infrastructures for medium-and-long-term prospects have been built.

6. Activities to support our technical strategy

6.1 Further strengthening of project management and improvement of capability required as a decommissioning executor

6.1.1 Significance and current status of project management

In order to smoothly proceed with the entire decommissioning project while coordinating and harmonizing, it is necessary to establish a management system in which the organizations involved in the project work together to achieve the goals and enhance their overall capabilities.

The individual work in each work area of a decommissioning project generally proceeds through the following processes: research and development, conceptual design, basic design, detailed design, manufacturing, on-site installation, inspection, and operation. In addition, the NRA will conduct reviews and inspections, as necessary. In order to carry out such a series of processes without omission or delay, it is effective to set up the major workflow defined in the long-term plan as individual projects, which are management units of an appropriate scale. It is then important to optimize the interrelationships and chronological relationships among the projects, and to proceed with overall consistency under a sophisticated project management system so that the risks inherent in the projects can be appropriately managed. From this perspective, TEPCO has been working to build and strengthen its project management system, and in April 2020, then it was reorganized, and the general framework of the management system and the structure was established. In the future, it is important to enhance and upgrade the management methods and rooted in the on-site operations as an effective system. The following are examples of major approaches until FY2020.

① Strengthening the authority of project managers through reorganization, and improving safety and quality levels

In April 2020, the Fukushima Daiichi Decontamination and Decommissioning Engineering Company was reorganized to establish a program[82]/project structure and the Decommissioning Safety & Quality Office was set up directly under the Chief Decommissioning Officer (CDO). As a result, the authority of project managers has been strengthened by assigning full-time project managers and granting budget implementation authority. In terms of operation, efforts were made to improve the operational propelling force of the decommissioning project by establishing a system in which management and other related parties share information on the progress, issues and risks of each program and project every month. Moreover, the Decommissioning Safety and Quality Office was set up to ensure the safety of decommissioning work and to maintain and improve the level of work quality in the face of the extremely uncertain and technically challenging tasks such as fuel debris retrieval.

However, there are many issues for project-based operations to reach a mature level. For example, workforce and work content should be properly controlled so as not to ruin motivation of

[82] Program is the upper level of project, and it is a project in which multiple projects are organically combined to realize the overall mission.

members due to fluctuations in workload between projects. Operations in the decommissioning safety/quality assurance office, a wide range of activities are needed, including establishing a system involved in engineering work by each lead group and developing members specializing in safety.

② Preparation of a plan focusing on long term perspective (Mid-to-Long-term Decommissioning Action Plan)

Since the accident at the Fukushima Daiichi NPS, TEPCO has conducted decommissioning projects in accordance with the requirements based on the Act on Special Measures Concerning Nuclear Emergency Preparedness and the Nuclear Reactor Regulation Law[83], and the target process of the Mid-and-Long-term Roadmap determined in the Inter-Ministerial Council for Contaminated Water, Decommissioning and Treated Water issues. In March 2020, looking at the target process, TEPCO made the decommissioning project transparent to local communities and society and clarified its proactive attitude toward decommissioning while clarifying the complex and long-term work outlook by preparing and publishing the Mid-and-Long-Term Decommissioning Action Plan 2020. In March 2021, TEPCO published the Mid-to-Long-Term Decommissioning Action Plan 2021, which was updated based on the progress of the work.

③ Formulation of R&D medium-to-long-term plan

The decommissioning of the Fukushima Daiichi NPS has entered a "strategic stage of foresight" and is entering a "phase of systematically tackling unexplored areas" such as the retrieval of fuel debris whose location and properties are unknown. For this reason, collaboration with R&D is becoming increasingly important, and TEPCO is preparing the R&D medium- to long-term plan jointly with NDF starting in FY2020. Going forward, TEPCO will need to strengthen its R&D planning system to promote the examination of technological development issues and implementation plans, in order to further strengthen its R&D planning and examination functions, etc.

④ Enhanced budget planning

Each fiscal year, budget plans for program/project work, non-program work (maintenance and utility facilities), and operating costs necessary for decommissioning are formulated, and the steady implementation of work necessary for decommissioning and the improvement of budget accuracy have been achieved through the appropriation of subjects based on the decommissioning mid- to long-term action plan and efforts to finalize designs at an early stage. On the other hand, there are some discrepancies between the budget and the actual results due to the gap between the plan and the actual results in the budget year and the weakness of the design forecast, etc. Therefore, it is necessary to continue the efforts to improve the budget accuracy.

[83] Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors

⑤ Addressing issues across projects

TEPCO constantly identifies and shares risks and issues, and each project deals with them, but for issues that do not fit into individual programs/projects, a cross-divisional system is established to deal with them. For example, in the decommissioning of the Fukushima Daiichi NPS, the prevention of the spread of alpha-nuclides to the environment and the prevention of internal exposure of workers have been identified as future issues, and a cross-program/project working group has been established to deal with these issues.

6.1.2 Key issues and strategies to be strengthened in the future

As shown in 6.1.1, various measures are being implemented to ensure the safe and reliable implementation of the decommissioning project, and it is necessary to continuously upgrade and improve the operation of each of these measures. The following is the important items to be strengthened in the future.

6.1.2.1 Safety and Operator’s Perspectives and promulgating the "Safety First

In April 2021, TEPCO received an order from the NRA to prohibit the transfer of specified nuclear fuel materials due to non-conformities in the protection of nuclear materials at the Kashiwazaki-Kariwa Nuclear Power Station (hereinafter referred to as “Kashiwazaki-Kariwa”). As a result, TEPCO's nuclear regulatory inspection response category became Category 4 ("the objectives of the activities in each monitoring area are satisfied, but there is a prolonged or significant deterioration in the safety activities conducted by the operator"), and many inspections will now be conducted under the supervision of the NRA.

The operation of Fukushima Daiichi NPS differs from that of the Kashiwazaki-Kariwa in terms of the form of business, and as for the management system, the full-scale project management has been implemented in the Fukushima Daiichi, which is also different compared to Kashiwazaki-Kariwa. In February 2021, on the other hand, a non-conformance event occurred, where information on the failure of the seismometer installed on a trial basis in the Unit 3 reactor building of the Fukushima Daiichi NPS was not shared within the organization, and it was not fixed/restored for a long period of time. Although there were no issues with work safety or exposure, etc., a series of events are receiving severe external criticism. At the Fukushima Daiichi NPS, the management themselves are engaged in conversation with all site personnel to understand the organizational issues behind the events. In order to establish safety as an organizational culture, not only a slogan, it is not enough to just ask employees to prepare themselves, but each and every employee needs educational materials and opportunities to learn about safety in a systematic way.

NDF has just started to prepare a safety evaluation guide for decommissioning of the Fukushima Daiichi NPS (a booklet summarizing the peculiarities of the decommissioning of the Fukushima Daiichi NPS, the concept of ensuring the safety of the decommissioning of the Fukushima Daiichi NPS, safety evaluation methods, decision criteria, examples, etc.) with reference to the safety evaluation guide of the U.S. DOE. NDF will consider how to roll out the guide to project activities once it is developed, in coordination with the Office of Decommissioning Safety and Quality, and the Project Management Office.

Ultimately, it is the people and organizations (operators) who handle all aspects of the "Fukushima Daiichi NPS site" (operation, maintenance, radiation control, instrumentation, analysis, etc.) that will realize safety. At Fukushima Daiichi NPS, the effectiveness and sufficiency of radiation exposure reduction and personal safety measures are confirmed at "ALARA Meeting[84]" and "Safety pre-evaluation Meeting[85]". However, it is important that operators (including not only TEPCO employees but also employees of subcontractors) who are familiar with the work site participate in these meetings, and that the work process is designed so that safety can be comprehensively checked from the "on-site perspective" (experience) of the various operators. In other words, it is important to establish the business process as a mechanism, such as incorporating into the gate process the business process of "systematizing and manualizing experience and related information from the field about procedures, rules, and precautions to ensure safety at work, and having the personnel and operators in charge of managing and operating the field work familiarize themselves with them. It is important to establish such a system.

6.1.2.2 Owner's engineering capability

In the case of large scale projects, where the technology is mature and the performance requirements are clear, such as the construction of nuclear power plants, waterfall-type [86] engineering is generally used. In this type of waterfall engineering, it is effective to clearly communicate the requirements of the client (TEPCO) to the potential contractors (manufacturers, general contractors, etc.) in the supply chain as far as possible upstream of the project to proceed with the project smoothly.

However, since the fuel debris retrieval is an inexperienced approach, the target setting and required specifications from TEPCO, the executor of the decommissioning project, are not always clear at the time of engineering, and the degree of performance requirement setting, physical feasibility of the method and equipment, and performance assurance must be trial and error. Therefore, in addition to “the sequential approach” that addresses “work in the initial stage” and then rolls out the information gained to the next stage, the operation executor’s performance requirements, the establishment of the supply chain functions, and its engineering process should be iteration approach-based, to some extent[87]. In this regard, it would be effective to proceed by

[84] The purpose of this meeting is to reduce the radiation exposure of workers and to discuss the effectiveness and sufficiency of engineering measures (physical measures) such as removal of radiation sources and installation of shielding.

[85] The purpose of this meeting is to prevent the occupational accidents and to deliberate on the effectiveness and sufficiency of personal safety measures using risk management methods (risk identification, risk analysis and risk assessment).

[86] It is a method of increasing the accuracy of the design in stages, from conceptual study to basic design and then to detailed design. Since the process proceeds in a single direction from upstream to downstream, it has the advantage of easy control of budget, process and resources (personnel), but it also has the disadvantage of not being able to go back once the next phase is underway. It is called the waterfall type, because it resembles a waterfall where water falls from the top to the bottom.

[87] A method of gradually increasing the percentage of completion of engineering by finding the next result based on a certain result and repeating this cycle.

referring to case examples in other industries that are also implementing the same iteration-type engineering. For iteration-type engineering, the contract between a project executor and a supply chain is not conventional[88]. Therefore, TEPCO, as a project executor, is strongly required to “make a judgment on engineering and is responsible for the results.” To do so, in addition to project management capability, TEPCO needs to improve the capabilities that the project executor should have, to optimize the entire supply chain, specifically the engineering ability that TEPCO, the project executor, proactively performs as the owner[89] (owner's engineering capability) such as the ability to make engineering judgments, the ability to evaluate business risks, and the ability to specify order specifications. For example, in the evaluation of fuel debris retrieval method selection, it is necessary to evaluate from the viewpoints of quality (fuel debris retrieval status, safety, etc.), project (cost, time and other visions), and technical feasibility. Alternatively, a project that would previously have been ordered by a single company can be divided among multiple companies, and TEPCO can oversee and manage the entire construction. It is also important for the enhancement of owner's engineering abilities to accumulate and feedback various field experiences, such as by these activities and the promotion of insourcing to be described later.

Fuel debris retrieval is not a job like the design and construction of a nuclear power plant, where the finished product is delivered with a performance guarantee. Therefore, unless TEPCO, the executor of the project, bears the technical and business risks at the end of the project, the cost would be astronomically inflated. The fact that the project executor bears the technical risk also means that the project executor itself must have the ability to assess the reliability of the functional settings and engineering design, which requires more technical skills. Here, the most important point is to incorporate “safety and operator's perspectives” as upstream as possible in engineering. (Fig.42)

[88] In the conventional construction of nuclear power stations, a supply chain has delivered completed products to a project executor after guaranteeing the performance (Full turnkey contract).

[89] An owner here has three positions of a party responsible for disaster occurrence, a specified nuclear facility licensee, and a facility owner. TEPCO is executing the decommissioning project from these three positions. (A project executor of decommissioning)

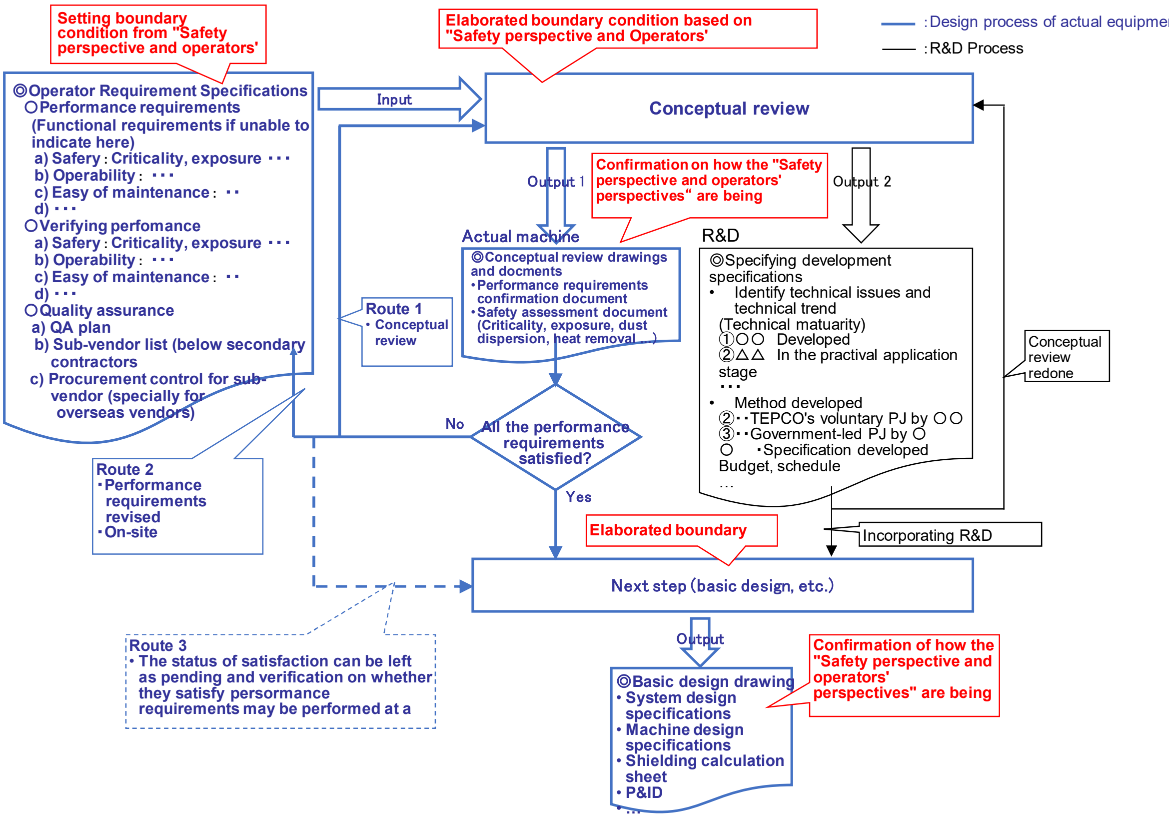


Fig.42 Design flow of fuel debris retrieval (conceptual)

6.1.2.2.1 Ability to assess and manage process risks

It is fundamental to project management to identify and respond to project risks such as cost, process, and safety, which have a significant impact on project execution, and to complete the project within a certain margin of error by making full use of forecasting skills and know-how, while giving top priority to safety and taking into account the constraints and uncertainties surrounding the project. It is necessary to complete the project within a certain margin of error in terms of period and cost. In particular, project risk management for work with high technical difficulty and uncertainty (e.g., fuel debris retrieval) will become even more important in the future.

As a precondition for this, it is essential for TEPCO employees to improve their skills. It is also necessary to eliminate elementary errors such as process delays due to interference from multiple construction work caused by a lack of consideration in the construction planning stage, or failure to assess the progress of processes simply due to insufficient on-site checking.

In the system aspect, TEPCO has commenced operation of the process management system, which enables assessment of the impact on important milestones in an objective way when any delay in a project process becomes apparent. TEPCO has also implemented a process in which each project evaluates the impact of changes, and an appropriate approver authorizes the changes if any revision in milestones is required. However, these initiatives are still in the early stage. It is necessary to further improve the ability to assess and manage process risks, and the operation of the process management system will be further developed to enable detailed input processes, linking of processes between projects and programs, critical path analysis, risk-based process planning, and EVM[90] (Earned Value Management). It is important to further develop the operation of the system to enable critical path analysis, risk-based process planning, and EVM (Earned Value Management). In addition, for systematic risk reduction, it is necessary to increase the sensitivity of each individual to identify risks, to understand the daily changes at the site, to understand the social situation, and to continue to improve technical capabilities. Given the large uncertainties inherent in the projects, including rubble removal on Unit 1operating floor, zeolite sandbags recovery, fuel debris retrieval, etc., are upcoming, the project management should be further strengthened in order to be able to grasp the status of delays and assess the impact more accurately.

6.1.2.2.2 Acquisition management capability

The decommissioning of the Fukushima Daiichi NPS must be carried out reliably over a long period of time under changing site conditions, and it will be difficult to cope with conventional contracts, especially for high-risk project work such as fuel debris retrieval. Therefore, it is necessary to prepare a contract method based on a new concept, in which both the contractor and the recipient cooperate, share the contractual risks, and aim for the agreed-upon goals. In terms of procurement, instead of one-way Buying from an ordering party to an order-receiving party, both parties should bear in mind the concept of Acquisition of the final result by Making, with consideration of all steps from development, manufacturing, to even operation/maintenance (Table 3).

To deal with such a making type project, it is necessary to improve the owner's engineering capability, such as the ability to concretize specifications, as well as to become familiar with acquisition management, which focuses on "acquisition.

Based on the shared awareness that conventional Buying-oriented project management alone is not enough to properly control projects with large uncertainties such as fuel debris retrieval, TEPCO and NDF are actively learning the method with the cooperation of external experts starting from the last fiscal year, and trying to proceed with a step-by-step approach the acquisition management adopted by the U.S. federal government as a benchmark to further develop current project.

Table 3 Difference between “Making” and “Buying”

	Making	Buying
	Acquisition of the outcome of the project （Acquisition）	Purchase products (things) that meet the specifications
What to call the order receiving parties and their roles	Contractor, a partner who is responsible for obtaining the outcome of the project	Vender, who supplies equipment that conforms to specifications

[90] It is a tool for analyzing the cost status of a project. It analyzes the current cost status and predicts how much the budget will be when the project is finally completed.

How to decide who will receive the order	Select based on proposal content and feasibility	Select by price
Contract method	Contracts in line with risk allocation	Fixed price contract
	Data-based cost estimation (analogy: analogy, integration, parametric : sensitivity, analysis, etc.)	Quotation/price list, etc.
	“Do the Right Things” * When doing things, always think about what the right purpose, the right goal, and the right means.	“Do Things Right” * Do things right, by the rules, by the procedures.

6.1.2.2.3 Promotion of insourcing

In order to meet the many challenges of decommissioning, it is essential to strengthen engineering capabilities, and TEPCO is promoting insourcing as a means to achieve this. “Insourcing” means to develop an ability that enables TEPCO to implement planning, design, maintenance, and operation on its own. It aims at reducing unreasonableness and waste, further deepening the level of productivity improvement, and improving the operation quality of TEPCO employees including the quality of design and procurement. The improvement of these qualities will ultimately contribute to the improvement of safety. While insourcing is effective for a wide range of issues, it is advisable to proceed with an awareness of the issues that are expected in the future to gain more benefits. Examples of future issues are listed below.

・Improvement of operation and facility quality through internal integration of operation and maintenance know-how

In terms of operation and maintenance of facilities, it is expected that the number of operation tasks will increase due to the installation of new facilities and that maintenance costs will increase due to the aging and deterioration of facilities. It is necessary to devise more "Smart Maintenance" methods to keep maintenance costs low while not degrading reliability. For example, in the case of a new system to be used for a long period of time, sensors (vibration, heat, sound, etc.) for online monitoring of the operating status can be mounted in advance, a large amount of accumulated maintenance data be processed by computer to be used for prediction maintenance, and maintenance data obtained from existing systems be utilized. In this way, systematization and sophistication of maintenance knowledge management facilitate accumulation of maintenance knowhow internally.

Furthermore, for long-term operations such as debris retrieval, where highly radioactive materials (and nuclear fuel materials) are handled, TEPCO employees, who are the license holders, should perform the operations themselves, and the operator should ensure radiation safety and strive to internally accumulate operational know-how.

・Enhanced capability to make an engineering judgment

For a task like fuel debris retrieval, which is inherently subject to great uncertainty and for which TEPCO has no previous experience, TEPCO needs to make engineering judgments on its own, and to do so, it must have a firm grasp of the details of design, construction, maintenance, and operation. Therefore, to take design as an example, TEPCO should improve its competence by working on system design and determining the required specifications of equipment from the current stage.

6.1.2.3 Securing and developing human resources

6.1.2.3.1 Securing and developing human resources for smooth implementation of decommissioning projects

(1) Securing and developing human resources based on the medium-and-long-term human resource development plan

The development of human resources is essential as a basis for the smooth implementation of long-term decommissioning projects. For this purpose, project management and engineering capabilities should be enhanced. In addition, it is important to assume occupational categories, the number of engineers and the time required in the future (design, operation, maintenance, chemical analysis, safety assessment, radiation control, etc.) in light of the Medium-and-Long-term Decommissioning Action Plan, to summarize them as the medium-to-long term human resources development plan, and to promote human resources development and securing of staff systematically.

As an initiative to enhance project management capabilities, TEPCO has given responsibility and authority to project managers so that they can devote themselves to project management. In addition to basic education for project management, TEPCO is expanding the efforts such as to newly introduce training program for systematic learning in accordance with international standards. As an initiative to enhance engineering capabilities, while collaborating with other electric power companies, as well as manufacturers, general contractors and engineering companies, TEPCO has been making use of the knowledge of external experts including overseas experts, and accumulating and transferring their technical capabilities and know-how including engineering capabilities. Based on the medium-and long-term decommissioning action plan, measures are being considered to develop human resources and to secure personnel, assuming the number of engineers and the time to be required in the future.

(2) Secure and develop human resources with a "safety and operator’s perspectives

It is necessary to secure and train human resources with "safety and operator’s perspectives" systematically and promptly.

TEPCO recognized early on that it is important to secure and train human resources who are well versed in the field and have acquired sufficient knowledge of the field to have an "operator’s perspective. The “core technologies” required for the decommissioning of the Fukushima Daiichi NPS have been established, and the “Decommissioning Core Technology Course” has been held to foster them. Since 2015, the company has been implementing initiatives to improve on-site capabilities with in-house veteran instructors. In addition to the above general familiarity with the Fukushima Daiichi NPS decommissioning site and acquisition of on-site knowledge, in case where special skills specific to the facility or work to be handled are required, it is necessary to secure and train personnel with such special skills individually. For example, in the case of remote handling of unsealed radioactive materials such as fuel debris retrieval, which has never been experienced in the operation of power plants, it is necessary to consider securing and training personnel with such special skills individually through operation at mock-up facilities in Japan and overseas, and actual work in removal of radioactive materials on a gradually expanding scale. It is necessary to consider how to secure and train them individually. With regard to human resource development for fuel debris retrieval operations, TEPCO has started efforts to learn about remote operation systems and training methods overseas (RACE, UK). It is also essential to secure and develop human resources capable of planning the necessary analyses according to analysis needs and also equipped with analytical techniques (including handling of samples) for analysis work associated with fuel debris retrieval. Given the prolonged decommissioning work, it is recommended to utilize local human resources. Assuming an analysis system by function of the facility for analysis, TEPCO is planning to secure techniques by accepting outside engineers, transfer and send their personnel to outside analysis institutions, and systematically allocate chemical and material careers and new employees, and is starting some of these initiatives.

It is the personnel (operator) or organization engaged in the entire site of the Fukushima Daiichi NPS (operation, maintenance, radiation control, instrumentation, analysis, etc.) that ultimately achieves the perspective of both “operator” and “safety” on site. For this reason, it is desirable that personnel with an "operator’s perspective" simultaneously acquire a "safety perspective" such as criticality management and confinement of alpha-nuclides. To this end, there is an urgent need to develop human resources who have the following qualities and who can play a central role in expanding the "safety perspective".

① To know the basic concept to achieve reactor safety.
② To know the general methods to achieve the basic concept
③ To know how the basic concept and general methods of safety are applied and realized in actual design.
④ To have a field-oriented viewpoint of safety assurance.
⑤ To have basic technical knowledge about handling of unsealed radioactive materials (including alpha nuclides), criticality, and measurement (detection) methods. As evidence of such knowledge, the person should have a qualification such as a chief nuclear fuel handling officer (or a qualification established internally as equivalent).
⑥ The person should have knowledge of international precedents.
⑦ The person should have knowledge and experience in negotiating with the NRA.

6.1.2.3.2 Fostering the next generation who will be responsible for the future decommissioning of Fukushima Daiichi NPS

In order to continue decommissioning of the Fukushima Daiichi NPS for a long period of time and to continue R&D activities necessary for that, it is important to secure solid technical capabilities on a constant basis. For that purpose, it is essential to train and secure future researchers and engineers, and ensure technical transfer.

Primarily, the assumption is that TEPCO will establish the decommissioning of the Fukushima Daiichi NPS as a planned and sustainable project, under which activities of human resource development and acquisition, including their employees and contractors, will be promoted independently and continuously. Human resources involved in decommissioning of the Fukushima Daiichi NPS are required not only to specialize in nuclear energy but also to possess science and technical knowledge in fields other than nuclear energy. TEPCO is required to secure and develop human resources with such diverse technical backgrounds in the course of decommissioning.

For this reason, in addition to the importance of strongly encouraging human resource development and recruitment directly within TEPCO and contractors, it is most important that excellent human resources who have graduated from universities, graduate schools, technical colleges, high schools, etc., and have specialized in science and technology are continuously sourced to these companies. In order to achieve this in a stable manner, it is necessary for higher and secondary educational institutions to create an opportunity for learning and acquiring peripheral knowledge in addition to professional knowledge, and to maintain associated systems and structures so that they can function as a whole, including teachers.

With this basic understanding, it is important for industrial-academic-governmental institutions, as a whole, to steadily promote efforts according to each level of higher and secondary education.

For students in higher education such as universities, graduate schools, and technical colleges, it is important for the industry and higher education institutions to cooperate and continuously implement activities to promote understanding of the nuclear industry. In recent years, due to the reorganization of faculties and departments related to nuclear energy, it has been pointed out that the human resource development function in higher education institutions in the nuclear field is weakening. Therefore, attention must be paid to maintaining and enhancing the human resource development function in the nuclear field nationwide. In particular, it is necessary to communicate the idea that the decommissioning of the Fukushima Daiichi NPS is an extremely advanced technical challenge, which is unprecedented in the world, and to establish various career paths for young researchers and engineers to thrive in the nuclear industry including decommissioning, and to specifically demonstrate and make them feel a career view. For the development of next-generation human resources, it is fundamentally important that young researchers/engineers are constantly produced from these higher education institutions.

In particular, the World Intelligence Project by MEXT and JAEA/CLADS has introduced a system in which students and young researchers are made to be aware of decommissioning as an important research area, and is engaged in decommissioning research. From the perspective of human resource development, support has been provided for young researchers and teachers in preparing and implementing lecture curriculums related to decommissioning. After more than five years since the launch of the World Intelligence Project, it has produced great results in terms of both research and human resource development. At “the Conference for R&D Initiative on Nuclear Decommissioning Technology by the Next Generation (NDEC)”, a conference for students to present their research findings as part of the World Intelligence Project, and the Creative Robot Contest for Decommissioning for technical college students, students present their research results, exchange views with researchers and engineers involved in the decommissioning of the Fukushima Daiichi NPS, and are given awards for excellent performance on a constant basis. As for an initiative for young researchers in the World Intelligence Project, in addition to the existing program represented by young researchers under the age of 39, a new program was launched in FY 2021 that requires young researchers to assume commensurate research responsibilities within R&D projects as a requirement for application. In order to encourage autonomous research activities, an initiative has been taken since April 2021 where young researchers engaged in the World Intelligence Project are allowed to be involved in independent research activities up to 20% of the efforts devoted to the project with the consent of the research representatives (relaxation of the obligation of full-time commitment).

It can be said that the mechanism of and implementation by the World Intelligence Project have produced some results for researchers and students in higher education institutions. Hereafter, it is important to implement this project so that the perspectives of decommissioning sites in TEPCO and those of the activities in higher education institutions can be more aligned.

For junior and high school students in the stage of secondary education, it is important to introduce appealing points of engaging in the nuclear energy field including decommissioning, and to make efforts to attract their technical interests with a focus on decommissioning, as well as to increase their understanding of decommissioning and reconstruction of the Fukushima Daiichi NPS, and in a broad sense, of the career path in science and technology fields. The secondary education stage is a stage in which, as a preparatory stage for participating in and contributing to society, students can develop their individuality and competence while being influenced by researchers, engineers, and science teachers who are active in society, and can make independent choices and decide their career paths. From this perspective, NDF holds the “International Mentoring Workshop Joshikai in Fukushima” for middle and high school female students mainly in Fukushima Prefecture in cooperation with the Organization for Economic Cooperation and Development and the Nuclear Energy Agency (hereinafter referred to as “OECD/NEA”) as an effort to increase interest in science and engineering, especially in decommissioning, etc., through exchanges with female researchers and engineers in order to enhance understanding among women and help increase their motivation to participate in development studies. In 2020, the "International Mentoring Workshop Virtual Joshikai in Fukushima 2020" was held using the online system, even in the midst of the global outbreak of the new coronavirus infection, and will continue to be held using the online system in 2021. In addition, “student sessions” are held for high school students, etc., to give thought to the reconstruction of Fukushima along with the International Forum on the Decommissioning of the

Fukushima Daiichi NPS (hereinafter referred to as “International Forum”). Through these efforts, high school students, etc., are given an opportunity to think about activities to achieve both decommissioning of the Fukushima Daiichi NPS and reconstruction, enabling them to increase their awareness that decommissioning is an important issue in reconstruction of local communities, and foster interest in and willingness to contribute to decommissioning and reconstruction efforts. Such activities have achieved some positive results. These sessions and forums will continue to be held in 2021 by utilizing an online system.

For the development of such technical personnel associated with decommissioning of the Fukushima Daiichi NPS, it is also necessary to expand into wider fields including fundamental research and related research. In the course of raising the level of the entire fundamental technological base in Japan, it is expected that initiatives to deal with the nuclear legacy and nuclear safety will take deeper root.

Institutions concerned are continuously required to promote and strengthen their efforts to secure and develop human resources for the next generation according to their respective roles and levels.

6.1.2.3.3 Dissemination of basic knowledge and promoting the people’s understanding for decommissioning and radiation safety involved in decommissioning

It is important for many citizens and local residents to acquire basic knowledge of the accident and decommissioning, disaster response, radiation safety, food safety, etc., related to the Fukushima Daiichi NPS from the perspective of future resilience of the whole country. This is because it will serve as basis for discussions on decommissioning, and related radiation safety, etc., based on accurate information and for promoting public understanding. In addition, although it is not directly aimed at fostering human resources who will play a leading role in the nuclear field in the next-generation, it is also an aspect of indirectly broadening the range of human resources who are interested in not only nuclear energy but also science in general. Particularly in the field of nuclear energy, it is necessary to learn about the relationships in local communities and society through various opportunities according to the developmental stage of children, as well as to acquire knowledge/experience on nuclear energy and decommissioning. In doing so, since it is important that children take an interest through the knowledge and experiences of adults around them, such as teachers and parents. Therefore, it is important to further spread knowledge on nuclear energy and decommissioning based on scientific evidence, which to a wide range of people including those involved in primary education institutions.

6.2 Strengthening international cooperation

6.2.1 Significance and the current status of international cooperation

In recent years, nuclear reactors and nuclear fuel cycle-related facilities built at the dawn of the use of nuclear energy have reached the end of their operational life, and decommissioning of these facilities is in full swing in many countries. Among the reactors that have experienced severe accidents are the Windscale Pile-1 reactor in the UK, the Three Mile Island Unit 2 reactor (TMI-2) in the US, and the Chernobyl Unit 4 reactor (ChNPP-4) in Ukraine. These facilities have been undergoing stabilization work and safety measures for many years. In addition, there are large uncertainties in the management of a wide variety of radioactive materials at legacy sites overseas, and decommissioning and environmental remediation efforts are expected to take a long time. Each country continues to face challenges such as "unknown unknowns," long-term project management, and securing large amounts of funding.

In order to steadily proceed with the decommissioning of the Fukushima Daiichi NPS, which deals with difficult engineering issues, it is important to learn lessons from precedent decommissioning activities, etc. and apply them to the decommissioning of the plant as a risk reduction strategy, and to utilize the world's highest level of technology and human resources, i.e., to gather and utilize the wisdom of the world.

In addition, the decommissioning of the Fukushima Daiichi NPS is a process to solve unexplored engineering problems by combining knowledge from various fields, not limited to the nuclear field, and it can be interpreted that the decommissioning of the Fukushima Daiichi NPS can be a powerful place to create innovation. The concentration of diverse knowledge and experience from around the world in Fukushima is, in the first instance, an important effort to steadily advance the decommissioning of the Fukushima Daiichi NPS itself. This is also an important initiative from the perspective of building a symbiotic relationship with the local community, which is essential for the long-term progress of decommissioning.

To bring together the wisdom of the world, it is important to maintain and develop the international community's continuous understanding, interest, and cooperation in decommissioning. Therefore, it is important to gain the confidence of the international community by disseminating accurate information on the progress of decommissioning, etc., and to promote decommissioning in a mutually beneficial manner that is open to the international community by actively and strategically returning to the international community the knowledge, etc., gained through the accident at the Fukushima Daiichi NPS and decommissioning.

Specifically, it is important to promote bilateral cooperation in line with the circumstances of each country and to utilize the framework of multilateral cooperation through the IAEA and OECD/NEA. These international organizations have an important role in establishing international standards for decommissioning. It is important to participate in the formulation of international standards based on Japan's experience in decommissioning to promote the decommissioning of the Fukushima Daiichi NPS in an internationally open manner, and it is also expected to fulfill part of Japan's responsibility to the international community by sharing Japan's experience with other countries. From this perspective, Japan has been holding an annual dialogue and establishing a conference body to share information with other countries as an intergovernmental framework. NDF has been working on disseminating information on decommissioning through participation in side events of the IAEA General Conference and speaking at major international conferences such as the OECD/NEA Steering Committee. By securing the confidence of the international community and promoting mutually beneficial decommissioning, we are trying to maintain and develop the international community's continuous understanding and interest as well as cooperative relationships. (Attachment 16)

In addition, although the global outbreak of new coronavirus infections is a major obstacle to the above-mentioned international cooperation, many meetings and events such as the IAEA General Conference and its side events, OECD/NEA, etc. are held online, and Japan has participated by utilizing the online system. NDF has also been actively utilizing the online system, etc., to gather the world's wisdom, maintain and enhance the international community's continuous understanding and interest, and maintain and develop cooperative relationships with international society. Specifically, opportunities for continuous exchange of information have been secured, for example, by conducting online exchanges of opinions and meetings with overseas experts invited to NDF every year, and by holding online annual meetings with decommissioning-related organizations in other countries. For example, in a short period of two weeks from May 17 to May 28, 2021, bilateral meetings between several domestic organizations concerned and the OECD/NEA were realized. In the future, it is important to further expand the opportunities for communication with other countries by taking advantage of the experience gained so far. Fig.43）

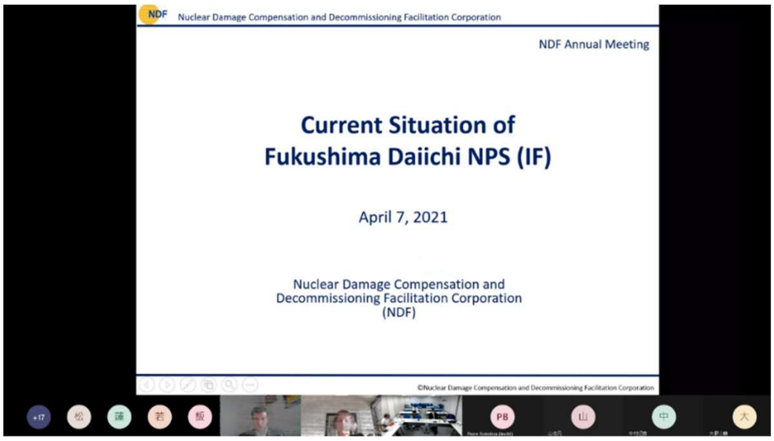


Fig.43 Annual meeting of NDF with foreign organizations concerned (held online in April 2021)

6.2.2 Key issues and strategies

6.2.2.1 Integrating and utilizing wisdom and knowledge from around the world

The decommissioning of Fukushima Daiichi NPS is expected to take a long time, and the decommissioning of legacy sites has many points of reference in terms of technology and operation as a leading model. In each country that has a legacy site, public organizations related to decommissioning are playing a central role in promoting decommissioning to cope with issues such as the necessity of new technologies and expertise that differ from the operation and maintenance of nuclear reactors and nuclear fuel cycle facilities. Therefore, it is necessary for Japan to conclude cooperative agreements and arrangements among domestic and overseas organizations under the intergovernmental framework, and to continue to implement regular information exchange. NDF will continue to maintain and develop long-term partnerships with public decommissioning organizations that play a central role in each country, such as NDA in the UK, CEA in France, and DOE in the US.

With regard to the decommissioning of the Fukushima Daiichi NPS, Japan needs to learn from the wisdom of the world not only in terms of technology but also in terms of operations, including systems and policies, strategy formulation, project planning and management, safety assurance, and regional communication. To this end, as we move forward with decommissioning, Japan has received support from the international community and has received various kinds of assistance from foreign governmental organizations and experts, through the dissemination of information on issues related to decommissioning to the international community and participation in international joint activities such as the DAROD project by the IAEA and the joint project by the OECD/NEA. More than ten years have passed since the accident, and it is necessary to continue the mutually beneficial relationship while also working to return the know-how and results accumulated so far to the internal community.

The difficulty in traveling to and from other countries due to the pandemic outbreak of new coronavirus infections has become an obstacle to the continuation of such mutually beneficial relationships. It is important to secure opportunities for communication and work to maintain and develop relationships by utilizing online systems and other means so that the unprecedented situation, not limited to the new coronavirus infection, does not dilute relationships with relevant organizations, experts, and international organizations in other countries.

Through a series of the International Forums, NDF has collected the wisdom including lessons learned and technologies derived from decommissioning around the world. It is important to continue to make the International Forum an effective opportunity to gather wisdom from all over the world by using the online system, even in the situation where direct participation from other countries is difficult.

The decommissioning of nuclear power plants, both in Japan and abroad, is being carried out under contracts between many companies and decommissioning executors, and the global market for decommissioning is expanding greatly. As the engineering of Fukushima Daiichi NPS is in full swing, it is important to grasp the latest status of excellent technologies and human resources in the world and to utilize them effectively. In this context, TEPCO has been actively engaged in technological exchanges with overseas private companies. It is necessary for TEPCO to continue to keep abreast of the latest information from around the world, including the situation in the private sector, and to engage in continuous communication with these private companies, sharing information on the progress of decommissioning work and forming an environment in which the necessary technologies can be accessed when needed.

6.2.2.2 Maintaining and developing the international community's understanding of and interest in decommissioning and cooperative relationships

In order to mobilize the wisdom of the world for the decommissioning of the Fukushima Daiichi NPS, it is important to maintain and develop the understanding, interest, and cooperative relationship of the international community. To this end, it is important for the government and other domestic organizations to disseminate accurate information on the decommissioning of the Fukushima Daiichi NPS, considering the fact that more than 10 years have passed since the accident and the interests of the recipients of the information have changed since the time of the accident. Specifically, for experts, it is necessary to provide information not only on the progress of decommissioning but also on the lessons learned from the decommissioning work. In addition, consideration should be given to providing easy-to-understand information not only for experts but also for non-experts, and to adding appropriate devices that consider the level of understanding of the recipients regarding the background of the accident. In addition, it is also important to participate in international joint activities and disseminate information to return the knowledge obtained in the course of decommissioning to the international community. In participating in international joint activities, it is necessary to work on the premise that decommissioning, which is Japan’s top priority, will be steady implemented, and the interests of the international community can also be secured. From the aspect of returning the results, it is important to maintain the level of interest while responding to the changes in the international community, such as the growing interest in not only the accident and decommissioning itself but also the application to other issues.

On the premise of ensuring safety and taking thorough measures against reputational damages, some countries, in fact, issued comments expressing concern about the impact on the environment and questioning the transparency in disposal in response to the Japanese government's announcement of a policy to discharge ALPS-treated water into the ocean. On the other hand, foreign governments, related organizations, and international organizations that understand the situation of Japan's decommissioning have issued comments supporting Japan's decision. In addition, the IAEA announced that it would actively cooperate with the discharge of ALPS-treated water into the ocean from a third-party standpoint by dispatching a review mission and supporting environmental monitoring. These actions are pushing to gain international understanding of offshore discharge, and the importance of ensuring transparency and building cooperative relationships through accurate information dissemination was reaffirmed.

It is important to continue to obtain the understanding and cooperation of the international community for the steady implementation of decommissioning, including the future discharge of ALPS-treated water into the ocean, and it is necessary for the government and other relevant domestic organizations to disseminate accurate and easy-to-understand information at international conferences, international joint activities, and on websites. It is also important for NDF, through various opportunities, to disseminate information that is accurate meets the changing interests of the recipients, and to return to the international community the knowledge obtained through the decommissioning process.

6.3 Local community engagement

6.3.1 Significance and the current status of local community engagement

6.3.1.1 Basic concept

The fundamental principle for the decommissioning of the Fukushima Daiichi NPS is "Balancing between reconstruction and decommissioning”. In the areas where the evacuation order has been lifted, progress toward reconstruction is gradually being made, not only by the return of residents and the resumption of business activities, but also by the promotion of migration and settlement from outside the area and new investment. While giving top priority to further reducing risks to the surrounding environment and ensuring safety. It is necessary to strengthen communication and promote coexistence with local communities to gain the trust of the community. Decommissioning should not be allowed to have a negative impact on the reconstruction process due to anxiety and distrust of decommissioning, in other words, decommissioning should never be a hindrance to reconstruction efforts.

Therefore, it is important to deepen the understanding of local residents and reassure them about the decommissioning through interactive communication: not one-way dissemination of information, but sincere listening to the concerns and questions of local residents and promptly providing them with accurate information in an easy-to-understand manner to eliminate them. In addition, to accomplish the decommissioning over a very long period of time, the continuous cooperation of companies, especially local companies, is essential. At the same time, the participation of local companies in the decommissioning project is an important pillar of TEPCO's contribution to the reconstruction of Fukushima, as it will not only revitalize decommissioning-related industries in the region and create employment and technology, but also lead to the spread of the results to other regions and industries. In light of this, TEPCO will contribute to job creation, human resource development, and the creation of industrial and economic infrastructure in the region through decommissioning, while collaborating with efforts to realize the "Fukushima Innovation Coast Framework Promotion Organization", which sets the accumulation of decommissioning-related industries in the Hamadori region as a priority field, and aims to achieve "Balancing between reconstruction and decommissioning”.

6.3.1.2 Specific measures under the current situation

(1) Communication initiatives

The government has been exchanging opinions with local related organizations at the "Fukushima Advisory Board on Decommissioning, Contaminated Water and Treated Water" and other meetings held by the government, disseminating information on the current status of decommissioning through videos, websites, brochures, etc., and holding briefings and roundtable discussions for local residents and related local governments.

NDF is holding the International Forums for the purpose of frank exchange of opinions on decommissioning with participants including local communities and organizations concerned, and for sharing of the latest knowledge, technical achievements, and issues on decommissioning with experts in Japan and overseas (the Forum was postponed in FY 2020 due to the COVID-19 infection). They also hold briefings on the progress of decommissioning at meetings hosted by the national and local governments. In order to promote the exchange of opinions at the International Forums, a “hearing activity” is held every year to hold a conversation with local communities including high school and technical college students before the International Forum is held. Then, their real voice is collected, summarized and edited as a booklet, and distributed as the “Voice from Fukushima” at the International Forums.

TEPCO has been making efforts to provide explanations and dialogue to regional representatives at conferences hosted by the government and Fukushima Prefecture, as well as to hold regular press conferences and lectures for the media, and to disseminate information through its website and brochures. In addition, the company is actively accepting visitors to the Fukushima Daiichi NPS and holding roundtable discussions, as it is very effective in forming a common understanding to have people see the current status of decommissioning as it is and to exchange frank opinions (number of visitors: 18,238 in FY2019, 4,322 in FY2020). In addition, the "TEPCO Decommissioning Archive Center" established in Tomioka Town as a place where people can learn about the process of the nuclear power plant accident and the progress of decommissioning has about 70,000 visitors as of the end of April 2021, and since last fiscal year, the center has been collaborating with the "The Great East Japan Earthquake and Nuclear Disaster Memorial Museum" opened by Fukushima Prefecture in Futaba Town.

In addition, a virtual tour of the decommissioning site of the Fukushima Daiichi NPS has been available on TEPCO's website since 2018. In the current situation where direct observation is limited due to the new coronavirus infection, such simulated experience programs are more useful than ever.

(2) Approach to create regional industrial and economic infrastructure through decommissioning

Based on their "commitment to the people of Fukushima to achieve both reconstruction and decommissioning” established in March end, 2020 (hereinafter referred to as “Commitment”), TEPCO has summarized their efforts for the accumulation of decommissioning work into the following 3 categories: (1) Increased participation of local enterprises, (2) Support for local enterprises to step up and (3) Creation of new local industries, and has started to implement them in a phased manner. With regard to (1) and (2), in cooperation with the Fukushima Innovation Coast Framework Promotion Organization and the Fukushima Soso Reconstruction Promotion Organization, TEPCO has set up and are operating a joint consultation service to support matching between local companies interested in participating in decommissioning projects and prime contractors who are considering placing orders with local companies. In addition, it is also conducting a survey of the needs of both prime contractors and local companies regarding human resource development, and has started joint research with several universities. Moreover, the contents of the "Medium- to-Long-Term Procurement Outlook in the Decommissioning" prepared in September 2020 are being updated as necessary to reflect the progress of decommissioning work, and briefing sessions are being held not only for prime contractors, but for local governments, commercial and industrial organizations, and local contractors, paying close attention to the spread of the new coronavirus infection.

In addition, with regard to (3), in order to build an integrated decommissioning project implementation system locally, from "development and design" to "manufacturing," "operation," "storage," and "recycling," TEPCO is planning to establish and operate several new facilities in the 2020s, so that technologies and products of relatively high difficulty and importance, which have been ordered outside Fukushima Prefecture, including overseas, can be completed in the Hama-dori region (announced on May 27, 2021). In particular, with regard to local manufacturing, TEPCO has set-up a joint venture with partner companies that have a proven track record in the field of high-function products that had to be manufactured outside of the prefecture, the plan is to set up a manufacturing base in the Hama-dori region with the aim of creating local employment, placing orders with local companies, and promoting collaboration.

At the same time, in order to steadily promote initiatives for local community engagement, TEPCO has reformed their organization as needed. The local partnership promotion group was established in the Fukushima Daiichi Decontamination & Decommissioning Engineering Company in April 2020, TEPCO set up the specialized department working to engage with local community at the Fukushima Daiichi NPS in October 2020, and the Hama-dori decommissioning industry project office that directly reports to the president was also established. Based on the division of roles, they are engaged in internal/external coordination, field response to local communities and consideration of medium-and-long-term direction.

6.3.2 Key issues and strategies

(1) Communication issues and strategies

Misunderstandings, concerns, and rumors caused by the inappropriate dissemination of information on the decommissioning of the nuclear power plant will lead to a loss of reputation and trust in the decommissioning of the nuclear power plant not only in the local community but also in society as a whole, which will not only delay the decommissioning of the nuclear power plant but also hinder the reconstruction of Fukushima. For this reason, TEPCO needs to continue to take various measures to promptly communicate the current status of decommissioning in an easy-to-understand manner. In this regard, while the impact of the new coronavirus infection are expected to continue for the foreseeable future, TEPCO will make active use of tools such as virtual tour programs and online conference systems, and it is also important to strengthen communication that is possible even in non-face-to-face and non-contact situations, such as by further enhancing photo and video content.

In addition, the government, NDF, and TEPCO must work to build trust with local communities by providing information more carefully under appropriate coordination. Therefore, capturing opportunities to hold round-table talks and join local meetings/events, it is necessary to have direct interaction with local communities. Efforts should also be made for two-way communication by conversation, including listening to their concerns and questions carefully through events such as

International Forums, and to deliver accurate information in an easy-to-understand and careful manner.

In particular, the disposal policy of the ALPS-treated water has been the subject of anxiety and concern not only from locally but also domestically and internationally, and local governments and related organizations are strongly urged to provide accurate information and take all possible measures to prevent rumors. In light of these circumstances, TEPCO must do its utmost to suppress rumors by steadily implementing the measures outlined in the "TEPCO's response based on the government's basic policy on the disposal of treated-water by multi-nuclear removal facilities" (released on April 16, 2021), and by making additions and revisions as necessary.

(2) Issues and strategies related to the creation of regional industrial and economic infrastructure through decommissioning

As mentioned in 6.3.1.2(2), TEPCO is making various efforts to realize the "Commitment," but these efforts will not produce visible results immediately and will require a certain period of time. In addition, for “(3) Creation of new local industries” which will be promoted in the future, it is a relatively large-scale investment and is expected to have a great economic effect on the Hamadori area. However, as advanced techniques are required to produce high-performance products, the issue is how to eliminate the technical gap between experienced prime contractors and local communities and connect this to promoting active participation in local companies. Therefore, for the time being, it is important to continue and strengthen the current activities in a credible manner, including "(1) Increased participation of local enterprises" and "(2) Support for local enterprises to step up". It is also important to carefully explain to local governments, commercial and industrial organizations, and other organizations concerned the location and scale of new decommissioning-related facilities, the schedule from construction to operation, and the status of considering engagement with local communities in terms of employment, cooperation and order placement, and to proceed with the activities while gaining understanding and cooperation.

Also, with the understanding of prime contractors, it is necessary to consider specific methods of ordering and contracting that will make it easier for local companies to receive orders, and to implement these methods on a trial basis. As a result of interviews conducted with local companies last year, it became clear that local companies do not necessarily want to be the main contractor, but tend to want to enter the market as a subcontractor to gain technology and experience. After properly understanding the intentions and needs of these local companies, a scheme can be established to benefit both parties by not only approaching local companies, but also encouraging existing prime contractors to place orders with local companies, including technical guidance. This will contribute to the promotion of orders from local companies by adopting methods that are beneficial to both parties. At the same time, with regard to human resource development, the Fukushima Decommissioning Engineer Training Center of the Fukushima Nuclear Energy Suppliers Council, which was established in 2018 and has been providing education of radiation protection and special education on specific matters such as low-voltage electricity handling, should be used to provide the training. In parallel, specific studies and preparations for training specifically for local companies should be accelerated. It is important to steadily promote these various efforts while responding to changes in the situation as appropriate, and to build a foundation for local industry and economy through the decommissioning project and to develop local companies and human resources.

In addition to research and development related to decommissioning, as companies from outside the region move into the region and provide technical guidance to local companies, the number of engineers and researchers visiting and staying in the region is expected to increase. Therefore, it is necessary to establish the necessary environment and support system so that such external personnel can integrate into the local community and play an active role as a member of it. In particular, it is necessary to take into consideration a wide range of functions such as daily life and education so that not only single people but also families can live together with peace of mind.

In this regard, in order not only to promote the return of residents to their homes, but to accelerate the reconstruction of the evacuated areas by encouraging migration and settlement in the wider area, Fukushima Prefecture has established the “Fukushima 12 Municipalities Migration Support Center”, and assists migration and settlement of people mainly from outside the prefecture to the 12 municipalities, disseminating information to people nationwide who are interested in migration and providing various types of support for those who wish to move to the 12 municipalities. It is important to consider the possibility of collaboration and cooperation with these local initiatives.

To steadily promote these efforts for coexistence with local community, it is essential to strengthen the organizational structure within TEPCO and to have close cooperation between each department. As mentioned in Section 6.3.1.2(2), TEPCO has been reorganizing itself to set up specialized departments for regional symbiosis, and efforts to promote local industries through decommissioning are gradually moving forward, and gaining a certain level of recognition from the local community. It is important to keep this trend going steadily, while further strengthening the internal structure as necessary.

In addition, it is necessary to further strengthen cooperation and collaboration with local governments, including Fukushima Prefecture, and local related organizations, including the Fukushima Innovation Coast Framework Promotion Organization and the Fukushima Soso Recovery Promotion Organization, which are operating a joint consultation service and co-hosting matching business meetings. NDF will provide appropriate support to TEPCO's efforts for regional symbiosis, and will strive to strengthen cooperation and collaboration with local governments and related organizations.

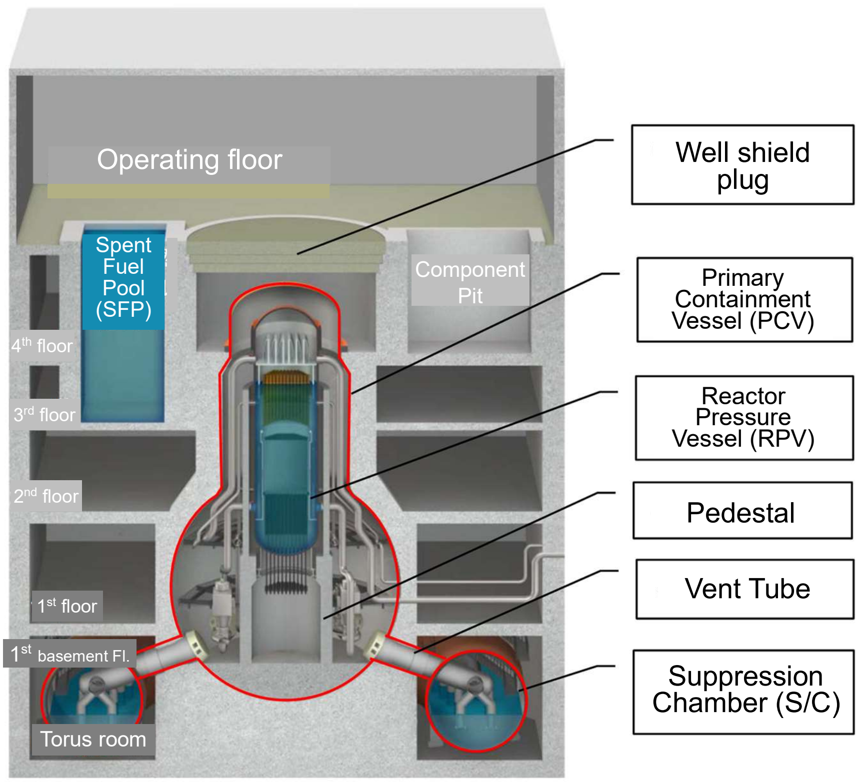
List of Acronyms/Glossaries

Acronym	Official Name
ALARP	As Low As Reasonably Practicable：Risk should be reduced as far as reasonably practicable including risk/benefit criteria or cost while taking feasibility of risk reduction measures into account.
ALARA	As Low As Reasonably Achievable：The principle of radiological protection in which it advocates that all radiation exposure must be maintained as low as reasonably achievable in consideration of social and economic factors.
D/W	Dry Well
DOE	United States Department of Energy
FP	Fission Products
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IRID	International Research Institute for Nuclear Decommissioning
JAEA	Japan Atomic Energy Agency
JAEA/CLADS	JAEA Collaborative Laboratories for Advanced Decommissioning Science
NDA	Nuclear Decommissioning Authority
NDC	Nuclear Development Corporation
NDF	Nuclear Damage Compensation and Decommissioning Facilitation Corporation
NFD	Nippon Nuclear Fuel Development Co., Ltd
OECD/NEA	OECD Nuclear Energy Agency
PCV	Primary Containment Vessel
RPV	Reactor Pressure Vessel
SA	Severe Accident
S/C	Suppression Chamber
SED	Safety and Environmental Detriment
SGTS	Standby Gas Treatment System
TMI-2	Three Mile Island Nuclear Power Plant Unit 2
Penetration X-2	Penetration X-2 of PCV
Penetration X-6	Penetration X-6 of PCV
Center of the World Intelligence project	The project that promotes nuclear science and technology and human resource development gathering wisdom and knowledge
Operating Floor	Operating Floor of the buildings
Commitment	The commitment to the people of Fukushima for achieving both reconstruction and decommissioning
Kashiwazaki-Kariwa	Kashiwazaki-Kariwa Nuclear Power Station

Technical Strategic Plan	Technical Strategic Plan for Decommissioning of the Fukushima Daiichi Nuclear Power Station of Tokyo Electric Power Company Holdings, Inc.
Technical Prospects	Prospects of processing/disposal method and technology related to its safety
International Forum	International Forum on the Decommissioning of the Fukushima Daiichi Nuclear Power Station
Submersible ROV	A remotely operated submersible survey vehicle (Remotely Operated Vehicle)
Mid-and-Long-term Roadmap	Government-developed “Mid-and-long-term Roadmap” toward the decommissioning of TEPCO’s Fukushima Daiichi Nuclear Power Station Units 1 to 4
TEPCO	Tokyo Electric Company Holdings, Inc.
Withdrawal Plan	Withdrawal plan for reserve fund
The Policy of Preparation of Withdrawal Plan	The Policy of preparation of withdrawal plan for reserve fund for decommissioning
Fukushima Daiichi NPS	Fukushima Daiichi Nuclear Power Station of Tokyo Electric Company Holdings, Inc.
Fuel removal from SFP	Fuel removal from spent fuel pool

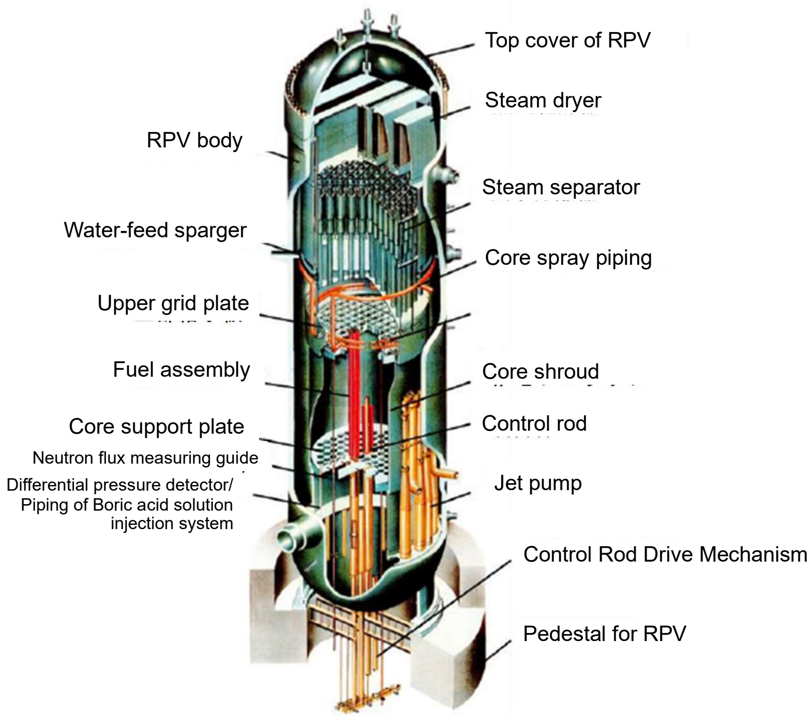
Glossary	Description
Inventory	Amount of radioactive material contained in the risk source (radioactivity, concentration of radioactive material, or toxicity possessed by the radioactive material)
Well plug (Shield plug)	A top cover to screen upper part of Primary Containment Vessel made of concrete (It is the floor face of the top floor of reactor building in operation)
Engineering	Design and other work to apply technical elements to the site
Cask	Special container used for transporting and storing spent fuel
Subdrain	Wells near the building
Sludge generated at decontamination device (waste sludge)	Sludge containing high level of radioactive material generated at the decontamination device (AREVA), which was operated for contaminated water treatment from June to September 2011
Spray curtain	Watering to contain dust and allow it to settle
Sludge	Muddy substance, dirty mud
Slurry	A mix of dirty mud and mineral, etc. in water
Zeolite	Sorbent used to recover radioactive materials such as cesium
Torus room	A room that houses a large donut-shaped suppression chamber that holds water for emergency core cooling system.
Fuel debris	Nuclear fuel material molten and mixed with a part of structure inside reactor and re-solidified due to loss of reactor coolant accident condition
Bioassay	A method for evaluating the types and amounts of radionuclides ingested into body by analyzing samples from the human body, such as excrement

Glossary	Description
Facing (paving)	Covering the ground surface in the power station with asphalt, etc.
Platform	Footing for work installed under RPV inside pedestal
Flanged tank	Bolted assembly tanks
Pedestal	A cylindrical basement that supports a body of reactor
Manipulator	Robot arm to support fuel debris retrieval
Mock-up	A model which is designed and created as close to real thing to possible
Boric acid, sodium chloride	Soluble neutron absorber (boric-acid solution)



（Courtesy of IRID）

Fig. 44 Structural drawing inside Reactor building



（Courtesy of IRID）

Fig. 45 Structural drawing inside Reactor Pressure Vessel (RPV)

List of Attachment

Contents

Attachment1 Revision of the Mid-and-Long-term Roadmap and the earlier published Technical Strategic Plan

[1st Edition of the Mid-and-Long-term Roadmap (December 21, 2011)]

- In response to completion of Step 2 described in "the Roadmap towards Restoration from the Accident at the Fukushima Daiichi NPS" compiled by the government and Tokyo Electric Power Company (TEPCO) after the accident, the necessary measures to be progressed over the mid-and-long-term, including efforts to maintain securely stable conditions, fuel removal from spent fuel pools (SFPs), fuel debris retrieval, etc. were compiled by three parties of TEPCO, Agency for Natural Resources and Energy, and Nuclear and Industrial Safety Agency and conclude at The Government and TEPCO's Mid-to-Long-Term Countermeasure Meeting.
- Basic principles towards implementation of mid-to-long efforts were proposed and targets with time schedules were established by dividing the period up to completion of decommissioning into three parts; the period up to spent fuel removal start (1st period), the period up to fuel debris retrieval start from completion of the 1st period (2nd period) and the period up to completion of decommissioning from completion of the 2nd period (3rd period).

[Mid-and-Long-term Roadmap Revised 1st Edition (July 30, 2012)]

- "Specific plan on the matters to be addressed with priority to enhance mid-and-long-term reliability" developed by TEPCO after completion of Step 2 was reflected and revised targets based on the state of work progress were clearly defined.

[Mid-and-Long-term Roadmap Revised 2nd Edition (June 27, 2013)]

- Revised schedule was studied (multiple plans were proposed) based on the situation of each Unit concerning fuel removal from SFP and fuel debris retrieval, and R&D Plan was reviewed based on the above.

[Technical Strategic Plan 2015 (April 30, 2015)]

- The 1st edition of the Technical Strategic Plan was published to provide a verified technological basis to the Mid-and-Long-term Roadmap from the viewpoint of proper and steady implementation of decommissioning of the Fukushima Daiichi Nuclear Power Station. (NDF was inaugurated on August 18, 2014 in response to reorganization of existing Nuclear Damage Compensation Facilitation Corporation)
- Decommissioning of the Fukushima Daiichi Nuclear Power Station was regarded as "Continuous risk reduction activities to protect human beings and environment from risks caused by radioactive materials generated by the severe accident", and Five Guiding Principles (Safe, Reliable, Efficient, Prompt, Field-oriented) for risk reduction were proposed.
- Concerning the field of fuel debris retrieval, feasible scenarios were studied by regarding the following methods as the ones to be studied selectively; the submersion-top entry method, the partial submersion-top entry method, and the partial submersion-side entry method.
- Concerning the field of waste management, policies for storage, control, etc. were studied from a mid-and-long-term viewpoint based on the basic concept for ensure safety during disposal or for a proper processing method.

[Mid-and-Long-term Roadmap Revised 3rd Edition (June 12, 2015)]

- While much importance was placed on risk reduction, priority-setting for actions was performed so that risks could definitely be reduced in the long term.
- Targets for several years from now were concretely established including policy decision on fuel debris retrieval (two years later from now was targeted), volume reduction of radioactive materials contained in the stagnant water in the buildings by half (FY2018), etc.

[Technical Strategic Plan 2016 (July 13, 2016)]

- In response to the progress state of decommissioning after publication of the Technical Strategic Plan 2015, concrete concepts and methods were developed based on the concept and direction of the efforts

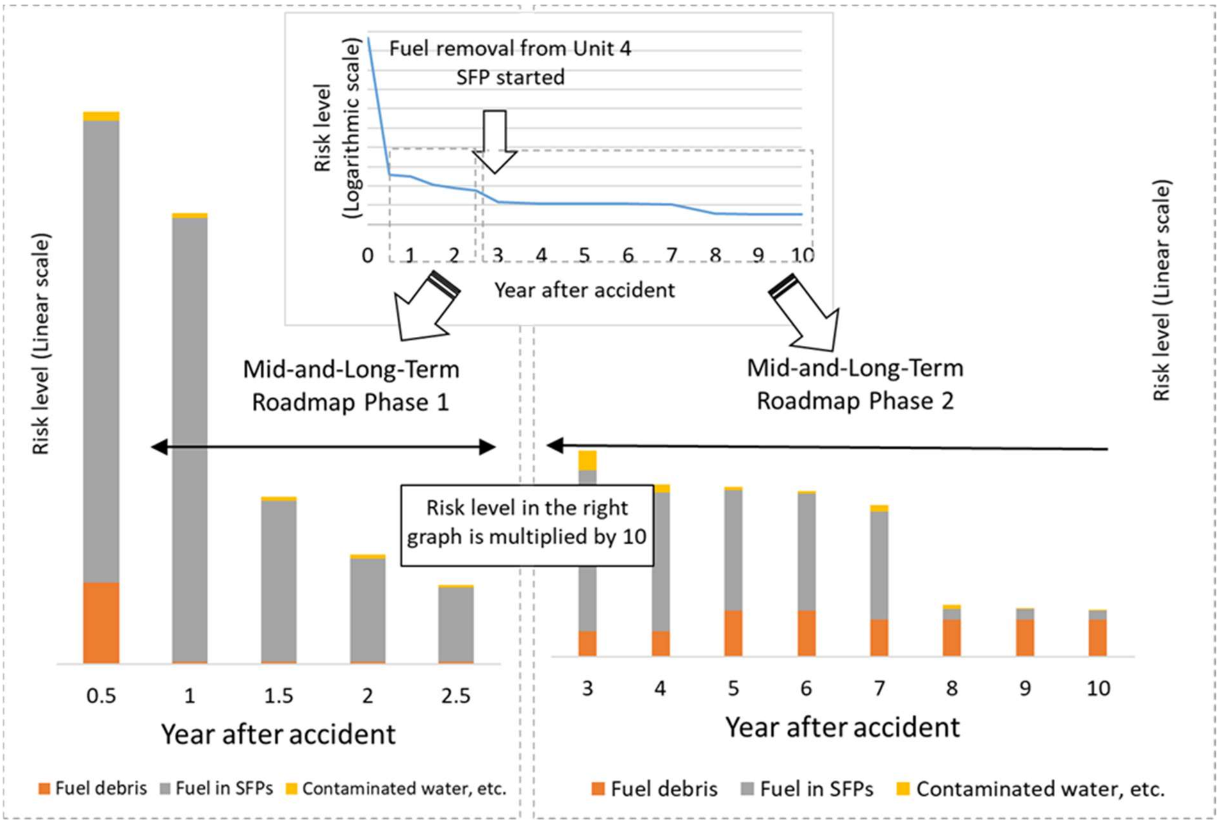
of the Technical Strategic Plan 2015 to achieve the target schedule specified in “Policy decision on fuel debris retrieval for each unit” which is expected to be completed by about summer 2017 defined in the Mid-and-Long-term Roadmap, “Compiling of the basic concept concerning processing/disposal of radioactive waste” which is expected to be complete in FY2017, etc.
[Technical Strategic Plan 2017 (August 31, 2017)] • Feasibility study was conducted on the three priority methods for fuel debris retrieval. Recommendations for determining fuel debris retrieval policy were made and efforts after policy decision including preliminary engineering were recommended as strategic recommendations. • Recommendations were made for compiling the basic concept concerning solid waste processing/disposal.
[Mid-and-Long-term Roadmap Revised 4th Edition (September 26, 2017)] • Policy on fuel debris retrieval and immediate efforts were decided based on NDF technical recommendations. • Basic concepts concerning solid waste processing/disposal were compiled. • Individual work was defined based on the viewpoint of “Optimization of total decommissioning work”.
[Technical Strategic Plan 2018 (October 2, 2018)] • The Plan added contaminated water management and fuel removal from SFP, and presented the direction from mid-to-long-term perspective that overlooks entire efforts of decommissioning of Fukushima Daiichi NPS.
[Technical Strategic Plan 2019 (September 9, 2019)] • The plan presented the strategic recommendation for determining fuel debris retrieval methods for the first implementing unit as well as the direction from mid-to long-term perspective that overlooks entire efforts of decommissioning of Fukushima Daiichi NPS including waste management, etc.
[Mid-and-Long-term Roadmap Revised 5th Edition (December 27, 2019)] • The first implementing unit and the method of fuel debris retrieval were determined. • The methods of fuel removal from SFP in Units 1 and 2 were changed. • TEPCO maintains the current target to suppress the amount of contaminated water generation to about 150m3/day within 2020, in addition, set the new target to less than 100m3/day within 2025.
[Technical Strategic Plan 2020 (October 6, 2020)] • The plan characteristically included providing of the Mid-and-Long-Term Decommissioning Action Plan, identifying of requirements for the study of fuel debris retrieval methods toward further expansion of the scale, clarifying of the concept for ensuring safety in decommissioning operations, and strengthening of management system in response to the growing importance of R&D.

Attachment2 Major risk reduction measures performed to date and future course of action

Change in the risk level over time assessed and expressed by SED for the entire Fukushima Daiichi NPS is shown in Fig. A 2-1. The vertical axis in the top graph in the figure shows the risk level in common logarithmic scale and the horizontal axis shows number of years after the accident.

Although the risk level at the time of zero year after the accident was at high level caused by the fuel in SFP which lost its cooling function and the molten nuclear fuel, over the time of 0.5 years after the accident the risk level has been reduced with a significant decrease in both Hazard Potential and Safety Management, because of implementation of safety measures including cooling function restoration of SFPs, cooling of fuel debris with water injection by core spray system, nitrogen injection, etc. (in 2011) as well as the contribution of inventory and decay heat decrease due to decay of radioactive materials.

The risk level in 0.5 to 2.5 years after the accident is shown in the enlarged graph (the vertical axis is in linear scale) with the breakdown of major risk source (fuel debris, fuel in SFP and contaminated water, and the others) at the bottom left in the figure and the similar graph since 3 years after the accident is given in the bottom right with the risk level multiplied by 10. These graphs demonstrate that a continuous risk reduction has been achieved.



Evaluation of fuel in SFP 8 years after the accident occurred reflects the results of water temperature rise in the testing on SFP cooling shutdown. (For detail, see Fig. 3 in Chapter 2 of main part.)

Fig. A2- 1 Reduction of risks contained in the Fukushima Daiichi NPS

Change in the risk level with further breakdown of major risk sources over time since 0.5 years after the accident is shown in Evaluation of fuel in SFP 8 years after the accident occurred reflects the results of water temperature rise in the testing on SFP cooling shutdown. (For detail, see Fig. 3 in Chapter 2 of main part.)

Fig. A2- 12. With a logarithmic scale, risk sources can be indicated that are too small to be displayed in the linear scale of Evaluation of fuel in SFP 8 years after the accident occurred reflects the results of water temperature rise in the testing on SFP cooling shutdown. (For detail, see Fig. 3 in Chapter 2 of main part.)

Fig. A2- 1. Fuel in the Common Spent Fuel Storage Pool and the Dry Cask Temporary Custody Facility are not shown which stay in the region of sufficiently stable management. The "stagnant water in buildings + zeolite sandbags" shown in Fig. A2- 2 was assessed based on the information on the stagnant water in buildings for the period of 0-8 years after the accident. However, since 9 years after the accident, the condition of zeolite-containing sandbags placed in the basement of the process main building and the high-temperature incinerator building has become clear, and this information was incorporated into the assessment.

Among the major risk sources, fuel debris, fuel in SFPs, and the stagnant water in buildings and zeolite-containing sandbags have relatively high risk levels. Although, in recent years, the treatment of the stagnant water in buildings has progressed and the risk level of the "stagnant water in buildings + zeolite sandbags" has been on a declining trend, attention should be paid to zeolite sandbags laid with a high dose because they may hinder future decommissioning work. In addition, as for the water stored in tanks (flanged tank and welded tank), the risk of leakage and consequently the risk level has been greatly reduced as a whole because the treatment of the water stored in the flanged tanks, which have a higher risk of leakage than welded tanks, has proceeded well.

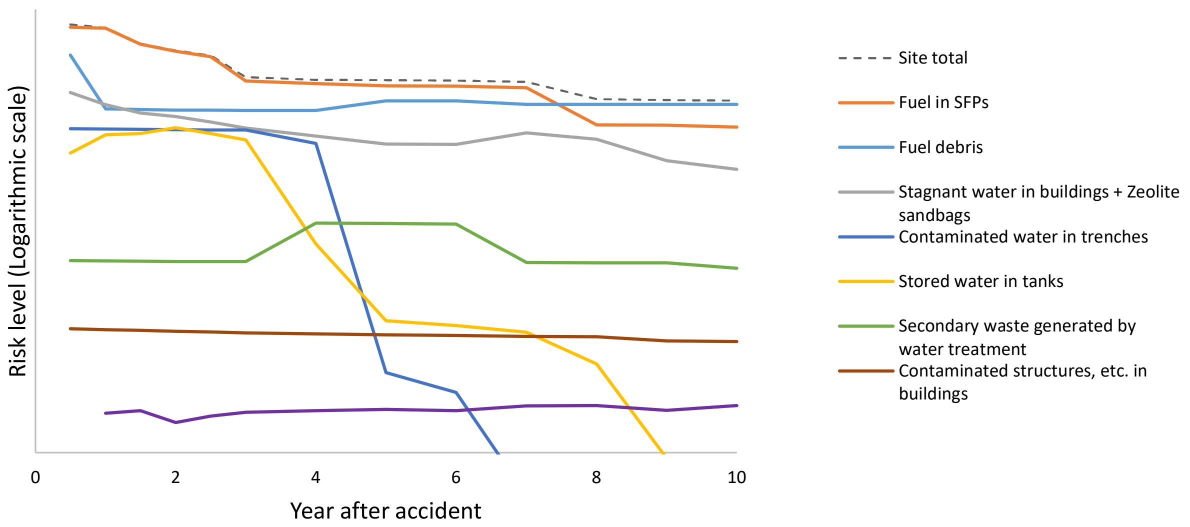


Fig. A2- 2 Change in the risk level for each major risk source

(1) Fuel in SFPs

From one year after the accident, rubble was removed and a cover for fuel removal was installed at Unit 4 in preparation for fuel removal, thereby enhancing the functions of reducing the risk of fuel damage by rubble in SFP and controlling the dispersion of damaged fuel. Further, 2.5 years after the accident, fuel removal was started and the fuel was transferred into the Common Spent Fuel Storage Pool with low Safety Management, and the risk level was lowered (completed in 2014)[91].

Although the effect of risk level reduction was observed due to the decrease in Safety Management through the diffusion control function of the building cover at Unit 1 (installed in 2011), this effect has been currently lost because the building cover was removed (in 2015) in preparation for removal of fuel in SFP[92]. In order to prevent dust scattering during rubble removal, a large cover will be installed by FY2023, and fuel removal from SFP is planned to start in FY2027 to FY2028.[93].

For Unit 2, a gantry for fuel removal will be installed on the south side of the reactor building, and the removal of the fuel in SFP is scheduled to start in FY2024 to FY2026[93].

In Unit 3, a cover for fuel removal was installed in 2018 after rubble removal was performed in preparation for fuel removal from SFP, then fuel removal from SFP was started from April 2019. After that, transfer to the Common Spent Fuel Storage Pool was completed in February 2021[94].

In case cooling fuel in SFPs is stopped, the pool water temperature may rise and the pool water level may lower due to decay heat. In and after the 8th year after the accident, as a result of incorporating the observation that the rise in water temperature after cooling shutdown of SFPs was slower than expected, the risk level of fuel in SFPs is lower than previously estimated, because the time margin before the risk of water level lowering becomes apparent increases.

(2) Fuel debris

Although fuel debris was at a high risk level just after the accident because it was at molten state, and in addition, radioactive materials were released, the risk level was reduced, not only by decay of the radioactive materials, but also by reduction of Hazard Potential and Safety Management because of restoration and strengthening of cooling function.

As described in (1), the diffusion control function of the building cover of Unit 1 reduced the risk associated with the dispersion of fuel debris, and lowered the risk level due to the decrease in Safety Management; however, this effect is currently lost.

(3) Stagnant water in buildings + Zeolite sandbags

Although stagnant water in buildings is generated by cooling of fuel debris and immersion of groundwater into the buildings, etc., the risk level has been lowered due to the start of operation of cesium sorption apparatus (KURION) and Second cesium sorption apparatus (SARRY), the effect of subdrains and land-side impermeable walls, water drainage in condensers, and the start of the

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91 Decommissioning project, Status of the decommissioning work, Fuel removal work of Unit 4, (Website), Tokyo Electric Power Company Holdings, Inc.

92 The 57th Study Group on Monitoring and Assessment of Specified Nuclear Facilities, Reference 7 “State of progress of Unit 1 of Fukushima Daiichi NPS and rubble removal on the north side of the operating floor”, Tokyo Electric Power Company Holdings, Inc.

93 Mid-and-Long-term Decommissioning Action Plan (March 25, 2021), Tokyo Electric Power Company Holdings, Inc.

94 Decommissioning project, Status of the decommissioning work, Fuel removal from spent fuel pool in Unit 3, (Website), Tokyo Electric Power Company Holdings, Inc.

operation of Third cesium sorption apparatus (SARRY-II). This stagnant water treatment in the buildings so far significantly contributes to risk level reduction of the total site following contribution by fuel removal in SFP.

(4) Contaminated water in trenches

Although the contaminated water of high concentration has been stagnated in the seawater pipe trenches in Units 2 to 4 since immediately after the accident, the trenches were blocked and the treatment of the stagnant water has been completed (in 2015)[95]. With regard to the seawater pipe trench of Unit 1, the concentration of which is lower than that of Units 2 to 4, purification of the stagnant water is under consideration [96]。

(5) Stored water in tanks

There are several types of stored water in the tank with different radioactive material concentrations depending on the stage of purification treatment. First of all, the strontium treated water generated from the purification process of the water in the buildings by KURION, SARRY and SARRY Ⅱ is stored as welded tank water. After that, the risk level is further reduced by multi-radionuclide removal equipment (ALPS), etc., and the water is stored in welded tanks as ALPS-treated water, etc. (ALPS-treated water and water under treatment). For the concentrated waste liquid generated from the evaporation-enrichment system, which operated only for a short period immediately after the accident, the precipitated slurry with a high concentration of radioactive materials (concentrated waste liquid slurry) was separated, and the remaining liquid (concentrated waste liquid) is transferred to welded tanks, thereby reducing the leakage risk and lowering the risk level.

The treatment of the concentrated salt water generated from the treatment with KURION before ALPS came into operation was completed in 2015 through the operation of ALPS and the advanced multi-nuclide removal equipment (Advanced ALPS)[97]。

Risk level of these stored water in the tanks are also lowered by raising and duplexing the weir (for the existing tanks completed in 2014), transferring from flanged tanks to welded tanks, and treating the Sr-treated water remaining at the bottom of the flanged tanks (in 2019), and treating ALPS-treated water (in 2020). The remaining concentrated salt water at the bottom of flanged tanks is being treated for dismantling the tanks.

(6) Secondary waste generated by water treatment

Many radioactive materials have moved from contaminated water to secondary waste through water treatment. What has been generated includes the sludge from decontamination device, the waste sorption vessels by operation of KURION and SARRY (in 2011) and by the SARRY-II (in 2019), ALPS slurry by operation of ALPS (in 2013), the waste sorption vessels by the advanced ALPS (in 2014), waste sorption vessels by the mobile-type treatment system that treated seawater

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95 Decommissioning project, Status of the decommissioning work, Removal of contaminated water in seawater pipe trenches, (Website), Tokyo Electric Power Company Holdings, Inc.

96 The 91st Study Group on Monitoring and Assessment of Specified Nuclear Facilities “Reference 1 ”Work Schedule for Instruction Items to be Considered Based on the Measures for Mid-and-long-term Risk Reduction at TEPCO’s Fukushima Daiichi NPS (Risk Map), (March 2021 Edition)”, Tokyo Electric Power Company Holdings, Inc.

97 Decommissioning project, Status of the decommissioning work, Purification of contaminated water, (Website), Tokyo Electric Power Company Holdings, Inc.

pipe trenches, etc. Although the sludge from decontamination device greatly contributes to the risk level, sludge is not newly generated at present, and thus, the risk level of the total secondary waste generated by water treatment is not on an increasing trend. As a tsunami countermeasure, the decontamination sludge stored in the main process building (T.P. 8.5m) will be extracted (planned for FY2023), placed in a storage container, and transferred to the elevated area (T.P. 33.5m)[98].

Although the concentrated waste liquid slurry separated from the concentrated waste liquid was stored in horizontal welded tanks without the weir and placed on the ground without the base, its risk level has been lowered due to the approach to safety taken by installing the reinforced-concrete base and the weir.

(7) Contaminated structures, etc., in the buildings

There is no significant change at the present moment in the risk level of contaminated structures, etc. in the buildings comprised of structures, piping, components, etc. (shield plug, piping of emergency gas processing system and the like) in the reactor buildings, PCVs or RPVs that are contaminated by dispersed radioactive materials caused by the accident.

(8) Rubble, etc.

Rubbles, etc. as solid waste are stored under a variety of conditions such as in solid waste storage, in temporary waste storage and by outdoor accumulation. Each has different Safety Management, and the rubbles stored in outdoor sheet covered storage and outdoor accumulation are of the highest risk level. In the past, the facilities with better management condition have been enhanced by soil covered temporary storage facilities (in 2012), felled tree temporary storage pool (in 2013), expansion of solid waste storage facilities (in 2018), etc. In addition, the rubble from temporary storage facilities was transferred to the better-controlled solid waste storage facility (in 2020). Furthermore, outdoor temporary storage is planned to be discontinued by the end of FY 2028 by increasing incinerators, volume reduction installations and solid waste storages, etc., in accordance with the Solid Waste Storage Management Plan[99].

[98] Safety Monitoring Council for the decommissioning of Fukushima Daiichi NPS (March 8, 2021), Reference 5-1

[99] The Solid Waste Storage Management Plan at the Fukushima Daiichi NPS (July 2021 Edition), Tokyo Electric Power Company Holdings, Inc.

Attachment3 Overview of SED indicator

Risk analysis targeting various risk sources, which have diverse characteristics and exist all over the site, was conducted in reference to the SED indicator[100] developed by the NDA. The SED indicator is an important factor to decide priority to implement risk reduction measures. It was partially modified (refer to the following pages) so that unique characteristics of the Fukushima Daiichi NPS could be easily reflected when it was applied to the Fukushima Daiichi NPS. Overview of the SED indicator and the modified part to be applied to the Fukushima Daiichi NPS are described below.

The SED indicator is expressed by the following formula. The first formula is the one widely used for waste assessment and the second is for contaminated soil assessment. In each formula, the first term is referred as to “Hazard Potential” and the second as “Safety Management” of risk sources.

$$SED = (RHP + CHP) \times (FD \times WUD)^4$$

or

$$SED = (RHP + CHP) \times (SSR \times BER \times CU)^4$$

Each indicator is explained below. Although CHP stands for “Hazard Potential” of the chemical substance, details are not given here as it is not used in this section.

(1) Hazard Potential

Radiological Hazard Potential (RHP) is an indicator representing the potential impact of radioactive materials and represents the impact to the public by the following formula when the total amount of radioactive materials is released.

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

Inventory is defined as shown below by Radioactivity of risk sources and the Specific Toxic Potential (STP) and corresponds to the effective radiation dose[101]. The STP is defined as the volume of water required to dilute 1TBq of radioactive materials and corresponds to the radiation dose coefficient. Ingestion of a certain amount of such diluted water throughout the year will result in a radiation exposure dose of 1mSv. The SED indicator conservatively uses the larger radiation dose coefficient between ingestion and inhalation.

$$Inventory(m^3) = Radioactivity(TBq) \times STP(m^3/TBq)$$

[100] NDA Prioritization – Calculation of Safety and Environmental Detriment score, EPGR02 Rev.6, April 2011.
[101] Instruction for the calculation of the Radiological Hazard Potential, EGPR02-WI01 Rev.3, March 2010.

Form Factor (FF), as shown in Table A3-1, is an indicator representing how much radioactive material is actually released depending on material form, such as gas, liquid, solid, etc. The indicator is set assuming that 100% of radioactive material is released in the case of gas and liquid when containment function is totally lost and that 10% of radioactive material is released in the case of powder based on the measurement data. Because of no clear basis, the indicator in case of solid is set to a sufficiently small value assuming that the solid materials are less easily released.

In Table A3-1, several expected forms, especially for fuel debris, are added to the definition used by the NDA. The scores for the form of No.4 and No.5 are newly established.

Control Factor (CF), as shown in Table A3-2, is an indicator representing time allowance available before restoration when safety functions maintaining current stable state are lost. CF is taking into account exothermicity, corrosivity, flammability, hydrogen generation, reactivity with air or water, criticality, etc. which are typical characteristics of risk sources. CF is the same as the one defined by the NDA.

(2) Safety Management – FD and WUD

Facility Descriptor (FD) is an indicator representing whether containment function of the facility is sufficient or not. Risk sources are ranked by score based on a combination of the factors including integrity of the facility, redundancy of containment function, safety measure condition, etc.

Waste Uncertainty Descriptor (WUD) is an indicator representing whether any impact is generated or not when the risk source removal is delayed. Risk sources are ranked by score based on a combination of the factors including degradation or activity of the risk source, packaging state, monitoring condition, etc.

As these indicators are difficult to be applied to the Fukushima Daiichi NPS if they are used as defined by the NDA, they are re-defined as shown in Table A3-3 and Table A3-4 respectively.

(3) Safety Management - SSR, BER and CU

The definition of SSR, BER and CU used for Safety Management assessment for contaminated soil is the same as the one defined by the NDA and each score is shown in Table A3-5.

Speed to Significant Risk (SSR) is an indicator concerning the time until the public is affected through such as distance to the site boundary, groundwater flow conditions, etc. and to assess urgency of taking measures.

Benefit of Early Remediation (BER) is an indicator to assess benefits obtained from early implementation of measures against risks.

Characterization Uncertainty (CU) is an indicator to assess reliability or uncertainty in the risk assessment model.

Table A3-1 Definition and score of FF

No.	Form	FF
1	Gas, liquid, watery sludge* and aggregated particles*	1
2	Other sludge	1/10 = 0.1
3	Powder and removable contaminants (surface contamination, etc.)*	1/10 = 0.1
4	Adhesive* or penetrating contaminants (surface penetrating contamination)*	1/100 = 0.01
5	Fragile and easily decomposable solid (porous MCCI (Molten Core Concrete Interaction), etc.)*	1/10,000 = 1E-4
6	Discrete solid (transportable size and weight by human power such as pellets, etc.)	1/100,000 = 1E-5
7	Large monolithic solid, activated component	1/1,000,000 = 1E-6

* : Form which is added to the NDA definition to enhance applicability to the case of the Fukushima Daiichi NPS

Table A3-2 Definition and score of CF

No.	Time allowance available before any risk is realized	CF
1	Hours	1
2	Days	10
3	Weeks	100
4	Months	1,000
5	Years	10,000
6	Decades	100,000

Table A3-3 Criteria and score of FD

Category	Criteria (NDA definition is modified to enhance applicability to the case of the Fukushima Daiichi NPS)	NDF Score
1	No component for diffusion control function exists. Therefore, no assessment for containment function is available.	100
2	"Safety assessment criteria*2" are not satisfied at "the time of assessment*1" caused by the accident effects, etc. The component for diffusion control function is single.	91
3	"Safety assessment criteria" are not satisfied at "the time of assessment" caused by the accident effects, etc. The component for diffusion control function is multiple.	74
4	"Safety assessment criteria" are not satisfied until "the time of work (such as transfer, treatment, recovery, etc.) *3" for the risk source contained in the component for diffusion control function. The component or diffusion control function satisfying "safety assessment criteria" exists at "the time of assessment".	52

5	Integrity of diffusion control function has been assessed and “safety assessment criteria” are satisfied until “the time of work (such as transfer, treatment, recovery, etc.)” for the risk source. Frequency of occurrence of “contingency*4” is high, and when contingency occurs countermeasures preventing diffusion of the risk source contained in the component are not sufficient. The component for diffusion control function is single.	29
6	“Safety assessment criteria” is satisfied until “the time of work (such as transfer, treatment, recovery, etc.)” for the risk source. Frequency of occurrence of “contingency” is high, and countermeasures preventing diffusion of the risk source contained in the component are not sufficient. The component for diffusion control function is multiple.	15
7	“Safety assessment criteria” are satisfied until “the time of work (such as transfer, treatment, recovery, etc.)” for the risk source. Facilities dissatisfying “safety assessment criteria” exist in the surrounding area, and the potentiality is high to make (receive) the diffusion impact*5 of the risk source to (from) these adjacent facilities. The component for diffusion control function is single.	8
8	“Safety assessment criteria” are satisfied until “the time of work (such as transfer, treatment, recovery, etc.)” for the risk source. The potentiality is high to make (receive) the diffusion impact of the risk source to (from) these adjacent facilities. The component for diffusion control function is multiple.	5
9	“Safety assessment criteria” are satisfied until “the time of work (such as transfer, treatment, recovery, etc.)” for the risk source. The potentiality is low to make (receive) the diffusion impact of the risk source to (from) these adjacent facilities. The component for diffusion control function is single.	3
10	“Safety assessment criteria” are satisfied until “the time of work (such as transfer, treatment, collection, etc.)” for the risk source. The potentiality is low to make (receive) the diffusion impact of the risk source to (from) these adjacent facilities. The component for diffusion control function is multiple.	2

*1. This refers to “at the time” of study on SED score, i.e., “at the present time” of assessment.
*2. “Safety assessment criteria” described in this sentence refer to “the matters for which measures should be taken” or “securing of diffusion control function within the scope of design basis event”.
*3. This refers to the time of “recovery” of the risk source for disposition and carrying out for which SED score shall be studied.
*4. External events (natural disasters, etc.) are postulated as contingencies.
*5. The potentiality of diffusion of the risk source exists to (from) adjacent facilities when facilities receive external impact caused by contingencies or impact caused by any events (fire, etc.), etc.

Table A3-4 Criteria and score of WUD

Category	Criteria (NDA definition is modified to enhance applicability to the case of the Fukushima Daiichi NPS)	NDF Score
1	The material is fuel (which contains fissile material) and active*1. Necessary information (existent amount, existent location, radioactivity, etc.) for work including treatment, recovery, etc. is insufficient (cannot	100

	be confirmed or estimated), and control and surveillance with monitoring, etc. are unavailable. Handling is impracticable for the current form or condition because of reasons where the form is not proper for handling, or that it is not stored in a special container.	
2	The material is fuel and active (which has fissile properties). Necessary information for work including treatment, recovery, etc. is insufficient, and control and surveillance are unavailable. Handling is practicable for the current form or condition because of reasons where the form is proper for handling or that it is stored in a special container.	90
3	Although the material is active, it is not fuel (but waste). Necessary information for work including treatment, recovery, etc. is insufficient.	74
4	The material is fuel and active (which has fissile properties). Necessary information for work including treatment, recovery, etc. is obtained (can be confirmed or estimated), and control and surveillance with monitoring, etc. are available. Handling is impracticable for the current form or condition.	50
5	The material is fuel and active (which has fissile properties). Necessary information for work including treatment, recovery, etc. is obtained, and control and surveillance are available. Handling is practicable for the current form or condition.	30
6	Although the material is active, it is not fuel (but waste). Necessary information for work including treatment, recovery, etc.	17
7	Although the material is inactive*2, it has physical or geometrical instability. Handling is impracticable for the current form or condition.	9
8	Although the material is inactive, it has physical or geometrical instability. Handling is practicable for the current form or condition.	5
9	The material is inactive and has no physical or geometrical instability or has sufficiently low level of instability. Handling is impracticable for the current form or condition.	3
10	The material is inactive and has no physical or geometrical instability or has sufficiently low level of instability. Handling is practicable for the current form or condition.	2

*1 “Active” refers to possession of activity defined by CF at such a significant level as that activity affects control and work.
*2 “Inactive” refers to non-possession of activity or possession of sufficiently low level of activity.

Table A3-5 Definition and score of SSR, BER and CU

<table>
<tr><th>Indicator</th><th>Score</th><th colspan="2">Criteria</th></tr>
<tr><td rowspan="3">SSR</td><td>25</td><td colspan="2">Risks may be realized within 5 years.</td></tr>
<tr><td>5</td><td colspan="2">Risks may be realized within 40 years.</td></tr>
<tr><td>1</td><td colspan="2">40 years or over (There is very little possibility that risks are realized.)</td></tr>
<tr><td rowspan="3">BER</td><td>20</td><td colspan="2">Implementation of measures can reduce risks by 2 or more orders of magnitude or can facilitate control stepwise.</td></tr>
<tr><td>4</td><td colspan="2">Implementation of measures can reduce risks by 1 or more order of magnitude, but cannot facilitate control.</td></tr>
<tr><td>1</td><td colspan="2">Implementation of measures can only bring negligible risk reduction effects, and cannot facilitate control, either.</td></tr>
<tr><td rowspan="3">CU</td><td>20</td><td>(1)+(2)= 5 to 6 points</td><td rowspan="3">(1) Assessment for the present state
1 point: Major nuclear types and diffusion pathways are monitored.
2 points: Monitored, but insufficient data for construction of assessment model
3 points: Not monitored
(2) Assessment on future prediction
1 point: Sufficient site characteristics are obtained for construction of assessment model.
2 points: Major characteristics representing the site are obtained.
3 points: There is no model usable for future prediction</td></tr>
<tr><td>4</td><td>(1)+(2)= 3 to 4 points</td></tr>
<tr><td>1</td><td>(1)+(2)= 2 points</td></tr>
</table>

Attachment4 Risk sources that are not explicitly addressed in the major risk sources

Major risk sources are listed in the Table 1 in Chapter 2 of the body part. Looking ahead to the decommissioning of the entire Fukushima Daiichi NPS, it is necessary to focus on risk sources that are not explicitly addressed in the major risk sources. Table A4-1 focuses on waste existed before the accident and radioactive materials with low concentration diffused by the accident, and is summarized with reference to Measures for Mid- term Risk Reduction at TEPCO's Fukushima Daiichi NPS (Risk Map)" provided by the NRA.[102].

Table A4- 1 Risk sources that are not explicitly addressed in the major risk sources (1/2)

Issue	Risk source	Descriptions
Liquid radioactive materials	Underground water tank	The residual water in all the underground water tanks were completely recovered [103]. Dismantling and removal policies are under consideration.
	Accumulated water on site	Extracted by the comprehensive risk inspection performed in 2015[104]. Since then, the concentration of radioactive materials and volume of water are being checked accordingly [105]。
	Drainage	In drainage A, Cs-137: lowered to ND ~ 23 Bq/L [106]. In drainage K, the contamination source on the roof of the Unit 2 Reactor building was removed, and the contamination level fell to 67 Bq/L. In addition, purification materials were installed[107], and measures such as operation of discriminating-type PSF monitors were taken[108].
	Sludge on the floor in the buildings	The floor surface of turbine buildings and radioactive waste disposal buildings of Units 1 to 4, waste process building and Unit 4 reactor building remain exposed, and radioactivity of sludge after the exposure was 1.9×10^{13}Bq[109]. For reactor buildings of Units 1 to 3, process main building and high temperature incinerator building, stagnant water processing is underway.
Spent fuel	Spent control rods, etc.	Spent control rods, etc.: 24,030. Shroud fragments, etc.: 193 m^3[110]. The major nuclide is Co-60.

102 NRA, Measures for Mid-term Risk Reduction at TEPCO's Fukushima Daiichi NPS (Risk Map)", (March 2021 Edition)

103 The 44th Meeting of the Secretariat of the Team for Countermeasures for Decommissioning and Contaminated Water Treatment "Reference 3-6: On-site Monitoring Status (Conditions of Water Discharge Channels in Units 1 to 3 and Underground Water Storage Tanks)"

104 Comprehensive Risk Inspection of Fukushima Daiichi NPS that impacts outside the Site Boundary - Review Results - (April 28, 2015) Tokyo Electric Power Co., Inc.

105 The 88th Meeting of the Secretariat of the Team for Countermeasures for Decommissioning and Contaminated Water Treatment "Reference 1: Status of contaminated water and other accumulated water on the premises (as of March 18, 2021)"

106 The 32nd Study Group on Monitoring and Assessment of Specified Nuclear Facilities "Reference 2: Status of measures for reducing the concentration of waste water in drainage K"

107 The 63rd Study Group on Monitoring and Assessment of Specified Nuclear Facilities "Reference 2: Measures for rainwater inflow control (Progress status of installing purification materials for rainwater drainage in turbine buildings)"

108 The 74th Meeting of the Secretariat of the Team for Countermeasures for Decommissioning and Contaminated Water Treatment "Reference 3-6: Starting of operation of PSF monitor in the drainage K"

109 The 87th Study Group on Monitoring and Assessment of Specified Nuclear Facilities "Reference 3-5 : Progress of the treatment of stagnant water in buildings, etc."

110 NRA, Material of interview with the licensee "Solid Waste at Fukushima Daiichi NPS" September 21, 2018, Tokyo Electric Power Company Holdings, Inc.

	In-pool water	Salt removal in Units 2 to 4 was completed in 2013.
	Fuel in Units 5/6 SFP	Unit 5 : 1,374, Unit 6 : 1,456[111]
Solid radioactive materials	Waste before the earthquake	185,816 drums are stored[112]. The major nuclide is Co-60.
	Contaminated soil	As a result of the topsoil analysis, more than half of the samples are in excess of the designated standards (8,000 Bq/kg) based on the Act on Special Measures Concerning the Handling of Environmental Pollution by Radioactive Materials[116]
	Rubbles around buildings	Dismantling of rubbles scattered on the roof floor of the buildings due to hydrogen explosions is now in operation and planned. The amount of rubbles has not been confirmed.
Counter-measures to external events, etc.	Exhaust stack	Exhaust stack of Units 1/2 : dismantlement work was carried out since August 2019, and the upper part of 61 m out of the total height of 120 m was divided into 23 blocks in total for dismantling. On May 1, 2020, a lid was installed on a barrel 59 meters above the ground to prevent rainwater inflow, and dismantling was completed[113]. Exhaust stack of Units 3/4 : Measured 3mSv/h at the base[114].
	Megafloat	The work of bottoming and internal filling was completed[115]. Revetment maintenance and embankment work are underway.
	Dust in operating floor	Below the target value of release control (1×10^7 Bq/h). Gradually declining[116].
	Rainwater inflow into buildings	Rubble on the roof was removed and waterproofing was newly provided. Purification materials were installed in the gutters. Check valves were installed in the drain pipes. The roof drain was repaired and closed[117]. Facing of the Elevation T.P.2m, T.P.6m and T.P.8.5m was completed[118].
Important issues to progress decommissioning	Radiation source on the 3rd and 4th floors of Unit 3 R/B	On the 3rd floor, beams at several locations were damaged. A maximum of 45 mSv/h was measured. On the 4th floor, 104 mSv/h was observed[119].

[111] The 89th Meeting of the Secretariat of the Team for Countermeasures for Decommissioning and Contaminated Water Treatment "Reference 4-2 : Storage status of spent fuel , etc."

[112] NRA, Material of Interview with Licensee "Restoration Status of Exhaust Radiation Monitor at Auxiliary Common Facilities for Common Spent Fuel Storage Pool and Ventilation & Air Conditioning System at Fuel Storage Area of Fukushima Daiichi NPS" September 21, 2018, Tokyo Electric Power Company Holdings, Inc.

[113] "Completion of Dismantling of Exhaust Stack of Units 1/2 at Fukushima Daiichi NPS" (May 1, 2020), Tokyo Electric Power Company Holdings, Inc.

[114] The 19th Committee on Accident Analysis of the Fukushima Daiichi NPS "Reference 4 : Interim report on the investigation and analysis of the accident at TEPCO's Fukushima Daiichi NPS (proposal)"」

[115] The 81st Secretariat Meeting of the Team for Countermeasures for Decommissioning and Contaminated Water Treatment, "Reference 3-1:Progress Status of Megafloat Project at Fukushima Daiichi NPS to Reduce Tsunami Risks

[116] Daily Analysis Results of Radioactive Materials at Fukushima Daiichi NPS, (Website), Tokyo Electric Power Company Holdings, Inc.

[117] The 78th Secretariat Meeting of the Team for Countermeasures for Decommissioning and Contaminated Water Treatment, "Reference 3-1: Progress Status in Rooftop Rainwater Measures"

[118] The 84th Study Group on Monitoring and Assessment of Specified Nuclear Facilities "Reference 1-3: Progress and status of studies on measures to control the generation of contaminated water, amount of groundwater and rainwater inflow per building"

[119] The 14th Study Committee on Accident Analysis of the Fukushima Daiichi NPS "Reference 3: Progress of on-site Investigation"

Attachment5 Change in risk over time

Overview of the concept of risk management in the UK is shown in Fig. A5-1. Even if the current risk level is plotted in the white region of the graph, it does not mean such risk level can always be accepted over time, but the time will come when such risk level cannot be accepted in the future (yellow region). In addition, as time passes, the risk level may increase caused by degradation of facilities and risk sources (represented by the dotted line). On the other hand, when risk reduction measures are taken, the risk level can be reduced so that it may not reach the unacceptable region (red region) with careful preparation and thorough management, although it may be temporarily increased. In this way the risk level shall be targeted to be sufficiently reduced (represented by the solid line) so that it may not reach into the unacceptable or intolerable region.

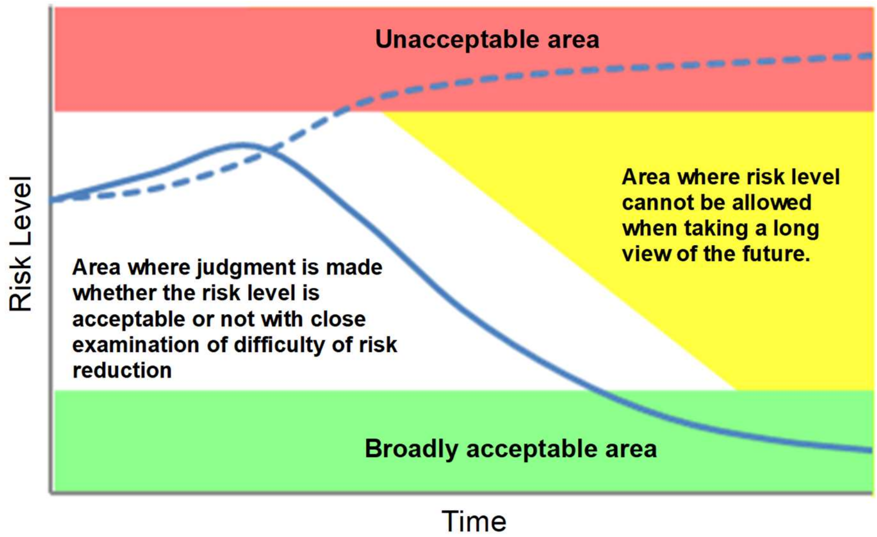


Fig. A5-1 Change in risk over time[120]

[120] 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” 1st International Forum on the Decommissioning of the Fukushima Daiichi Nuclear Power Station (April 2016)..

Attachment6 Coverage of fuel debris retrieval

In the Mid-and-Long-term Roadmap issued on December 21, 2011, fuel debris is described as "material in which fuel and its cladding tubes, etc. have melted and re-solidified", namely, fuel debris is "fuel assembly, control rod and structures inside reactor have melted and solidified together" according to the report by IAEA[121],[122].

The condition inside PCV is as shown in Fig. A6-11, as the comprehensive estimations from the inside investigation of reactor, the past accidents including TMI-2 or ChNPP-4, and the result of the simulation test. It does not show any of specific unit. For more detail, as shown in the Fig A6-1, fuel debris can be classified by form such as damaged pellets, debris, crust, etc.

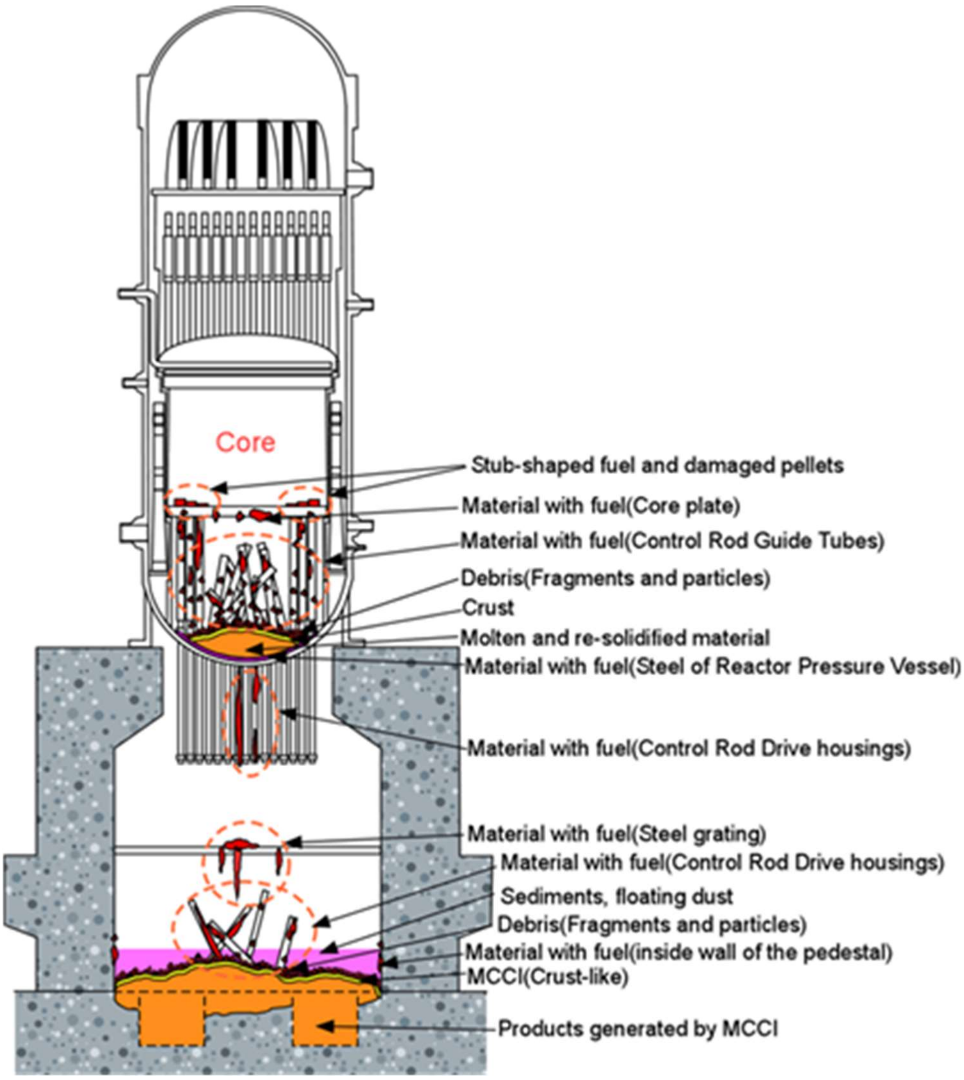


Fig. A6-1 Estimated inside of the PCV of the Fukushima Daiichi NPS

[121] International Atomic Energy Agency 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)

[122] Managing the Unexpected in Decommissioning, IAEA Nuclear Energy Series No. NW-T-2.8, Vienna (2016)

Since nuclear fuel material requires considerations to prevent criticality, it is rational that objects which exist inside PCV should be broadly sorted into two from the viewpoint of retrieval, containment, transfer and storage. The one includes nuclear fuel material and the others. The one that does not include nuclear fuel material is to be treated as a radioactive waste in case radioactive cesium or cobalt are contained or adhered.

Based on this, an example of fuel debris concept as a retrieval target of fuel debris is as shown in Fig. A6-22. Objects generated by core damage have been classified depending on necessity of criticality measures and the content of fuel, in spite that a lot of names are used according to the content of fuel component or form in appearance.

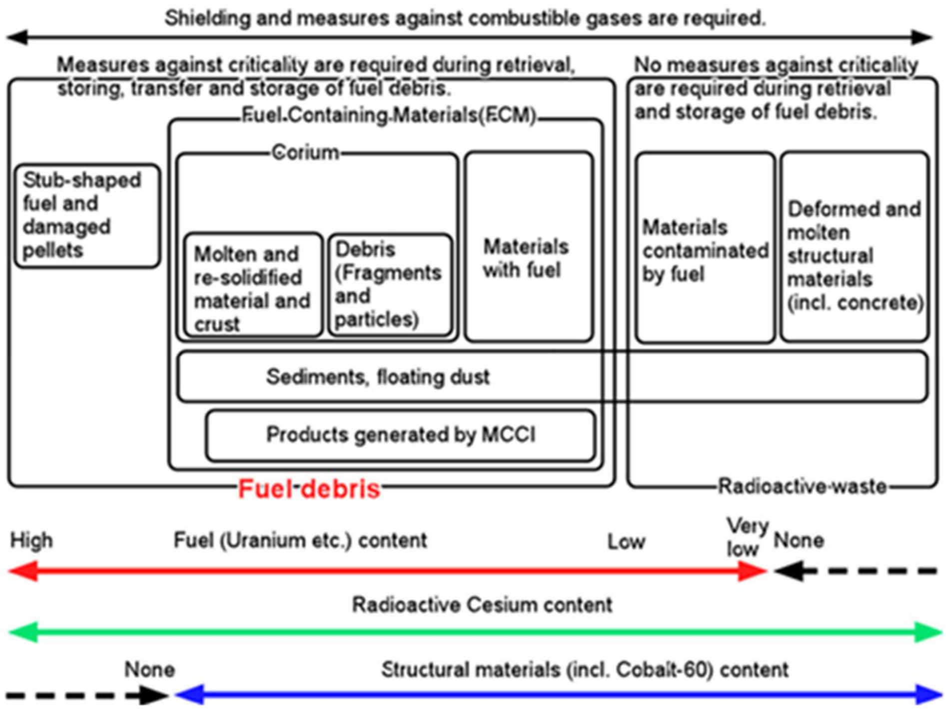


Fig. A6-2 An example of organized concept of fuel debris as fuel debris retrieval target at the Fukushima Daiichi NPS

【Glossaries and Terms】

FCM : Fuel Containing Materials. It refers broadly that molten fuel component comes to solidify in conjunction with structural materials. It is also called lava-like FCM due to its appearance.

Corium : A substance that mainly fuel assembly and component of control rod as core component have molten and solidified.

Crust : A hard outer layer or shell on the surface. When molten fuel is solidified, it may become a hard solid state of shell because of higher cooling speed on the surface layer.

MCCI product : A product generated by Molten Core Concrete Interaction, that includes calcium, silicone, etc. which are concrete component.

Material with fuel : A substance that molten fuel has adhered to material that does not include fuel component originally, like CRD housing, grating and s, then solidified. It is possible to confirm fuel adhesion state by sight.

Material contaminated by fuel : A substance that adhering molten fuel cannot be confirmed by sight, but fuel component can be detected with α ray detector. It is impossible to locate fuel component other than using by electron microscope because particle of adhered fuel component is extremely small and whit.

Attachment7 Concept of Safeguards

Safeguards are verification activities undertaken to ensure that nuclear materials are used only for peaceful purposes and not diverted to nuclear weapons, etc.

Japan has concluded the Safeguards Agreement between Japan and the International Atomic Energy Agency (IAEA) pursuant to the “Treaty on the Non-Proliferation of Nuclear Weapons” (NPT). In accordance with this Agreement, relevant domestic laws (the Law for the Regulations of Nuclear Source Materials, Nuclear Fuel Materials and Reactors (the Nuclear Reactor Regulation Law) and others) have been developed to establish a domestic safeguards system and accept the IAEA’s safeguards.

Specifically, the Safeguards Office of the Nuclear Regulatory Agency (JSGO) has confirmed that all nuclear materials in Japan have not been diverted to nuclear weapons, etc., by conducting the following activities, and the IAEA has acknowledged this authorization through inspections, etc.

(1) Material accountancy by nuclear operators

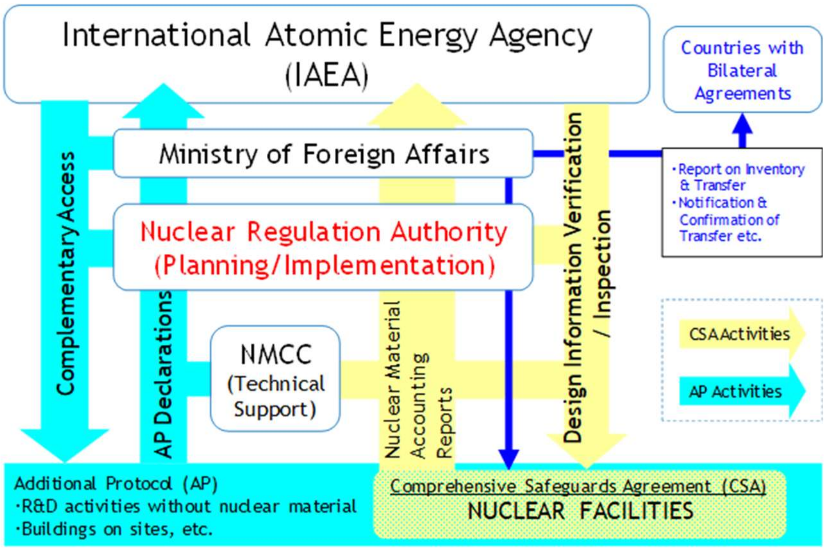
A nuclear operator shall determine the places where nuclear materials are handled at its facilities, shall perform a strict and accurate material accountancy including the amount of nuclear materials to be entered or transferred from its premises and the periodic inventory of nuclear materials, and shall report to the NRA. The JSGO will compile these reports received from operators and submit them to the IAEA.

(2) Containment/Monitoring

To ensure that nuclear materials have not been secretly transferred, the JSGO and IAEA have attached a “seal” on the lids of containers in which nuclear materials are stored and on the gateway for fuel at nuclear power plants. At nuclear power plants, etc., “surveillance cameras” are installed to constantly monitor the movement of nuclear materials.

(3) Inspection

Inspectors from the JSGO and the IAEA will visit the nuclear facilities in person and conduct inspection activities such as verifying the consistency of reports and records, checking the quantity and volume of nuclear materials on site, collecting samples for analysis, and assessing the containment/monitoring data.



*1 : Except for complementary access occurred during regular domestic inspection
*2 : Based on Nuclear Power Plant Regulation, specify Nuclear Material Control Center (authorized foundation) as "Specified safeguards inspection Agency" and "Specified information processing Agency"

Fig. A7-1 Safeguards implementation system in Japan

Sited from : "Safeguards" in NRA website, https://www.nsr.go.jp/activity/hoshousochi/index.html

The safeguards shall be applied to the Fukushima Daiichi NPS without exception. TEPCO, as a nuclear operator, will consult with the JSGO and IAEA to determine the method of implementing the safeguards (material accountancy, containment/monitoring, inspection) to be applied to Fukushima Daiichi NPS, and practically apply them on site.

Attachment8 Terms related to radioactive waste management

IAEA Safety Requirements GSR-Part 5[123]explains that predisposal of radioactive waste encompass all stages of radioactive waste management from generation to disposal, including processing, storage and transportation. Terms related to the management of radioactive waste as defined in the IAEA glossary are shown in Fig A8-1. Within the pre-disposal management, processing of radioactive waste is classified into pretreatment, treatment and conditioning. Processing is carried out to be in the form of waste suitable for selected or anticipated disposal options. Radioactive waste may also be stored in for its management, therefore it is thought to be necessary that the form is suitable for transportation and storage.

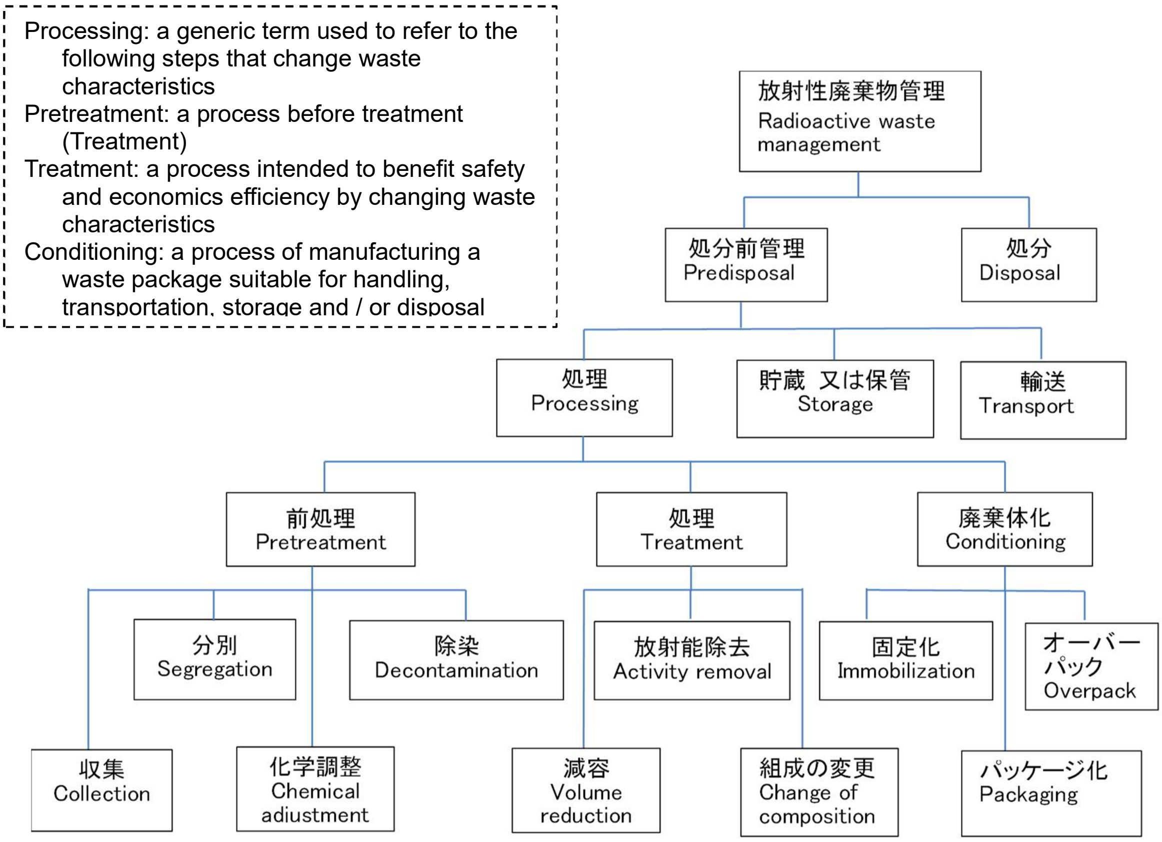


Fig. A8-1 Terms related to radioactive waste management (IAEA)[124] and their translation examples (For the Japanese translation example, refer to the materials of the Japan Atomic Energy[125,126])

[123] IAEA, Predisposal Management of Radioactive Waste, IAEA Safety Standards Series No. GSR Part 5, (2009). (NSRA, IAEA Safety Standard/Predisposal of Radioactive Waste/General Safety Requirement 5, No. GSR-Part5, July, 2012)

[124] IAEA, IAEA Safety Glossary Terminology Used in Nuclear Safety and Radiation Protection 2007 Edition, p.216, (2007).

[125] AESJ, The Report of 2013, - Organizing information of radioactive waste and matters to be considered for solving the issues (p.7), March 2014, the Expert Committee, “Processing /disposal of radioactive waste generated by the accident of Fukushima Daiichi NPS”

[126] AESJ, Seiya Nagao and Masafumi Yamamoto, “Introduction to radioactive waste - Management of radioactive waste from operation and decommissioning of nuclear and other facilities (!) Perspective of radioactive waste management, the 56 of (9) of Journal of the Atomic Energy Society of Japan, p.593, (2014).

Attachment9 Disposal of radioactive waste[127,128,129]

1. International classification of radioactive waste

Radioactive waste contaminated with radioactive materials is generated through operation and dismantling of nuclear power plants and the use of radioisotopes in medical and industrial applications. Radioactive waste shall be classified appropriately according to the radioactivity level and properties of waste, types of radioactive materials, etc., and strictly controlled, and then shall be reasonably processed and disposed of so as not to affect the human living environment.

The IAEA's Specific Safety Requirements SSR-5 “Disposal of Radioactive Waste” (2011)[130] specifies that a preferred strategy for the management of radioactive waste that is internationally agreed is to contain the waste and isolate it from the living environment, while minimizing the generation of radioactive waste. The required isolation and containment depend on the magnitude of the hazards of the waste and the time, thereby a disposal option (design and depth of facilities) being selected accordingly.

The IAEA’s General Safety Guide GSG-1 “Classification of Radioactive Waste”[131] indicates the relationship between the classification of radioactive waste and disposal options depending on the magnitude of the hazards (amount of radioactivity) and the duration (the half-life) of the radioactive waste, as shown in Fig. A9-1. Each classification is also shown in Table A9-1.

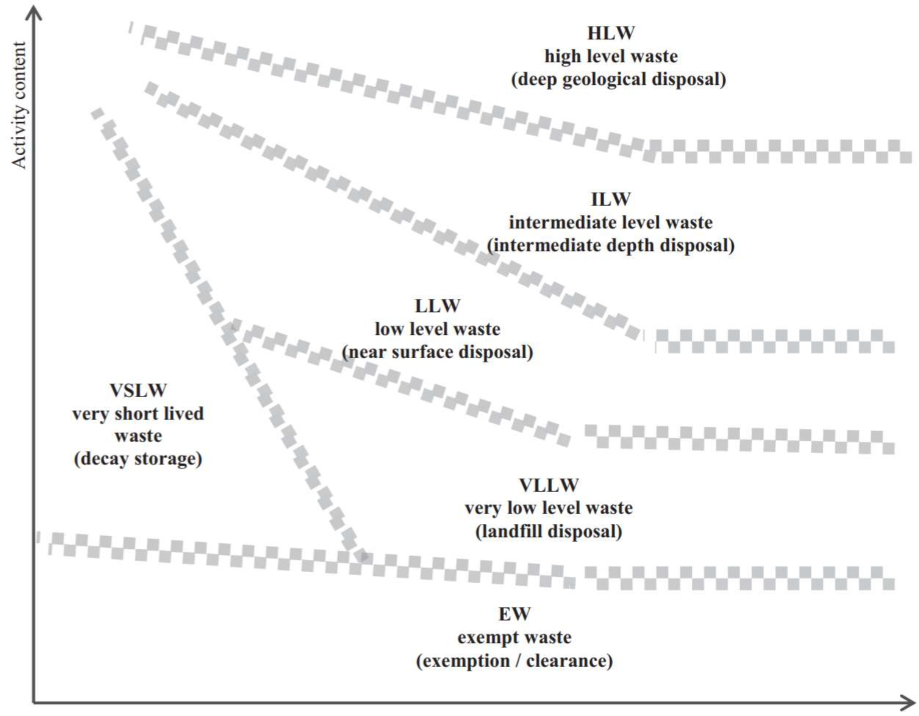


Fig. A9-1 Conceptual diagram of waste classification

[127] Osamu Tochiyama, Principles and Basics of Radioactive Waste Disposal, Radioactive Waste Management Funding and Research Center (a Public Interest Incorporated Foundation) (2016)
[128] https://www.enecho.meti.go.jp/category/electricity_and_gas/nuclear/rw/
[129] https://www.fepc.or.jp/nuclear/haikibutsu/index.html
[130] IAEA SSR-5 "Disposal of Radioactive Waste"(2011)
[131] IAEA GSG-1" Classification of Radioactive Waste" (2009)

Table A9-1 Classification of radioactive waste in GSG-1

Classification	Description of Classification
Exempted waste (EW)	Waste satisfying the criteria for clearance, exclusion and exemption from regulatory control for radiation protection purposes
Very short-lived waste (VSLW)	Waste that is decay-stored for a limited period of time up to several years and then exempted from regulatory control, as approved by the regulatory body.
Very low level waste (VLLW)	Waste that does not necessarily satisfy EW standards but does not require high-level containment and isolation. Suitable for disposal in shallow landfills where regulatory control is limited.
Low level waste (LLW)	Waste that exceeds clearance levels but has a limited amount of long-lived nuclides. Rigid isolation and containment are required for periods of up to several 100 years and are suitable for disposal in engineering facilities in shallow soils.
Intermediate level waste (ILW)	Waste that requires higher-level containment and isolation than the near surface disposal because of the nuclides it contains, especially long-lived nuclides. However, considerations on heat removal are hardly required. Because ILW may contain concentrations of long-lived nuclides (especially α-nuclide) that are not manageable in near surface disposal, a depth of tens to hundreds of meters are required for disposal.
High Level waste (HLW)	Waste with a large amount of heat generation at high activity concentration levels or waste containing large amounts of long-lived nuclides for which a design equivalent to a disposal facility for such waste needs to be considered. Generally, waste is disposed of in a stable stratum at the depth of several hundred meters or more from the ground surface. In some countries, spent fuel is classified as HLW.

2. Classification and disposal in Japan

In Japan, radioactive waste is broadly divided into “low-level radioactive waste” (equivalent to VLLW to ILW in GSG-1), which is generated through the operation of nuclear power plants, and “high-level radioactive waste” (equivalent to HLW of GSG-1), which is generated through the reprocessing of spent fuel that is generated through the operation of nuclear power plants and is vitrified with a high level of radioactivity. When disposed of, waste shall be classified appropriately according to its radioactivity level and properties, types of radioactive materials, etc., and shall be strictly controlled, and reasonably processed and disposed of under the principle that responsibilities lie with those who have generated the waste.

“High-level radioactive waste” is a vitrified waste liquid with a high radioactivity level that is produced in the process of reprocessing spent fuel generated through the operation of nuclear power plants. In Japan, the act (the Designated Radioactive Waste Final Disposal Act (the Final Disposal Act)) stipulates that radioactive waste shall be disposed of in strata more than 300 meters deep underground.

The term “low-level radioactive waste” refers to all types of radioactive waste other than “high-level radioactive waste”, and is further divided into several categories depending on where it is generated and the level of radioactivity.

The types of radioactive waste generated by the operation of nuclear power plants and the disposal methods assumed are shown in Table A9-2.

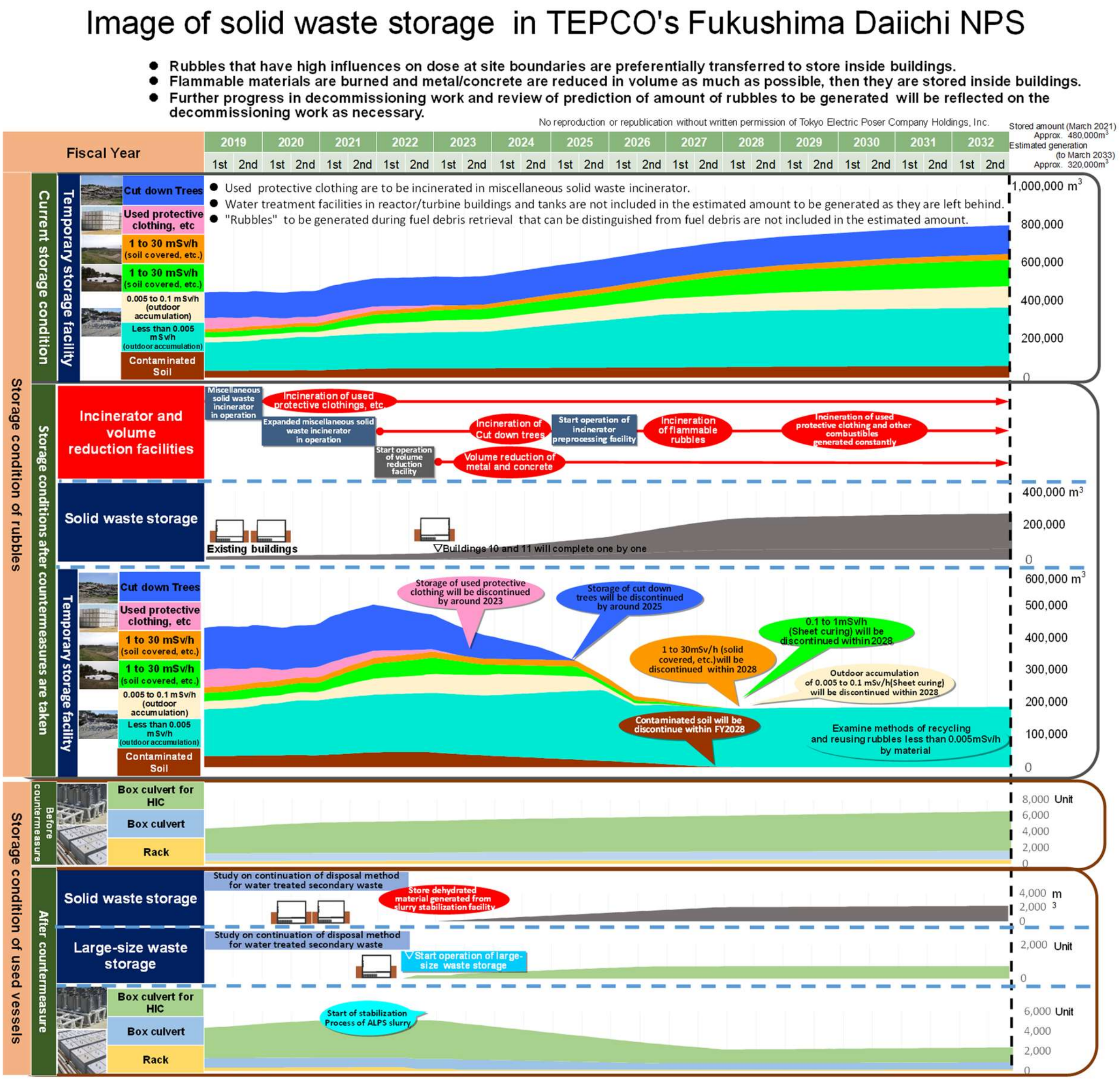
Of these, only waste with relatively low-level radioactivity generated through the operation of nuclear power plants has been subject to disposal in pits since 1992 at the Rokkasho Low-level Radioactive Waste Disposal Center of Japan Nuclear Fuel Limited in Rokkasho Village, Aomori Prefecture. Including the existing facilities, approximately 1 million drums of waste contained in 200-liter drums are planned to be buried, and eventually the scale will be enlarged to approximately 3 million drums using 200-liter drums.

Table A9-2 Types of Radioactive Waste Generated by the Operation of Nuclear Power Plants

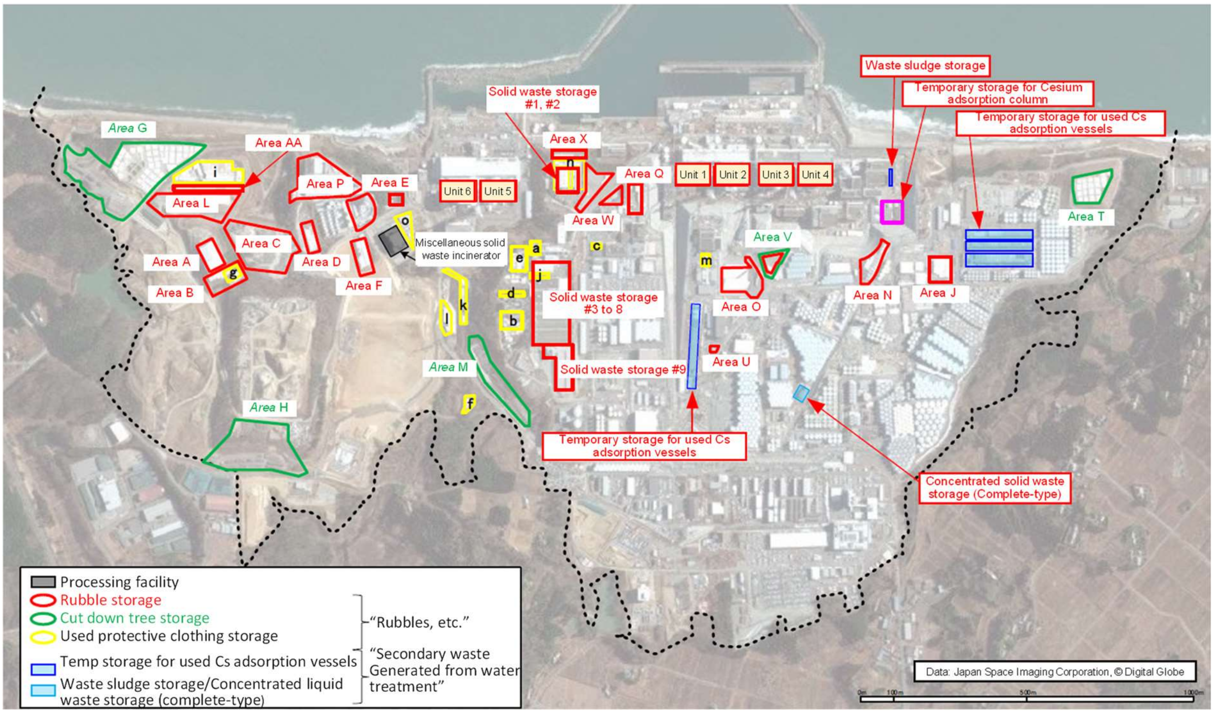
Types of Radioactive Waste			Examples of Waste	Site generated	Disposal Method (example)
Low-level Radioactive Waste	Waste from Nuclear Power Plants	Waste with extremely low radiation level	Concrete, metal, etc.	Nuclear Power Plant	Trench disposal (Near surface disposal (L3))
		Waste with relatively low radiation level	Effluent, filter, waste equipment, solidification of consumables		Pit disposal (Near surface disposal (L2))
		Waste with relatively high radiation level	Control rod, structures inside reactor		Intermediate depth disposal (L1)
	Uranium Waste		Consumables, sludge, waste equipment	Uranium enrichment and fuel processing facility	Intermediate depth disposal, Pit disposal or Trench disposal, geological disposal in some cases
	Radioactive Waste includes Transuranic Nuclide (TRU Waste)		Parts of control rod, effluent, filter	Spent fuel reprocessing facility, MOX fuel fabrication facility	Geological disposal, Intermediate depth disposal or Pit disposal
High-level Radioactive Waste			Vitrified waste	Spent fuel reprocessing facility	Geological disposal

Waste below the clearance level	Most of demolition waste of nuclear power plants	All the above sites	Reuse / Disposal as general goods

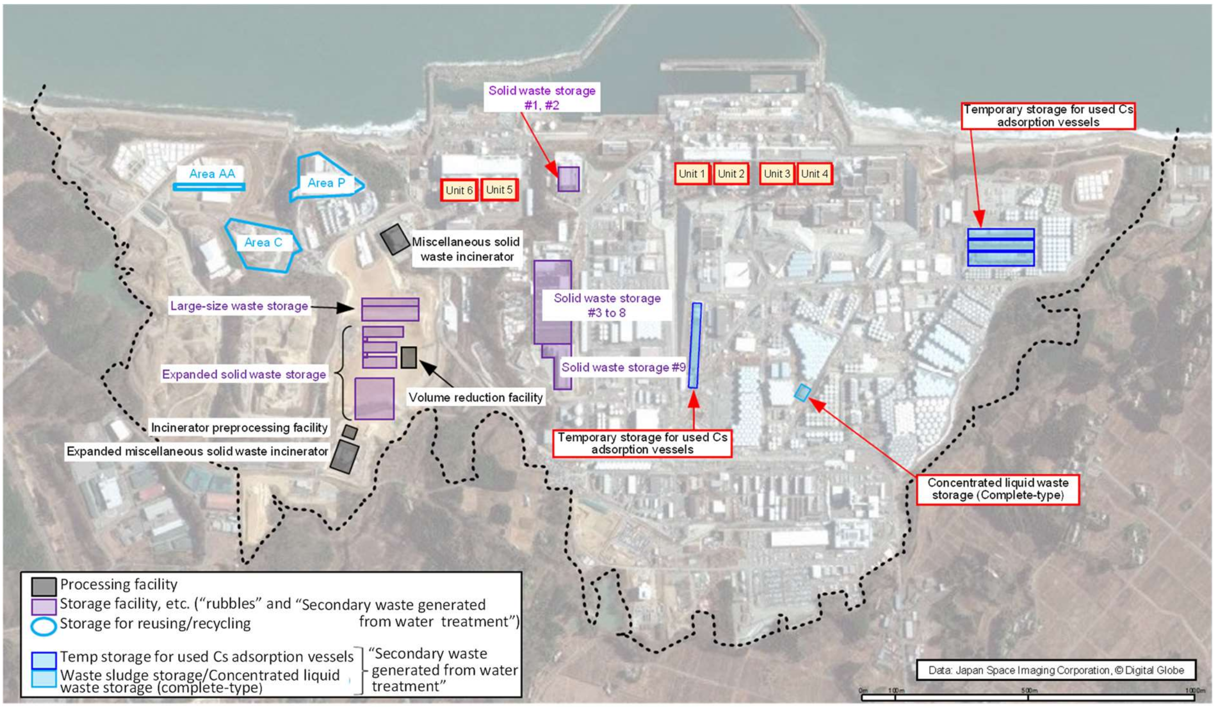
Fig. A9-2 Japan Nuclear Fuel Ltd. Low-level Radioactive Waste Disposal Center



[132] TEPCO, Solid Waste Storage Management Plan for Fukushima Daichi Nuclear Power Station of Tokyo Electric Power Company Holdings, Inc., July 2021 edition (issued on July 29, 2021)



(a) Present storage condition of "rubble, etc." and "secondary waste generated by water treatment"



Future storage condition of "rubbles, etc." and "secondary waste generated by water

Fig. A10-1 Present and future storage conditions of "rubble, etc." and "secondary waste generated by water treatment" on site of the Fukushima Daiichi NPS

Attachment11 Prospects of processing/disposal method and technology related to its safety

Table of contents

1. Overview

Waste management is a long-term effort that must attain the prospect of the implementation of final disposal, while reducing risks in every stage. Since a large amount of solid waste with differing properties is generated in association with the decommissioning of the Fukushima Daiichi NPS[133], efforts are being carried out based on the basic concept of solid waste management summarized in the Medium-and-Long-term roadmap for decommissioning of the Fukushima Daiichi Nuclear Power Station of Tokyo Electric Power Company Holdings Inc. (hereinafter referred to as the “Medium-and-Long-term roadmap”). TEPCO is required to thoroughly implement safe and reasonable storage of the generated solid waste. With NDF taking the lead, the concerned organizations are examining integrated management of solid waste from characterization to processing/disposal in a professional manner based on their respective roles. In addition to improving analytical abilities for characterization, a flexible and reasonable waste stream (integrated flow of the measures from characterization to processing/disposal) is under development. Regarding processing/disposal strategies among these activities, the “Technical Strategic Plan 2020 for Decommissioning of the Fukushima Daiichi Nuclear Power Station of Tokyo Electric Power Company Holdings, Inc.” (hereinafter referred to as the “Technical Strategic Plan”) specifies that the prospects of processing/disposal method and technology for solid waste related to its safety are to be provided by around FY 2021, and the summary results are shown below.

<Key points of "basic concepts on solid waste”>

(1) Thorough containment and isolation
Thoroughly contain and isolate radioactive materials to prevent human access to them, in order not to cause harmful radiation exposure.

(2) Reduction of solid waste volume
Reduce the amount of solid waste generated by decommissioning as much as possible.

(3) Promotion of characterization
Have appropriate characterization for addressing the increase in the number of analysis samples for the purpose of advancing studies on processing/disposal methods of solid waste.

(4) Thorough storage and management
Store generated solid waste safely and reasonably according to characteristics of the solid waste.
Secure storage capacity to ensure that the waste can be stored within the site of the Fukushima Daiichi NPS.

(5) Establishment of method for selecting the preceding processing methods in consideration of disposal
Before the technical requirements of disposal are established, establish a method for

[133] There is a possibility that some rubble generated after the accident may not be classified as waste or radioactive waste due to reuse on site. However, such rubble, secondary waste generated by water treatment, and radioactive solid waste stored at the Fukushima Daiichi NPS since before the accident are hereinafter referred to as “solid waste”.

selecting processing for stabilization and immobilization (preceding processing) and then select preceding processing methods.

(6) Promotion of effective R&D with a comprehensive view of overall solid waste management
Work with each R&D field in characterization and processing/disposal to form an overview of the overall management of solid waste and then identify the required R&D tasks.

(7) Development of continuous operational framework
In order to continue safe and steady solid waste management, establish a continuous operational framework that includes development of relevant facilities and human resources.

(8) Measures for reducing radiation exposure of workers
Thoroughly implement radiation exposure control, and health and safety management based on the relevant laws/regulations.

2. Policy for developing Technical Prospects

The Medium-and-Long-term roadmap states that Technical Prospects will be provided in the strategic plan by around FY 2021. Specifically, it will “present how to proceed with volume reduction of solid waste”, “develop analytical/evaluation methods for efficient characterization”, and “establish a method for reasonably selecting safe processing/disposal methods for solid waste once necessary information including characterization is obtained”.

For “volume reduction”, the basic policy for solid waste specifies that TEPCO should reduce the burden of overall waste management (for the entire process from generation, storage, processing to disposal) to the extent possible. Strategic plan 2020 considered the possibility of further efforts based on precedents in other countries. Then, example cases for volume reduction of dismantled waste in other countries were investigated (Chapter 3).

“Developing analytical/evaluation methods for efficient characterization” and “establishing a method for reasonably selecting safe processing/disposal methods for solid waste once the necessary information including characterization is obtained” means to develop the methods necessary for disposing of materials that become waste even after undergoing volume reduction. NDF has organized and examined the following specific goals in strategic plan 2018. Of the wide variety of solid wastes, secondary waste generated by water treatment, which has high mobility and for which there is no precedent for processing or disposing of it in Japan, has been selected as the main object for examination.

(1) Establish a safe and reasonable disposal concept based on the characteristics and volume of the solid waste generated in the Fukushima Daiichi NPS with its applicable processing technology, and develop safety assessment methods that apply the features of the disposal concept, with consideration of examples from other countries.
(2) Clarify radiological analysis and evaluation methods for characterization.
(3) Clarify processing technology for which practical application can be expected for stabilization and immobilization, with consideration of the disposal of several important waste streams such as secondary waste generated by water treatment.
(4) Before the technical requirements for disposal are determined (i.e., preceding processing), establish a method for reasonably selecting processing technology for stabilizing and immobilizing waste based on the above methodology.
(5) Have the prospect of setting processing/disposal strategies for solid waste for which the processing technology considering disposal is not clarified, using a series of methods to be developed by around FY 2021.
(6) Clarify issues and measures concerning storage of solid waste until it is conditioned.

Based on these results, Chapter 4 provides Technical Prospects.

3. Example cases from other countries for volume reduction of dismantled waste

In countries that have been promoting the use of nuclear power from early on, decommissioning of facilities and environmental remediation of sites are in progress for reasons such as aging and economic efficiency. Along with dismantling and remediation work, a large volume of low-level waste has been or will be generated in the near future.

A variety of management policies for radioactive waste are being implemented in various countries according to their circumstances. This chapter provides case examples from three countries, the UK, US and France, where waste management including volume reduction has been promoted for diverse and large-volume low-level radioactive waste (hereinafter referred to as “LLW”) generated by decommissioning and environmental remediation activities especially at legacy sites related to nuclear development. The scope of this chapter does not include high and medium-level radioactive waste whose generation is limited compared with LLW.

3.1 Case examples in the UK

3.1.1 Background

In the UK, against a backdrop of issues such as an increase in the estimated volume of dismantled waste associated with decommissioning of government-owned legacy sites and tight disposal capacity at an existing low-level radioactive waste repository (hereinafter referred to as “LLWR”) located near Drigg, Cumbria County, the government reviewed their previous policy and announced their long-term management policy of solid low-level radioactive waste[134] (hereinafter referred to as “LLW management policy”) in 2007. With this reviewed policy, the government presented their basic approach to properly account for the nature of the subject LLW, and allow flexibility in selecting cost-effective LLW management policies that not only ensure safety but that are also environmentally acceptable.

In response to that, the Nuclear Decommissioning Authority (hereinafter referred to as the "NDA"), which is responsible for the decommissioning of legacy sites and the management of radioactive waste, developed the low-level radioactive waste management strategy[135] (hereinafter referred to as the “LLW management strategy”) in 2010 to present strategic initiatives for considering possible flexible options for volume reduction of waste and LLW management in accordance with the waste hierarchy and has implemented specific measures.

3.1.2 LLW waste management policy based on the waste hierarchy concept

The LLW management strategy provides the management policy for LLW based on the concept of the waste hierarchy (refer to Fig. A11-6) as described below.

[134] Policy for the Long-Term Management of Solid Low Level Radioactive Waste in the United Kingdom, By Defra, DTI and the Devolved Administrations, 26 March 2007.
[135] UK Strategy for the Management of Solid Low Level Radioactive Waste from the Nuclear Industry, NDA, August 2010.

3.1.2.1 Reduction of waste generation

The LLW management strategy places highest priority on minimizing the generation of radioactive waste in the waste hierarchy with the ultimate goal of reducing the volume of radioactive waste. For example, for facilities, the strategy points out the necessity of considering the reduction in the volume of potential radioactive waste through the lifecycle from designing to decommissioning. For existing contaminated materials generated as a result of past activities, implementing consistent management in accordance with the radioactive waste management plan, which was developed to manage the process up to the disposal of each type of radioactive waste at each site, is required to reduce the secondary waste that may be generated in the course of management.

3.1.2.2 Minimization of waste volume

For radioactive waste whose generation cannot be avoided despite efforts to reduce it, the amount of radioactivity and physical volume should be reduced to as low as reasonably practicable. Minimization of the waste volume is promoted by three activities: classification and segregation according to the radioactive waste properties, etc.; reduction of radioactivity by decontamination of facilities before decommissioning and of contaminated materials before transportation as radioactive waste; and characterization of radioactive waste for appropriate classification.

3.1.2.3 Re-use

In the LLW management strategy, some materials can be reused after maintenance is performed as required, such as temporary storage containers of radioactive waste and lead-shielding bricks. Moreover, facilities, systems and buildings may still be worth reusing for other uses even after serving their initial purpose, and therefore radioactive waste producers should explore utilizing such opportunities to the extent practicable.

The NDA operates a website for assisting the identification of potential users of facilities and systems no longer in use for their initial purpose and the transfer of such facilities and systems to them, and this initiative has resulted in the reuse of a considerable number of NDA assets that are no longer needed, and has even contributed to cost reduction.

Soil and rubble can also be reused as materials for site ground leveling and back-filling of voids if appropriate permission by the regulatory authorities is obtained upon evaluation of the site in accordance with the guidance of the environmental regulatory authorities described below.

3.1.2.4 Recycling

Further use of materials through recycling is seen as an important opportunity to reduce the volume of radioactive waste. Metal is the main object of recycling, but other materials such as concrete and rubble are also being recycled. Processing technologies such as surface decontamination to decontaminate the target objects, separation of radioactive materials and impurities by melting metals, crushing, and segregation are used for recycling. Characterization, monitoring, and evaluation of substances are also carried out so that they can be recycled effectively.

In the UK, metal recycling is an option that should be adopted from the perspective of resource conservation, optimum utilization of disposal sites and environmental protection, etc. Blast decontamination of surface contamination is employed in facilities in the UK, and melting processing is employed in other countries (Sweden, US, Germany).

The cumulative volume of metal treated since the launch of the National Waste Program (hereinafter referred to as the “NWP), which was established to implement the LLW management strategy, is 32,000 m^3 as of the end of March, 2021 (Fig. A11-1).

3.1.2.5 Waste volume reduction

Volume reduction of radioactive waste is not positioned as a formal stage of the waste hierarchy, but in the LLW management strategy, volume reduction and reforming of LLW, for which there is no option except disposal, by compression or heat processing (incineration or other heat processing) plays an important role in the optimum utilization of disposal sites.

As of the end of March 2021, the cumulative volume of waste incinerated since the launch of the NWP exceeds 27,000 m^3 (Fig. A11-1).

3.1.2.6 Waste disposal

One of the objectives of the LLW management policy/strategy in the UK is to make LLWR usable into the future. For LLW that can be subject to management policies other than disposal at an LLWR, it encourages consideration of management options that account for risks, and takes the policy of avoiding disposing of waste at LLWR to the extent possible.

As one of the measures, the government has established the category of very low level waste (hereinafter referred to as “VLLW”), and allowed disposal by controlled burial in industrial disposal facilities if the environmental regulatory authorities approve it based on an environmental safety case evaluation. Similar to VLLW, waste in the LLW category whose radioactive concentration is below a certain level (200 Bq/g) can be disposed of if the environmental regulatory authorities approve it based on an environmental safety case evaluation. At present, three private facilities dispose of VLLW, etc. As of the end of March, 2021 the cumulative volume of VLLW disposed of since the launch of the NWP exceeds 72,000 m^3 (Fig. A11-1).

	Metal Treatment (m^3)	Incineration (m^3)	VLLW Disposal (private facility) (m^3)
FY2020-21 Total volume	1,263	1,960	9,337
Cumulative volume from the beginning of NWP	32,062	27,506	72,312

Fig. A11-1 Cumulative volume of waste processed and disposed of VLLW since the launch of NWP pursuant to the LLW management strategy[136]

[136] March 2021 Waste Metrics Dashboard, Period 12: 21st February to 31st March FY20/21, LLW Repository Ltd.

In addition to exploring the possibilities of constructing a new LLW disposal site in or near the licensed nuclear sites in the future, in the LLW management policy, a management policy in which contaminated underground structures and soil in licensed nuclear sites are not collected and instead consideration is given to in-situ disposal (in-situ disposal) with permission by the environmental regulatory authorities based on an evaluation is listed as a possible option.

In response to this, in 2018 the environmental regulatory authorities published a guidance document on the management of radioactive waste generated by decommissioning and environmental remediation of nuclear sites. The document calls for the establishment of safety cases covering the entire site, including on-site disposal facilities, in-situ disposal, backfilling of underground cavities by contaminated rubble and contaminated soil, etc., and the waste management planning which serves as the basis. It also specifies that the safety cases and waste management plan are to be evaluated to determine whether or not to approve the release of the site from the environmental regulations (

Fig.).

At three sites (Winfrith, Trawsfynydd, Dounreay), the NDA has been engaged in preliminary studies on the possibility of in-situ disposal and backfilling, etc., from the viewpoint of controlling the generation of new waste and promoting environmental remediation with economic reasonability while ensuring safety[137].

Promoting such flexible policies for LLW management has significantly reduced the volume of LLW disposed of in LLWR. As a result, there is the prospect that LLW to be generated from decommissioning of legacy sites can be managed with the remaining available capacity in LLWR.

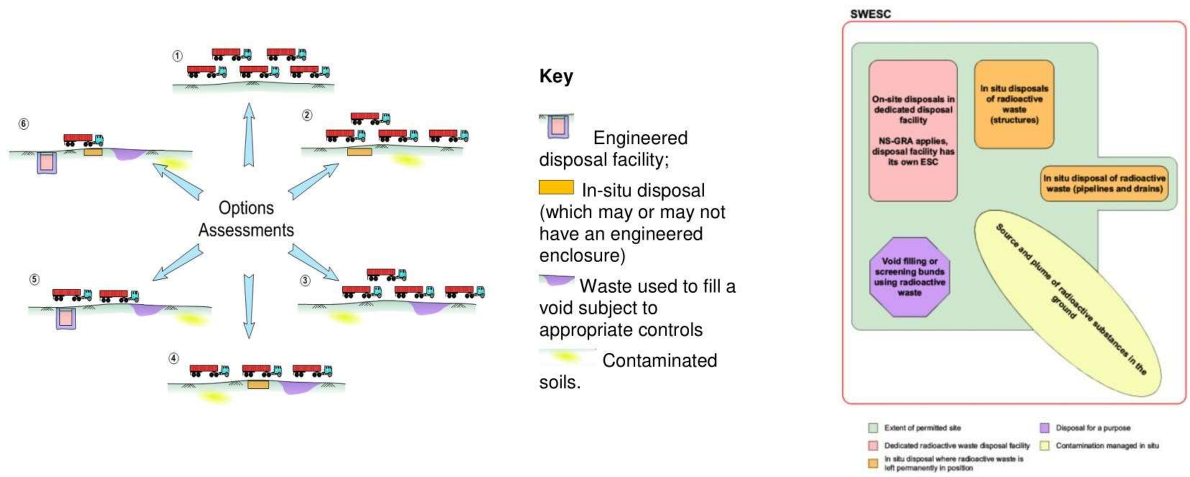


Fig. A11-2 Possible disposal strategy options in a licensed nuclear site[138] (left), and safety case concept covering the entire site[139] (right)

[137] Strategy Effective from March 2021, NDA, March 2021.

[138] Nuclear Decommissioning, Consultation on the Regulation of Nuclear Sites in the Final Stages of Decommissioning and Clean-Up, BEIS, May 2018.

[139] Management of radioactive waste from decommissioning of nuclear sites: Guidance on Requirements for Release from Radioactive Substances Regulation, Version 1.0, SEPA, EA, NRW, July 2018.

3.2 Case examples in US

3.2.1 Background

In US, there are a number of national defense-related legacy sites that were established and operated during the Cold War. The US Department of Energy (hereinafter referred to as the "DOE") is responsible for the decommissioning, environmental remediation of these sites, and the resulting waste. It provides site management in accordance with its regulations enforced by DOE orders under the agreement with the US Environmental Protection Agency and state regulatory authorities.

However, in nuclear power plants owned and operated by private electric utilities, their operators are promoting decommissioning in their sites that have ceased operation due to aging, etc. LLW generated is disposed of at disposal sites under the responsibility of the state subject to the regulations developed by the US Nuclear Regulatory Commission (hereinafter referred to as the “NRC”).

3.2.2 DOE's initiatives to reduce radioactive waste volume

In promoting the decommissioning and environmental remediation of legacy sites since 1989, the DOE has provided waste management that includes reducing a large volume of radioactive waste generated at each site. Especially for the volume of contaminated metal scrap generated from dismantling uranium enrichment facilities at the Oak Ridge site, the DOE has supported and promoted metal recycling by private companies. As a result, at its peak in 1995, 13,600 tons of contaminated carbon steel and stainless steel per year were melted and recycled to the open market. In 2000, its release to the open market was suspended, but metal recycling for effective use within DOE sites continues[140].

At the Bear Creek processing facilities of EnergySolutions, one of the major metal processing facilities, steel, lead, and other metals received from home and abroad are melted and recycled into shielding blocks for use at DOE sites.

Consignor (by Country)	Year Started to Receive	Total Amount Received (ton) (as of July 2015)
United States	1991	62,380
Belgium	1996	304
Canada	2006	2,033
Germany	2000	1,153
Spain	2001	99
United Kingdom	2006	307
Subtotal (not including United States)		3,896
Total		66,276

Fig. A11-3 Melting processing at EnergySolutions[141], shielding blocks, and total volume of accepting metal processing by country[142]

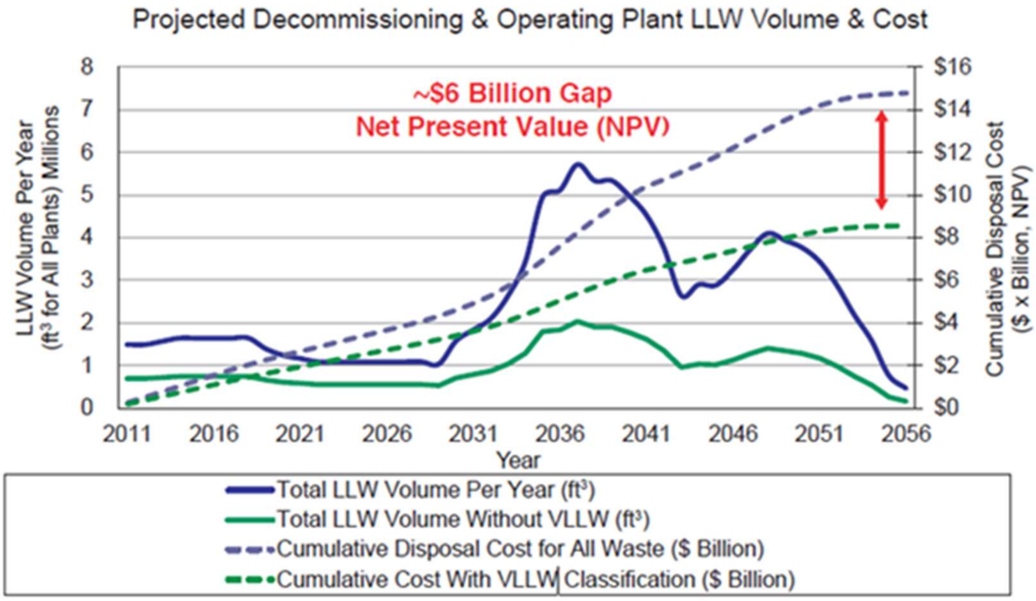
[140] Waste Management for Decommissioning of Nuclear Power plants: An EPRI Decommissioning Program Report, EPRI, Symposium on Recycling of Metals Arising from Operation and Decommissioning of Nuclear Facilities, April 2014.
[141] DOE Promotes Internal Recycle and Reuse of Its Metals in a Big Way!, 1995InSite, Vol.26 GTS DURATEK.
[142] Recycling and Reuse of Materials Arising from the Decommissioning of Nuclear Facilities, OECD/NEA, 2017.

3.2.3 Initiatives to develop regulatory systems for dismantled waste from private nuclear power plants

In US, there are considered to be limitations to the economic benefits of metal recycling LLW generated by decommissioning private nuclear power plants because disposal costs are relatively low; large equipment such as steam generator, etc. can be buried in shallow ground disposal sites for LLW; and a clearance system has not been introduced.

Since there is no category of radioactive waste in US that corresponds to VLLW, it is necessary to dispose of waste equivalent to VLLW as Class A waste, which is one of the categories of LLW, at LLW disposal sites. However, in terms of risk-informed regulations aimed at the flexible application of regulations according to the degree of risk, the NRC permits VLLW-equivalent waste to be disposed of at state facilities for hazardous waste disposal on a case-by-case basis[143],[144].

With the outlook of an increasing volume of low-level dismantled waste generated from planned decommissioning of nuclear power plants (Fig. A11-4), the NRC is discussing introducing a new category for VLLW and a clearance system in terms of safety for transportation as well as economic efficiency.



(Source : EPR material)

Fig. A11-4 Estimated LLW volume and cost at nuclear power plants in US (waste generated from decommissioning and operation) [145]

3.3 Case examples in France

3.3.1 Background

In France, radioactive waste is classified according to the place of generation in nuclear facilities based on the concept of zoning, and this concept is also applied to dismantled waste, etc. generated by decommissioning and environmental remediation in nuclear sites. Since France does not have a clearance

[143] 10 CFR 20.2002 Method for obtaining approval of proposed disposal procedures.
[144] Backgrounder on Disposals of Very Low-Level Waste Under 10 CFR 20.2002, NRC Website in US
[145] EPRI Research Summary: Very Low Level Waste, US NRC's 30th Annual Regulatory Information Conference, March 2018.

system, radioactive waste categorized as VLLW is disposed of at the VLLW disposal site established in Morvilliers in 2003 (hereinafter referred to as “CIRES”) (Refer to Fig. A11-5).

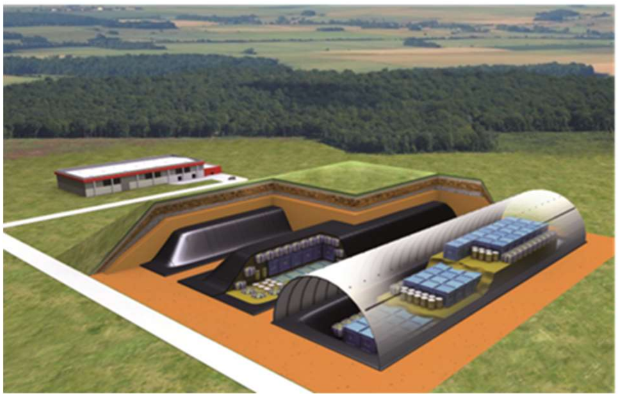


Fig. A11-5 Conceptual drawing of disposal at CIRES[146] (left), and bird-eye view of the facility[147] (right)

The French national radioactive waste management agency (hereinafter referred to as “ANDRA”), which is responsible for the disposal of radioactive waste, estimates the volume of VLLW generated from decommissioning the existing nuclear facilities in France to be 2,200,000 m^3. As of the end of FY 2014, 280,000 m^3 of VLLW are placed in CIRES, which is approximately 43% of the permitted disposal capacity (650,000 m^3), had been placed at CIRES, and it is necessary to establish a new VLLW disposal site or expand its permitted disposal capacity in 2025[148].

3.3.2 Movement toward law amendment for metal recycling

Under these circumstances described above, in “the national plan on management of radioactive materials and waste (PNGMDR)” developed triennially, the Nuclear Safety Authority (hereinafter referred to as “ASN”) as the nuclear regulatory authority requires waste producers such as ORANO and Électricité de France (EDF) to examine conditions for recycling metal and rubble in light of volume reduction of VLLW disposal, and to report technical/safety options for facilities for metal melting processing and management policies.

In France, while the necessity of constructing a second VLLW disposal site (centralized or distributed) is being considered, law amendment is under discussion for a proposal to allow recycling of some metals contained in VLLW by clearance as an exception, if such recycling is considered to be appropriate. Discussions are in progress toward ASN’s approval of using rubble as a filler for voids at the CIRES disposal site.

[146] The surface disposal concept for VLL waste, Andra.
[147] Source : Google Map
[148] French National Plan for the Management of Radioactive Materials and Waste (PNGMDR) 2016-2018, ASN.

3.4 Summary

As described above, handling of VLLW generated in large volume even from decommissioning of normal nuclear power plants has become a common issue for the nations. While ensuring safety, various measures have been implemented according to the situation in each country, such as the nature of radioactive waste generators and disposal entities (public/private), prospects for securing disposal sites, and differences in regulatory systems concerning radioactive waste disposal.

Especially for recycling by metal processing, the UK and the US have a track record. Therefore, various implementation conditions for processing characteristics, etc. should be investigated/examined.

In light of the differences in systems and approaches, and based on the waste hierarchy rules/principles, feasible measures for decommissioning the Fukushima Daiichi NPS and policies that includes regulation to realize it should be materialized.

4. Prospects of processing/disposal method and technology related to its safety

4.1 Approach for volume reduction

If a large volume of solid waste exists, not only does segregation and analysis take time, but the number of storage containers and the scale of storage facilities increase, and so does the load of solid waste management. Therefore, it is extremely important to reduce the physical volume of solid waste as much as possible.

At the Fukushima Daiichi NPS, it is important to permeate initiatives on volume reduction, aimed at decreasing the burden of overall solid waste management, into the overall decommissioning activities by referring to examples from other countries that have implemented the waste hierarchy concept as show in Chapter 3.

Specifically, the priorities for measures to be taken as waste management are (1) prevention of waste generation, (2) minimization of waste volume, (3) reuse, (4) recycling, and (5) disposal. In solid waste management, it is important to prioritize (1) as much as possible, and consider (5) disposal as the last option (Fig. A11-6) for volume reduction of solid waste to be stored, processed, and disposed of.

In terms of reducing waste generation, it is important to consider the reduction of the volume of materials used in the design and construction plan. It is also important not to bring in substances that affect processing/disposal as much as possible. To minimize physical volume, consideration on preventing contamination through strict segregation, maintenance/management of manufactures, extension of product life, and waste volume reduction is important. Reuse should be promoted after contamination checks, decontamination, repair and parts replacement, and it is useful to consider ease of reuse from the design stage. Considering alternative uses is also beneficial. In recycling, it is important to consider the state of contamination of contaminated valuable sources, separate and process recyclable materials, and use them as new materials and products.

As shown in Fig. A11-6, TEPCO has also been implementing initiatives corresponding to this concept. As new measures to be implemented, of the rubble accumulated outdoors (surface radiation dose rate $\leqq$ 0.1 mSv/h), the reuse/recycling of metals, concrete, and scrapped flanged tanks with extremely low surface radiation dose rates is being considered. As part of this activity, decontamination methods for recycling metals are being examined. In promoting safe and reasonable waste management, it is important to examine further possibilities based on the characteristics of solid waste at the Fukushima Daiichi NPS in reference to precedent cases in other countries.

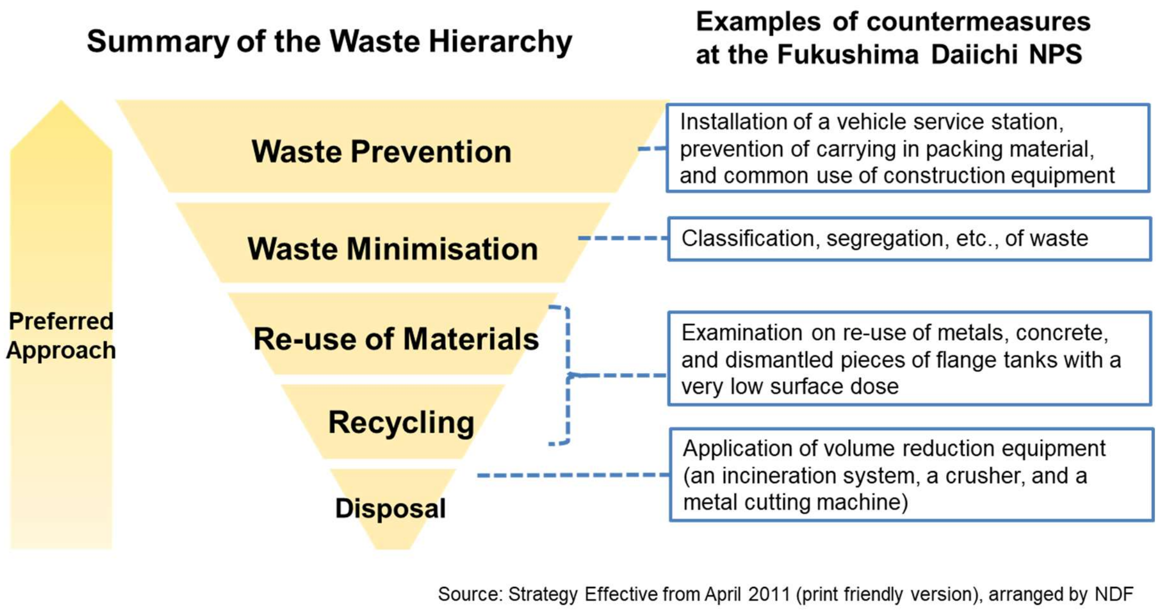


Fig. A11-6 Summary of waste hierarchy at the NDA, UK, and countermeasures at the Fukushima Daiichi NPS

4.2 Development of analytical/evaluation methods for implementing efficient characterization

Efficient characterization is necessary as solid waste at the Fukushima Daiichi NPS is characterized by diverse nuclide compositions and activity concentrations, and a large physical volume. With the specific goal, "(2) Clarify radiological analysis and evaluation methods for characterization" described in Chapter 2, analytical methods for simplified and speed-up data acquisition, as well as characterization methods using statistical methods have been developed in the Project of Decommissioning and Contaminated Water Management[149] and other programs. For the latter, a statistical inventory estimation method was developed by incorporating a method for efficiently identifying inventory, which combines the nuclide relocation models based on the contamination mechanism with the analytical data, with a method for quantifying uncertainties in evaluated values by using statistical methods.

4.2.1 Development of analytical methods for simplified and speed-up data acquisition

The "basic concept for solid waste management" indicates that appropriate characterization responding to an increase in the number of analytical samples should be promoted to proceed with examination for solid waste processing/disposal. Therefore, the subject waste and nuclides to be evaluated in the characterization were clarified, and efficient analytical methods with the required

[149] IRID, supplementary budget in FY 2018, "Subsidies for the Project of Decommissioning and Contaminated Water Management (R&D for Treatment and Disposal of Solid Waste), FY 2019 results, December 2020https://irid.or.jp/wp-content/uploads/2021/01/2019011kotaihaikibutsu_02.pdf
IRID, supplementary budget in FY 2018, "Subsidies for the Project of Decommissioning and Contaminated Water Management (R&D for Treatment and Disposal of Solid Waste), FY 2020 results (in preparation for publication)

measurement accuracy and the associated manual were developed. Then, the possible issues were examined to incorporate them into analysis in the Radioactive Material Analysis and Research Facility Building #1 (hereinafter referred to as “Okuma Building #1).

Solid waste generated from the decommissioning of the Fukushima Daiichi NPS undergoes primary, secondary, and tertiary classification (Fig. A11-7), and is temporarily stored, stored, and conditioned according to its classification. In characterization, the physical and chemical properties (particle size distribution, viscosity, quality of materials, surface radiation dose rate, nuclide concentration, etc.) of waste required for proper classification of generated waste are identified.

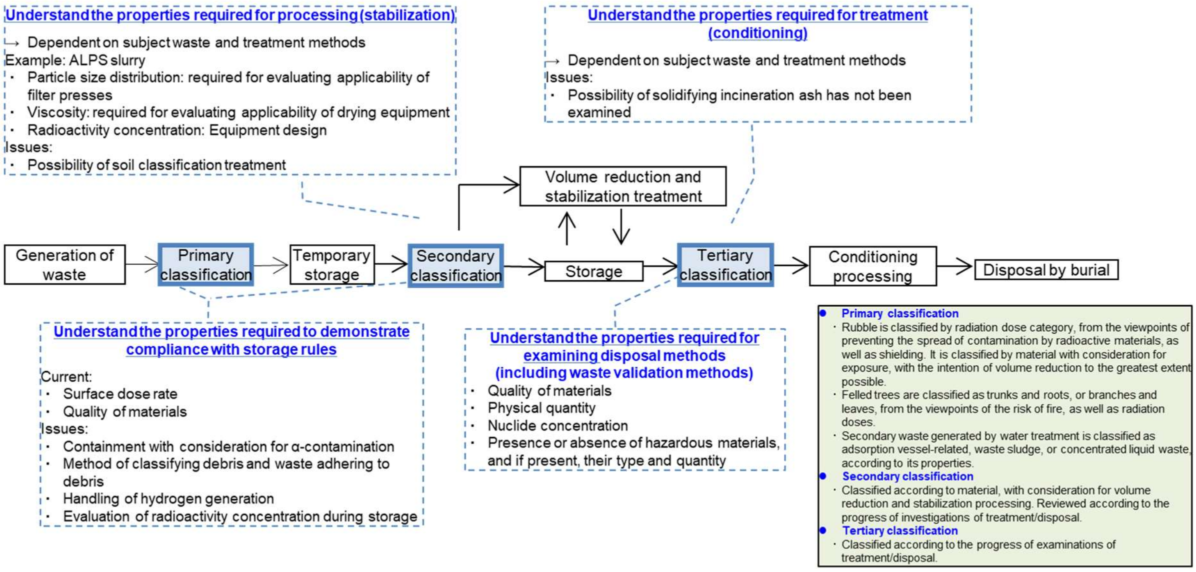


Fig. A11-7 Characterization according to waste classification at the Fukushima Daiichi NPS[149]

In terms of reducing risks in decommissioning the Fukushima Daiichi NPS, the types of waste subject to characterization were selected from rubble, incineration ash, soil, waste generated from fuel debris retrieval (hereinafter referred to as “waste from debris retrieval”), and secondary waste generated by water treatment and other substances with different properties based on the physical volume, radiation dose, nuclide composition and disposal characterization, etc. As subjects (nuclides to be evaluated) for analyzing their nuclide concentration of waste properties, short half-life nuclides have been excluded from those important for radiation protection, and those important for domestic disposal safety assessment have been selected. For nuclides whose concentration can be estimated based on fuel combustion calculation in terms of isotopes and chemical similarity, nuclides to be examined and analyzed have been summarized, taking the importance of safety in disposing of waste generated from the accident into account.

The progress of characterization of the target waste is as follows. Of the secondary waste generated by water treatment, characterization of high-risk sludge generated by decontamination devices and concentrated waste liquid with the aim of secondary classification for volume reduction, stabilization and storage has been completed. The sludge generated by decontamination devices is being examined with the aim of retrieval from the process building and storage based on the information obtained from characterization. Characterization of ALPS slurry is underway for secondary classification. For rubble, analysis of the nuclide composition of samples (111 samples as of March end, 2020) taken for tertiary classification as well as examination of statistical inventory evaluation methods are being carried out. Upon the completion of the Okuma Building #1, the data will be accumulated in order to develop disposal validation methods for the tertiary classification, and the application of a scaling factor method will be considered.

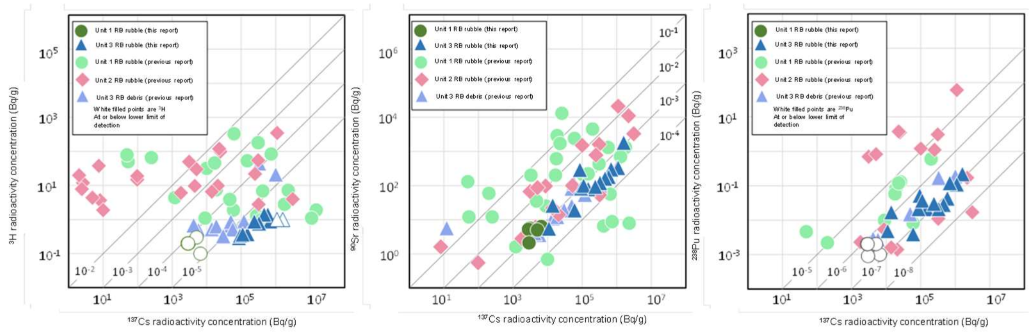


Fig.A11-8 Examination of nuclide composition of rubble (reactor building) (correlation with Cs-137 concentration)[149]

4.2.1.1 Technology development for automating and accelerating characterization

4.2.1.1.1 Development of reasonable sampling technology for analytical samples

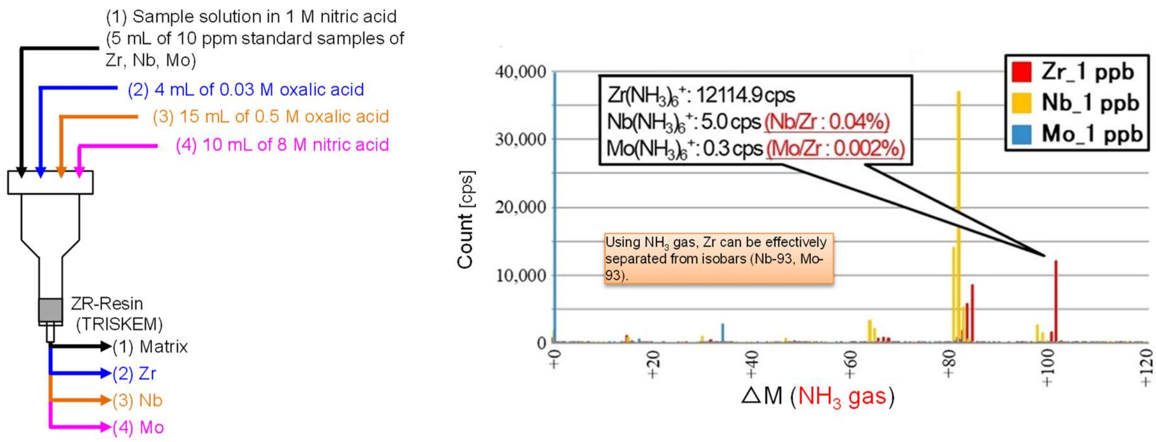
In order to efficiently collect a small number of analytical samples from rubble sampled on site while ensuring representativeness, technology was developed that evaluates the contamination distribution of the collected rubble (sample mapping) and collects a small number of analytical samples from the positions selected according to the contamination distribution (sampling), and its mock-up machine was manufactured. Based on a performance evaluation test, areas for improvement with a view to practical application were summarized.

4.2.1.1.2 Development of analytical methods using ICP-MS/MS

In determining the quantity of nuclides by measuring β-rays and characteristic X-rays, a time-consuming and precise chemical separation process is required in advance. It takes a long time to measure long half-life nuclides with low specific activity. However, analysis using ICP-MS/MS (triple quadrupole inductively coupled plasma mass spectrometry) enables the quantity of multiple nuclides to be determined simultaneously because the substances can be sorted by mass. As separation is possible by using reactant gas even if nuclides (isobar) of the same weight coexist in a sample, the separation process can be economically streamlined. In order to apply ICP-MS/MS to the analysis of rubble in the Okuma Building #1, automation technology was developed for a separation process using ICP-MS/MS and analytical samples.

The separation process using ICP-MS/MS was developed for difficult-to-measure nuclides (Zr-93, Mo-93, Pd-107, Sn-126) contained in rubble. An automatic solid-phase extraction device was developed, which removes the sample matrix from an analytical sample, and separates isobars by one-step extraction chromatography (Fig. A11-9a). With the use of NH_3 gas as reactant gas for measuring by ICP-MS/MS, it was confirmed that Zr-93 (Fig. A11-9b) can be efficiently separated from the isobars (Nb-93, Mo-93), Ag-107 from the isobar (Pd-107), and Te-126 from the isobar (Sn-126).

The workload of analytical technicians is high for chemical separation of analytical samples, causing errors in the recovery rate of the nuclides depending on the analytical technician. Therefore, an automatic solid-phase extraction device with small errors in the recovery rate of the nuclides was developed for solid-phase extraction separation of Zr-93, Mo-93, Pd-107, Sn-126. The separation process of Zr-93 and Mo-93, which are difficult-to-measure nuclides, was applied to the automatic solid-phase extraction device. As a result of optimizing the dipping rate in each step, the recovery rate of Zr and Mo, and the mix rate of the isobar (Nb-93) were almost the same as those of analytical technicians. In addition, the working time using the automated solid-phase extraction system was reduced to 60 to 70 minutes (120 min by analytical technician).



a. Separation process for measuring Zr-93/Mo-93 (1 g concrete dissolved sample)

b. Nb/Zr separation by NH_3 gas

Fig. A11-9 Examination of analytical methods using ICP-MS/MS[149]

4.2.1.1.3 Development of a chemically reasonable separation process for radiometry

For β-rays and characteristic X-rays (Cl-36, Ca-41, Ni-63, Sr-90), for which an analytical method using ICP-MS/MS is not appropriate in terms of detection sensitivity, etc., radiometry is used for analysis. Among these nuclides, a process for Ca-41, Ni-63 and Sr-90, which have the potential to economically streamline a series of separation processes, was developed. As a result of applying this process to the mock rubble, a good recovery rate was obtained. It has high separation performance for Cs-137 and Sr-90, which coexist in large quantities in solid waste, and the working time is 30 hours (half the conventional separation process). It was also found that the amount of highly corrosive HCl used was greatly reduced, resulting in reducing the load on the system.

4.2.1.1.4 Development of a standard analytical method

Analytical samples from solid waste vary in shape and chemical composition, requiring validation of analytical methods for multiple nuclides for each analytical sample according to the nuclide. Therefore, requirements by analytical step (dissolution, separation and measurement) were clarified, and then a standard analytical method was developed for validating all the analytical methods that can handle different nuclides by verifying each analytical step. The validation conditions of the standard analysis method using ICP-MS/MS were examined for mock concrete to provide the specific evaluation items (e.g., repeatability, influence of different sample properties, evaluation of decontamination factors for interfering radionuclides, influence of measurement sample matrix, etc.) by analytical procedure. As a result of validating the analytical flows for Zr-93/Mo-93, Pd-107/Sn-126 and Ca-41/Ni-63/Sr-90 analyzed by radiometry, it was confirmed that the conditions qualified for the standard analysis method were satisfied. In addition, as a result of evaluating the lower detection limit in a method using mock concrete in each analysis flow, it was confirmed that the value was considerably lower than the upper limit of the concentration in the case of trench disposal. Based on the above results, a quality assurance manual (draft) was developed that takes into account validation of the standard analytical method using ICP-MS/MS and radiometry in accordance with the quality management system.

4.2.2 Development of characterization using statistical methods

4.2.2.1 Establishment of statistical inventory estimation methods

4.2.2.1.1 Examination of a waste contamination mechanism at the Fukushima Daiichi NPS (1F)

With an attempt to estimate the inventory necessary for waste classification, the focus was placed on air and contaminated water as nuclide migration media, and to divide into source term, migration process and immobilization of contamination (Fig. A11-10). Then, in terms of importance of waste classification, the contamination mechanism was examined for Unit 2 for which there is a large amount of analytical data on dismantled waste in its reactor building.

As a result of evaluating the transport ratio (the concentration ratio to Cs -137 was standardized by fuel composition), where the elements with similar chemical properties show similar values, it was estimated that the source term released through the air was H-3 and C-14 for damaged cladding tubes, other FP and TRU nuclides for damaged fuel, and Co-60 and Ni-63 for activated products. Since the radiation dose rate near the shield plug was high and contamination was significant, and the Cs-137 concentration in the ceiling was more than 2 orders of magnitude higher than that on the floor, it was presumed that the migration process through the air was diffusion from the shield plug which forms the boundary with the PCV. The transport ratios of major nuclides in the ceiling, wall, and rooftop were similar, and the nuclide composition did not change during spatial migration. In addition, the concentration of rubble on the east side was higher than that on the west and north sides of the rooftop. Therefore, the implication is the diffusion of the contaminated air due to the falling of the east-wall blowout panel. Imaging plate measurement confirmed that contaminants that traveled through the air and affixed to waste was spot contamination which would not be likely to penetrate into concrete or metal.

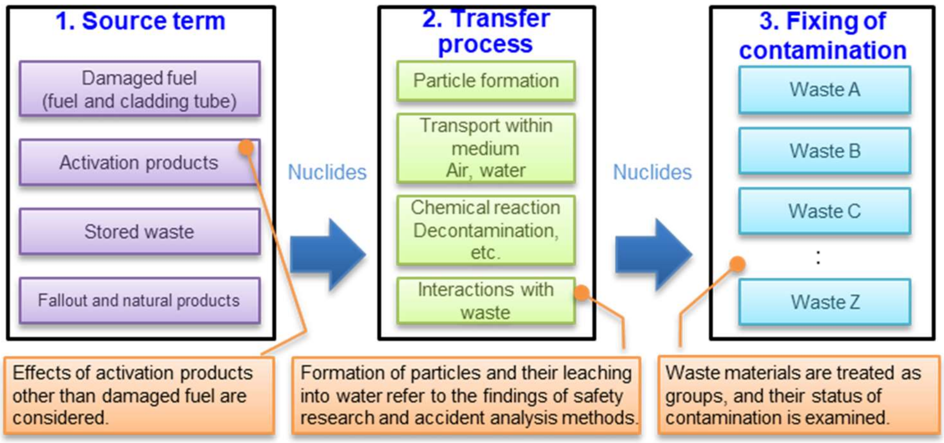


Fig. A11-10 Outline of examining a contamination mechanism of waste at 1F[149]

4.2.2.1.2 Examination of reducing uncertainty in nuclide migration parameters

With in-core fuel and structures in the buildings (activation products) as a source term based on the contamination mechanism examined, a nuclide relocation model was developed to calculate the migration process in which nuclides released into the air inside the buildings affix to waste through the air, and the other nuclide migration process in which nuclides migrate from fuel debris and affix to sorption vessels through the stagnant water (Fig. A11-11). For nuclide migration parameters such as the rate of release to outside the buildings, the distribution of the transport ratios was analyzed by Bayesian statistics based on the analytical data of migration to the stagnant water and to inside/outside the buildings. Then, by expressing the uncertainty of the parameters (mean and standard deviation) quantitatively, the migration ratio was set by the probability distribution (lognormal distribution). The accuracy of the parameters was improved by accumulating analytical data. Some analytical data on ALPS-sorbent was obtained and it demonstrated a good match with the values estimated by the statistical inventory method.

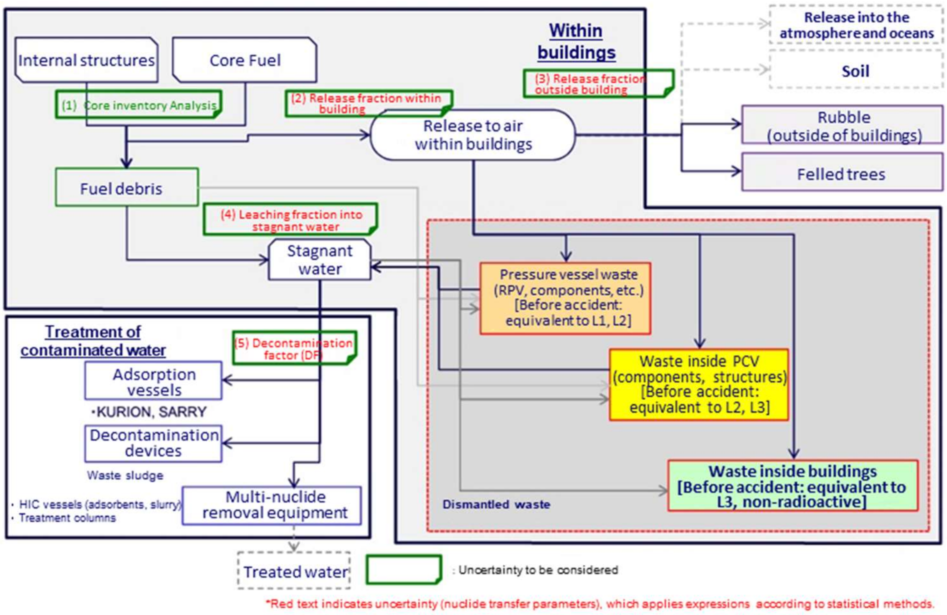


Fig. A11-11 Radionuclide migration model to calculate migration to waste[149]

4.2.2.2 Consideration of analysis planning using statistical methods

In reference to the Data Quality Objectives (hereinafter referred to as the "DQO") process[150] used by the National Nuclear Laboratory in the UK, analytical planning methods (combination of the DQO process and Bayesian estimation method), which evaluates the number of analytical samples required for the goal established by statistical methods according to waste properties, were tested for sludge generated by decontamination devices and dismantled waste (concrete). In a trial run for the sludge generated by decontamination devices, it was assumed that the sludge was disposed of after dehydration processing, and sampled during this processing. As a result of calculating the probability of remaining below the standard concentration assumed at the time of disposal for Co-60, Sr-90, and Cs-137, it was presumed that the assumed criteria were met at a probability of 90% or higher by analysis of four or more samples.

As described above, based on the statistical inventory estimates and the safety assessment on disposal, the objective of analysis could be specified by waste to obtain the number of analysis points, and the effectiveness of the analysis planning method using statistical methods was confirmed.

4.2.2.3 Establishment of an analysis database management system

An analysis database management system called FRAnDLi (Fukushima Daiichi Radwaste Analytical Data Library) has been developed, which contains and discloses the analytical data on solid waste obtained from the Project of Decommissioning and Contaminated Water Management

[150] DQO process: Process developed by the US Environmental Protection Agency (US EPA) for planning the sampling of analytical samples for decision-making. A sampling plan will be developed in seven steps. Planning and analysis are repeated as necessary to achieve the objectives. This process can be applied to various problems that can be solved by analysis.

as well as the information on waste-related analytical data provided by TEPCO (sample information (type, sampling location, date/time, etc.), analytical values of radioactivity concentration)). The system is utilized to review analysis strategies, and is widely used with over 1000 monthly accesses.

4.2.3 Summary

The following results were obtained from activities aimed at “clarifying radiological analysis and evaluation methods for characterization” as a specific goal for providing Technical Prospects.

- An efficient analytical method with the required analytical accuracy was established. The established analysis method is expected to be made into a manual in FY 2021.
- The technical applicability of the analysis methods for automation/acceleration (sampling technique for analytical samples, analytical method using ICP-MS/MS, automation technology for chemical separation, separation process for radiation measurement) was confirmed, and R&D results will be incorporated into Okuma Building #1.
- FRAnDLi that contains information on solid waste properties has been developed, leading to establishment of a statistical inventory estimation method using the analytical data collected. In addition, examination of an analysis plan according to the collected waste properties, etc. became possible.

As described above, analytical and evaluation methods have been developed for efficient characterization, including the development of methods for segregating and reasonably analyzing substances by mass, the rationalization and automation of separation processes, and the development of statistical methods for estimating the inventory of solid waste with less analytical data. In the future, the inventory estimation of solid waste necessary for disposal safety assessment, etc. will be promoted by the statistical inventory estimation method, while reducing the uncertainty in nuclide migration parameters, etc. by sample analysis using the developed analytical methods. For solid waste that can be adequately sampled and evaluated, the statistical inventory estimates will be replaced with the inventory calculated from analytical data.

4.3 Establishment of methods to reasonably select safe processing/disposal methods for solid waste

In selecting processing/disposal methods in a reasonable manner, based on the waste properties, an appropriate combination of processing (waste form) and disposal (disposal facility) methods should be clarified so that the risk of buried solid waste to the public and environment can be maintained sufficiently low in the future.

In the case of solid waste from a normal reactor, its properties can be estimated to some extent by the previous findings (data) and analytical methods. Accordingly, the appropriate combination of processing (waste form) and disposal (disposal facility) methods can sufficiently reduce the risk to avoid a significant impact on the public and surrounding environment.

Even in the case of solid waste from the Fukushima Daiichi NPS, molten nuclear fuel is a major source of contamination, and the radioactivity concentration does not exceed that of spent fuel.

Therefore, the risk can be sufficiently reduced by understanding the overall picture of the target solid waste (properties such as nuclide composition, activity concentration by waste, waste volume), and selecting a proper combination of processing (waste form) and disposal (disposal facility) methods, while utilizing the experience and knowledge on radioactive waste processing/disposal accumulated in domestic and overseas.

However, the overall picture of solid waste to be disposed of, including that which will be generated, will gradually become clear as the progress and plans for fuel debris retrieval, contaminated water management, and other decommissioning work are clarified. Therefore, it is necessary to repeatedly examine processing/disposal methods and safety assessments, starting from the waste for which properties have been clarified; to give consideration to making processing/disposal methods more appropriate; and to accumulate knowledge to consider safe and reasonable processing/disposal methods for diverse solid waste collectively. Aiming for safer and not extremely conservative storage of waste with high mobility such as slurry waste, processing (preceding processing) for stabilization/immobilization may be required before determining the disposal method (disposal facility). Reprocessing would be necessary if the specifications of the waste form even after preceding processing did not conform to those required by the disposal method (disposal facility) to be determined. Therefore, in order to minimize such a possibility, a selection method for the preceding processing method with disposal in mind is needed.

As mentioned below, study on an appropriate combination of processing and disposal methods, or preceding processing methods, is considered for the waste for which properties have been identified to some extent.

- Establish several feasible disposal methods suitable for waste characteristics (without specifying the feature of facilities such as their locations and sizes)
- In parallel, establish several processing methods suitable for waste characteristics to be considered, and set the specifications of the waste form after applying each processing method.
- Evaluate the safety of several selected disposal methods based on the specifications of the waste form after processing to verify whether risk to the public and environment is sufficiently low, and to consider more effective processing/disposal methods based on the evaluation results.

The above examination steps are repeated to narrow down disposal methods and specifications for the waste form after processing. Clarifying the overall characteristics of solid waste concurrently with characterization helps identifying an appropriate combination of processing/disposal. When preceding processing becomes necessary, candidate processing methods will be selected in consideration of the status of examination and open issues at that point.

It is also important to consider the period during which pre-disposal management is to be implemented, taking into account the risk reduction during that period, and examining necessary and feasible technologies. Since storage is important to provide flexibility to respond to the progress of processing/disposal, and to reduce radiation exposure of workers due to the decay of the radioactive materials, it is important to consider storage strategies as part of this examination process.

A series of these studies is represented as a flowchart shown in Fig. A11-12. In establishing the methods as a series of methods for selecting processing/disposal methods in a reasonable manner, technical knowledge and evaluation methods necessary for these studies (establishment of processing technology and conditioned waste specifications appropriate for waste, safe, reasonable and feasible disposal methods, disposal safety assessment) have been developed through research/development (verification of applicability of processing methods using engineering-scale test equipment conducted mainly on the secondary waste generated by water treatment, establishment of safe, reasonable and feasible disposal methods based on the waste properties and applicable processing technologies, and development of safety assessment methods) by the Project of Decommissioning and Contaminated Water Management[149].

The results of these studies are shown below.

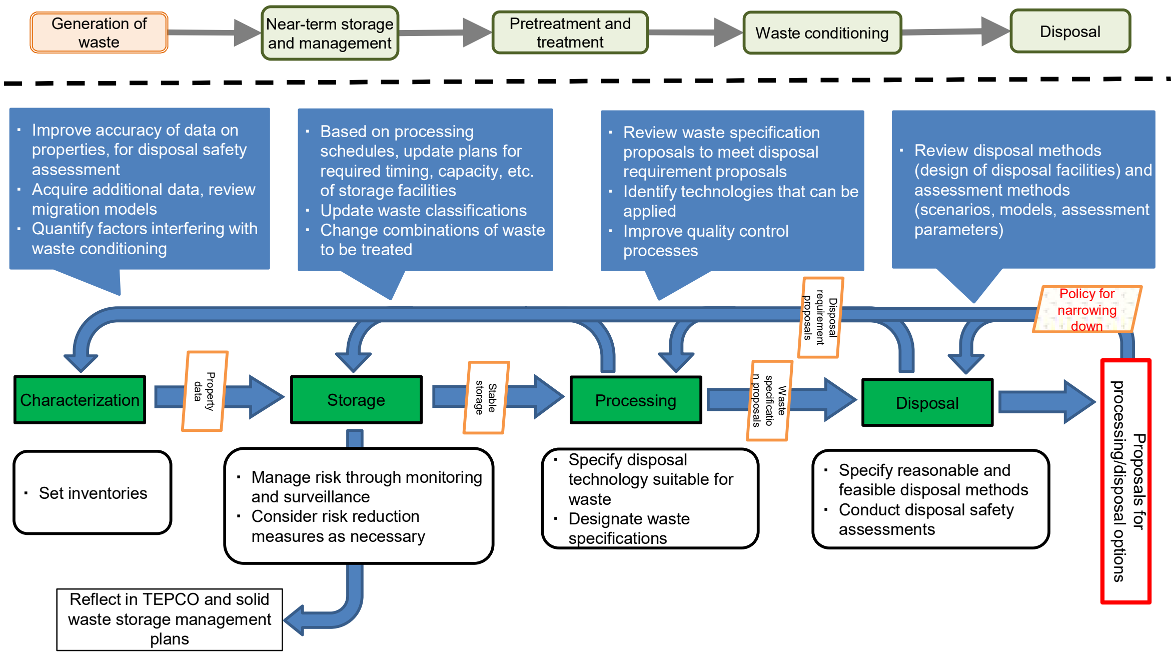


Fig.A11-12 Development of methods for reasonably selecting safe processing/disposal methods of solid waste

4.3.1 Clarification of issues and measures for storage and management

As some solid waste materials are high-level waste that contain water or to which water is adhered, the hydrogen generation by radiolysis is of concern during storage. Therefore, regarding the storage of high-level waste for which consideration of hydrogen management is required, issues and measures with respect to the approach to safety were examined in the Project of Decommissioning and Contaminated Water Management and other programs.

High-level waste includes waste generated by fuel debris retrieval (hereinafter referred to as “waste from fuel debris retrieval”) and secondary waste generated by water treatment. The secondary waste generated by water treatment is temporarily stored in box culverts, etc., but storage facilities will be established according to TEPCO's storage plan to eliminate temporary storage as much as possible. For this purpose, the Project mainly targets the waste from fuel debris retrieval for conducting the following studies on storage. The waste from fuel debris retrieval was temporarily divided into removed objects and other waste (that retains its original shape and has no possibility of criticality). The removed objects were further divided into those above the fuel loading position and the external structures of the pedestal.

4.3.1.1 Safety function requirements during storage of high-level waste

As a result of examining hazards from major accidents and safety function requirements anticipated during storage of high-level waste (containment, transfer and storage), it was determined that the most important safety function requirement was prevention of dispersion of radioactive materials (ensuring airtightness) and hydrogen release (ensuring air permeability). Survey results of domestic and overseas storage cases of high-level waste showed that Zion and TMI-2 nuclear power plants in US used dry storage, Paks-2 nuclear power plants in Hungary and those in Japan (shroud replacement) used wet storage. The TMI-2 and Paks-2 nuclear power plants used storage containers with filter vents. Since wet storage requires securing a pool for storage and pool water management, dry storage was selected for high-level waste and examined for storage containers with filter vents.

4.3.1.2 Examination of high-level waste storage methods

4.3.1.2.1 Concept of high-level waste storage

It is assumed that the high-level waste removed from the reactor buildings is stored in inner containers in the reactor buildings, storage containers in storage container handling cells in the expanded building, and then in shipping casks in delivery cells (Fig. A11-13). The assumption is that the shipping casks are transferred to the temporary storage facility on site, and the waste is taken out from the shipping casks for drying in the drying area, and then brought into a dry storage condition in the storage area.

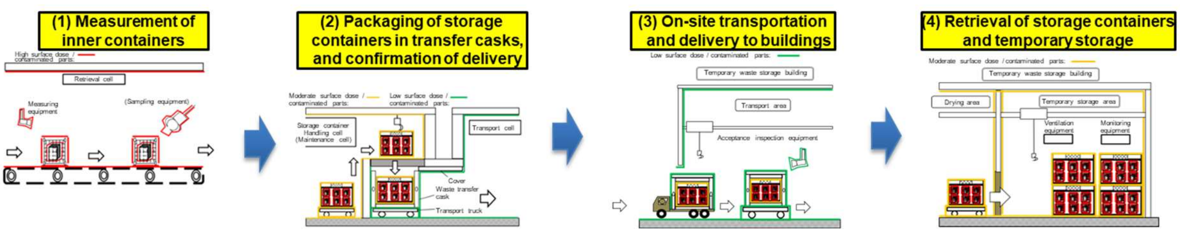


Fig. A11-13 Example flow of handling high-level waste[149]

Assuming the timing of implementing hydrogen management for high-level waste, i.e., during on-site transportation and at the time of temporary storage, consideration was given to hydrogen management for each case. Possible hydrogen management during on-site transportation include installation of vent filters in storage containers, sealing of shipping casks and installation of catalysts inside shipping casks. The vent filter discharges hydrogen generated in the storage container to the shipping cask and maintains the hydrogen concentration inside the shipping cask below the lower limit of explosion (< 4 vol%) to prevent dispersion of radioactive materials. The shipping cask is provided with a sufficient void space to maintain the hydrogen concentration below the lower limit of explosion during on-site transportation.

Hydrogen management during temporary storage includes vent filters installed in storage containers during transportation and a ventilation system in temporary storage facilities. The vent filter discharges hydrogen generated in the storage container to the temporary storage facility and maintains the hydrogen concentration inside the storage container below the lower limit of explosion to prevent dispersion of radioactive materials. The hydrogen concentration inside the temporary storage facility should be maintained below the lower limit of explosion with the ventilation system. As a measure for preventing corrosion of storage containers, drying treatment is applied to free water adhered to the surface of high-level waste before temporary storage.

4.3.1.3 Examination of container and storage facility requirements

4.3.1.3.1 Preconditions for the properties and storage of high-level waste

Although different types of high-level waste to be retrieved along with debris retrieval are taken into consideration, it is assumed that the maximum radiation dose of high-level waste to be retrieved eventually is 400 Sv/h for more than 1,500 tons of contaminated metal and concrete, and its surface density (in the reactor well) is 1.2×10^8 Bq/cm^2. It is assumed that high-level waste, except for some large-sized waste, is cut and stored in inner containers. It is assumed that storage containers, each containing an inner container, are stored in shipping casks, and transported to temporary storage facilities on site. It is also assumed that the storage containers are then removed and stored in temporary storage facilities.

The surface radiation dose of the shipping cask during on-site transportation is assumed to be < 30 mSv/h in reference to on-site transportation at the Fukushima Daiichi NPS, and the shipping cask is assumed to be spill-resistant in terms of prevention of spread of α-contamination. As the conditions for acceptance in the temporary storage facilities, it is assumed that the surface radiation dose of the storage container is set to < 10 Sv/h and the weight (including shielding and contents) is set to < 7.5 tons in reference to the solid waste storage facility building #9. Based on the properties of high-level waste and conditions for on-site transportation and storage in temporary storage facilities, the provisional dose classification of high-level waste, requirements of acceptance in temporary storage facilities and shielding of the storage container are assumed as the basis for considering storage methods.

4.3.1.3.2 Examination of reasonable container shape

The shapes of the inner container, storage container (storing an inner container) and shipping cask were examined based on the required functions for high-level waste containers (shape, material, containment and shielding performance, hydrogen management, operability, structural strength). It is assumed that the inner container will have a mesh structure for draining the removed objects. Depending on the debris retrieval method, large and small inner containers will be prepared. It is also assumed that one large inner container or several small inner containers can be stored in one storage container.

Storage containers (1.42 x 1.42 x 1.00 mm) were examined from the perspective of handling such as its surface radiation dose and weight including waste. According to the radiation dose level of the high-level waste, four types of shielding thicknesses were considered so that the surface radiation dose rate of the storage container could be maintained at < 10 Sv/h. The total weight was considered so as not to exceed the weight limit of the forklift.

The shipping cask is assumed to have a sealed structure to suppress the release of radioactive materials during transportation.

4.3.1.3.3 Examination of hydrogen management

Vent filters should be installed in storage containers to maintain the hydrogen concentration in the storage container below 4 vol%. The shipping cask should be spill-resistant to prevent the spread of contamination during on-site transportation, and its shape will be determined from the void ratio, etc. for hydrogen management. The essential functions required for vent filters can be divided into the aspects of safety and handleability. In terms of safety, the functions required for vent filters include those related to containment, criticality, shielding and hydrogen explosion. In terms of handleability, however, those related to remote operability, drying processing and condensation during storage are required. The essential functions for vent filters and their requirements were summarized in terms of safety and handleability. In terms of safety, the functions required for vent filters include those related to containment, criticality, shielding and hydrogen explosion. The essential functions for vent filters in terms of handleability are remote operability, drying treatment, and dew condensation during storage. At WIPP disposal sites in the US, the air flow rate, collection efficiency and hydrogen diffusion performance are set for each container as the minimum specifications for the vent filter as requirements for accepting waste containers.

4.3.1.3.4 Examination of drying treatment

Drying treatment conditions (water adhered to waste, drying area) and drying treatment methods, and nozzle installation and air supply/exhaust methods were examined. Based on the survey results of existing drying technology, requirements of drying treatment systems (applicability, treatment for temporary storage, applicability to the retrieval flow), influence during drying treatment, technical requirements for drying, container-by-container drying treatment and operating indexes were examined and summarized.

In order to prevent contamination in temporary storage facilities, storage container lids shall not be opened. Therefore, a drying treatment method for high-level waste in which a nozzle is installed on the storage container and connected to a drying device was assumed. After completion of the drying processing, the nozzle portion may be contaminated, and thus its decontamination is required. Drying processing methods include the convection (static) method and conduction (static) method. With the convection (static) method with connected nozzle, a blower and a heater are installed on the air supply side. Hot air (dehumidified air or dry nitrogen) is supplied to heat the high-level waste, and exhausted by the blower on the exhaust side. With the conduction (static) method with connected nozzle, the bottom and side surface of the storage container are heated by a heater, and exhaust is exhausted out by a blower on the exhaust side or vacuum pump. A hot air supply nozzle was installed on the body of the storage container, and the blow-out end was installed on the lower part of the container. The hot air supply provides dry air or dry nitrogen, and exhaust methods include normal pressure exhaust and depressurized exhaust.

4.3.1.4 Examination of measurement methods during high-level waste storage

Measurement and inspection items necessary during high-level waste storage were examined. Confirmation items that may be measured and inspected during storage of high-level waste (hereinafter referred to as "measurement/inspection items") depend on the timing in the handling flow of high-level waste.

Measurement/inspection items during storage in inner containers are assumed to be the basic properties of waste, physical properties of waste, radiochemical properties, calorific properties, and hydrogen generation characteristics. Measurement/inspection items during storage in storage containers are assumed to be appearance, weight, filling rate, surface radiation dose rate, surface contamination, surface temperature, hydrogen generation rate, and nuclide release volume. Measurement/inspection items during storage in shipping casks are assumed to be appearance, weight, surface radiation dose rate, surface contamination, surface temperature, and sealing.

The priority of the measurement/inspection items was examined in view of the purpose of measurement, difficulty of measurement, estimability from measured values, avoidability by operation, evaluation and prior investigation. The priority measurement and inspection items are considered to be weight, surface contamination, and surface radiation dose. The priority measurement/inspection items when estimation by the surface radiation dose is not feasible are considered to be calorific values, surface temperature, radioactivity, amount of nuclear materials, and hydrogen generation rate. Measurement/inspection items and timing in the handling flow of high-level waste were examined (Fig. A11-14). It is assumed that high-level waste is separated from fuel debris when it is removed from reactor buildings.

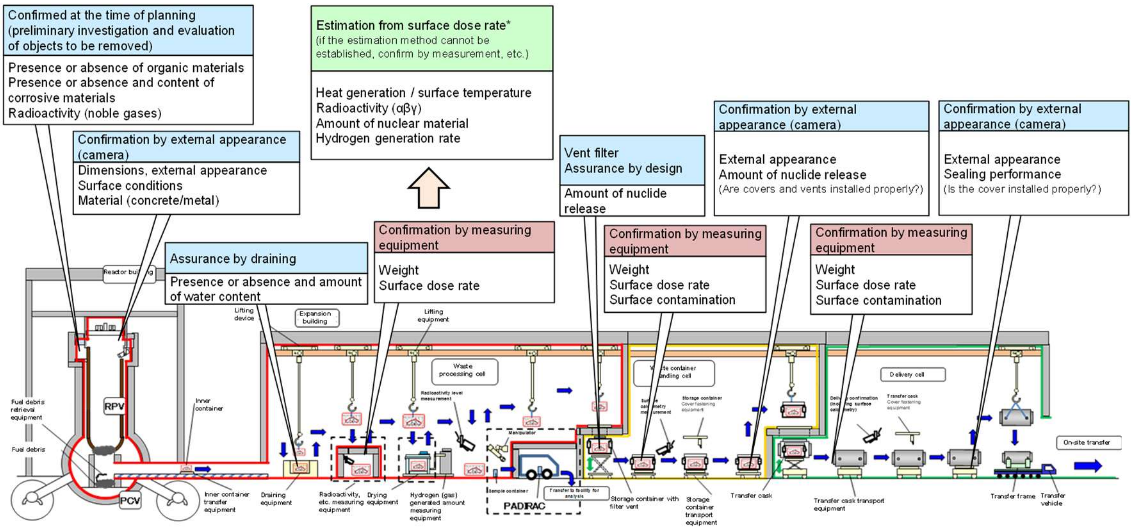


Fig. A11-14 Examination example on measurement/inspection items and timing in the handling flow of high-level waste[149]

4.3.1.5 Examination on long-term storage of high-level waste

Defects and countermeasures assumed during prolonged temporary storage of high-level waste, other than hydrogen management, were examined in reference to countermeasures in Japan and abroad. The following defects are assumed during prolonged storage: A. Container breach due to corrosion, B. Prolonged generation of hydrogen, C. Occurrence of external events, and D. Deterioration of consumables. Possible measures include use of corrosion-resistant materials, drying of waste, temperature and humidity control for A; solidification process, gas venting, installation of hydrogen storage materials, etc., and monitoring of gas generation for B; protection of structures, fall-prevention, and use of high-strength containers for C; and monitoring/inspection, reduction of consumables, and securing of containment performance for D. Since waste other than high-level waste is stored for a long period of time, further investigation of the actual problems and countermeasures is planned in FY 2021.

During prolonged storage of high-level waste in storage containers, it is anticipated that the hydrogen diffusion performance will not be satisfied due to corrosion or deterioration of the vent filter, and that the contents will tend to scatter. As a countermeasure, it is necessary to verify the progression of corrosion by changing to corrosion-resistant materials and using test pieces. It is necessary to maintain the vent filter functions for prolonged storage of high-level waste, therefore, degradation factors of the vent filter will be investigated in FY 2021, and methods for verifying the vent filter functions will be examined..

4.3.1.6 Summary

As a result of the above activities, clarification has been achieved as described below on the issues and measures concerning storage and management of solid waste before being conditioned.

- By examining storage methods, the handling flow up to temporary storage of high-level waste and the timing for implementing hydrogen management have been examined. Hereafter, the handling flow from temporary storage to disposal should be considered. It has become possible to consider hydrogen management, but it is still necessary to examine the evaluation conditions of the hydrogen generation rate and validate the evaluation results.

- By examining the requirements of containers and storage facilities, the containers and drying processing systems required for hydrogen management were considered. Verification of the drying processing system and its drying performance is required. In accordance with examination of the fuel debris retrieval method, it is necessary to review the shape of the container, shielding properties, site for applying drying treatment, measurement items and timing, etc.
- The possibility of the occurrence of an event of concern other than hydrogen management, such as leakage and dispersion due to corrosion of storage containers, has been evaluated and measures have been proposed. Since it is important to maintain the vent filter functions for the prolonged storage of high-level waste, the degradation factors of the vent filter will be investigated in FY 2021, and methods for verifying corrosion and degradation of the vent filter will be examined.

4.3.2 Clarification of processing technology with disposal in mind

The following efforts have been made with the goal of “clarifying processing technology which practical application could be expected for stabilization and immobilization, considering disposal for several important waste streams such as secondary waste generated by water treatment”.

- Of the solid waste, processing technology with the prospect of practical application will be extracted at least for ALPS slurry and sludge generated by decontamination devices.
- The prospect of not only the initial performance of waste packages but also the long-term stability after disposal based on the mechanism will be shown.

Then, processing technologies with possible applicability were investigated, taking into account the types and properties of secondary waste generated by water treatment. Of the technologies already in practical use in Japan and abroad, for those that have reached the level of practical application and those under research/development, the applicability of low-temperature solidification processing technology (cement, AAM (alkali-activated materials)) and high-temperature solidification processing technology (CCIM (cold crucible induction melting), In-Can, GeoMelt ICV) to secondary waste generated by water treatment was examined (Table A11-1).

Table A11-1 Processing technology overview[149,151,152,153]

	Classification		Technology Name	Overview
Low-temperature processing	Cement solidification	--	OPC	Solidification of waste by utilizing the setting properties of cement hydration reactions. Includes in-drum and out-drum methods.
	AAM solidification	--	AAM solidification	In AAM, a high-concentration alkali solution (mixture of NaOH, KOH, and sodium silicate) is added to an inorganic powder (metakaolin, etc.) containing an alumina silicate, and is solidified by the dissolution of raw materials and polycondensation reactions. The solidified waste has an amorphous network structure.
High-temperature processing	Vitrification	Downflow method	Cold crucible induction melting (CCIM)	Glass frit is melted by ignition metals, and waste and glass raw materials are heated by high frequency induction. The molten materials flow downward and are solidified by cooling them in vessels. The mixture of molten materials is accelerated by methods such as bubbling. By water-cooling the furnace wall, its corrosion is suppressed by a skull layer formed between the molten materials and the furnace wall. Organic waste is thermally decomposed by supplying oxygen from the upper part of the furnace.
		InCan method	Joule heating (GeoMelt)	Waste and glass raw materials are supplied into a container and an electric current is applied between graphite electrodes to gradually melt the materials in the vicinity of the electrodes. After melting, they are cooled in containers and solidified. The molten materials are agitated by thermal convection.
		InCan method	External heating (InCan)	Waste and glass raw materials are supplied into a container, which is heated by an external heater. After the materials are melted, they are cooled in melting vessels and solidified. The molten materials are agitated by thermal convection generated in vertically-divided heating regions.

OPC: Ordinary Portland Cement

4.3.2.1 Examination of applicability of low-temperature solidification processing to secondary waste generated by water treatment

4.3.2.1.1 Examination of applicability of cement/AAM solidification processing technology

The applicability of low-temperature solidification processing technology to secondary waste generated by water treatment was examined in small-scale testing (Table A11-2).

Table A11-2 Applicability of low-temperature solidification processing[149]

Waste			Volume Reduction and Pretreatment Technology	Cement solidification	AAM solidification
Secondary waste generated by water treatment	Waste sludge		Dehydration or drying + heat treatment or melting	○	○
	Carbonate slurry		Dehydration or drying	◎	○
	Iron coprecipitation slurry		Dehydration or drying	○	◎
	Inorganic adsorbents	Zeolite	Unnecessary	○	○
		Silicotitanate	Unnecessary	◎	○
		Titanate	Unnecessary	◎	◎
		Titanium oxide	Unnecessary	◎	○
		Sb adsorbents	Unnecessary	◎	○
	Organic adsorbents	Chelate resin	Incineration or gasification	◎	○
		Resin-based adsorbent	Incineration or gasification	◎	◎
		Cs colloid filter	Incineration or gasification	○	○
		Sr colloid filter	Incineration or gasification	◎	○
	Ferrocyanide		Incineration or gasification	◎	○

◎: Solidification technology is highly applicable to accident waste, ○: Applicable, ×: Not applicable

Next, solidified cement and solidified AAM were prepared for carbonate slurry as ALPS slurry (solidified waste content (WL) of 30 wt%) and iron coprecipitated slurry (solidified waste WL of 20%) to obtain their characteristic data (fluidity, caking, compressive strength, void ratio, etc.) and

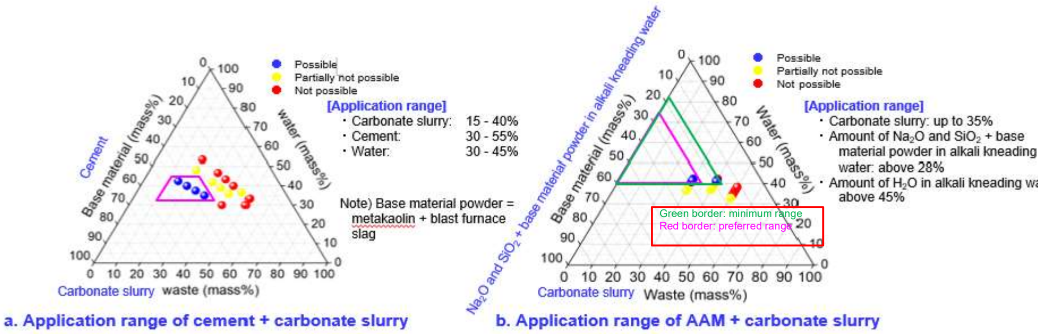
151 IHI Corporation, The Summary results of subsidy program for the Project of Decommissioning and Contaminated Water Management, "R&D for processing/disposal of solid waste" (FY 2019 to 2020) in the FY2018 Supplementary Budget, being ready for release in https://dccc-program.jp/category/result

152 Orano ATOX D&D SOLUTIONS, The Summary results of subsidy program for the Project of Decommissioning and Contaminated Water Management, "(R&D for processing/disposal of solid waste) Applicability evaluation of the In-can vitrification process to Fukushima waste (FY 2019 to 2020) in the FY2018 Supplementary Budget, being ready for release in https://dccc-program.jp/category/result

153 Kurion Japan K.K, Subsidy program for the Project of Decommissioning and Contaminated Water Management, "(R&D for processing/disposal of solid waste) Project of HOLISTIC EVALUATION OF GEOMELT ICV™ FOR TREATMENT OF 1F WATER TREATMENT SECONDARY WASTES" in the FY2018 Supplementary Budget, being ready for release in https://dccc-program.jp/category/result)

examine the applicability of low-temperature solidification processing technology based on solidified package performance and operability as evaluation axes (Fig. A11-15).

In FY 2021, application testing of low-temperature solidification processing technology using dehydrated ALPS slurry to homolytic/homogeneous solidification and filling/solidification was performed on an engineering scale (200 L), and its practical application will be verified.



Applicability of low-temperature solidification processing technology to carbo

Fig. A11-15 nate slurry[149]

4.3.2.1.2 Investigation of the possibility of expanding applicability by using special cement

Cement with a high volume of blast furnace slag (CB90, normal cement : blast furnace slag = 90% : 10%) and alumina cement containing fly ash (AF20, alumina cement : fly ash = 80% : 20%) were selected as special cements with potential applicability in terms of high fluidity during solidification (feasibility of solidification process and suppression of hydrogen generation) and high heat resistance (increased radioactive inventory and WL) in accordance with the characteristics of the secondary waste generated by water treatment that contains chemical components (sodium carbonate, boric acid, sodium chloride) which adversely affect cement solidification. While CB can be applied to sodium carbonate mixed waste, AF may have superior heat resistance and fluidity. The influence of their chemical composition is different from that of OPC.

The solidification characteristics of CB and AF according to their chemical composition were tested. Then, the applicability of ALPS slurry was examined based on fluidity, setting time, breeding and compressive strength as evaluation criteria. The maximum WL for carbonate slurry was 25% for CB90 and 35% for AF20, and the WL for iron coprecipitated slurry was 25% for both CB90 and AF20. The effectiveness of CB90 was confirmed for sodium carbonate and boric acid which affected the setting time.

4.3.2.1.3 Examination of inspection methods on solidification possibilities by low-temperature solidification processing

Physical properties (shape and size, particle size distribution, density, bulk density, water content, chemical composition, radiation dose, properties when liquid is added) that affect the feasibility of low-temperature solidification processing (influence of processes and solidified waste) were summarized and mock materials with these properties (silica sand, fine silica powder, calcium carbonate, sodium carbonate, magnesium sulfate, iron oxyhydroxide, barium sulfate, metallic aluminum, coal ash, montmorillonite) were selected. Using these mock materials, a two-step screening method for inspecting the potential of waste solidification in a small scale before low-temperature solidification processing of the secondary waste generated by water treatment was developed. In the primary screening, the waste properties are measured with a 50 mL centrifuge tube to evaluate the feasibility of the process as brief assessment. In the secondary screening, the properties during solidification processing are measured with a 2 cm square sample to evaluate the performance of the kneaded/mixed products and solidified waste. This screening method was tested using the screening method developed by using mock ALPS slurry, and it was confirmed that solidification by low-temperature solidification processing technology was possible.

4.3.2.2 Examination of applicability of high-temperature solidification processing technology

4.3.2.2.1 Examination of vitrification of secondary waste generated by water treatment by crucible testing

WL, for which high-temperature solidification processing of the secondary waste generated by water treatment is possible, was examined through crucible testing. (Table A11-3). In order to suppress volatilization of Cs generated during high-temperature processing of secondary waste generated by water treatment, examination was also made on WL for which solidification processing is possible at a low melting temperature. WL applicable to engineering-scale testing of high-temperature solidification processing technology was examined in crucible testing, and it was confirmed that each processing technology enables the secondary waste generated by water treatment to be melted and solidified at the set melting temperature (Table A11-4).

Table A11-3 High-temperature solidification processing test in crucibles

Waste	No.	WL(wt%)	Melting Temperature (°C)	Glass Type
Silicotitanate	IC-01	52	1200, 1100	Borosilicic acid
Silicotitanate	AC-04	30	850→1100	Borosilicic acid
Silicotitanate (embedded glass)	AC-05	70	900	Borosilicic acid
AREVA sludge	IC-02	6	1200, 1100	Borosilicic acid
AREVA sludge	AC-06	30	900	Borosilicic acid
AREVA sludge	IC-07	23	1200, 1150	Iron phosphate
Zeolite	IC-05	62	1050	Borosilicic acid
Zeolite	AC-03	70	850→1100	Borosilicic acid
Iron coprecipitation slurry	IC-06	47	1200, 1150	Iron phosphate
Iron coprecipitation slurry	AC-02	30	850→1100	Boroclinic acid
Carbonate slurry	AC-01	50	850→1100	Borosilicic acid
Ion-exchange resin	AC-07	30	900	Borosilicic acid
Resin-based adsorbent	IC-03	20	1200, 1100	Borosilicic acid
Activated carbon	IC-04	5	1200, 1100	Borosilicic acid
Activated carbon	AC-08	30	900	Borosilicic acid

Table A11-4 Filling rate of waste in engineering-scale testing (WL) [151,152,153]

No.	Waste	WL (wt%)	Melting Temperature (°C)	Glass Type
IE-01	Carbonate slurry	20	1200, 1100	Borosilicic acid
IE-02	Iron coprecipitation slurry	35	1200, 1100	Borosilicic acid
IE-03	Zeolite, carbonate slurry	72	950→1000	Borosilicic acid
IE-04	Zeolite, AREVA sludge	67	1050	Borosilicic acid
AE-01	Zeolite, silicotitanate, carbonate slurry, iron coprecipitation slurry, sand	80	850→1100	Borosilicic acid
AE-02	Zeolite, silicotitanate, sand	80	850→1100	Borosilicic acid
AE-03	Carbonate slurry, iron coprecipitation slurry	41	850→1100	Borosilicic acid
KE-01	Zeolite, carbonate slurry, iron coprecipitation slurry	82	700→1250(1415)	Borosilicic acid
KE-02	Zeolite, silicotitanate	83	700→1250(1535)	Borosilicic acid
KE-03	Zeolite, AREVA sludge	70	700→1250(1411)	Borosilicic acid
KE-04	Soil, carbonate slurry	80	-- (1600)	Borosilicic acid

( ) : Melting temperature during engineering scale test

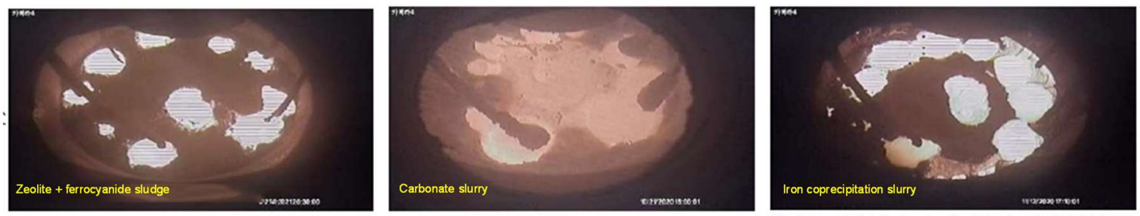
4.3.2.2.2 Engineering-scale testing of high-temperature solidification processing technology

Fig. A11-16 shows the condition in the melting furnace during engineering-scale testing with single and mixed waste in CCIM. Characteristics data of the light-emitting, circular section at the center and the periphery of the molten metal surface was obtained using an engineering-scale system (CCIM, In-Can, GeoMelt ICV) for the high-temperature solidification processing technology, to produce solidified waste through high-temperature processing of the secondary waste generated by water treatment. Due to bubbling (the molten metal is stirred by air supplied to the inside), the circulation of molten metal is active in this section. The light-colored section between circular light-emitting sections at the center and the periphery is a calcinated layer.

Testing using In-Can was conducted[152] to produce solidified waste. Mock waste and glass additives were placed in a can before melting, and other mock waste and glass additives were added while heating at 1100°C. Fig. A11-17 shows the cross-sectional photo of the solidified waste produced[153] by In-Can engineering-scale testing.

A test using GeoMelt ICV was conducted to produce solidified waste. Mock waste was placed in a melting vessel before melting, and heated at approx. 1400°C from the bottom. After the mock waste was left in the vessel for approx. 1.5 to 15 hours, top-off frit (low-melting-point glass material that does not contain mock waste) was added for approx. 0.5 hour. Fig. A11-18 shows the GeoMelt ICV engineering-scale testing condition.

In FY 2021, engineering-scale testing using dehydrated carbonate slurry, examination of supply methods, promotion of high-filling of carbonate slurry, suppression volatilization of Cs, and test restarting of the furnace after shutdown are planned. The temperature distribution in the furnace and the Cs volatilization mechanism will also be studied.



(Perimeter: 10 L/min, Center (1): 10 L/min, (2): 10 L/min) (Perimeter: 15 L/min, Center (1): 50 L/min, (2): 100 L/min) (Perimeter: 10 L/min, Center (1): 10 L/min, (2): 10 L/min)

Fig. A11-16 Engineering-scale testing condition in CCIM (4.5 hours after placing in service)

a. Solidified products produced from all mixtures b. Solidified products produced from Cs-rich waste c. Solidified products produced from HIC slurry

Fig. A11-17 Condition of solidified waste produced by In-Can engineering-scale testing

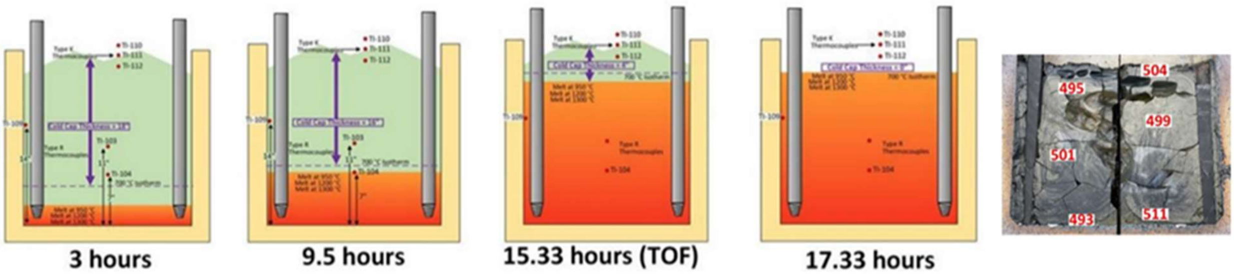


Fig. A11-18 GeoMelt ICV engineering-scale testing and condition of solidified waste

4.3.2.3 Examination on stability of waste packages

In order to verify stability at disposal of the solidified waste produced by the selected processing technology, leachability of the solidified waste, deterioration of the solidified waste after low-temperature solidification processing caused by temperature, and the influence of the deterioration of the solidified waste over the long term were examined[149].

4.3.2.3.1 Examination of leachability of solidified waste

The leaching rate of the mock nuclide in the solidified waste obtained by low-temperature solidification processing of the ALPS slurry (evaluated with the leaching rate of all the mock nuclides leached as 100%) was determined from the concentration of ion species in the solidified waste and the amount of ion species leached from the solidified waste after soaking the solidified waste with a curing period of 28 days in pure water for 91 days (Table A11-5).

Table A11-5 Leaching rate of mock nuclides in solidified waste after low-temperature solidification processing[149]

Processing Technology	Solidified Material	Waste Loading Rate (wt%)		Leaching Rate (%)							
				Cs		Sr		Sn		Ce	
		Carbon	Iron	Carbon	Iron	Carbon	Iron	Carbon	Iron	Carbon	Iron
Cement	OPC	30	20	95	80	5	15	<0.1	0.1	--	--
AAM	M	30	20	23	12	0.5	<0.1	3-5	1-3	<1	<1
AAM	MB20	30	20	18	15	0.5	<0.1	3-5	1-3	<1	<1
AAM	MB40	30	20	27	20	0.5	<0.1	3-5	1-3	<1	<1

Carbon: Solidified carbonate slurry, iron: Solidified iron coprecipitated slurry
OPC: Ordinary Portland Cement, M: Metakaolin
MB20: Metakaolin + 20% of blast furnace slag, MB40: Metakaolin + 40% of blast furnace slag
"- -": No leaching

The MCC-1 test was conducted on the solidified waste produced by engineering-scale testing of the high-temperature solidification processing technology to calculate the leaching rate of the solidified waste (Table A11-6).

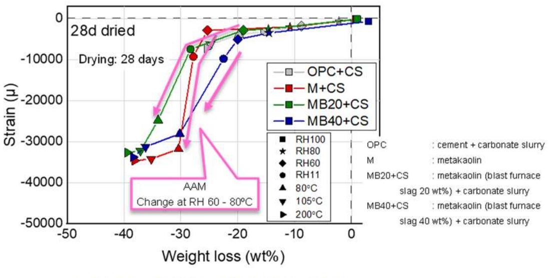
Table A11-6 Leaching rate of solidified waste produced by engineering-scale testing (MCC-1 test)[14]

Waste	Evaluation Period	B ($g/m^2/d$)	Na ($g/m^2/d$)
Zeolite, carbonate slurry, iron coprecipitation slurry, silicotitanate, sand	7-28d	1.4-3.2	--
Zeolite, silicotitanate, sand	7-28d	--	0.18-0.50
Zeolite, carbonate slurry, iron coprecipitation slurry	28-365d	--	0.012
Zeolite, silicotitanate	28-365d	--	0.014
Zeolite, waste sludge	28-365d	--	0.012

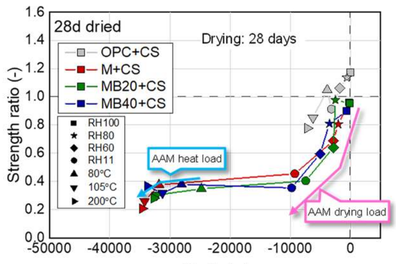
4.3.2.3.2 Examination on deterioration by temperature of solidified waste after low-temperature solidification processing

Temperature increase due to the decay heat of the nuclides contained in the solidified waste was examined[149]. The highest temperature of solidified cement (1×10^5 - 1×10^{10} Bq/cm^3) was evaluated in cylindrical and square containers, and by radiation transfer and thermal analyses. The result of calculating the calorific value using the measured value of the solidified AAM showed that the calorific value of the solidified AAM was lower than that of the solidified cement. The calorific value was calculated by several layout models, in which solidified waste containers were aggregated, to derive a relational expression between activity concentration and maximum temperature. The upper limit of activity concentration for the temperature limit of cement deterioration (60°C) was 6.8 × 199 Bq/cm^3. Evaluation of changes in temperature of the solidified carbonate slurry cement (6 m^3 of square container) at the maximum WL (30 wt%) using a single layout model based on the inventory (1×10^7 Bq/cm^3 : analyzed value) contained in it showed that the temperature rise was about 3°C and that the possibility of the performance of the solidified waste decreasing was low.

Next, the deterioration of solidified waste by changes in temperature of storage facilities was examined. Heating and drying load testing was conducted on the solidified waste after low-temperature solidification processing to assess the compressive strength and strain amount of the solidified waste. The mass depletion rate, shrinkage strain and compressive strength of the solidified waste by drying and heating were also assessed (Fig. A11-19). As a result, the solidified cement had higher drying/heat resistance than the solidified AAM, while the strain amount of the solidified AAM was extremely high. Although the solidified cement and solidified AAM satisfied the standard value of compressive strength (1.47 MPa), it was proven that their performance drastically decreased under high-temperature and drying conditions.



a. Mass reduction rate and shrinkage strain of solidified waste



b. Shrinkage strain and compressive strength ratio of solidified waste

Fig. A11-19 Heating and drying load testing on solidified carbonate slurry[149]

4.3.2.3.3 Examination of influencing factors on long-lasting deterioration of the solidified waste after low-temperature solidification processing

For deterioration of the solidified carbonate slurry cement, the change of its mineral phase due to WL or temperature was evaluated by thermodynamic equilibrium calculation[149]. Based on the chemical composition of cement hydrate and solidified waste, carbonate minerals and chloride minerals to be generated were extracted from a thermodynamic database. The calculated results could basically reproduce the changes in liquid-phase chemical composition, pH and mineral-phase composition in experiments. However, hydrotalcite, the Mg-Al mineral that was not identified in experiments, was formed at all liquid-solid ratios in calculations. It was assumed that the calculated Al concentration of the liquid phase might become much lower than the experimental results due to the formation of hydrotalcite. From these results, it was found that the long-lasting deterioration behavior of solidified cement could be analytically captured by thermodynamic equilibrium calculation. It is necessary to expand the thermodynamic database in order to accurately grasp the long-term behavior of diverse types of solidified waste. Currently, it is difficult to evaluate the long-lasting deterioration behavior of solidified AAM by thermodynamic equilibrium calculation. Therefore, in order to examine the applicability of the solid phase of the solidified AAM to the thermodynamic equilibrium calculation, the basic testing of AAM reaction with waste was conducted. The solidified ALPS slurry AAM before/after immersion was analyzed by SEM and TEM to obtain phase-change data of peripheral waste components. From SEM observation, the joint between slurry and AAM matrix components was identified, and then TEM observation was conducted. As a result of TEM observation of the interface between the waste and AAM matrix on the section determined by SEM observation, no reaction was observed at the interface between slurry and base materials. Deterioration due to the migration of free components or traces of crystallization of the AAM base materials was observed. From this fact, it was found that the possibility of the mineral phase formation by the interaction between base AAM materials and slurry would be low at this time. It was found that consideration of the mineral phase formation by the interaction between slurry and base materials would not be required in thermodynamic equilibrium calculation.

In order to understand the long-lasting deterioration behavior of solidified waste, it is necessary to select an appropriate acceleration test method and evaluate the long-lasting deterioration behavior by acceleration testing of solidified waste.

4.3.2.4 Summary

The following results were obtained for low-temperature (cement and AAM solidification) and high-temperature (CCIM, In-Can, GeoMelt ICV) solidification processing technologies.

- It was confirmed through small-scale, low-temperature solidification processing testing and high-temperature solidification processing testing using crucibles that the selected processing technology would be able to solidify secondary waste generated by water treatment. A screening technique for determining the possibility of solidifying ALPS slurry with low-temperature solidification processing technology was developed.

- The applicability of ALPS slurry by low-temperature solidification processing was given. It will be necessary to verify the applicability of the low-temperature solidification processing to sludge generated by decontamination devices containing cyanide.
- Through engineering-scale testing in high-temperature solidification processing facilities (CCIM, In-Can, GeoMelt ICV), the prospect of the practical application of the high-temperature solidification processing technology to ALPS slurry and sludge generated by decontamination devices was verified. As for the low-temperature solidification technology (cement and AAM solidification), engineering-scale testing using ALPS slurry will be performed in FY 2021 to verify the prospect of its practical application.
- The leaching ratio of mock nuclides contained in the solidified waste after low-temperature solidification processing and the leaching rate of elements as main components of some solidified waste after high-temperature solidification processing were given. Deterioration of the solidified waste after low-temperature solidification processing affected by temperature and long-lasting deterioration of the solidified waste was also examined. Going forward, it will be necessary to examine the leaching rate of the solidified waste associated with elements and under testing conditions with disposal in mind as well as evaluate the long-term deterioration behavior of the solidified waste by acceleration testing.

These things have enabled clarification of the processing technologies for which practical application can be expected for stabilizing and immobilizing several important waste streams such as secondary waste generated by water treatment.

4.3.3 Establishment of methods to reasonably extract candidate technologies applicable to preceding processing

The following was examined in an attempt to “establish methods of reasonably selecting processing technology to stabilize and immobilize waste based on the expected processing technology before the technical requirements for disposal are determined (i.e., preceding processing)”:

- Provide waste package specifications based on the identified candidate processing technologies
- Enable judgment on whether the candidate processing technologies can be extracted taking into account uncertainties in characterization of the waste stream
- Clarify the applicability of processing technologies being studied

4.3.3.1 Examination on the influence of waste composition on performance of solidified waste

4.3.3.1.1 Examination of waste composition during low-temperature solidification processing

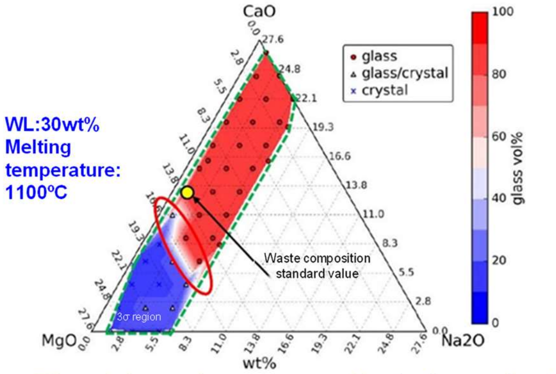
The waste composition (water/solid ratio) during low-temperature solidification processing was evaluated from the test result of fluidity, caking and compressive strength[149]. The water/solid ratio (W/P) of the waste composition in cement solidification processing was 54 wt% in the selected carbonate slurry (WL of 30 wt%) and 48 wt% in the selected iron coprecipitated slurry (WL of 20 wt%). Here, W/P is the weight ratio of water to cement + waste. It was found that the compressive strength of solidified carbonate slurry cement decreased as the filling ratio and W/P increased. W/P, the waste composition in AAM solidification processing was 86 wt% in the carbonate slurry (WL: 30 wt%, blast furnace slag: 20 wt%) and 93 wt% in the iron coprecipitated slurry (WL: 20 wt%, blast furnace slag: 20 wt%). Here, W/P is the weight ratio of alkali mixed water to metakaolin + blast furnace slag + alkali mixed water. It was found that the compressive strength of the solidified carbonate slurry AAM decreased with increasing WL and W/P, and decreased with a decreasing Na/Si ratio by adding blast furnace slag to the waste composition.

4.3.3.1.2 Examination of waste composition during high-temperature solidification processing

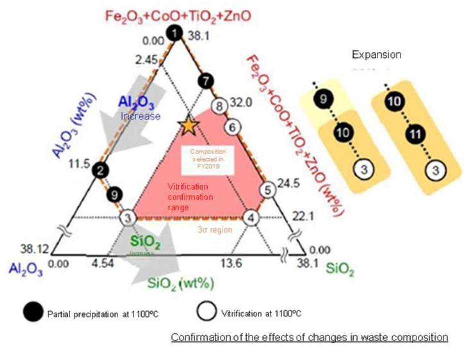
A glass property model (PNNL-DB) for determining a composition that enables vitrification was used to evaluate the waste composition during single and mixing (combination of nine types of waste) processing of carbonate slurry, zeolite, silicone-titanate, etc., which satisfy the limiting conditions of the melting furnace (melting temperature, viscosity, electrical conductivity, etc.) [149]. The maximum WL in high-temperature processing of the carbonate slurry was 12 wt% for GeoMelt ICV, CCIM, and In-Can, and the maximum WL in high-temperature processing of the iron coprecipitated slurry was 20 wt% for GeoMelt ICV, CCIM, and 15 wt% for In-Can. It was found that the applicability of the waste composition in high-temperature solidification processing was expanded in mixing processing in which easy-to-vitrify zeolite and iron coprecipitated slurry, etc. were combined.

4.3.3.1.3 Influence of changes in waste composition on high-temperature solidification processing

Due to the large uncertainty in the analysis results of waste, the possibility of solidification by high-temperature solidification processing (CCIM, melting temperature at 1100°C) in case of changes in waste composition was examined[149]. For CaO, MgO and Na_2O, which are the main components of carbonate slurry (WL of 30 wt%), and Fe_2O_3 + CoO + TiO_2 + ZnO, SiO_2, and Al_2O_3, which are the main components of iron coprecipitated slurry (WL of 40 wt%), the influence on high-temperature solidification processing in case of changes in the composition of the main components was verified through crucible testing to provide the fluctuation range of solidifiable waste composition (Fig. A11-20).



a. Effect of changes in waste composition (carbonate slurry)



b. Effect of changes in waste composition (iron coprecipitation slurry)

Fig. A11-20 Fluctuation range of waste composition eligible for high-temperature solidification processing[149]

4.3.3.2 Examination on suppressing Cs volatilization during high-temperature solidification processing

Volatilization suppression measures of Cs during high-temperature solidification processing were investigated to examine Cs volatilization characteristics and Cs volatilization suppression mechanism based on processing methods and operating conditions. Items affecting volatilization of Cs include cold cap (cold portion consisting of the waste supplied on the upper part of the molten section, etc.), top-off frit, bubbling and boron addition.

The influence of cold cap, top-off frit and bubbling were evaluated by crucible testing[149]. As a result, it was found that when a cold cap was formed at a surface temperature of 800°C or lower, the volatilization volume of Cs was reduced by more than 2 orders of magnitude. For top-off frit, it was found that the volatilization volume of Cs decreased, but the effect was limited when the melting progresses quickly. As for bubbling, the Cs volatilization volume almost doubled compared with the case without bubbling but the increase was small compared to that in the bubbling area of the molten surface.

In an attempt to investigate the mechanism of Cs volatilization suppression with a cold-cap when solid waste is solidified at high temperatures in a GeoMelt ICV, crucible testing was used to calculate the TG/DTA curves of zeolite + sludge generated by decontamination devices in the case of undergoing high-temperature solidification processing. The test results showed that weight loss ended at 730°C, and then vitrification was triggered. Since there was no significant Cs migration around the calcinated layer during melting, it was confirmed that no Cs diffusion that could cause significant compositional changes would not occur.

Regarding the addition of boron, Cs-sorbed zeolite or silicic titanium acid + zeolite was heated with a vitrifying agent, and the weight was measured in each temperature range to verify the Cs volatilization volume for different temperatures. As a result, the Cs volatilization volume increased with the temperature rise at 950°C or higher[149]. Mass spectrometry of the volatile components and mass balance of samples before/after heating showed that Cs and B volatilized at a molar ratio of 1:1. It was found that the concentration of B in the vitrifying agent might affect the Cs volatilization volume.

4.3.3.3 Examination on the waste package performance produced by processing technology

Solidified waste produced by low-temperature solidification processing is either a uniform solidified waste package or a container-filled and solidified waste package stored in a 200 L drum can or a large rectangular-shaped container. A solidified waste container produced by high-temperature solidification processing is either one of two types of 200 L containers for CCIM, a universal canister (UC, used for waste package for high-level waste in France) or a cylindrical shielding container for In-Can, or an ICV container (square-shaped container used during melting in GeoMelt ICV) for GeoMelt ICV (Table A11-7). It is assumed that a solidified waste container produced by high-temperature solidification processing is used with a waste package container as it is or is contained in another waste package container.

Table 4.3-1 Solidified waste container produced by high-temperature solidification processing technology[155,156,157]

	Solidified products Container	Weight	Solidified products Container	
Solidified products container produced by CCIM	200 L drum (single container)	400kg (solidified product weight)	200 L drum (wall thickness) Internal volume: 160 L	
	200 L drum (double container)	400kg (solidified product weight) One 160 L container / 200 L container	160 L drum	Package 160 L drum in 200 L drum. Fill space between containers with mortar, etc.
Solidified products container produced by In-Can	UC (Φ0.43x1.34m)	0.5ton One CAN / UC	110LCAN (Φ0.4 × 1.2m)	No mortar or other filling between containers. Cs-rich waste.
	Cylindrical shielding container (Φ1.2 × 1.2m)	Approximately 10 tons Seven CANs / cylindrical shielding container	110LCAN (Φ0.4 × 1.2m)	Fill space between containers with mortar. Sr-rich waste.
Solidified products container produced by GeoMelt ICV	ICV container (2.23 × 2.23 × 2.62m)	10 ton (solidified product weight)	ICV container	Solidified products and refractory container and refractory sand are packaged in ICV container. Solidified products include electrodes.

UC: Universal canister

Next, the specifications (density, compressive strength, and G-value) of the solidified waste package produced by low-temperature solidification processing of carbonate slurry and iron coprecipitated slurry were examined (Table A11-8). The specifications (WL, density, compressive strength, and composition) of the solidified waste package produced by engineering-scale testing of the high-temperature processing technology were also examined (Table A11-9).

Table A11-2 Specifications of the solidified waste package produced by low-temperature solidification processing of carbonate slurry[149]

Processing Technology	Solidified Material	Waste Loading Rate (wt%)		Density (g/ml)		Compressive Strength (28 days) (N/mm^2)		G value (H_2) (molecules/100 eV)	
		Carbon	Iron	Carbon	Iron	Carbon	Iron	Carbon	Iron
Cement	OPC	30	20	1.73	1.9	6.81	34.4	0.16	0.07
AAM	M	30	20	1.62	1.73	5.3	17.3	0.18	0.16
AAM	MB20	30	20	1.61	1.74	8.3	14.4	0.09	0.11
AAM	MB40	30	20	1.63	1.75	7.0	19.9	0.11	0.17

Table A11-9 Specifications of the solidified waste package produced by high-temperature solidification processing of secondary waste generated by water treatment[151,152,153]

Waste	WL (wt%)	Weight (kg)	Density	Compressive Strength (MPa)	Main Composition (wt%)			
					Al_2O_3	B_2O_3	Na_2O	SiO_2
Carbonate slurry	20	175-414	$2.6g/cm^3$	44-133	13.1	--	9.0	49.1
Iron coprecipitation slurry	35	206-325	$2.7g/cm^3$	133	6.8	9.0	14.3	40.0
Zeolite, waste sludge	67	249-334	$2.5g/cm^3$	62-131	--	--	--	--
Zeolite, carbonate slurry, iron coprecipitation slurry, silicotitanate, sand	84	102	$2.75g/cm^3$	42-200	6.07	6.44	13.76	35.78
Zeolite, silicotitanate, sand	80	103	$1.43\text{-}2.59g/cm^3$	8-301	11.52	15.0	12.35	48.75
Carbonate slurry, iron coprecipitation slurry	40	72	- -	- -	0.6	15.2	11.9	34.9
Zeolite silicotitanate	77-79	156-166	$2.42\text{-}2.76g/cm^3$	103-148	10.83	6.25	10.68	51.45

4.3.3.4 Examination on practical-scale systems for solidification processing technology

A practical-scale system concept for processing technology was examined to obtain economic performance data, such as system configuration and processing efficiency, details of maintenance, types of consumables and replacement frequency, and secondary waste volume.

The low-temperature processing technology has in-drum and out-drum mixing methods. In the in-drum mixing method, waste and cement are kneaded and mixed inside the container. In the out-drum mixing method, waste and cement are kneaded and mixed in a kneader outside the container. In the case of solid waste, cement that is kneaded/mixed outside the container is fed into the container containing waste[149].

For high-temperature processing technology, it is assumed that the existing secondary waste generated by water treatment will solidify in a decade. For CCIM, consideration is underway to install two melting furnaces (to produce 200 kg of glass), and use a 200 L container depending on the waste inventory to be treated as a container for solidification (single container or double container) to which molten materials flow down. For In-Can, it is planned to install a Cs-rich unit and a Sr-rich unit according to the waste properties, and to set up two Dem & Melt units each (to produce 300 kg of glass). For GeoMelt ICV, assuming that the existing KURION and SARRY sorbent will be solidified in a year and the ALPS slurry in 4 years, it is planned to install two GeoMelt ICV melting furnaces (to produce 10 tons of glass).

4.3.3.5 Summary

As a result of examination for establishing methods to reasonably extract a set of candidate processing technologies that can be applied to the preceding processing, the following outcome was obtained:

- The waste composition that can solidify an ALPS slurry at low temperature was given In addition to the method for inspecting the possibility of solidification shown in 4.3.2.1.3, it became possible to determine whether low-temperature processing can be applied to ALPS slurry. Going forward, it is necessary to verify the applicability to sludge generated by decontamination devices containing cyanide.
- The carbonate slurry is produced in a large amount, but the WL during solidification processing is small. For such waste, it is necessary not only to aim at improving WL, but also to examine the measures required for optimizing all entire waste such as by processing several types of waste together.
- The waste composition in the case of single and mixing processing of the secondary waste generated by water treatment by applying each high-temperature solidification processing technology was evaluated and provided in a glass property model. Comparative verification with engineering-scale test results will be needed in the future.
- The range of waste composition of ALPS slurry to which high-temperature solidification processing can be applied was given.
- The Cs volatilization volume by suppression during high-temperature solidification processing was evaluated. The suppression mechanism of Cs volatilization with a cold-cap was also investigated, demonstrating the effect of Cs volatilization suppression. Additional tests will be conducted in FY 2021 to examine in detail the effect of high-temperature solidification processing technology on the reduction of Cs volatilization.
- The specifications (density, compressive strength, G-value, and composition) of the solidified waste container and the solidified waste produced by each processing technology were given.
- Examination was made on full-scale configuration of low-temperature and high-temperature solidification processing systems, details of maintenance, processing speed, amount of secondary waste generation, and economic performance.

As a result, it has become possible to extract the candidate processing technologies based on the uncertainty of the waste composition in characterization by providing the applicability of the solidifiable waste composition. Waste package specifications after solidification were also provided based on the identified candidate processing technologies applicable for preliminary processing.

4.3.4 Consideration of disposal concept and development of safety assessment method

4.3.4.1 Solid waste characteristics and development of disposal concept

4.3.4.1.1 Solid waste characteristics, and needs for safe and reasonable disposal

In order to develop technology for the safe and reasonable disposal of such waste, there is a need for a research approach to disposal that takes into account the characteristics of waste.

This solid waste is characterized by a wide variety of contamination conditions and chemical substances as well as a large physical volume (large amounts). There are also uncertainties in terms of contamination conditions and selection of processing/segregation/decontamination options. These characteristics raise the need for the diverse acceptability of waste, capacity enhancement, and measures to address the risk of uncertainty depending on the disposal concept.

As an approach to research on the disposal of such waste characterized by diversity, large amounts, and uncertainty, it is not appropriate to simply apply the existing disposal concept for normal radioactive waste in Japan to solid waste because this concept was not developed based on a sufficient understanding of these characteristics and the needs arising therefrom. Given this, a research approach was adopted that establishes a safe and reasonable disposal concept (a basic concept including depth of disposal, barrier structure, and waste package) for the characteristics of solid waste, and that develops safety assessment methods (scenario development, evaluation model development, and evaluation parameter setting) for such disposal concept, while also investigating promising technology, reference cases, and research results in Japan and overseas and incorporating the knowledge gained.

4.3.4.1.2 Needs based on the characteristics of solid waste, and reference cases identified

Fig. A11-21 links overseas reference cases with the needs for diverse acceptability of waste, capacity enhancement, and measures to address the risk of uncertainty raised by diverse, large-volume, and uncertain characteristics of waste. The reference cases were selected with respect to whether they comply with environmental conditions, including the geography and geology of Japan; comply with regulations and are related to processing/disposal concepts in Japan; have a proven track record; have high economic performance; and have low enough levels of uncertainty to achieve the processing/disposal concepts in detail.

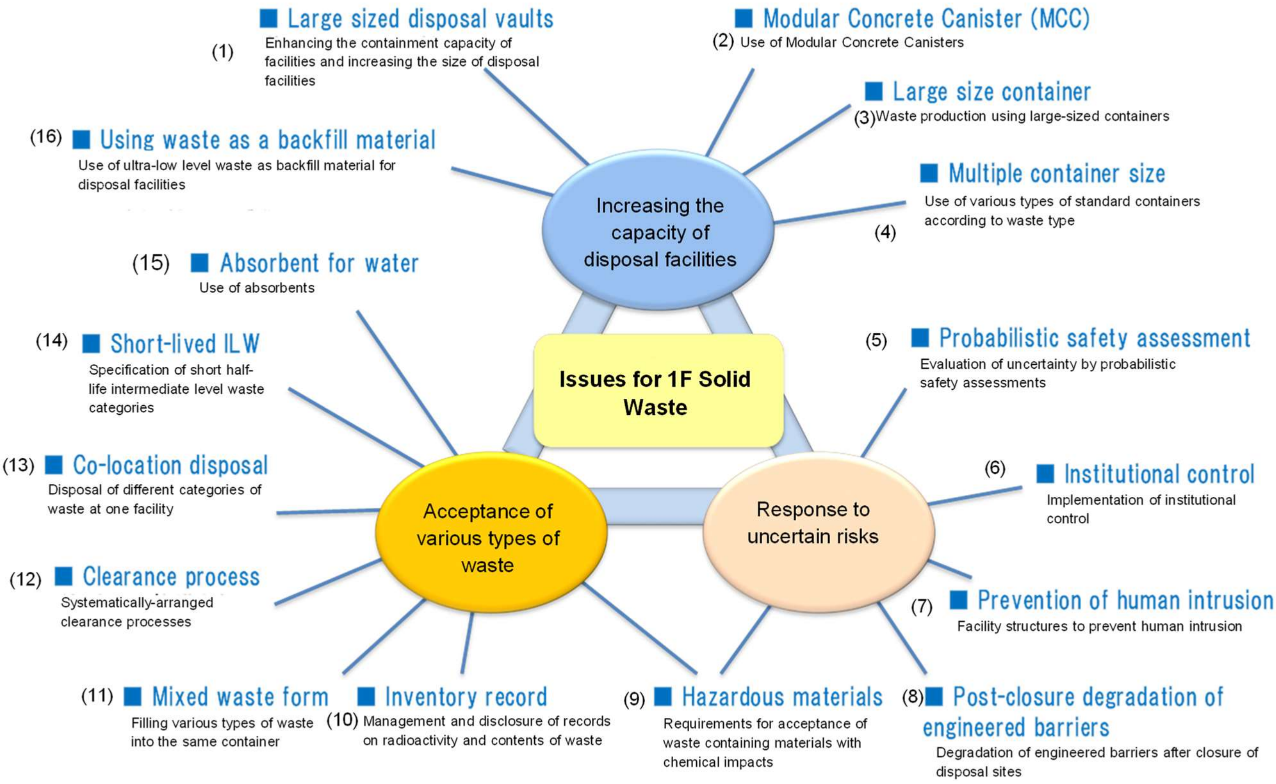


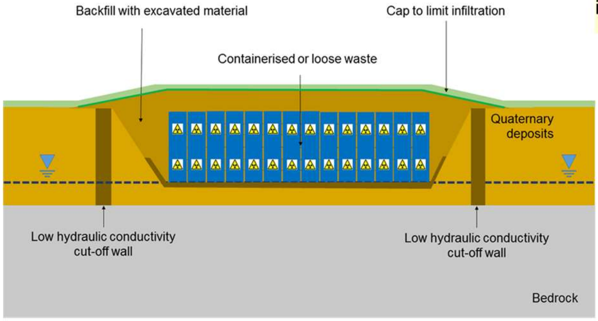
Fig. A11-21 Needs arising from solid waste characteristics, and matching with overseas reference cases149

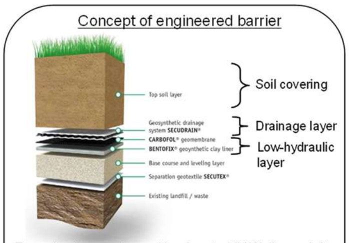
The disposal concepts were examined based on the reference cases (numbering in Fig. A11-21) corresponding to the needs. Then, as potential disposal options, the disposal concepts equivalent to trench disposal, pit disposal, and mid-depth disposal were clarified to the extent that would allow model and parameter setting for safety assessment (Fig. A11-22)149. The status of incorporation of the reference cases into the disposal concept are shown in association with the disposal concept in Fig. A11-22, with reference cases shown and numbered in Fig. A11-21.For the disposal category equivalent to trench disposal (L3), in reference to VLLW disposal sites in UK, the barrier structure is likely to have clay-covered liners with a low-permeability layer in the upper part of the disposal area surrounded by the low-permeable floor and sidewalls. For the disposal category equivalent to pit disposal (shallow ground disposal site (L2)), in reference to LLWR in UK, the barrier structure is likely to be such that the disposal area of a reinforced concrete vault consisting of a floor covered with low-permeability clay and walls with bentonite on the surface is backfilled with cement grout, and water-shielding barriers using geomembrane, etc. are installed in the upper part. For the disposal category equivalent to mid-depth disposal (mid-depth disposal facility (L1)), in reference to the silo type disposal site for short-life, low- to medium-level radioactive waste in Sweden, the barrier structure is likely to be such that the disposal area of the reinforced concrete pit is placed inside a tunnel pit several dozen meters below ground or deeper, with the outside covered with low-permeability liners and the void space backfilled with cement grout and closed with plugs.

Concept of ground surface disposal (L3)

(1)(3)(4)(6)(7)(8)

The burial facility will be backfilled with soil. The intention is to use low-permeability floor surfaces and side walls, and a low-permeability layer at the upper area to reduce water infiltration, with a soil covering also applied.





Example of an engineered barrier at a VLLW disposal site (Kings Cliffe) in the United Kingdom (From results of 2017-2018 survey on overseas disposal sites)

Concept of near surface disposal (L2)

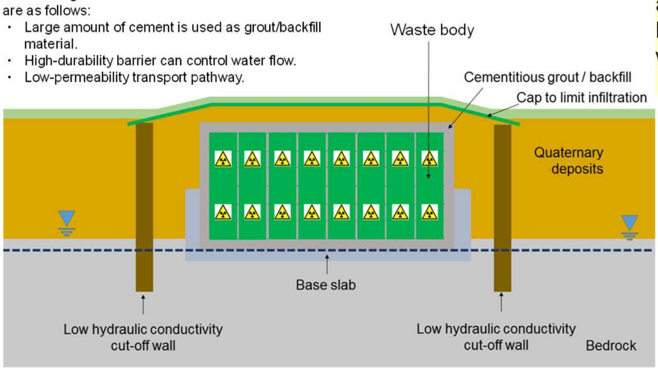
(1)(3)(4)(6)(7)(8)(11)

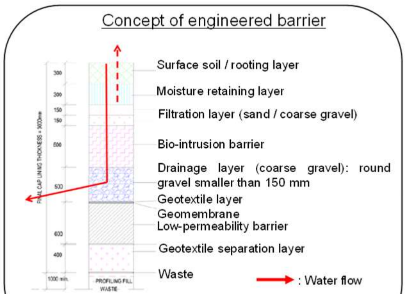
Consists of a reinforced concrete vault in which low-permeability clay is laid on the floor surface and a multi-layer soil covering is applied on the upper area. The burial facility is backfilled with cement grout*. The bentonite wall surface suppresses water infiltration.

*Bentonite is used in backfilling for burial of vitrified waste.

Similar to ground surface facilities. Possible differences are as follows:

- Large amount of cement is used as grout/backfill material.
- High-durability barrier can control water flow.
- Low-permeability transport pathway.





Example of an engineered barrier at a LLW disposal site (Drigg) in the United Kingdom (From results of 2017-2018 survey on overseas disposal sites)

Concept of medium-depth disposal facility (L1) (1)(3)(4)(6)(7)(8)(14)

Installed roughly 50 meters underground. It is composed of a concrete-walled tunnel which is plugged and closed off. The voids are backfilled with cement grout.

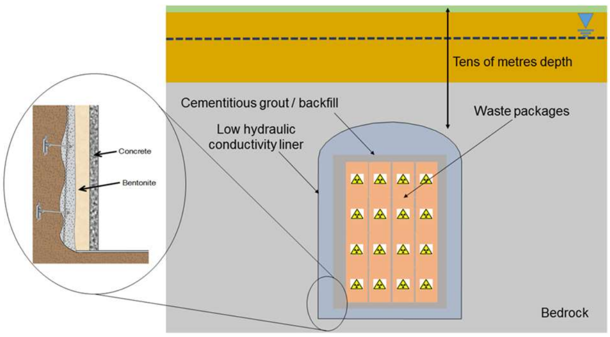


Fig. A11-22 Disposal concepts equivalent to trench disposal, pit disposal, and mid-depth disposal clarified as potential disposal options based on reference cases[149]

4.3.4.2 Safety assessment technique for solid waste disposal

In order to identify priority issues to be discussed, parameters for assessment were defined based on the disposal concepts assumed above, and safety assessment was performed[149]. An outline is provided below.

4.3.4.2.1 Selection of representative solid waste

In order to provide Technical Prospects for a safety assessment technique and methods for examining disposal concepts for solid waste, the following types of waste, (1), (2), and (3), which have "characteristics that make safety assessment difficult," were selected for further examination. One or more types of solid waste were selected from three categories: rubble, secondary waste generated by water treatment, and dismantled waste consisting of solid waste.

(1) Large-volume: The feasibility of disposal is greatly affected because the capacity required for disposal facilities also changes and the (economical) reasonability varies greatly depending on the selection of disposal options. This makes it difficult to eliminate excessive conservatism in accordance with the ALARA concept[154].

(2) Diversity and uncertainty of the types, concentrations and chemical substances contained in contaminated nuclides: These factors increase the uncertainty in determining the nuclide inventory in disposal. It is difficult to determine the nuclide inventory given this uncertainty. As the contained chemical substances may have a direct/indirect influence on the nuclide relocation behavior, it is also difficult to properly incorporate its influence into safety assessment.

(3) Uncertainty of processing/segregation/decontamination options: The selection of processing options changes the nuclide release behavior of the waste package, and it is difficult to establish a source term model according to the waste package characteristics.

- For rubble, Rubble 1, Rubble 2, and waste inside buildings (concrete) were selected in terms of Characteristics (1) and (2).
- For secondary waste generated by water treatment, KURION and carbonate slurry were selected based on the above Characteristics (1), (2) and (3), and Decontamination device sludge from Characteristics (2) and (3).
- For dismantled waste, RPV waste and waste inside PCVs were selected in terms of the above Characteristic (2) different from rubble (contamination overlap caused by radioactive materials from activated materials and those from fuel).

[154] The principle of optimizing radiological protection advocated by the International Commission on Radiological Protection (ICRP), and an abbreviation for "as low as reasonably achievable". That is, the magnitude of individual doses, the number of people exposed, and the likelihood of incurring exposures "should all be kept as low as reasonably achievable, economic and social factors being taken into account" (Japan Radioisotope Association, 1990 Recommendations of the International Commission on Radiological Protection, (1991)).

4.3.4.2.2 Setting of evaluation parameters

(1) Example of hydraulic parameter setting for a standard geological environment and engineered barriers

The evaluation parameters were defined based on the disposal concept clarified in Fig. A11-22. Table A11-10 shows parameter setting examples based on the standard geological environmental conditions, functions/characteristics of the disposal system, and their changes. In setting the evaluation parameters, in light of uncertainties, a pessimistic case with pessimistic parameters, and a basic case with realistic parameters are provided.

Table A11-10 Parameter setting examples based on the standard geological environmental conditions, functions/characteristics of the disposal system, and their changes[149]

Radiation dose evaluation parameters			Parameter characteristics			Evaluation	
Item (unit)	Conservative setting value	Realistic setting value	(1) Distribution form	(2) Sensitivity to radiation dose (linear/nonlinear)	(3) Magnitude of impact to radiation dose (sensitivity)	Importance (comprehensive evaluation)	Action policy
Natural barrier permeability coefficient, surface layer (m/s)	3.0×10^{-5}	1.0×10^{-5}	Lognormal Site-specific	Exponential	Large impact on short half-life nuclides	High impact and site dependency	Conservative setting Multiple-condition setting
Natural barrier permeability coefficient, shallow layer (m/s)	1.0×10^{-6}	1.0×10^{-7}	Lognormal Site-specific	Exponential	Large impact on short half-life nuclides	High impact and site dependency	Conservative setting Multiple-condition setting
Natural barrier permeability coefficient, medium depth (m/s)	1.0×10^{-7}	1.0×10^{-8}	Lognormal Site-specific	Exponential	Large impact on medium and long half-life nuclides	High impact, with site dependency	Conservative setting Multiple-condition setting
Bentonite mixed soil, L2 (m/s)	1.0×10^{-8}	1.0×10^{-10}	Lognormal Depends on composition, construction, deterioration, etc.	Linear against seepage water volume	Large impact also on medium and long half-life nuclides	High impact, depends on design and construction	Conservative setting Multiple-condition setting
Bentonite L1 (m/s)	1.0×10^{-10}	1.0×10^{-11}	Lognormal Depends on composition, construction, deterioration, etc.	Linear against seepage water volume	Large impact on medium and long half-life nuclides as well	High impact, depends on design and construction	Conservative setting Multiple-condition setting
River flow rate (m^3/y)	2.0×10^{-7}	1.0×10^{-8}	Site-specific National distribution close to lognormal	Linear	Inverse proportion	High impact and site dependency	Conservative setting Multiple-condition setting

(2) Incorporation as evaluation parameters with time variation in the degradation process of engineered barriers

The degradation process of engineered barriers was incorporated as evaluation parameters with time variation. Specifically, parameters with time variance were defined with the assumption of degradation in engineered barriers such as geomembrane, concrete, and metal containers. With the disposal concept equivalent to pit disposal, referencing LLWR in the UK, evaluation was conducted under the assumption that the water permeability coefficient of water-shielding barriers consisting of clay and a geomembrane would rise linearly from 10^{-12} m/s to 10^{-9} m/s from 150 to 1,000 years after installation due to the degradation of the geomembrane. For the disposal option equivalent to mid-depth disposal, the assumption for the evaluation was that the water permeability coefficient of the disposal facility would change linearly with time from 10^{-11} m/s equivalent to reinforced concrete to 10^{-6} m/s over 5,000 years due to loss of barrier functions.

Regarding the containment performance of disposal containers, according to research results from the UK, evaluation was performed based on the assumption that loss of containment performance of ISO containers, stainless steel containers, and HICs would occur 10, 300, and 1,000 years after their placement, respectively, and then the containment performance would be lost completely over the same period.

(3) Impact of chemical substances contained in solid waste

First, the chemical substances contained in solid waste that may affect the nuclide relocation behavior were extracted. Organic substances, seawater components, and boric acid solution, which are common to most types of solid waste, were selected. Although they are not very similar, ferrocyanide, sulfates, and carbonates were also selected as components, which are contained in the priority secondary waste generated by water treatment and whose influence on nuclide migration is a concern. The influence of these chemical substances shall be evaluated as a sorption reduction factor (coefficient indicating the decrease of sorption distribution coefficient, which is a parameter of nuclide migration. Hereinafter referred to as “SRF”). Barrier materials (cement-based materials and bentonite) were selected as substances to be sorbed, as they are common to all disposal concepts, placed in spatial proximity to waste packages, and considered to be greatly affected by chemical substances. The nuclides to be evaluated were determined from those for the existing disposal concept and a grouping concept based on chemical similarities. Table A11-11 shows the substances of influence (framed in red) and the waste that may contain them.

Table A11-11 Substances of influence and waste in which they are contained[149]

Raw waste			Secondary waste generated from contaminated water												Rubble / felled trees, etc.						Dismantled waste		
			Cesium adsorption vessels	Secondary cesium adsorption vessels (SARRY)	Waste sludge	Iron coprecipitation slurry	Carbonate slurry	Ag rooted activated carbon	Titanate	Titanium oxide	Ferrocyanide	Chelate resin	Resin-based adsorbent (column)	Filter	Rubble (concrete)	Rubble (metal)	Rubble (other)	Felled trees	Soil	Used protective clothing	Dismantled waste (concrete)	Dismantled waste (metal)	Dismantled waste (other)
Impacting materials	Raw waste components	Zeolite	○	○																			
		Silica-based material	○		○																		
		Iron hydroxide			○	○																	
		Carbonate					○																
		Magnesium hydroxide					○																
		Sulfate			○																		
		Activated carbon						○															
		Organic material			○	○	○	○	○	○	○	○	○	○			○	○	○	○			○
		Titanium-based material		○					○	○													
		Ferrocyanide			○						○												
		Metal	○	○		○	○	○	○	○	○	○	○		○	○					○	○	
		Concrete													○						○		
	Coexisting materials	Seawater components	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○		○	○	○
		Boric acid solution	○	○	○	○	○	○	○	○	○	○	○	○							○	○	○
		Oil content	○	○	○	○	○	○	○	○	○	○	○	○							○	○	○
		Organic material	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○	○		○	○	○
		Silica-based material													○	○	○						
		Material with fuel																			○	○	○

○ Selected 8 types of wastes and the classification of the original waste in the table are as follows.

- Rubble1 and Rubble 2 : rubble (concrete), rubble (metal) and rubble (other)
- Waste concrete in the buildings : part of dismantled waste (concrete) [including all waste in the PCVs and in the buildings, because dismantled waste is the entire waste generated by future dismantling operations other than the rubbles currently generated]
- KURION : secondary cesium sorption vessels
- Carbonate slurry : carbonate slurry
- Iron coprecipitation slurry : iron coprecipitation slurry
- Sludge generated by decontamination devices : sludge generated by decontamination devices
- RPV waste and waste inside PCVs : part of dismantled waste

With the objective of preparing data on the influence of the substances affecting disposal on nuclide migration and other data necessary for evaluation, how to deal with the effects of domestic/overseas disposal methods and substances of influence on nuclide migration was investigated. As a result, the following information was collected and organized: handling examples of organic substances in Sweden, Switzerland and Japan; domestic and overseas examples of setting SRFs for isosaccharinic acid (hereinafter referred to as "ISA"), which has the greatest negative impact on the sorption disposition coefficients of actinoid, etc.; and the same examples for seawater components, etc.

While utilizing the above knowledge collected and organized from sources inside and outside Japan, the method shown in Fig. A11-23 was used to assess the direct influence of chemical substances, and the indirect influence of changes in pH in gap water between barriers and the concentration of chemical species due to thermal (T), hydrological (H), mechanical (M), and chemical (C) changes (hereinafter referred to as "THMC") on the environmental side.

After experimentally obtaining the data that had been lacking in the domestic and overseas findings, together with the thresholds, the SRFs by substances affecting disposal were indicated by element family grouped on the basis of chemical similarities (Table A11-12). The columns highlighted in yellow show where, despite the experiment, the SRFs cannot be defined with sufficient accuracy because the equilibrium concentration of dissolved and precipitated nuclides is very low. In response to this, an attempt will be made to obtain data and understand the extent of the impact through retesting and theoretical studies in FY 2021. Through these studies, it is anticipated that the influence of chemical substances on the sorption of nuclides can be evaluated by the end of FY 2021.

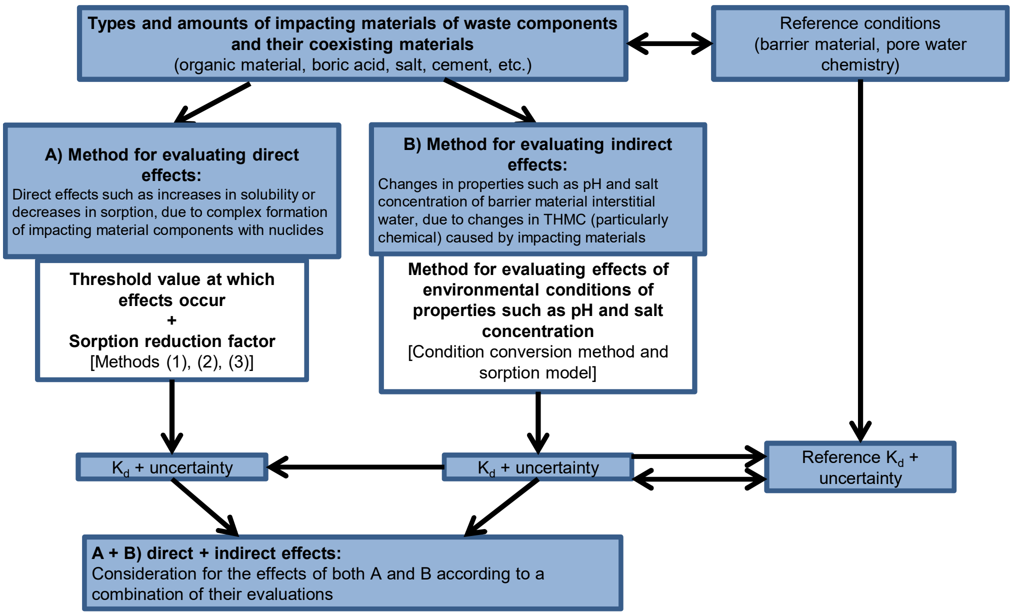


Fig. A11-23 Method to quantitatively evaluate the influence of substances affecting disposal on nuclide sorption[149]

Table A11-12 Sorption reduction factor (SRF) based on the combination of substances affecting disposal with elements that group nuclides according to their chemical properties[149]

Cement-based

Element Group	Representative Element	Impact-causing Concentrations (Threshold Value) and Sorption Reduction Factors (SRF)						
		Organic material (ISA)		Seawater	Boric acid		Ferrocyanide	
		Threshold	SRF	SRF*3	Threshold	SRF	Threshold	SRF
Alkali metals	Cs	-	1	2	-	1	-	1
Alkali earth metals	Sr	1×10^{-2} M*1	10	8	-	1	-	1
Divalent transition metals	Ni	-	1	1	-	1	-	1
Quadrivalent transition metals	Sn	1×10^{-4} M	100	10	-	1	Issues exist	Issues exist
Pentavalent transition metals	Nb	1×10^{-4} M	100	10	Issues exist	Issues exist	Issues exist	Issues exist
Trivalent actinides	Am/Eu	1×10^{-4} M	10	10	-	1	-	1
Quadrivalent actinides	Th	1×10^{-4} M	100	10	Issues exist	Issues exist	Issues exist	Issues exist
Pentavalent actinides	Np	1×10^{-4} M	10	10	-	1	1×10^{-3} M	3
Hexavalent actinides	U	5×10^{-4} M	10	10	1×10^{-2} M	500	1×10^{-3} M	3
Halogens	I	—*2	1	—*2	1×10^{-4} M	1.3	-	1
Anionic species	Se	—*2	1	—*2	Issues exist	Issues exist	Issues exist	Issues exist

*1 M: mol/L, *2 Set $K_d = 0$ (m^3/kg), *3 SRF between precipitation conditions and seawater conditions

Bentonite-based

Element Group	Representative Element	Impact-causing Concentrations (Threshold Value) and Sorption Reduction Factors (SRF)			
		Boric acid		Ferrocyanide	
		Threshold	SRF	Threshold	SRF
Alkali metals	Cs	-	1	-	1
Alkali earth metals	Sr	-	1	-	1
Divalent transition metals	Ni	-	1	Issues exist	Issues exist
Quadrivalent transition metals	Sn	1×10^{-2} M	100	1×10^{-3} M	2.7
Pentavalent transition metals	Nb	1×10^{-4} M	2.6	-	1
Trivalent actinides	Am/Eu	1×10^{-4} M	1.3	-	1
Quadrivalent actinides	Th	Issues exist	Issues exist	Issues exist	Issues exist
Pentavalent actinides	Np	-	1	1×10^{-3} M	4
Hexavalent actinides	U	-	1	1×10^{-3} M	2
Halogens	I	Issues exist	Issues exist	-	1
Anionic species	Se	Issues exist	Issues exist	1×10^{-4} M	2.0

* M: mol/L, ** Set $K_d = 0$ (m^3/kg).

* The direct impacts of sulfates and carbonates were evaluated to be insignificant based on past findings on complex formation.

: Specified based on results of past informative investigations.

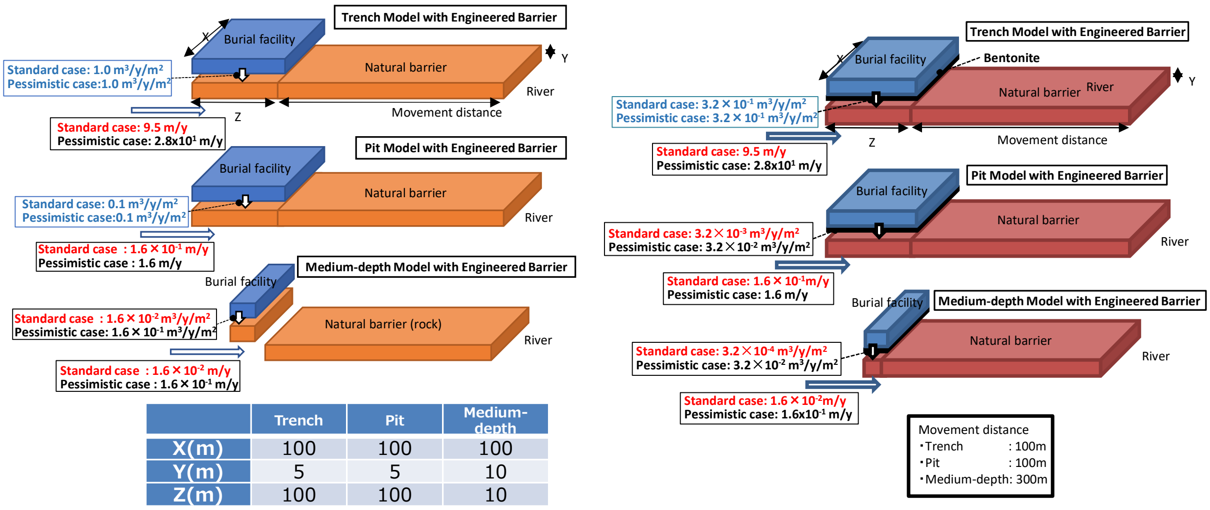
: Specified based on data obtained through sorption tests.

: Data was acquired but partition coefficients could not be obtained with sufficient accuracy (however, those effects are estimated to be insignificant).

4.3.4.3 Safety assessment for disposal

4.3.4.3.1 Evaluation system for nuclide migration models in a groundwater scenario

Fig. A11-24 shows an evaluation system for nuclide migration models in a groundwater scenario. In order from top to bottom, this figure provides concepts equivalent to trench disposal, pit disposal, and mid-depth disposal. The left side of the figure shows the case without engineered barriers, and with engineered barriers on the right side. The disposal facility is shown in blue and the natural barrier in orange. Regarding the Darcy flow rate of the natural barrier and the underground flux from the disposal facility to the natural barrier, the pessimistic cases are written in black, the basic cases in red, and the cases common to both in blue. Using these evaluation systems, nuclide release from waste, sorption distribution to cement-based materials, and sorption distribution to buffer materials were calculated for the disposal facility area, and sorption distribution to and advection/dispersion in the natural barrier were calculated for the natural barrier. The right end of each evaluation system is connected to the river.



	Trench	Pit	Medium-depth
X(m)	100	100	100
Y(m)	5	5	10
Z(m)	100	100	10

Fig. A11-24 Evaluation system for nuclide migration models[149]

4.3.4.3.2 Examples of processing effects (improvement in leaching rate) based on safety assessment at the time of disposal

Based on the disposal concepts shown in Fig. A11-24, radiation dose evaluation models were established to set evaluation parameters using the method shown in 4.3.4.2.2 and to perform nuclide migration analysis149. Table A11-13 shows an evaluation example of the basic cases. The leftmost columns of this table show the disposal categories based on the assessment of the isolation failure scenario after 400 years. The leaching rates of each waste package is reduced from instantaneous release, release at 10-3/y, 10-4/y and 10-5/y (set close to the standard values, which are defined as the leaching rate of vitrified waste in nuclide migration analysis of high-level radioactive waste) from left to right to evaluate the radiation dose in the groundwater scenario, and show the disposal categories based on the radiation dose assessment. Table A11-13 are dominated by the groundwater scenario. Looking into the waste types with letters highlighted in red in this table (the rows with red letters), the groundwater scenario is dominant on the left side of the row, where the leaching rate of the waste package is high. On the right side of the row, however, the leaching rate decreases and so does the radiation dose in the groundwater scenario, i.e., disposal as a shallower disposal category becomes possible. Then, the radiation dose is eventually reversed from that in the isolation failure scenario, which is irrelevant to the leaching rate, and the red letters no longer exist, making the isolation failure scenario dominant in determining the disposal category. Such examination makes it possible to quantitatively verify the effect of the improvement in the leaching rate by processing upon disposal for each waste type. However, the range of the columns with red letters is narrow for the waste types evaluated this time. This indicates that, for the waste types for which the inventory was defined based on the available knowledge, the isolation failure scenario plays a greater role in determining the disposal category than the groundwater scenario, and that the extent to which the improvement of leaching characteristics by processing is effective in enhancing safety in disposal or relaxing the disposal category is limited. Whether the groundwater scenario or the isolation failure scenario will dominate depends on the nuclide composition of the waste, i.e., how the nuclide inventory is given, and it is not deterministic at this point.

Table A11-13 Example effect (improvement in leaching rate) of processing by safety assessment on disposal (isolation failure and groundwater scenarios) (Basic case; Conservative parameter values are used for the groundwater scenarios)[149]

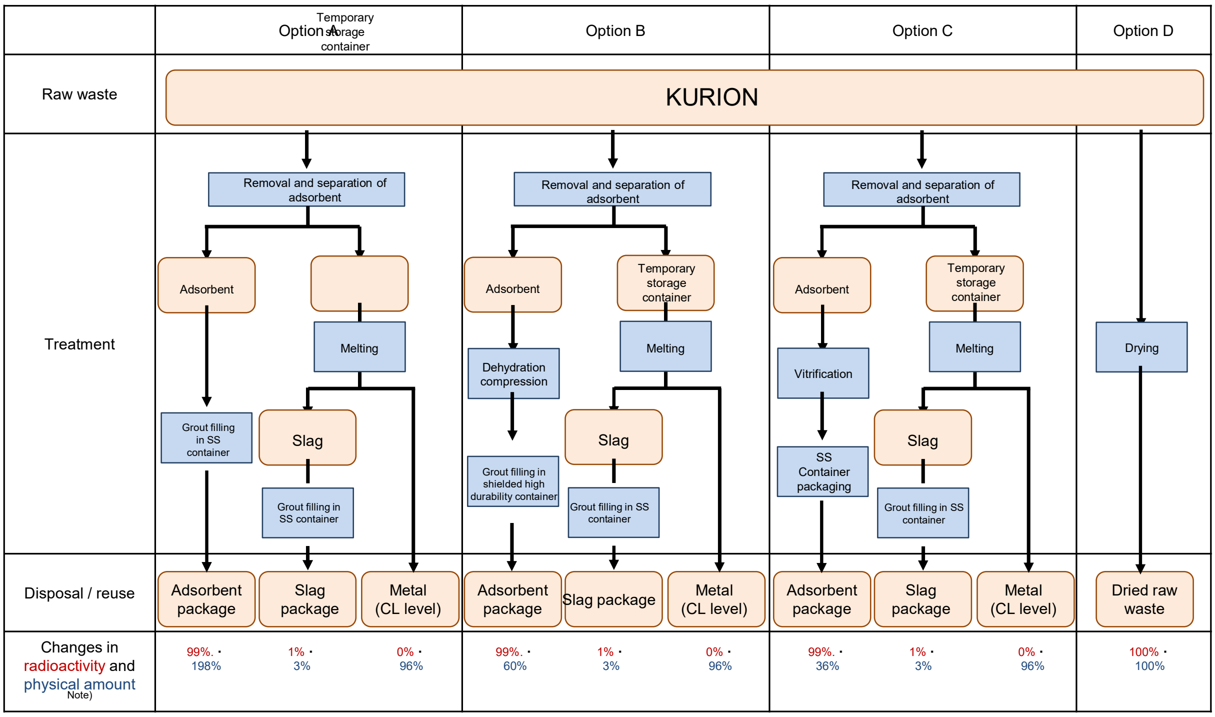
Waste	Isolation failure scenario (L2, L3 : 400 y, L1: 1 × 10^5 y)	Basic case of the groundwater scenario															
		Leaching rate of 1.0 × 10^{-3}/y				Leaching rate of 1.0 × 10^{-3}/y				Leaching rate of 1.0 × 10^{-3}/y				Leaching rate of 1.0 × 10^{-3}/y			
		-	SRF for natural barriers	SRF for facilities	SRF for facilities/ natural barriers	-	SRF for natural barriers	SRF for facilities	SRF for facilities/ natural barriers	-	SRF for natural barriers	SRF for facilities	SRF for facilities/ natural barriers	-	SRF for natural barriers	SRF for facilities	SRF for facilities/ natural barriers
KURION	L1	L3/L3	L2/L3	L3/L3	L2/L2	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3
Waste sludge	L1	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3
Carbonate slurry	L1	L2/L2	L2/L2	L2/L2	L2/L2	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3
Rubble 1	L3	L2/L2	L2/L2	L2/L2	L2/L2	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3
Rubble 2	L3	L2/L2	L2/L2	L2/L2	L2/L2	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3
RPV waste	Geological layer	L1/L1	L1/L1	L1/L1	L1/L1	L1/L1	L1/L1	L1/L1	L1/L1	L1/L2	L1/L2	L1/L2	L1/L2	L2/L2	L2/L2	L2/L2	L2/L2
Waste inside PCV	Geological layer	L2/L2	L1/L2	L2/L2	L1/L2	L2/L2	L2/L2	L2/L2	L1/L2	L2/L2	L2/L2	L2/L2	L1/L2	L3/L3	L2/L3	L3/L3	L2/L3
Waste in buildings (concrete)	L2	L2/L2	L2/L2	L2/L2	L2/L2	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3	L3/L3

*In this assessment, the rows with red letters are where the groundwater scenario evaluation results are dominant in determining the disposal category.

4.3.4.3.3 Relationship between processing/decontamination/segregation options and disposal concept based on safety assessment (disposal category)

Fig. A11-25 illustrates the treatment/decontamination/segregation options for Kurion, and the corresponding disposal categories[149]. It may be possible to conduct disposal equivalent to L1 by adopting Option C (segregate sorbents and temporary storage containers, and vitrify sorbents).

As with the Kurion case above, the disposal categories of the selected eight types of waste were evaluated for each processing, decontamination, and segregation option. As a result, it was determined to classify the RPV waste and the waste in PCVs as equivalent to geological disposal, regardless of the processing/decontamination/segregation options. However, it was also determined that, except for these two, other types of waste including secondary waste generated by water treatment by Kurion could fall into the disposal category equivalent to L1 depending on the selected processing, decontamination, and segregation options.



Disposal Concept Option	Processing Option A Cement solidification and packaging in SS containers		B Dehydration compression and packaging in high-durability containers		C Vitrification and packaging in SS containers		D Drying in containers	E Raw waste disposal, no packaging
	Adsorbent	Temporary storage container	Adsorbent	Temporary storage container	Adsorbent	Temporary storage container		
L3	Well users River users	Site residents	Well users River users	Site residents	Site residents	Site residents	Well users River users	Well users River users
L2	Well users River users	Site residents	Well users River users	Site residents	Site residents	Site residents	Well users River users	Well users River users
L1	Well users	3%	Well users River users	3%	36%	3%	Well users River users	Well users River users

(green) : In all scenarios, the dose is below the dose reference value, and burial is expected to be possible as a result of dose evaluations. Figures indicate amounts based on raw waste.

(pink) : In even one scenario or more, the dose is above the dose reference value, and burial is expected to be impossible as a result of dose evaluations. The main reasons are also described.

Fig. A11-1 Treatment/decontamination/segregation options for the secondary waste generated by water treatment by Kurion, and corresponding disposal categories[149]

As with the Kurion case above, the disposal categories of the selected eight types of waste were evaluated for each processing, decontamination, and segregation option. As a result, it was determined to classify the RPV waste and the waste in PCVs as equivalent to geological disposal, regardless of the processing/decontamination/segregation options. However, it was also determined that, except for these two, other types of waste including secondary waste generated by water treatment by Kurion could fall into the disposal category equivalent to L1 depending on the selected processing, decontamination, and segregation options.

4.3.4.3.4 Establishment of methods for reasonably selecting processing methods for stabilization and immobilization before technical requirements of disposal are established (preceding processing)

The associated requirements for the methods for appropriately selecting processing methods for stabilization and immobilization before technical requirements of disposal are established (preceding processing) are (1) to demonstrate safety using safety assessment models corresponding to the conditioned waste specifications and disposal concepts, and (2) to determine whether the candidate processing methods can be selected in consideration of waste properties and uncertainties in characterization. As shown in Table A11-13, it can be said that the goal of (1) has been achieved if a technique is available that compares the models with the prescribed radiation dose constraint based on the dose assessment in isolation failure and groundwater scenarios to determine the disposal category.

However, with regard to (2), upon understanding the impact to safety assessment of uncertainty in characterization, it is necessary to extract the treatment, decontamination, and segregation options as shown in Fig. A11-25, and establish a method for selecting the applicable candidate technologies in the waste management process by using evaluation techniques and results to determine the disposal category according to the radiation dose assessment for each option, as shown in Fig. A11-26. According to this method, and in association with the applicable candidate processing technologies, the consistency of the waste package specifications with the disposal system and their safety in case of disposal are evaluated. As a result, an evaluation report will indicate “Results of disposal safety assessment” and the resultant “Requirements on waste package specifications”, or “Requirements on disposal options, etc.” if it is more desirable to perform handling on the disposal side than on the processing side.

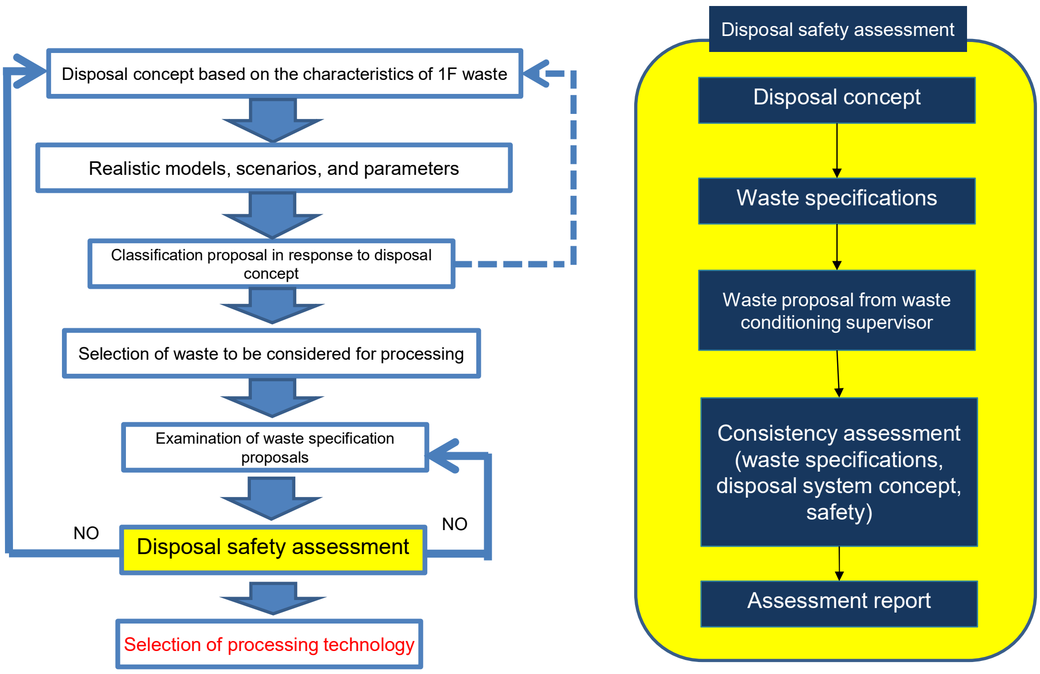


Fig. A11-2 Method for selecting applicable processing technologies and disposal options in the waste management process

4.3.4.4 Summary

As a result of examination for establishing safe and reasonable disposal concepts and safety assessment methods based on the characteristics of solid waste, the following outcome was obtained:

- Based on the characteristics of solid waste and the need for safe and reasonable disposal, and in reference to Japan and overseas reference cases, examples of disposal concepts (basic concepts such as disposal methods, barrier structure, and conditioned waste) appropriate for solid waste were provided.
- Representative types of solid waste were selected, and parameters for safety assessment were defined based on domestic/overseas cases and the disposal concepts shown in 4.3.1.1.3. In this process, examination was performed to incorporate the influence of chemical substances contained in solid waste. The target waste and radionuclides and the corresponding sorption reduction factors were given to evaluate their influence on the radiation dose. In addition, realistic parameters incorporating degradation behavior were defined, accounting for functions such as the low-permeability property of concrete, the corrosion resistance of disposal containers (physical containment performance), and water-shielding barriers (capping) in safety assessment. The influence of parameter changes by processing on the disposal categories, and the relationship between the processing/decontamination/ segregation options and the disposal categories were exemplified.

Based on the above, safe and reasonable disposal concepts were given, and safety assessment methods incorporating their characteristics could be developed.

Moreover, a series of methods for reasonably selecting processing methods for stabilization and immobilization before technical requirements of disposal were established (preceding processing) as described below.

- Safety can be provided using safety assessment models corresponding to waste package specifications and disposal concepts.
- As a result of 4.3.3, the candidate processing technologies can be extracted in light of the uncertainty in characterization, and associated waste package specifications be provided.
- Further, according to the method shown in Fig. A11-26, the consistency of the waste package specifications with the disposal system and their safety in case of disposal will be evaluated. If inappropriate, “Requirements on waste package specifications” will be indicated, and if it is more desirable to perform handling on the disposal side instead of the processing side, “Requirements on disposal options, etc.” will be indicated.

1. Now that these requirements have been incorporated and proper measures have been implemented in the waste management process in light of solid waste characterization and its uncertainties, it becomes possible to select applicable candidate processing technologies.

4.3.5 Prospects of establishing processing/disposal strategies for solid waste for which processing technology with consideration for disposal has not been clarified

As shown in 4.3.2, for the secondary waste generated by water treatment, including ALPS slurry, it has become possible to clarify the processing technology for which there are expectations for practical application with consideration for disposal. The following describes the establishment of examination methods to present the prospects for defining a strategy for each step from generation to disposal for other types of solid waste as well[149].

4.3.5.1 Method for examining the integration of R&D results

An input chart was developed to summarize the information, issues and research results necessary for examining each step from waste generation to disposal. Using this chart, the entire waste stream was examined in a comprehensive manner, and mutual feedback was provided on research results in order to develop methods for efficient examination.

A general overview of the method for examining the waste stream is shown in Fig. A11-27. The figure shows input information, integration of examination and research results, and output, from left to right.

A processing flow with multiple options can be developed based on information such as the research results on inventory evaluation by characterization, the waste list and waste management sheet. Providing feedback from R&D results to the input chart enables the information and issues required to be examined in each step in the waste stream to be summarized. The timing at which these should be considered can be summarized as a timeline chart in reference to the Medium-and-Long-term roadmap and TEPCO's storage plan for solid waste.

Based on the above, matters to be considered and their details, status of efforts, and necessary actions can be summarized in light of the waste stream examination as a hub. Also, for a processing flow with many options, the flow that is currently proposed was given, and it was refined by the concept of narrowing down the flow in accordance with the required technical requirements.

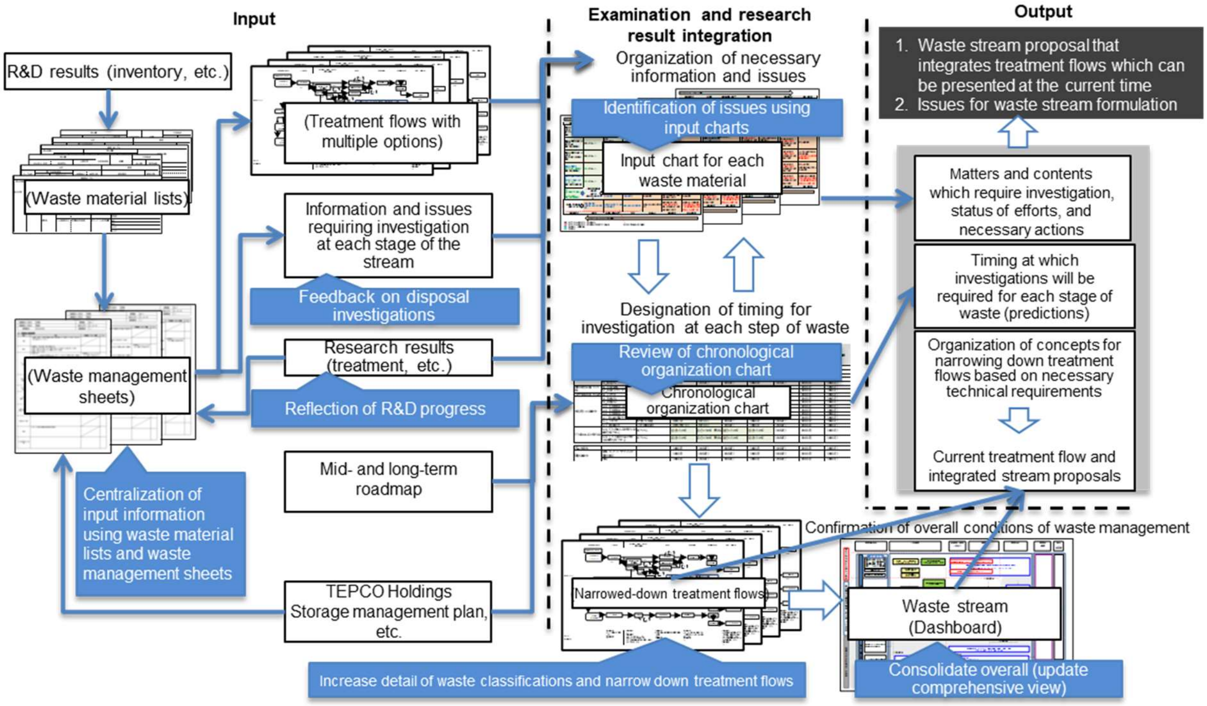


Fig. A11-3 Method for examining integration of R&D results[149]

4.3.5.2 Refinement of several flows of the waste stream

Fig. A11-28 takes ALPS slurry (multi-nuclide removal equipment slurry) as an example to show the concept of refining multiple processing flows.

Example of Stream S10-1 ALPS slurry (Multi-nuclide Removal Equipment Slurry)

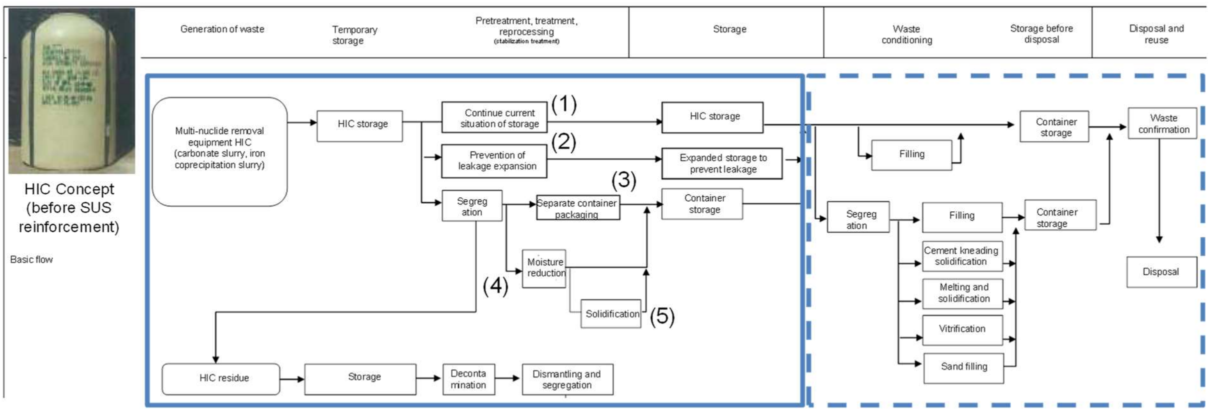


Fig. A11-4 Example concept of refining multiple processing flows (ALPS slurry)[149]

Currently, the carbonate slurry and iron coprecipitated slurry are generated from ALPS and temporarily stored in high-performance container HIC. For the next steps of pretreatment, treatment, and reprocessing, several processing flows from (1) to (5) can be used. “Continue current situation of storage” corresponds to (1), “Prevent spread of leakage” to (2), “Separate container packaging by segregation” to (3), “Moisture reduction” to (4), and “Store after additional solidification” to (5). TEPCO’s solid waste storage plan already provides a plan for dehydrating solid waste using a filter press to reduce risks such as leakage, and to contain the dehydrated materials in containers and store them in storage facilities. Here, a method for examining the validity of the concept of narrowing down from several flows to (4) “Moisture reduction” based on the basic policy for solid waste management and the required technical requirements is developed, leading to contributing to materializing the concept of such narrowing down.

4.3.5.3 Basic policy for solid waste management and required technical requirements

The technical requirements required by the International Atomic Energy Agency (hereinafter referred to as the “IAEA”.) Safety Standards, GSR-Part 5, “Predisposal Management of Radioactive Waste,” were summarized (Fig. A11-29).

According to Requirements 9, “Characterization and classification of radioactive waste,” characterization is cited as a technical requirement. Therefore, the determination of whether actions at different stages of predisposal management are implemented should be made based on the characterization research results.

“Radiation safety for the public” and “radiation safety for workers” were derived from Requirement 4, “Responsibilities of the operator”, and the corresponding technical requirements of “radiation protection and shielding” and “containment of radioactive materials” were developed, respectively. “Preventing the spread of accidents involving the release of large amounts of radioactive materials” and “maintaining containment” were derived from Requirement 11, “Storage of radioactive waste”, and the corresponding technical requirements were developed as “measures against waste-specific accidents/events such as explosions, fires, leakage, and diffusion” and “securing boundaries,” respectively.

“Understand the impact on later stages, and have prospects for satisfying the acceptance criteria” was also defined from Requirement 6 “Interdependences”, Requirement 10, “Processing of radioactive waste” and Requirement 12 “Radioactive waste acceptance criteria.” Based on the input chart and timeline chart, the interdependence between all stages in predisposal management should be duly considered, as well as the influence of the anticipated disposal options.

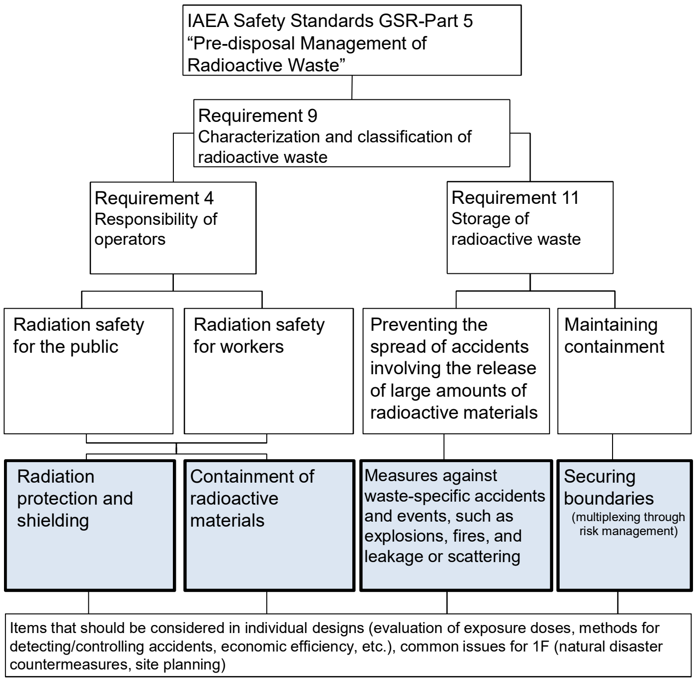


Fig. A11-5 Summarization of required technical requirements[149]

4.3.5.4 Concept for narrowing down to storage of ALPS slurry

As ALPS slurry is contained in high-performance container HIC, and then stored at temporary storage facilities, the risk of dispersion/leakage is kept sufficiently low. There was an event in the past where radiolysis of water in a container containing high radiation-dose ALPS slurry caused hydrogen generation but it was stagnating without being diffused, resulting in volume expansion of the slurry, and leakage of the supernatant water from the upper lid of the container. As a countermeasure, the flooding of supernatant water is controlled by lowering the water level in the high-performance container HIC. However, some slurry may still contain hydrogen, and even if hydrogen is degassed, the risk of recurrence of events may remain in high-radiation dose slurry. Therefore, further reduction of risk during storage will be considered in this study.

Fig. A11-28 shows an evaluation example where the technical requirements are applied to several processing flows to storage of ALPS slurry as shown in Table A11-14. Especially from the perspective of "Prevention of hydrogen retention in waste," there would still be the risk that hydrogen would be contained in the current storage conditions. The evaluation showed that (4) "Moisture reduction" and (5) "Moisture reduction and solidification" would be promising as countermeasures. Of these, the technical requirement for (5) "Understand the influence on later stages, and have prospects for satisfying the acceptance criteria" is not satisfied, and thus (5) is excluded because the elemental technology for solidification related to (5) is still in the stage of research/development, and there are no prospects yet. Therefore, the options can be narrowed down to (4) "Moisture reduction and storage in storage facilities."

As described above, it was confirmed that it is possible to narrow down from several flows by clarifying the required technical requirements.

The scope of this study is until storage, and development of a concept for the entire waste stream including disposal is planned to be completed by the end of FY 2021.

Table A11-3 Processing options until storage, and evaluation examples[149]

No.	Processing options up until storage	Shielding	Containment	Prevention of hydrogen retention in waste	Securing boundaries
(1) (3)	Continuation of current storage conditions / Packaging and storage in separate containers	○	○	△ Slurry assimilates hydrogen	△ (Temporary storage)
(2)	Prevention of leakage expansion (storage in storage facilities)	○	○	△ Slurry assimilates hydrogen	○ Containers and storage facilities
(4)	Moisture reduction and storage in storage facilities	○	○	○ Hydrogen is released from voids of dehydrated material	○ Containers and storage facilities
(5)	Moisture reduction, solidification, and storage in storage facilities	○	○	○ Generally stable, but depends on the solidification method	○ Waste materials themselves, containers, and storage facilities

4.3.5.5 Action status by waste stream from generation to disposal, and prospects for measures to be taken

Table A11-15 shows the items to consider from generation to disposal in each waste stream, and the status of action. Currently, characterization is mainly promoted as shown on the left side of the table. For some types of secondary waste generated by water treatment, examination is being performed on low- and high-temperature processing as part of the elemental technology research for processing, and on the disposal strategies shown on the right side of the table. The area in gray will be addressed by acquiring data on waste properties.

As described in 4.3, the overall picture of solid waste to be disposed of, including waste that will be generated in the future, will gradually become clear with the clarification of the progress and plans for fuel debris retrieval, contaminated water management, and other decommissioning works. Therefore, it is necessary to repeatedly examine processing/disposal methods and safety assessments, starting from waste for which properties have been clarified; to give consideration to making processing/disposal methods more appropriate; and to accumulate knowledge to consider safe and reasonable processing/disposal strategies for diverse solid waste collectively.

Based on these characteristics, characterization is insufficient for some waste streams, which need the overall picture to be captured and examination to continue. Therefore, as a method for reasonably selecting a safe processing/disposal method for solid waste, examination will be performed with the waste stream as a hub as shown in Fig. A11-12.

Table A11-154 Action status by waste stream from generation to disposal[149]

Waste	Stream Number	Waste					Pretreatment, treatment, reprocessing					Waste conditioning treatment						Storage before disposal	Disposal	
		Amount of waste (including the possibility of generation)	Radioactive inventory	Chemical properties (including materials impacting disposal)	Physical properties	Degree of segregation and container specifications	Specifications of received waste	Feasibility of essential technology	Specifications of delivered waste (including methods of ensuring performance)	Basic processes and specifications of equipment (including considerations for safety)	Material and radioactivity balances	Specifications of received waste	Feasibility of essential technology	Specifications of delivered waste (including methods of ensuring performance)	Basic processes and specifications of equipment (including considerations for safety)	Material and radioactivity balances	Waste verification methods	Waste verification methods and reuse verification methods	Waste acceptance criteria	Disposal management
Dismantled waste																				
Pressure vessels	S1																			
Metal from containment vessels	S2																			
Concrete from containment vessels	S3																			
Metal in buildings	S4																			
Concrete in buildings	S5																			
Rubble																				
Metallic rubble	S6	Generation ongoing																		
Concrete rubble	S7	Generation ongoing																		
Combustible materials (felled trees, protective clothing, etc.)	S8	Generation ongoing						Incineration												
Secondary waste generated by water treatment																				
Adsorption vessels (1) (SARRY, KURION)	S9	Generation ongoing											Low- and high-temperature treatment							
Adsorption vessels (2) (mobile purification equipment)		Generation completed																		
Multi-nuclide removal equipment (1) (slurry)	S10	Generation ongoing						Dehydration					Low-temperature treatment							
Multi-nuclide removal equipment (2) (adsorbents)		Generation ongoing																		
Multi-nuclide removal equipment (3) (treatment columns)		Generation ongoing																		
Sludge generated by decontamination devices	S11	Generation completed											Low-temperature treatment							
Filters	S12	Generation ongoing																		
Concentrated waste liquid	S13	Generation completed																		
Waste associated with debris retrieval																				
Top retrieval	S14																			
Side retrieval																				
Contaminated soil	S15																			
Waste generated during operation																				
Granular used resin and sludge	S16	Generation completed																		
Metal (used control rods, etc.)		Generation completed																		
Solidified substances (pellets, etc.)		Generation completed																		

Investigation completed (from generation to disposal)
Investigation partially completed (provide design specifications for some processes such as storage, treatment, and disposal)
Research in progress (design specifications have not been provided)
Not investigated (cannot be investigated due to a lack of input information, such as data on properties)
Investigation not started (investigation has not been started due to considerations for priority, etc.)

4.3.5.6 Summary

The following outcome was obtained as a result of the above examination:

- The entire waste stream was examined in a comprehensive manner, and mutual feedback was provided on research results, using the input chart to develop methods for efficient examination of the waste stream.
- In an attempt to narrow down several flows, the technical requirements in the IAEA Safety Standards, GSR-Part 5, “Predisposal Management of Radioactive Waste,” were summarized and applied to narrow down the flow until storage through evaluation using it. Then, it was confirmed that this could serve as one of the approaches for narrowing down.
- As the scope of this study up to FY 2020 is until storage, development of a concept for narrowing down for the entire waste stream including processing/disposal is planned to be completed by the end of FY 2021.
- For waste streams with insufficient characterization, the prospects for waste stream development have been obtained by future characterization data, continuous R&D activities, and in accordance with the examination method to integrate the outcomes.

As a result, with regard to waste streams for which processing technology has not yet been clarified for practical application with consideration for disposal due to insufficient characterization, that it was possible to offer prospects for setting processing/disposal strategies using the series of methods developed could be offered.

5. Issues based on Technical Prospects and technical strategies to achieve them

After providing for future issues based on the prospects of processing/disposal methods and technology related to their safety, the technical strategies by category to realize them are show below.

5.1 Issues based on Technical Prospects

Volume reduction is extremely important for the safe and reasonable management of solid waste according to the progress of decommissioning work in the future, so the measures in progress should be continued steadily. Since solid waste continues to be generated, it is important to continuously examine further possibilities by referring to advanced cases of overseas' for more volume reduction. It is recommended to realize volume reduction in consideration of the expected outcome and feasibility.

For the development of analytical/evaluation methods for efficient characterization, it is necessary to improve evaluation methods and continuously incorporate them into solid waste management, including processing/disposal, while accumulating analytical data using efficient analytical methods established through achievements in research/development. In this case, efforts should be made for low-activity waste such as rubble, as well as high-activity waste such as secondary waste generated by water treatment and waste generated from fuel debris retrieval, according to the characteristics of each type of waste.

To establish methods for selecting safe processing/disposal methods in a reasonable manner, the methods shown in Fig. A11-12 shown in 4.3 should be used to proceed with examination toward determining waste form specifications and manufacturing methods for Phase 3, as specified in the Medium-and-Long-term roadmap. Specifically, through these methods, the trial examples of optimization/rationalization of processing/disposal methods will be accumulated by waste stream according to the progress of characterization and under the assumption that safety is ensured to widely acquire findings on optimization by waste stream. Moreover, consideration will be given to specify strategies for optimization/rationalization of the overall picture covering the entire waste stream, allowing clarification of approaches toward such purposes. In doing so, it is important to consider the most appropriate measures, taking the actual use and economic feasibility into account by reflecting to the latest findings and applying the Best Available Techniques concepts. As the examination progresses and the processing/disposal measures for the overall picture of waste are finalized, it will be important to share the examination process for optimization, such as by sharing the awareness of problems with local communities and society.

5.2 Technical strategy by sectors

5.2.1 Characterization

For low-activity waste such as rubble, the analysis work itself is not so challenging, but an immense amount of time is required to measure entire quantity because of the enormous volume of waste, and so there are needs for volume reduction and a corresponding efficient analysis strategy. To that end, it is important to take an approach that efficiently ensures the required accuracy. In order to achieve this, promote efficient analyses by making them simplified/speed-up, and establish inventory evaluation methods that combine the DQO process with statistical methods.

For high-activity waste, there are difficulties inherent in sampling and analysis, and the amount of analysis data to be obtained is limited, which makes inventory assessment based on the transition model more important. It is necessary to obtain actual sample data, such as by ongoing efforts for sampling from cesium sorption vessels and its analysis, which is currently in progress. The application of inventory evaluation methods, which combine the DQO process with statistical methods, and the priority of data to be collected should also be considered to enhance the accuracy of the transition model.

Following the phase of analyzing samples that are easy to collect, characterization is now in the phase of collecting/analyzing samples that are important for waste management. Going forward, it is important to develop a medium-to-long-term analysis strategy that defines the solid waste to be analyzed, its priority, and quantitative targets for analysis, etc., and to proceed with analysis/evaluation accordingly. It is useful to accumulate trial results and verify their validity in order to establish the flow from the development of a medium-to-long-term analysis plan using statistical methods; analysis and data acquisition; the incorporation of the acquired data into examination of processing/disposal methods and evaluation of the outcome; to the development of the next medium-to-long-term analysis plan based on the evaluation results.

As for facilities for analysis, in addition to the existing facilities in the JAEA’s Ibaraki area, etc., it is planned to establish Radioactive Material Analysis and Research Facilities currently under construction, as well as facilities for analysis by TEPCO, allowing characterization of a variety of solid waste in parallel. Since the target nuclides, analysis items, accuracy, and the number of samples for analysis depend on the target solid waste are different, a structure should be established based on the appropriate division of roles and according to the characteristics of facilities.

5.2.2 Storage

For storage of all waste, it is important to reconsider measurement items and timing, etc., in terms of diverse information for characterization, while acquiring necessary information through continuous monitoring and surveillance of the storage status commensurate with the risks involved.

With regard to high-activity waste, such as waste generated from fuel debris retrieval, the issues and countermeasures assuming the further expansion of the fuel debris retrieval scale have been clarified according to the results of research/development as of FY 2021. Going forward, reviews should be performed along with the examination of the fuel debris retrieval methods. Reliable measures should be taken to ensure storage of the solid waste that is expected to be generated during fuel debris retrieval (trial retrieval and gradual expansion of the retrieval scale) before full-scale retrieval.

The site also has solid waste stored since before the accident, and a large volume of dismantled waste is expected to be generated after the completion of fuel debris retrieval. If the site only increases the storage capacity for solid waste, it will eventually reach the limit, so efforts should be made to reduce the volume of solid waste to be generated as much as possible.

In considering further possibilities of volume reduction, with the aim to reuse/recycle metals with an extremely low surface radiation dose, chemical decontamination (decontamination by phosphoric acid), physical (mechanical) decontamination (steel blasting), and decontamination by melting (decontamination by melting slag) are under consideration as metal decontamination methods for recycling.

As metal recycling with decontamination by melting slag has already been used in many Western countries, it is considered a promising candidate technology. Thus, it is important to focus on the areas where the conditions are different between such Western countries and Fukushima Daiichi NPS (target nuclides, etc.), and to evaluate the applicability of the method.

5.2.3 Processing/disposal

The objective is to establish safe and reasonable processing/disposal methods for all solid waste in which diverse waste streams exist, and to widely obtain knowledge for optimizing each individual stream. Therefore, it is necessary to continue the development/research of processing/disposal technologies required for the series of studies shown in Fig. A11-12.

Regarding the processing technology, outstanding issues in low- and high-temperature processing technology, for which research/development is promoted, should be addressed. Waste streams for which the possibility of solidification has not been investigated will be evaluated as necessary, and performance such as leachability for solidified substances to be produced will be evaluated. As for low-temperature processing technology, consideration is given to the transformation of solidified substances as well as inspection methods to verify the possibility of solidification. In the case of high-temperature processing technology, the feasibility of the whole processing system, including supply and exhaust systems, is an issue, in addition to the solidification process, and therefore it is necessary to carry out examination in a timely manner according to the start time of processing. Furthermore, in order to expand technological options, it is important to examine the possibility of low-temperature solidification after interim treatment such as steam reforming.

Regarding disposal technologies, in order to establish reliable safety assessment techniques, important issues specific to solid waste at the Fukushima Daiichi NPS will be explored and identified based on the understanding of the sensitivity structure of parameters to radiation dose and the long-term transition behavior of disposal facilities. Then, priorities will be examined and incorporated into research plans. Development and improvement of the proposed disposal options will also be promoted in reference to case examples in Japan and abroad. Furthermore, the target of waste streams on which disposal safety assessments will be performed will be expanded, group of the disposal options and categories from a bird-eye-viewpoint of all solid waste at Fukushima Daiichi NPS will be examined, and contributions will be made to considering appropriate measures for overall waste management in coordination with areas other than disposal, such as presenting targets for waste form performance and the accuracy required for characterization.

The main technical issues and plans in the future described above are summarized as shown in Fig. A11-30.

The Mid-and-Long-Term Roadmap states that properties of solid waste will be analyzed and specifications of waste form and their production method of conditioned waste will be determined in the third phase. As a systematic approach toward this goal, further examination will be conducted to present appropriate measures for overall management of solid waste in the (1) of third phase. Specifically, the first step is to develop processing/disposal options by examining the unimplemented issues related to processing technology, interim treatment, and disposal options. Then, the options will be compared and evaluated using the property data and other data that are becoming clear, and the examination will be conducted to identify waste streams suitable for the characteristics of solid waste.

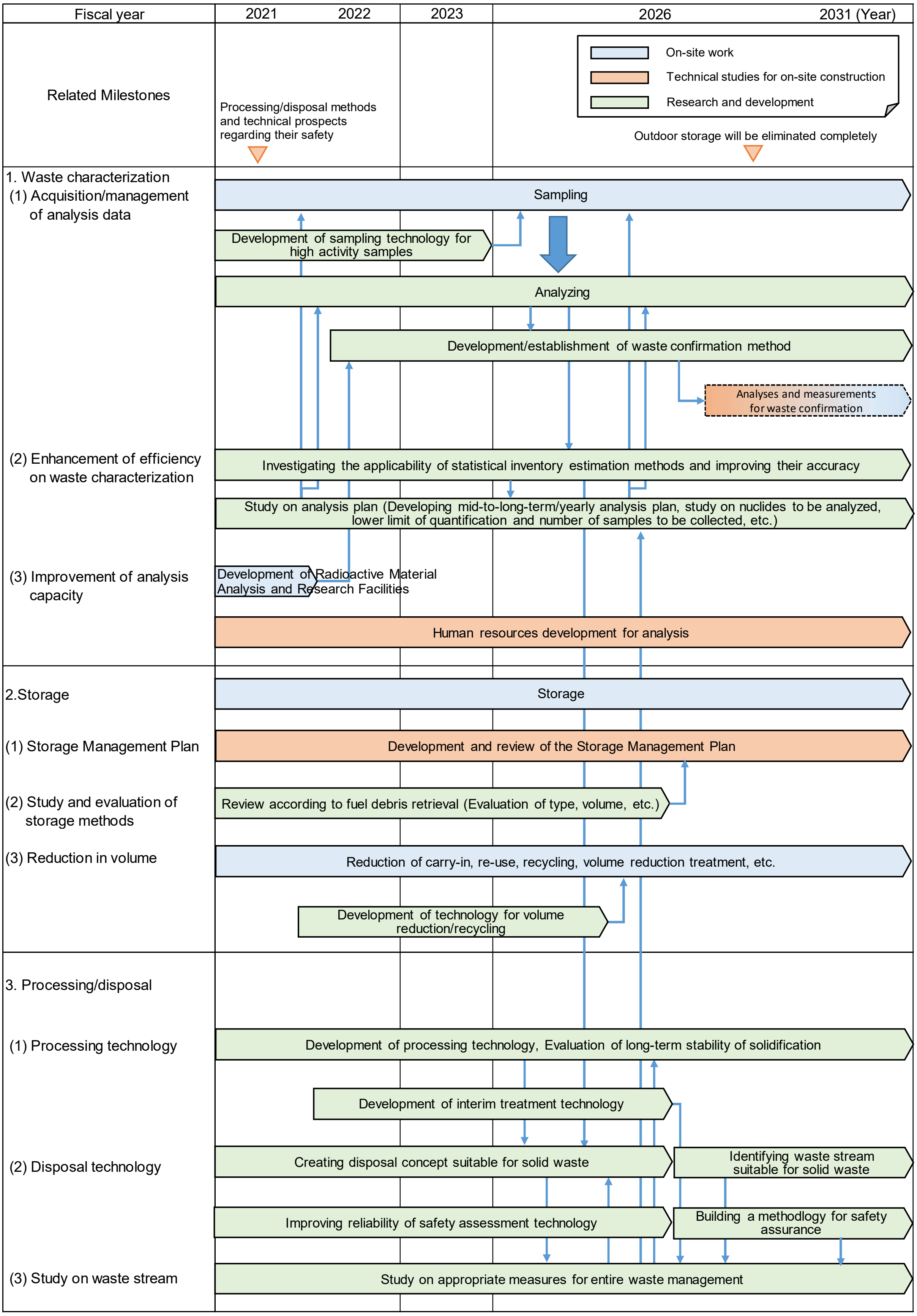


Fig. A11-30 Main technical issues and future plans on waste management (progress schedule)

Attachment 12 Efforts toward discharging ALPS-treated water into the ocean

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1. Introduction

On April 13, 2021, under the overriding principle of Balancing between Reconstruction and Decommissioning, and on the premise of ensuring safety and implementing comprehensive measures to prevent reputational damage, the Government announced the basic policy for discharging ALPS-treated water into the ocean from the Fukushima Daiichi Nuclear Station (1F) after comprehensive discussions at expert meetings for more than six years[155,156]. In addition, the Government requested TEPCO to proceed with preparations, including the installation of a specific discharge system, aiming to start the discharge of ALPS-treated water into the ocean in about two years[157].

In response to this policy, on April 16, 2021, TEPCO indicated their approach to ensuring safety through further efforts and in compliance with regulatory standards pursuant to laws and regulations; minimizing reputational damage; providing compensation in the event of reputational damage; and addressing issues for the future[158]. TEPCO also working to provide a briefing for stakeholders, and to obtain permission for the implementation plan.

This Attachment describes the significance of the policy for disposing of ALPS-treated water, and summarizes past discussions and studies from the viewpoints of the background of announcing the disposal policy; the effects of tritium on the human body; examples of discharge cases and regulatory standards in Japan and overseas; and systems and operation associated with discharge into the ocean. It also provides technical points for achieving offshore discharge.

[155] Tritiated Water Taskforce Report, June 3, 2016

[156] Report by the subcommittee dealing with water treated with multi-nuclide removal equipment, February 10, 2020

[157] Inter-Ministerial Council for Contaminated Water, Treated Water and Decommissioning Issues (5th meeting), Material 1, “Basic policy for disposing of treated water by multi-nuclide removal equipment at the TEPCO Fukushima Daiichi Nuclear Power Station (draft)”, April 13, 2021

[158] TEPCO, Attachment 1. Actions by TEPCO in response to the Government’s basic policy for disposing of treated water by multi-nuclide removal equipment, press release, April 16, 2021

2. Significance of the policy for disposing of ALPS-treated water

Waste liquid generated from nuclear reactor facilities in Japan is allowed to be discharged to the environment in a controlled manner "by reducing the concentration of radioactive materials in the discharged water as much as possible by filtration, evaporation, sorption by the ion-exchange resin method, etc., decay of radioactivity with time, and dilution with a large amount of water, and by preventing its concentration from exceeding the concentration limit established by the NRA" [159,160]. This operation is in accordance with the concept of "Dilution and Dispersion" as specified in Publication 81[161], the International Commission on Radiological Protection (ICRP) and the General Safety Requirement, Part 5, International Atomic Energy Agency (IAEA)[162163], and follows the guide of the IAEA General Safety Requirements, WS- GSG-9[164]. These guidelines require monitoring in order to ensure that discharged water concentrations meet the regulatory limits and that there are no unanticipated discharged. This controlled discharge avoids storing large amounts of waste liquid in the facility, enabling safe operation and management of the facility.

At the Fukushima Daiichi NPS, on the other hand, most of the radioactive materials contained in the contaminated water generated by the accident is removed by multi-nuclide removal equipment, and then the treated water is stored in tanks located on higher ground as ALPS-treated water containing large-volume tritium (ALPS-treated water[165], water under treatment[166]). As most of the contaminated water is stored in welded tanks, the possibility of leakage is very low, but the risk is not zero. At present, however, the number of these tanks exceeds 1000, which makes the space on the plant site tight and poses a serious problem on the sustainability of decommissioning work ahead. In particular, the following issues must be solved.

- Contaminated water is continuously generated[167] by various factors for the time being, and eventually it becomes difficult to secure a transfer destination, which hinders decommissioning work.

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159 Ministerial Ordinance for Commercial Nuclear Power Reactors concerning the Installation, Operation, etc., Article 90, Ordinance of the Ministry of International Trade and Industry 77 of 1978

160 Article 16, Rules on Safety and Physical Protection of Specific Nuclear Fuel Materials in Reactor Facilities at the TEPCO Fukushima Daiichi Nuclear Power Station, Ordinance 2 of the Nuclear Regulation Authority, 2013

161 ICRP, Radiation protection recommendations as applied to the disposal of long-lived solid radioactive waste. ICRP Publication 81. Ann. ICRP 28 (1998).

162 IAEA, Predisposal Management of Radioactive Waste, IAEA Safety Standards, No. GSR-Part5, Vienna (2011)

163 Compliance with the concept of "Dilution and Dispersion" means verifying that the concentration of radionuclides is at a level that does not have a significant health impact, and then discharging ALPS-treated water into the environment where dilution is expected to occur

164 IAEA, Regulatory Control of Radioactive Discharges to the Environment, IAEA Safety Standards, No. GSG-9, Vienna (2018).

165 Water that is treated with multi-nuclide removal equipment until the level of radioactive materials other than tritium falls below the regulatory limit on safety, without fail (the sum of ratios of concentrations required by law, except for tritium, is less than 1).

166 Water that is treated with multi-nuclide removal equipment, and does not meet the regulatory limit on safety (the sum of ratios of concentrations required by law, except for tritium, is 1 or higher).

167 Various factors include inflow of rainwater/groundwater into buildings, transfer from the ground at T.P.+2.5 m, injection of chemicals during purification by ALPS, transfer along with decommissioning work, and emergency transfer. The target of reducing the amount of contaminated water generated in 2025 is 100 m^3/day or less. Refer to "3.3 Contaminated and treated water management" in the main document for details.

- The risk of leakage from tanks due to aging or natural disasters increases with long-term storage.
- It is impossible to secure enough space for systems and facilities necessary for stable, temporary storage of spent fuel as well as safe fuel debris retrieval.

As such, simply increasing the number of tanks does not solve the fundamental problem [Note)]. Therefore, the Government's basic policy has the following significance.

- A risk of not being able to secure a transfer destination for contaminated water can be eliminated permanently, and the sustainability of decommissioning work can be ensured.
- Presenting an outlook for the disposal of ALPS-treated water can result in reducing the risk caused by long-term storage of more than 1,000 tanks.
- Along with more effective use of the entire site, resources spent for tank inspection and maintenance management due to the prolonged storage of ALPS-treated water can be concentrated on more risky decommissioning work.

The announcement of the basic policy for the disposal of ALPS-treated water ensures the sustainability of the overall decommissioning work and contributes to its steady progress.

Though the basic policy for 1F decommissioning is “to continually and quickly reduce the risks associated with the radioactive materials caused by the accident and that do not exist in normal nuclear power plants”, this disposal policy intends to bring the risk level of the water stored in welded tanks, including ALPS-treated water, from the “Sufficiently stable management” region into the level that does not require storage itself as a permanent measure.

Note) Options for waste liquid disposal

Article 16, “Rules on Safety and Physical Protection of Specific Nuclear Fuel Materials in Reactor Facilities at the TEPCO Fukushima Daiichi Nuclear Power Station” specifies as follows:

6 Radioactive waste liquid shall be disposed of by any of the following methods:

(a) Discharge by drainage facilities.

(b) Store and dispose of in waste liquid tanks with radiation hazard prevention.

(c) Enclose in containers or solidify with containers, and then store and dispose of in storage/disposal facilities with radiation hazard prevention.

(d) Incinerate in incineration facilities with radiation hazard prevention.

(e) Solidify in solidification systems with radiation hazard prevention.

7 In the case of disposal by the method described in (a) of the preceding item, the concentration of radioactive materials in the discharged water shall be reduced as much as possible by filtration, evaporation, sorption by the ion-exchange resin method, etc., decay of radioactivity with time, and dilution with a large amount of water in drainage facilities. In this case, the concentration of radioactive materials in the discharged water at drainage outlets and in discharged water monitoring systems shall not exceed the concentration limit specified by the NRA.

On the other hand, contaminated water is stored in tanks as ALPS-treated water after the concentration of radioactive materials is reduced by filtration, and sorption by the ion-exchange resin method, etc., in a water treatment system. Of the above, (b) means transferring liquid from tanks to waste liquid tanks, and “underground burial” compared/evaluated by the Tritiated Water Taskforce falls under this case. (c) and (e) can reduce the risk of leakage by solidification, but constructing additional storage facilities is required. (d) is a measure for organic solvents, etc. Based on the above, a remaining option for ALPS-treated water management is (a). However, when implementing (a), Article 16, Item 7 of the aforementioned Rule should be observed.

3. Discharge of ALPS-treated water into the ocean

3.1 Background of announcing the disposal policy

For the purpose of evaluating various options for the handling of ALPS-treated water, the "Tritiated Water Task Force[155]" was established under the Committee on Countermeasures for Contaminated Water Treatment, and started examination from December 25, 2013. This Task Force consolidated scientific information on tritium as basic data for determining the long-term handling of ALPS-treated water, and reviewed basic requirements (regulatory and technological feasibility) and possible constraints (period, cost, scale, secondary waste, work exposure, etc.) for the five disposal methods of geosphere injection, offshore discharge, vapor discharge, hydrogen discharge and underground burial). Concerning the tritium separation technique, a confirmation test project on the tritium separation technique was implemented in FY 2015. The conclusion was, "Taking the amount and concentration of ALPS-treated water into consideration, there was no technique that could be put to practical application immediately." These results were submitted in a report dated on June 3, 2016.

Based on the findings compiled in the Task Force Report, the Committee on Countermeasures for Contaminated Water Treatment decided to establish "The subcommittee on handling of ALPS-treated water (ALPS Subcommittee)[156]" with the aim of comprehensive review of the handling of ALPS-treated water from a social perspective, including reputational damage, and started examination from November 2016. In August 2018, public briefing/hearing sessions were held in Fukushima Prefecture and Tokyo to hear opinions on how to dispose of ALPS-treated water, as well as concerns about disposal. Then, discussions were held on the issues raised, while verifying the facts from a scientific perspective. Subcommittee meetings were held 17 times over a period of about three years and, on February 10, 2020, a report was compiled as a reference for the Government to evaluate the ALPS-treated water disposal method.

In this report, the proven methods of "vapor discharge" and "offshore discharge" were identified as realistic options. Despite the difference in scale, there is a precedent for "vapor discharge" from an accident reactor. Vapor containing radioactive materials is discharged in a controlled manner even from normal reactors during ventilation. For the purpose of disposing of radioactive waste liquid, however, it was pointed out that there was no case in Japan, where waste liquid was evaporated to gas and discharged as vapor. As for "offshore discharge", on the other hand, it was concluded that this method would be more reliable for reasons that radioactive waste liquid containing tritium has been diluted with seawater used for cooling and discharged into the ocean from nuclear facilities in Japan and abroad, and in terms of track records in normal reactors, ease of handling the discharge system and monitoring methods. However, it was noted that the quantitative relationship between the amount of discharged water and that of tritium discharge would not be equivalent to that before the accident at the Fukushima Daiichi NPS. In addition, from a social point of view, the offshore discharge could have a negative impact on the fishery and tourism industries in Fukushima Prefecture and the surrounding sea areas. Especially, the fish catch from the trial fishing in Fukushima Prefecture has not even recovered to 20% of the level before the Earthquake, and it is necessary to consider countermeasures based on this situation. Concerning the tritium separation technique, while keeping the conclusion that there is no technique that could be put to practical application immediately, studies on new techniques are underway. Thus, it has been determined that continued attention should be paid to technological trends. The International Atomic Energy Agency (IAEA) has evaluated that the conclusion of the ALPS Subcommittee mentioned above is based on scientific and technical grounds[168].

While receiving recommendations from the ALPS Subcommittee, the Government then selected "offshore discharge" as the basic policy, based on opinions obtained through exchanges with local governments and agricultural, forestry and fishery operators, public hearings, and written requests for public comment, in terms of past results of offshore discharge in Japan, and potential of reliable and stable monitoring. This conclusion is supported by the IAEA, as "The offshore discharge is technically feasible and consistent with international practices" and "The offshore discharge in a controlled way is constantly performed in countries operating nuclear power plants around the world"[169].

As described above, "offshore discharge" was announced by the Government as the disposal policy after comprehensive discussions at expert meetings and based on the opinions of a wide range of people. The IAEA, as a third party, has evaluated the policy as reasonable[169].

On the other hand, it is also a fact that there have been concerns about reputational damage due to the discharge of ALPS-treated water into the ocean. Therefore, efforts should be continued to obtain understanding to eliminate such concerns. In addition, the reliability of TEPCO has declined due to inappropriate incidents in terms of physical protection of nuclear materials at the Kashiwazaki-Kariwa Nuclear Power Station and insufficient provision of information during earthquakes at the Fukushima Daiichi Nuclear Power Station. TEPCO needs to take this reality seriously and respond more carefully than before. With this matter, for example, the lack of two-way communication may be one of the reasons for insufficient understanding from local stakeholders. One of the possible factors of the concern from commercial distributors about reputational damage is that information to suppress a negative impact has not been sufficiently provided. In addition, South Korea and China have expressed concern and criticism of Japan's policy, we need to continue our efforts to disseminate accurate information based on scientific evidence to the international community, such as clearly communicating the difference between contaminated water and ALPS-treated water. Therefore, greater transparency is required, for example, by repeatedly providing explanations in an easy-to-understand and careful manner,

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[168] IAEA Review Report (translation for reference), MOFA website, https://www.mofa.go.jp/mofaj/dns/inec/page24_001364.html, April 2020
[169] Statement by IAEA Director General Grossi, MOFA website, https://www.mofa.go.jp/mofaj/dns/inec/page24_001364.html, April 2021

mainly by TEPCO, in order to increase understanding of the basics for safe offshore discharge, i.e., (1) an operation plan for offshore discharge; (2) the effects of tritium contained in the water to be discharged to the ocean on the human body; and (3) the method for verifying the operation status, and by verifying safety through reliable third parties such as IAEA in cooperation with organizations concerned, and by delivering accurate information.

3.2 Impact of tritium contained in the water discharged to the ocean on the human body

Tritium (^3H, T) is a radioisotope of hydrogen whose nucleus consists of 1 proton and 2 neutrons, and decays to ^3He with a half-life of 12.3 years. At that time, it emits faint β-rays (18.6 keV at maximum, 5.7 keV on average), but the maximum range is very short at 5 mm in air and 6 μm in water. On earth, 70 quadrillion Bq of tritium is produced annually by nuclear reaction of neutrons produced by cosmic rays with nitrogen and oxygen in the atmosphere[170]. Tritium exists primarily as water molecules, and is contained in any water, including water vapor in the atmosphere, rainwater, seawater, tap water, and even in the human body. Water molecules containing tritium (tritiated water) have the same chemical properties as ordinary water molecules, and tritium is not concentrated in specific living organisms or organs. The human body also contains tens of Bq of tritium due to ingestion of drinking water.

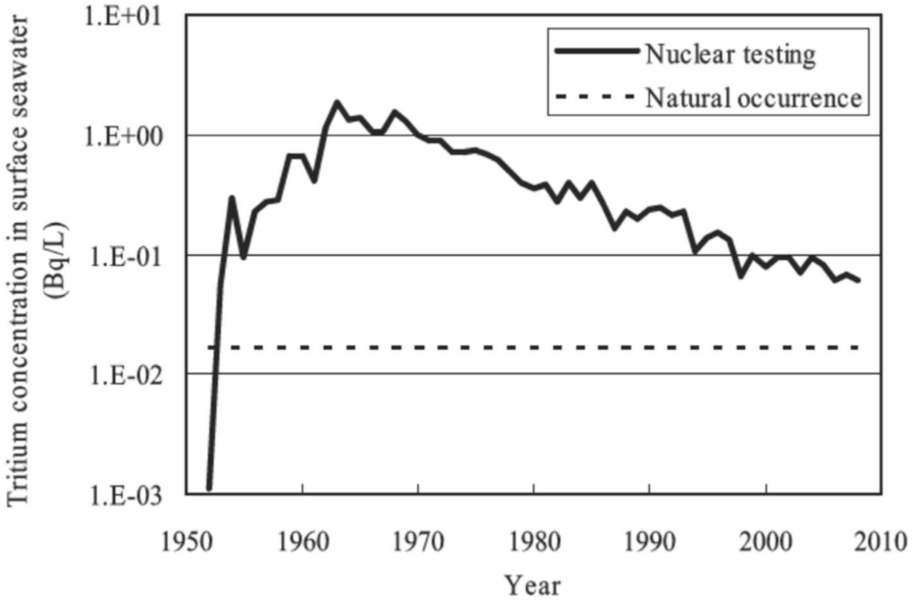
In nuclear power plants and nuclear fuel reprocessing plants, tritium is produced by ternary fission of ^{235}U in fuel, and by reaction of deuterium (^2H, D) and lithium contained in cooling water as water molecules, and boron and neutron in control rods. Since tritium exists as water, unlike other radioactive materials, it is technically difficult to recover and contain, and the effect of β-rays from tritium on the human body is very small compared with other radioactive materials. Therefore, tritium is discharged into the ocean, rivers, lakes/reservoirs, and the atmosphere according to the regulations of each country.

Fig. A12-1 shows the result of evaluating the tritium concentration in surface seawater based on the "Long-term assessment model of radionuclides in the sea (LAMER)[171]" [172]. As a result of atmospheric nuclear tests conducted in the 1950s - 1960s, it is estimated that $(1.8 - 2.4) \times 10^{20}$ Bq was discharged into the environment[170]. In 1963, the tritium concentration in surface seawater is estimated to have risen to a level exceeding 1 Bq/L. Subsequently, the tritium concentration caused by nuclear tests decreased year by year due to the decay of the radioactive materials and transfer to seawater, and it is considered that the tritium concentration has recently fallen below 0.1 Bq/L. The concentration of naturally occurring tritium is estimated to be about 0.02 Bq/L.

170 Hideki Kakiuchi, Tritium in the environment and its evaluation methods, Journal of the Atomic Energy Society of Japan, Vol. 60, No. 9, pp.537-541 (2018)

171 Masanao Nakano, LAMER; Long-term assessment model of radionuclides in the ocean, JAEA-Data/Code 2007-024(2008)

172 Masanao Nakano et al., Tritium Concentration and Diffusion in Seawater Discharged from Tokai Reprocessing Plant, Japanese Journal of Health Physics, 44(1), pp.60 - 65 (2009)



(Source : Masanao Nakano, et al., Concentration and diffusion of tritium in seawater discharged from Tokai Reprocessing Plant, Physics in Health)

Fig. A12-1 Changes in tritium concentration in surface seawater[172]

UNSCEAR 2016 Report, Annex C - Biological effects of selected internal emitters - Tritium complies a wealth of systematic information on the effects of tritium on the human body. As a model for exposure assessment, this report indicated a metabolism model that takes into account the fact that a portion of tritiated water changes its form to organically bound tritium (OBT) when ingested into the human body[173]. Tritiated water has an average biological half-life of about 10 days and is eliminated from the body, but about 3 - 6% of tritiated water changes to OBT in the body. The biological half-life of OBT is estimated to be about 40 days for the short half-life component and about 350 days for the long half-life component. Therefore, the committed effective radiation dose of OBT is estimated to be about 2 - 5 times that of tritiated water.

As described above, in the tritium exposure assessment model, tritiated water and OBT derived from tritiated water discharged from nuclear facilities are evaluated as not concentrating in specific living organisms or organs, although the period until they are eliminated from the body is longer[1773]. In the UK, tritium concentrations in bivalves and flounders have been found to be several 1000 times higher than in seawater[174]. However, subsequent investigations have revealed that this is due to artificially synthesized tritiated organic compounds contained in the discharged water from a nearby chemical plant[175] that produces research medicines containing radioactive materials[176,177], tritium discharged from nuclear power plants is not the cause.

173 UNSCEAR 2016 Report, Annex C - Biological effects of selected internal emitters-Tritium

174 Environment Agency, Food Standards Agency, Northern Ireland Environment Agency and Scottish Environment Protection Agency, Radioactivity in Food and the Environment 2002, RIFE-8 (2003)

175 In this chemical plant, various tritiated organic compounds such as amino acids, fatty acids, and carbohydrates labeled with tritium were produced for medical research, etc.

176 McCubbin D. et al., Incorporation of organic tritium (3H) by marine organisms and sediment in the Severn Estuary/Bristol Channel (UK) Mar. Poll. Bull. 42 852–63 (2001)

177 A. Hunt et al., Enhancement of tritium concentrations on uptake by marine biota: experience from UK coastal waters, J. Radiol. Prot. 30, 73–83(2010)

A correlation between tritium discharges and newborn infant mortality was reported[178] in local governments near a nuclear power plant in Ontario, Canada, from 1974 to 1985. However, the Canadian nuclear regulatory authorities subsequently expanded the data from 1971 to 1988 for analysis, and found no correlation between tritium discharges and newborn infant mortality[179]. Moreover, there have been no reports of effects on the human body that could be caused by tritium in the vicinity of nuclear facilities that discharge tritium[156].

In France, the regulatory authorities issued the Tritium White Paper in coordination with the Local Information Committee (CLI: Commission Local d'Information), including fishery workers, and by incorporating the views of various stakeholders, including operators, local residents, environmental groups, and congress members[180]. The White Paper confirms that tritium has little impact on the human body, while providing an action plan on tritium concentration measurement, discharge control, and environmental monitoring, which is constantly checked by parties concerned including above members[181]. In France, these activities by CLI have continued for over 40 years, and played an important role in helping local residents understand the effects of tritium on the human body and the environment.

Table A12-1 shows a list of reports on the effects of tritium on the human body. The Tritium Taskforce[155] and ALPS Subcommittee[156] refer to updated status in the world. In addition, opinions received at public briefings/hearings on the handling of treated water by multi-nuclide removal equipment are discussed based on scientific information, and their findings are summarized[156156]. Examples of typical questions and answers on the effects of tritium on the human body are shown in Table A12-2.

Using the exposure assessment model presented in UNSCEAR 2016[182], the effects of radiation were estimated in the case of continued disposal of all ALPS-treated water stored in tanks every year. The result showed that the radiation exposure dose associated with offshore discharge would be approximately 0.071 - 0.81 μSv per year, which is less than 1/1,000 of the natural exposure of 2.1 mSv (2,100 μSv) per year[183].

178 McArthur, D., Fatal Birth Defects, Newborn infant fatalities and tritium emissions in the town of Pickering, Ontario: A Preliminary Examination. A study by David McArthur, Toronto, Ontario, November 30 (1988)

179 Tritium releases from the Pickering Nuclear Generation Station and birth defects and infant mortality in nearby communities 1971-1988, Atomic Energy Control Board, Canada (1991)

180 ASN, The Tritium White Paper (2010)

181 Tritiated Water Taskforce Report (7th meeting), meeting minutes, April 9, 2014

182 UNSCEAR 2016 Report, SOURCES, EFFECTS AND RISKS OF IONIZING RADIATION

183 17th ALPS Subcommittee meeting, Material 3-2, Explanation on the UNSCEAR 2016 model, January 31, 2020

Table A12-1 Reports on the effects of tritium on the human body

Time of publication	Publishing organization	Document name	Objective
2007	International Commission on Radiological Protection (ICRP)	ICRP Publication 103	To update radiation and tissue weighting factors at equivalent and effective doses based on the latest scientific information in biology and physics on radiation dose exposure, and thereby update information on radiation damage.
2007	Health Protection Agency (HPA)	Reviews of risks from Tritium	To provide scientific opinions on internal radiation exposure doses and risks due to tritium, and to account for diverse point of views.
2010	Canadian Nuclear Safety Commission (CNSC)	CNSC 2010 Report	To perform an independent review of the scientific literature to assess the health risks to workers and the public caused by tritium exposure, to evaluate Canadian and international dosimetry on tritium intake, and to review approaches to limit tritium exposure.
2010	French Nuclear Safety Authority (ASN)	The Tritium White Paper	To provide detailed analysis of evaluation methods for the environmental dynamics of tritium and biological effects of tritium on the human body.
2016	United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR)	UNSCEAR 2016 Report Annex C. Biological effects of selected internal emitters - Tritium	To underpin radiation risk assessment and international radiation protection standards with the scientific knowledge compiled by UNSCEAR.

Table A12-2 Example questions and answers on the effects of tritium on the human body[156]

Q. Does tritium bioaccumulate?
A. Tritium does not bioaccumulate because it has the same properties as ordinary water.
Q. If tritiated water changes to organically bound tritium (OBT), does it become a harmful substance?
A. At about 3 - 6%, the rate of change to OBT in the body is small, but as the biological half-life becomes longer in the body, the health effects are likely to be two to five times greater. However, the health effects of tritium are originally as small as 0.000000019 per Bq, and it cannot be said that the health effects are especially large compared with other radioactive materials. This is about 1/300 of the effect on health from cesium.
Q. If the tritium atoms that compose genes change into helium atoms, will the genes be damaged?
A. Genes are constantly damaged and repaired daily by repair enzymes. The damage received by radiation of about 2 mSv per year is very small, less than 1 million times the frequency of the damage from the ultraviolet rays of the sun.
Q. Does tritium intake increase carcinogenicity?
A. In experimental carcinogenesis on mice, it has been reported that the incidence of cancer is within the range of the spontaneous incidence rate even if mice continue to drink tritiated water at a concentration of approximately 140 million Bq/L, and therefore the effect is small.
Q. Are there any examples of damage to health that could be caused by tritium in the vicinity of nuclear facilities that discharge tritium?
A. No examples of possible tritium-induced effects have been found.

(Source : METI)

3.3 Case examples of discharge in Japan and overseas, and regulatory standards

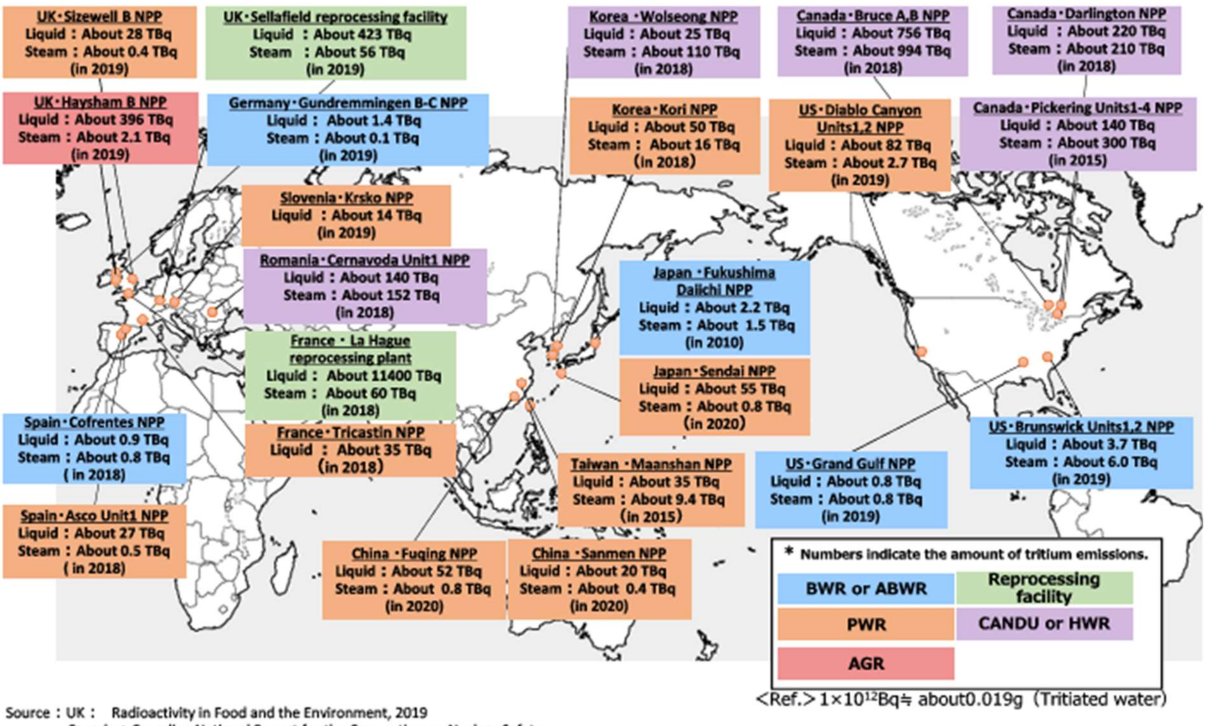
(1) Current situation of discharging radioactive waste liquid containing tritium in Japan and abroad

Radioactive waste liquid generated from nuclear facilities is allowed to be discharged to the environment in a controlled manner by reducing the concentration of radioactive materials as much as possible by removing and decay of the radioactive materials and dilution with a large amount of water, and by satisfying the regulatory standards in various countries. There are many case examples of discharge in Japan and abroad[159,164].

Fig. A12-2 shows the annual discharge of tritium from major nuclear power plants and reprocessing facilities in Japan and abroad. Based on the characteristics of each facility, the annual discharge tends to be: Reprocessing plants > Heavy water reactors/advanced gas reactors >

Pressurized water reactors > Boiling water reactors. In nuclear power plants in Japan, about 18 - 83 trillion Bq/year of tritium is discharge d into the ocean from pressurized water reactors, and about 31.6 billion - 1.9 trillion Bq/year from boiling water reactors. Since deuterium and boron are contained in the core cooling water of heavy water reactors and pressurized water reactors, the amount of tritium generated is larger than that of boiling water reactors. At 1F, before the accident (FY 2010), about 2.2 trillion Bq of tritium was discharged into the ocean annually.

In foreign countries, tritium is discharged from nuclear facilities as in Japan. From the La Hague Reprocessing Plant in France, about 10 quadrillion Bq of tritium is discharged into the ocean annually. Fig. A12-3 shows examples of liquid tritium discharge from neighboring Asian countries and regions. Some sites discharge around 100 trillion Bq per year[184].



Source : UK : Radioactivity in Food and the Environment, 2019
Canada : Canadian National Report for the Convention on Nuclear Safety
France : Tritium White paper
Other countries and regions : Prepared from reports published by electricity providers in various countries and regions.

Fig. A12-2 Annual discharge of tritium from nuclear facilities in Japan and abroad[185]

[184] Detailed explanatory material on ALPS-treated water, available on METI’s website https://www.meti.go.jp/earthquake/nuclear/hairo_osensui/alps.html, April 2021
[185] The 1st Expert Meeting for marine monitoring on ALPS-treated water, Reference 2, June 18, 2021

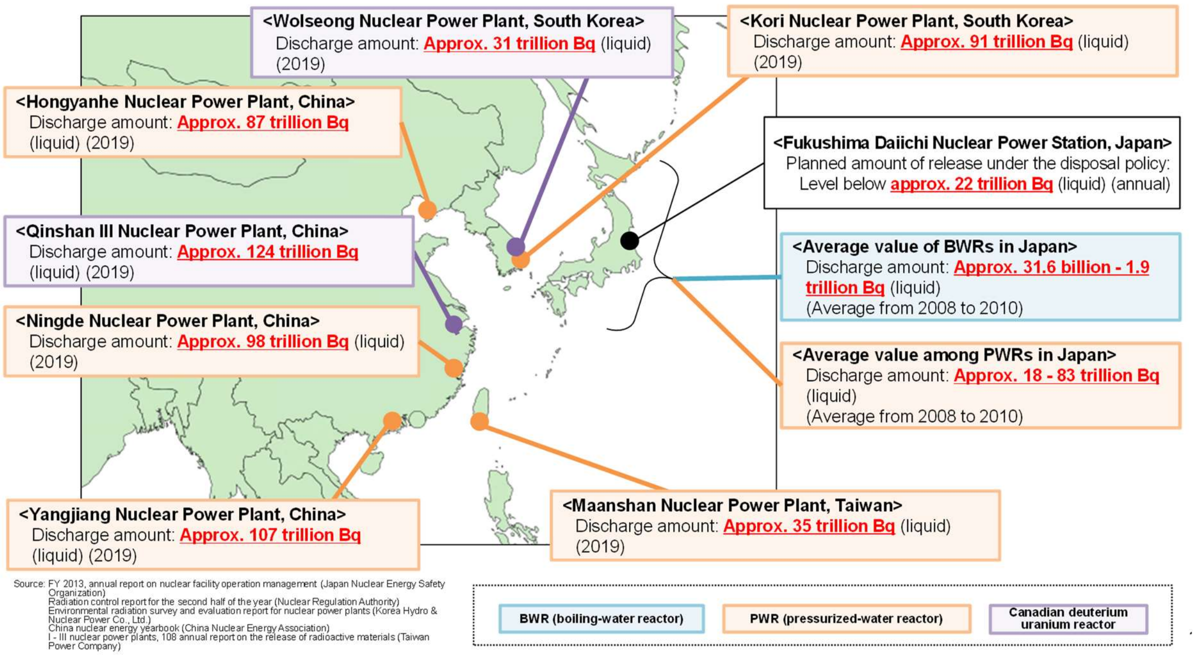
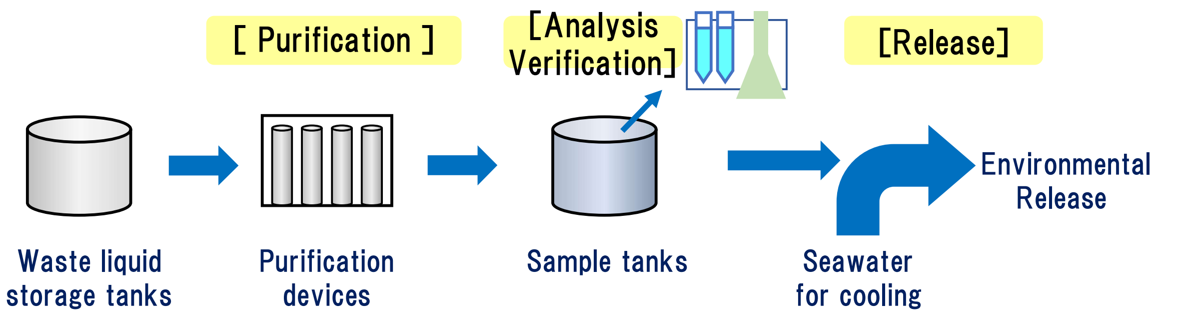


Fig. A12-3 Case examples of annual discharge of tritium in neighboring Asian countries/regions[184]

These nuclear facilities discharge tritium by means appropriate to the characteristics of each facility in order to minimize the radiological impact on the surrounding environment as much as possible. Fig. A12-4 shows the treatment flow up to discharging waste liquid containing tritium in nuclear power plants in Japan. Waste liquid generated in a power plant is [purified] to reduce the concentration of radioactive materials in the waste water as much as possible by filtration, evaporation, sorption by the ion-exchange resin method, etc., and decay of radioactivity with time. The purified liquid is collected in a sample tank where the concentration of the radioactive material is [analyzed and verified] by monitoring and sampling. Subsequently, waste liquid containing tritium is mixed with seawater used for cooling and then discharged to the environment.

As such, [purification] → [analysis/verification] → [discharge] is the basic process, and the discharge is in compliance with the regulatory standards on the concentration of radioactive materials. In nuclear power plants in Japan, radiation monitors are installed to measure γ-rays before radioactive materials are discharged into the environment to monitor concentration.

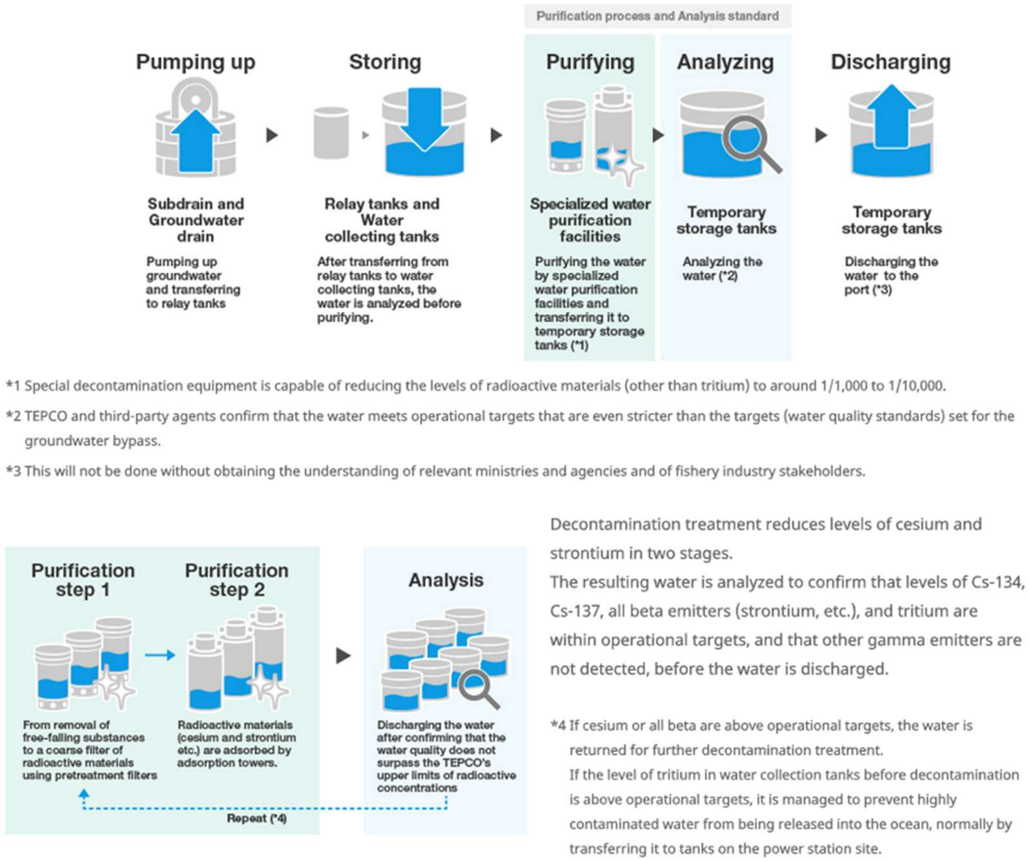


Source : Prepared by NDF with reference to materials on domestic nuclear power plants

Fig.A12-4 Treatment flow up to discharging waste liquid containing tritium in nuclear facilities

(2) discharging pumped water from sub-drains at 1F into the ocean

At 1F after the accident, in order to reduce the amount of groundwater flowing into the reactor buildings and to reduce the amount of contaminated water generated, the groundwater pumped up from the sub-drain installed in front of the buildings, and the groundwater pumped up from the groundwater drain installed in the revetment area to prevent overflow of the groundwater by sea-side impermeable walls, are purified before being discharged into the ocean (Fig. A12-5). At that time, the groundwater pumped up from the sub-drain and groundwater drain is purified, and its concentration is measured to confirm that the concentration is less than the target concentration in all tanks before being discharged. As in (1), the process of [purification] → [analysis/verification] → [discharge] is applied.



(TEPCO material edited by NDF)

Fig. A12-5 Flow from pumping from sub-drains to discharge [186]

(3) Regulation and management of tritium in nuclear power plants

a. Regulation on concentration

Regulations on radioactive materials in Japan are provided in accordance with the recommendations of the International Commission on Radiological Protection (ICRP) (basis for international laws and regulations). According to the ICRP, 1990 Recommendations, “the public

[186] Overview of wells (sub-drains) near the buildings, TEPCO website. https://www.tepco.co.jp/decommission/progress/water management/subdrain/index-j.html

dose during normal times should be less than 1 mSv per year”. The regulatory standards for nuclear power plants in Japan specify the concentration limits (concentration limits required by law) of radioactive materials contained in liquid and gas to be released to the environment according to the type. The concentration limit in water required by law is the concentration at which the average radiation dose rate reaches 1 mSv per year when a person continues to drink approximately 2 liters of water at the discharge outlet every day from birth until age 70. The concentration limit in air required by law is the concentration at which the average radiation dose rate reaches 1 mSv per year when a person continues to breathe air at the site boundary every day from birth until age 70. Based on these standards, the concentration limits of tritium required by law are 60,000 Bq/L in water and 5 Bq/L in air. However, the target concentration of tritium in the water generated by purifying the groundwater pumped up from sub-drains at 1F is set at 1,500 Bq/L, one-fortieth of the concentration limit required by law (60,000 Bq/L), which is sufficiently low.

In the US, the liquid concentration is defined at 37,000 Bq/L and the air concentration (water vapor) at 3.7 Bq/L, both of which are evaluated so that the radiation exposure dose does not exceed 1 mSv/year, taking into account the daily habits and other factors in the US. Similar values (in water: 40,000 Bq/L, in air: 3 Bq/L) have been established in South Korea. France, the UK, and Canada, on the other hand, have not established uniform concentration limits.

International standards for drinking-water, including those of the World Health Organization, are shown in Table A12-3[170]. For example, the WHO Guidelines for Drinking-Water Quality specifies the concentration of 10,000 Bq/L as the guidance level, which would result in the 0.1 mSv annual dose if ingested for a year (730 L/year: Equivalent to 2 L/day). There are no specific regulatory standards for tritium in drinking water and food in Japan (The regulatory limit is specified for the concentration of tritium at the time of discharge for control). The ICRP recommends an annual radiation dose limit of 1 mSv for public exposure under normal conditions. Based on this recommendation, each country has established the limits with different margins[170]. The EU has defined a tritium concentration of 100 Bq/L as a screening value, requiring to survey the presence of artificial radionuclides other than tritium, if exceeded.

Table A12-3 Concentration limits of tritium in drinking-water and exposure radiation doses if ingested for a year[170]

Country/Organization*	Tritium concentration limits (Bq/L)	Exposure dose (mSv/yr)
USA	740	0.01
Canada	7,000	0.09
Russia	7,700	0.1
Switzerland	10,000	0.13
WHO	10,000	0.13
Finland	30,000	0.4
Australia	76,103	1

* The screening value of 100Bq/L is used as an indicator to determine the need for additional investigation in EU.

(Atomic Energy Society of Japan material edited by NDF)

b. Total volume management

Prior to the accident at 1F, “the Discharge Control Target” for tritium for discharge into the ocean was 22 trillion Bq per year. “The Regulatory Guide for the Annual Dose Target for the Public in the Vicinity of Light Water Nuclear Power Reactor Facilities” (Decision of the Atomic Energy Commission of 1975) defines the target value for the discharge control within the range that enables it to meet the Target Radiation Dose of 0.05 mSv/year, which is a goal to maintain the low level of the radiation dose to the surrounding public due to the release of radioactive materials to the environment during normal operation. The guide also requires efforts to not exceed this Discharge Control Target. The Target Radiation Dose does not substitute for the regulatory concentration limit, and failure to achieve it should not be construed as a safety hazard. If 22 trillion Bq of tritium were to be discharged into the ocean from 1F, the radiation effects would be conservatively estimated at 0.00001 mSv/year (sufficiently lower than 0.05 mSv/year)[156]. After the accident at 1F, no discharge control targets or limits have been specified.

Though the discharge control target does not substitute for the regulatory limit, it is the target value that should not be exceeded under the concept of As Low As Reasonably Achievable without being satisfied that it is enough if it falls below the legal regulatory limit. While respecting this philosophy, at 1F, it is necessary to provide management using appropriate methods according to the facility conditions.

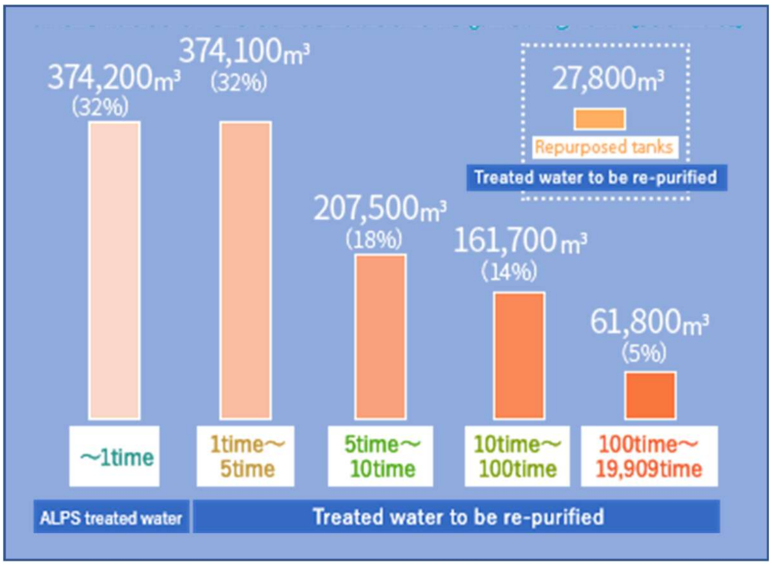
3.4 System and operation for offshore discharge

(1) Properties of ALPS-treated water

ALPS-treated water, etc. (ALPS-treated water, water under treatment) is the water after purifying the contaminated water in buildings with purification systems such as ALPS. Though its properties greatly differ from those of the contaminated water in buildings, it contains tritium that cannot be removed physically or chemically by purification devices. ALPS is a device that performs purification of radioactive materials other than tritium to bring them below the concentration limit required by law. About 70% of the ALPS-treated water currently stored in tanks contains water (water under treatment) exceeding the legal limit (the sum of concentration ratios required by law is less than 1) for discharging radioactive materials, except for tritium, into the environment.

This is due to the fact that in FY 2013 during the initial period of ALPS operation before performance was improved, there were cases where the legal limit for discharge was exceeded. Another reason was that the effective radiation dose (additional radiation exposure dose) at the site boundary exceeded 1 mSv per year due to highly contaminated water, etc., generated and stored in tanks after the accident, and thus priority was given to reducing the additional radiation exposure dose by reducing the frequency of ALPS sorbent replacement and increasing the amount of treatment, instead of satisfying the effluent standard. Specifically, from 2013 to the end of 2015, priority was given to reducing the additional exposure dose to less than 1 mSv/year as soon as possible, and the frequency of ALPS sorbent replacement was reduced to increase the amount of treatment. In addition, although the additional radiation exposure dose of less than 1 mSv/year has been achieved since FY 2017, priority was given to treating the strontium-treated water stored in flanged tanks with a high risk of leakage, by the end of 2018, and the amount of treated water was increased by reducing the frequency of sorbent replacement. As a result, the water under treatment stored in tanks exceeds the limit for discharging into the environment.

Therefore, the concentration of the ALPS-treated water stored in tanks varies depending on the operation of ALPS (frequency of sorbent replacement, etc.) and the time of treatment. Fig. A12-6 shows the storage volume of treated water by ratios of concentrations required by law. Approx. 30% of the entire storage volume has the ratios of concentrations required by law less than 1, and 1 or higher for approx. 70%. For water with a ratio of 1 or higher, it is planned to undergo secondary treatment to reduce it to less than 1. Detailed data on such ALPS-treated water are available on the TEPCO website and are updated regularly.

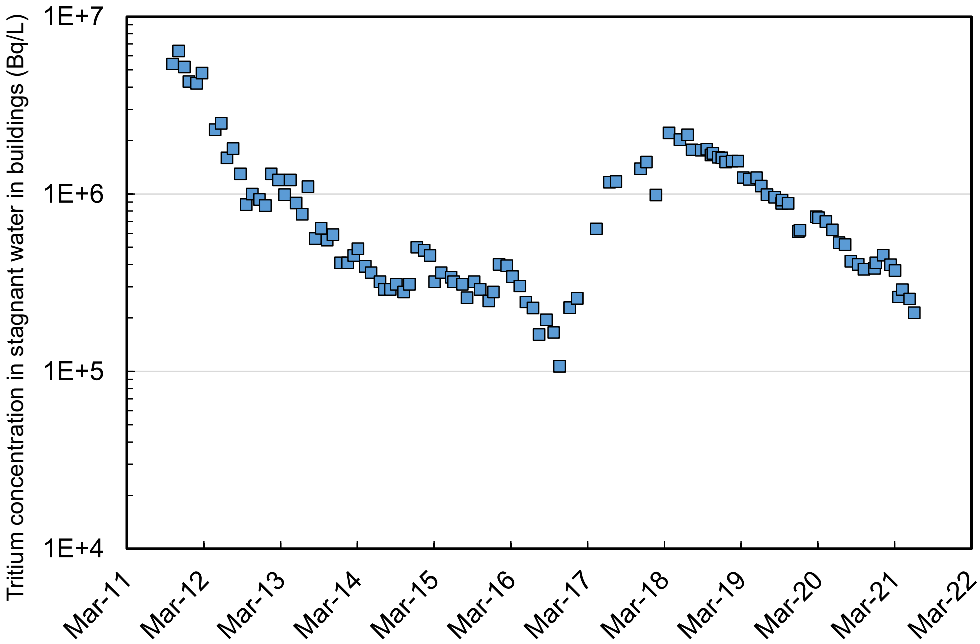


(Source : TEPCO)

Fig. A12-6 Storage volume of treated water by ratios of concentrations required by law (as of March 31, 2021)[187]

In addition, Fig. A12-7 shows changes in tritium concentration in stagnant water in buildings (results of concentrated water analysis of desalination unit RO). The tritium concentration, which was high immediately after the accident, decreased monotonously until the middle of 2016 due to the injection of reactor cooling water and the inflow of rainwater and groundwater. Later, as the water level in the buildings decreased, the concentration increased due to the pumping up of stagnant water containing highly-concentrated tritium that existed at the bottom of the buildings. However, the tritium concentration peaked in March 2018 and has been declining again. As described above, the ALPS-treated water contains tritium of different concentrations depending on the time of treatment.

[187] TEPCO, Treated Water Portal Site, website, https://www.tepco.co.jp/decommission/progress/watertreatment/



(Source : TEPCO)

Fig. A12-7 Changes in tritium concentration in stagnant water in buildings (results of concentrated water analysis of desalination unit RO)[188]

(2) Outline of the discharge system for offshore discharge

Fig. A12- 8 and (Source : TEPCO)

Fig. A12- 9 show conceptual drawings of the discharge system used for offshore discharge and the overall system image, announced by TEPCO, respectively[189,190]. Of the ALPS-treated water stored in reservoir tanks on site, the water under treatment that does not satisfy the legal limit for discharge (the sum of concentration ratios of nuclides required by law is less than 1) undergo secondary treatment and purification as many times as necessary until it becomes ALPS-treated water that satisfies the legal limit for discharge. Such ALPS-treated water is transferred to the measurement/verification system (approx. 10,000 m^3/tank group x 3) and stirred there, the concentrations of radioactive materials, specifically, tritium, the 62 nuclides (nuclides removed by ALPS) and carbon 14 are measured in order to verify whether the concentrations of these materials, except for tritium, satisfy the legal limit. Then, the ALPS-treated water is transferred to the header pipe, and mixed with seawater taken separately from the outside of the port so that the tritium concentration after dilution becomes less than 1,500 Bq/L. The mixed and diluted water is discharged into the ocean through a drainage pit equipped with a partition through an underwater tunnel about 1 km long. Emergency shut-off valves are installed to suspend discharge in the event

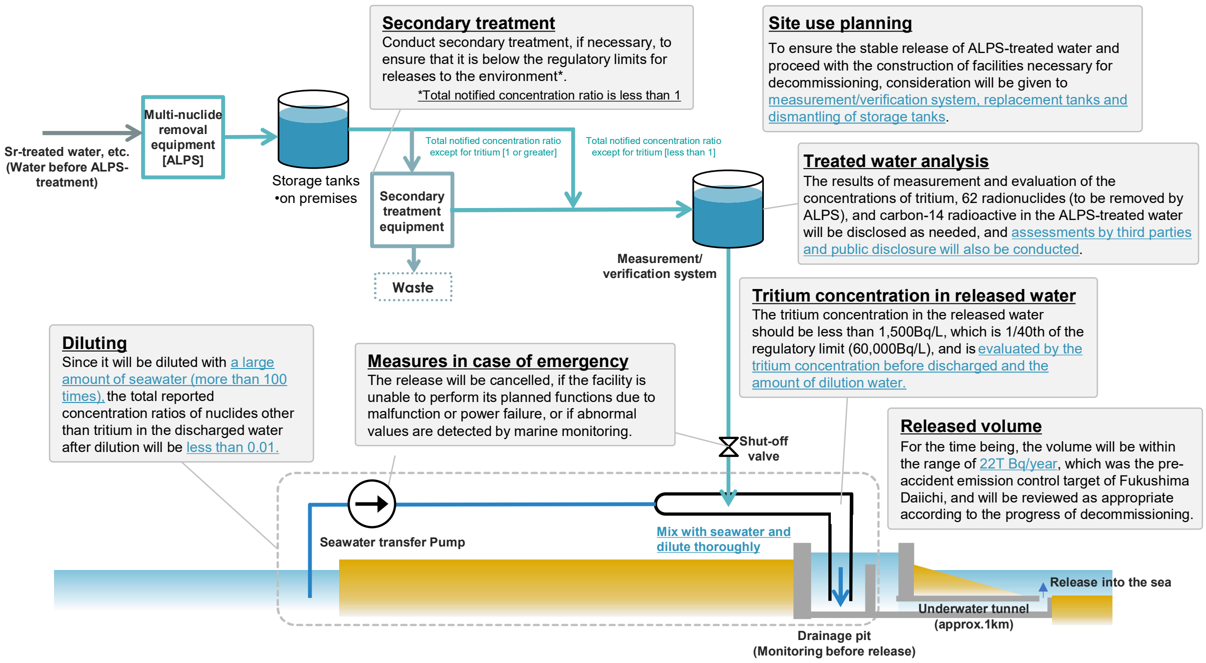
[188]TEPCO, Results of daily analysis of radioactive materials at the Fukushima Daiichi NPS, https://www.tepco.co.jp/decommission/data/daily analysis/

[189]TEPCO, Attachment 2: TEPCO Holdings' Action in Response to the Government's Policy on the Handling of ALPS-Treated Water [Digest version], press release, April 16, 2021

[190]TEPCO, Status of Review Regarding the Handling of ALPS-Treated Water at the Fukushima Daiichi Nuclear Power Station, press release, August 25, 2021

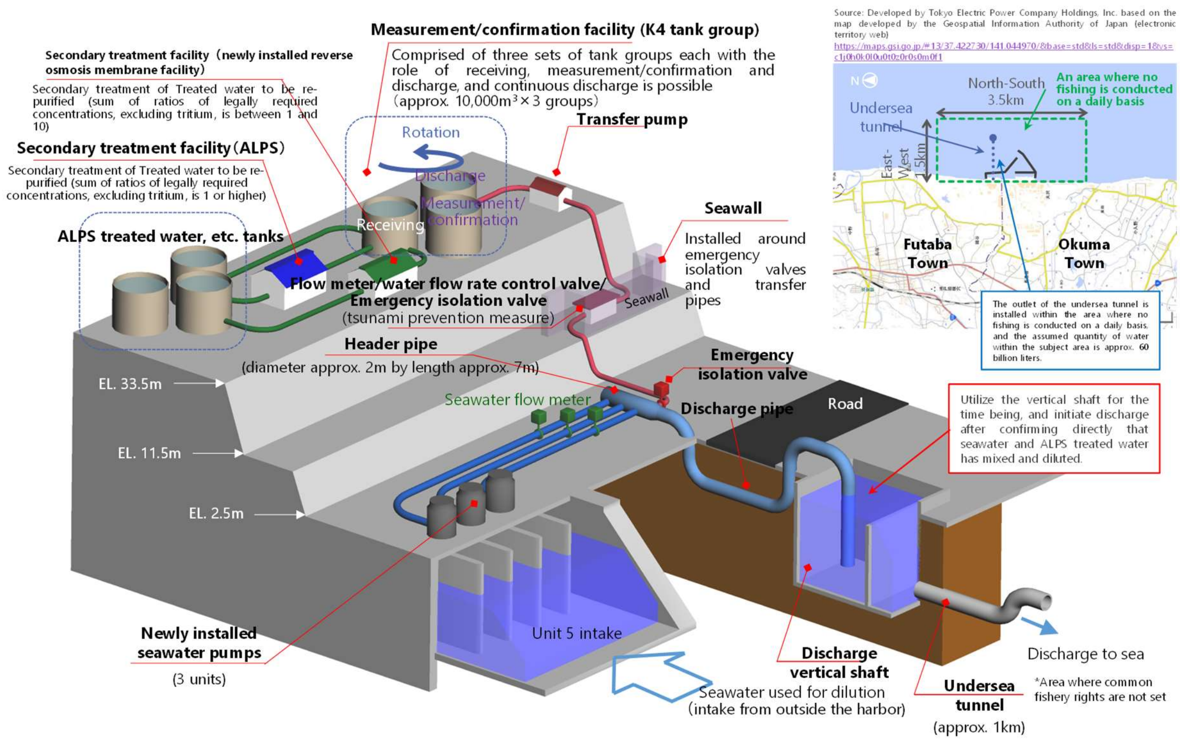
of any system failure of the seawater transfer pumps, etc. or any abnormal value is confirmed by marine monitoring.

[Conceptual diagram of the offshore releasing system]



(TEPCO material edited by NDF)

Fig. A12- 8 Conceptual diagram of the discharge system for offshore discharge [190]



(Source : TEPCO)

Fig. A12- 9 Overall image of the discharge system for offshore discharge [190]

This system used for offshore discharge consists of the process from [purification] → [analysis/verification] → [dilution] → [discharge]. Technically, the utilization of technology and systems with past results enables the establishment a highly reliable system. After the discharge, marine [monitoring] is planned to oversee whether the ALPS-treated water is discharged into the ocean in a safe way.

These systems will be approved by the NRA. The following shows the details and essential technical points of the secondary treatment, analysis, dilution and discharge, and emergency measures of this discharge system for offshore discharge.

a. Secondary treatment of ALPS-treated water

As described in the section on the properties of ALPS-treated water, since the water under treatment contains 62 nuclides and carbon 14, whose sum of ratios of concentrations required by law is 1 or higher, except for tritium, purification by secondary treatment is required. As shown in Fig. A-12-6Fig. A12-6, in particular, there are some tanks whose sum of ratios of concentrations required by law exceeds 10,000 at the highest. For these tanks, it is planned to repeat secondary treatment by purification by ALPS or reverse osmosis membrane device until the ratio becomes less than 1. In order to confirm the feasibility of the secondary treatment, purification testing (performance confirmation test for secondary treatment) is underway, using the water stored in the tank groups whose sum of ratios of concentrations required by law is high[191]. As part of this test, of the water under treatment, performance confirmation testing for the secondary treatment using the expanded ALPS has been carried out since September 2020, for the tank group (Group J1-C) of comparatively high concentration and the tank group (Group J1-G) of comparatively low concentration. The analysis of 62 nuclides, carbon 14, and tritium has also been performed for the

[191] The 85th Secretariat Meeting of the Team for Countermeasures for Decommissioning and Contaminated Water Treatment, Material 3-1, Status of performance verification testing on secondary treatment of treated water by multi-nuclide removal equipment, December 24, 2020

water after secondary treatment.

■ J1-C group (62 nuclides + C-14 + H-3)

		Concentration limits required by law [Bq/L]	Before secondary treatment (System inlet)*1 Analysis results [Bq/L]	Before secondary treatment (System inlet)*1 Concentration ratios required by law*3	After secondary treatment (Sample tank)*2 Analysis results [Bq/L]	After secondary treatment (Sample tank)*2 Concentration ratios required by law*3
Seven major nuclides	Cs-134	60	2.93E+01	0.49	<7.60E-02	0.0013
Seven major nuclides	Cs-137	90	5.99E+02	6.7	1.85E-01	0.0021
Seven major nuclides	Co-60	200	3.63E+01	0.18	3.33E-01	0.0017
Seven major nuclides	Ru-106	100	<5.00E+00	0.050	1.43E+00	0.014
Seven major nuclides	Sb-125	800	8.30E+01	0.10	2.26E-01	0.00028
Seven major nuclides	Sr-90	30	6.46E+04	2,155	3.57E-02	0.0012
Seven major nuclides	I-129	9	2.99E+01	3.3	1.16E+00	0.13
	C-14	2,000	1.53E+01	0.0076	1.76E+01	0.0088
	H-3	60,000	8.51E+05	14.2	8.22E+05	13.7

	Before secondary treatment (System inlet)*1	After secondary treatment (Sample tank)*2
Sum of the concentration ratios of the seven major nuclides required by law	2,165	0.15
Sum of the concentration ratios of the 62 nuclides*4 + C-14 required by law	2,406	0.35

Of the 0.35, the sum of the concentration ratios of the nuclides (51 nuclides) required by law, which analysis and evaluation confirmed was below the lower detection limit, is 0.19.

*1: Composite analysis was performed on the samples collected on September 19, 20 and 21.
*2: Analysis was performed on the samples collected on September 27.
*3: The value of the lower detection limit was used for calculating nuclides whose analysis results are below the lower limit of detection.
*4: See Reference for details of analysis results and the concentration limits required by law.

Numerically expressed, ○.○○E ±XX means ○.○○ × $10^{\pm XX}$.

(Source : TEPCO)

Table A12-5 and A12-5 show the results of the performance confirmation testing on secondary treatment. It is confirmed that the sum of ratios of concentrations required by law decreases from 2406 to 0.35 in the J1-C group tank water and from 387 to 0.22 in the J1-G group tank water by a single secondary treatment, and that the sum of ratios of concentrations required by law can be reduced to less than 1.

In addition, in order to verify the reliability of the analysis, a third-party organization has conducted analysis of the water generated by secondary treatment of the J1-C group tank water. As a result, the sum of ratios of concentrations required by law was 0.28 compared to 0.35 reported by TEPCO, and the outcomes were almost equivalent[192]. In this way, third-party analysis also confirmed that the sum of ratios of concentrations required by law became well below 1 after the secondary treatment.

As described above, the removal technology has already been verified for water under treatment whose sum of concentration ratios required by law is 100 or higher. It is concluded that it is possible to implement secondary treatment to sufficiently lower the concentration ratios to below the regulatory limit on safety, except for tritium.

192 The 91st Secretariat Meeting of the Team for Countermeasures for Decommissioning and Contaminated Water/Treated Water Treatment, Material 3-1, Results of performance verification testing on secondary treatment of treated water by multi-nuclide removal equipment (third-party organization), June 24, 2021

Table A12-4 Secondary treatment test results of J1-C tank group[191]

■ J1-C group (62 nuclides + C-14 + H-3)

		Concentration limits required by law [Bq/L]	Before secondary treatment (System inlet)*1 Analysis results [Bq/L]	Before secondary treatment (System inlet)*1 Concentration ratios required by law*3	After secondary treatment (Sample tank)*2 Analysis results [Bq/L]	After secondary treatment (Sample tank)*2 Concentration ratios required by law*3
Seven major nuclides	Cs-134	60	2.93E+01	0.49	<7.60E-02	0.0013
	Cs-137	90	5.99E+02	6.7	1.85E-01	0.0021
	Co-60	200	3.63E+01	0.18	3.33E-01	0.0017
	Ru-106	100	<5.00E+00	0.050	1.43E+00	0.014
	Sb-125	800	8.30E+01	0.10	2.26E-01	0.00028
	Sr-90	30	6.46E+04	2,155	3.57E-02	0.0012
	I-129	9	2.99E+01	3.3	1.16E+00	0.13
	C-14	2,000	1.53E+01	0.0076	1.76E+01	0.0088
	H-3	60,000	8.51E+05	14.2	8.22E+05	13.7

	Before secondary treatment (System inlet)*1	After secondary treatment (Sample tank)*2
Sum of the concentration ratios of the seven major nuclides required by law	2,165	0.15
Sum of the concentration ratios of the 62 nuclides*4 + C-14 required by law	2,406	0.35

Of the 0.35, the sum of the concentration ratios of the nuclides (51 nuclides) required by law, which analysis and evaluation confirmed was below the lower detection limit, is 0.19.

*1: Composite analysis was performed on the samples collected on September 19, 20 and 21.
*2: Analysis was performed on the samples collected on September 27.
*3: The value of the lower detection limit was used for calculating nuclides whose analysis results are below the lower limit of detection.
*4: See Reference for details of analysis results and the concentration limits required by law.

Numerically expressed, ○.○○E ±XX means ○.○○ × $10^{\pm XX}$.

(Source : TEPCO)

Table A12-5 Secondary treatment test results of J1-G tank group[191]

■ J1-G group (62 nuclides + C-14 + H-3)

		Concentration limits required by law [Bq/L]	Before secondary treatment (System inlet)*1 Analysis results [Bq/L]	Before secondary treatment (System inlet)*1 Concentration ratios required by law*3	After secondary treatment (Sample tank)*2 Analysis results [Bq/L]	After secondary treatment (Sample tank)*2 Concentration ratios required by law*3
Seven major nuclides	Cs-134	60	5.94E+00	0.099	<6.65E-02	0.0011
	Cs-137	90	1.18E+02	1.3	3.29E-01	0.0037
	Co-60	200	1.31E+01	0.065	2.33E-01	0.0012
	Ru-106	100	<2.27E+00	0.023	4.83E-01	0.0048
	Sb-125	800	3.23E+01	0.040	<1.37E-01	0.00017
	Sr-90	30	1.04E+04	347	3.18E-02	0.0011
	I-129	9	2.79E+00	0.31	3.28E-01	0.036
	C-14	2,000	1.26E+01	0.0063	1.56E+01	0.0078
	H-3	60,000	2.73E+05	4.6	2.72E+05	4.5

	Before secondary treatment (System inlet)*1	After secondary treatment (Sample tank)*2
Sum of the concentration ratios of the seven major nuclides required by law	349	0.048
Sum of the concentration ratios of the 62 nuclides*4 + C-14 required by law	387	0.22

Of the 0.22, the sum of the concentration ratios of the nuclides (53 nuclides) required by law, which analysis and evaluation confirmed was below the lower detection limit, is 0.17.

*1: Composite analysis was performed on the samples collected on October 5, 6 and 7.
*2: Analysis was performed on the samples collected on October 13.
*3: The value of the lower detection limit was used for calculating nuclides whose analysis results are below the lower limit of detection.
*4: See Reference for details of analysis results and the concentration limits required by law.

Numerically expressed, ○.○○E ±XX means ○.○○ × $10^{\pm XX}$.

(Source : TEPCO)

b. Analysis

ALPS-treated water that satisfies the regulatory limit, except for tritium, is transferred to the measurement/verification system, and 64 nuclides (tritium, 62 nuclides (nuclides removed by ALPS) and carbon 14) are always analyzed before dilution and discharge.

> Note) Treatment of the lower detection limit in analysis (ND value)
>
> In nuclide analysis, there is a case where no significant concentration is detected, and the limiting value is referred to as the lower detection limit (ND (Not Detected) value)
>
> The lower detection limit varies depending on the detection accuracy of the instrument, the measurement time, the amount of sample, the natural background, etc. When distillation or chemical separation is involved, an advanced analytical technique is also required. Therefore, the lower detection limit is not a constant value, and fluctuates depending on the institution and facility for analysis. When the measurement result is below the lower detection limit, TEPCO does not evaluate that the nuclide of concern is not contained (does not evaluate that the concentration is zero), but conservatively evaluates that the concentration of the lower detection limit exists.
>
> For a nuclide whose concentration is considered to be low, the lower detection limit should be lowered in advance. In particular, when analyzing a large number of nuclides and evaluating the sum of ratios of concentrations required by law from nuclide concentrations, it is important to select appropriate analytical conditions such as detection limit setting so that sufficient accuracy can be obtained for each nuclide analysis.

Table A12-6 shows nuclide measurement and evaluation methods employed by TEPCO. Table A12-7 gives details of the analytical methods used in the above secondary treatment tests.

Of the total 64 nuclides, 45 nuclides are analyzed/evaluated by γ-ray nuclide analysis, and 10 nuclides are by total-α activity. The other 9 nuclides are analyzed by β-ray measurement with pretreatment and inductively coupled plasma mass spectrometry (ICP-MS). Analysis takes time, especially if distillation or chemical separation is required (For example, in the case of tritium, it takes about half a day to a day.)。

Fig. A12-10 shows a standard analytical flow for tritium concentration[193], but water samples at 1F are generally contaminated with salt, and must first be desalted by distillation. Then, pretreatment using a concentration device is required according to the target lower detection limit. Even after the completion of the pretreatment, mixing with the emulsion scintillator and homogenization also take time. Similarly, analysis of carbon 14, nickel 63, cadmium 113m, etc., require a large amount of time for pretreatment.

In the performance confirmation test for secondary treatment, it took a long time, 2 months, to analyze nickel 63 and cadmium 113 m in particular. In order to secure this period and achieve an

[193] Tritium Analysis Method revised in 2002, Office of Emergency Planning and Environmental Radioactivity, Nuclear Safety Division, Science and Technology Policy Bureau, MEXT

efficient discharge in future operations, it is planned that three tank groups of about 10,000 m^3 will be prepared. As shown in Fig. A12-11, with this method the tank groups have 3 roles, receiving, measurement/evaluation and discharge are used every two months by rotation. For analysis, ALPS-treated water is circulated and stirred in the tank group to make it uniform, and then the solution is sampled.

At present, it takes about two months for the analysis, which is one of the conditions limiting the amount of water that can be discharged into the ocean. For this reason, in order to achieve more stable discharge into the ocean, it is necessary to rationalize the pretreatment and analysis procedures, and to further reduce the time required for the analysis. However, when demonstrating that the sum of concentration ratios required by law for 63 nuclides, except for tritium, is less than 1, it is necessary to conservatively evaluate nuclides, whose analysis results are less than the lower detection limit, using the lower detection limit. Therefore, it is important to select appropriate analysis conditions in consideration of factors affecting the lower detection limit [Note)]. The results obtained from these analyses should also be disclosed each time before dilution and discharge, and measurement/evaluation by third-party analytical institutions be considered. For third-party involvement, it is important to develop a quality assurance system because the results may vary depending on the institution and facility performing analysis due to fluctuation of the lower detection limit caused by the detection accuracy of measuring instruments or analysis techniques. For example, it is effective to analyze common samples in an integrated way before discharge to see the fluctuation range of the results, and verify and disclose the quality of each analysis institution beforehand. It is also necessary to consider a structure for verifying the operation status of the developed system.

Note) Treatment of the lower detection limit in analysis (ND value)

In nuclide analysis, there is a case where no significant concentration is detected, and the limiting value is referred to as the lower detection limit (ND (Not Detected) value)

The lower detection limit varies depending on the detection accuracy of the instrument, the measurement time, the amount of sample, the natural background, etc. When distillation or chemical separation is involved, an advanced analytical technique is also required. Therefore, the lower detection limit is not a constant value, and fluctuates depending on the institution and facility for analysis. When the measurement result is below the lower detection limit, TEPCO does not evaluate that the nuclide of concern is not contained (does not evaluate that the concentration is zero), but conservatively evaluates that the concentration of the lower detection limit exists.

For a nuclide whose concentration is considered to be low, the lower detection limit should be lowered in advance. In particular, when analyzing a large number of nuclides and evaluating the sum of ratios of concentrations required by law from nuclide concentrations, it is important to select appropriate analytical conditions such as detection limit setting so that sufficient accuracy can be obtained for each nuclide analysis.

Table A12-6 Measurement and evaluation methods for the concentration of 62 nuclides, tritium and carbon 14 removed by ALPS[191]

● Nuclides to be quantified and evaluated based on the results of γ-ray nuclide analysis using a Ge semiconductor dete

	Nuclides	Nuclide measurement or evaluation method
1	Rb-86	γ-ray nuclide analysis
2	Y-91	γ-ray nuclide analysis
3	Nb-95	γ-ray nuclide analysis
4	Ru-103	γ-ray nuclide analysis
5	Ru-106	γ-ray nuclide analysis
6	Rh-103m	Ru-103 and radioactive equilibrium
7	Rh-106	Ru-106 and radioactive equilibrium
8	Ag-110m	γ-ray nuclide analysis
9	Cd-115m	γ-ray nuclide analysis
10	Sn-119m	Evaluation from Sn-123
11	Sn-123	γ-ray nuclide analysis
12	Sn-126	γ-ray nuclide analysis
13	Sb-124	γ-ray nuclide analysis
14	Sb-125	γ-ray nuclide analysis
15	Te-123m	γ-ray nuclide analysis
16	Te-125m	Sb-125 and radioactive equilibrium
17	Te-127	γ-ray nuclide analysis
18	Te-127m	Evaluation from Te-127
19	Te-129	γ-ray nuclide analysis
20	Te-129m	γ-ray nuclide analysis
21	Cs-134	γ-ray nuclide analysis
22	Cs-135	Evaluation from Cs-137
23	Cs-136	γ-ray nuclide analysis

	Nuclides	Nuclide measurement or evaluation method
24	Cs-137	γ-ray nuclide analysis
25	Ba-137m	Cs-137 and radioactive equilibrium
26	Ba-140	γ-ray nuclide analysis
27	Ce-141	γ-ray nuclide analysis
28	Ce-144	γ-ray nuclide analysis
29	Pr-144	Ce-144 and radioactive equilibrium
30	Pr-144m	Ce-144 and radioactive equilibrium
31	Pm-146	γ-ray nuclide analysis
32	Pm-147	Evaluation from Eu-154
33	Pm-148	γ-ray nuclide analysis
34	Pm-148m	γ-ray nuclide analysis
35	Sm-151	Evaluation from Eu-154
36	Eu-152	γ-ray nuclide analysis
37	Eu-154	γ-ray nuclide analysis
38	Eu-155	γ-ray nuclide analysis
39	Gd-153	γ-ray nuclide analysis
40	Tb-160	γ-ray nuclide analysis
41	Mn-54	γ-ray nuclide analysis
42	Fe-59	γ-ray nuclide analysis
43	Co-58	γ-ray nuclide analysis
44	Co-60	γ-ray nuclide analysis
45	Zn-65	γ-ray nuclide analysis

● Nuclides to be quantified and evaluated based on the results of total-α activity measurement

	Nuclides	Nuclide measurement or evaluation method
46	Pu-238	Total-α activity
47	Pu-239	Total-α activity
48	Pu-240	Total-α activity
49	Pu-241	Evaluated value from Pu-238
50	Am-241	Total-α activity
51	Am-242m	Evaluated value from Am-241
52	Am-243	Total-α activity
53	Cm-242	Total-α activity
54	Cm-243	Total-α activity
55	Cm-244	Total-α activity

● Nuclides to be quantified and evaluated by other methods

	Nuclides	Nuclide measurement or evaluation method
56	H-3	β-ray measurement after separation by distillation
57	C-14	β-ray measurement after chemical separation
58	Sr-90	β-ray measurement after chemical separation
59	Sr-89	β-ray measurement after chemical separation
60	Y-90	Sr-90 and radioactive equilibrium
61	Tc-99	ICP-MS measurement
62	Cd-113m	β-ray measurement after chemical separation
63	I-129	ICP-MS measurement
64	Ni-63	β-ray measurement after chemical separation

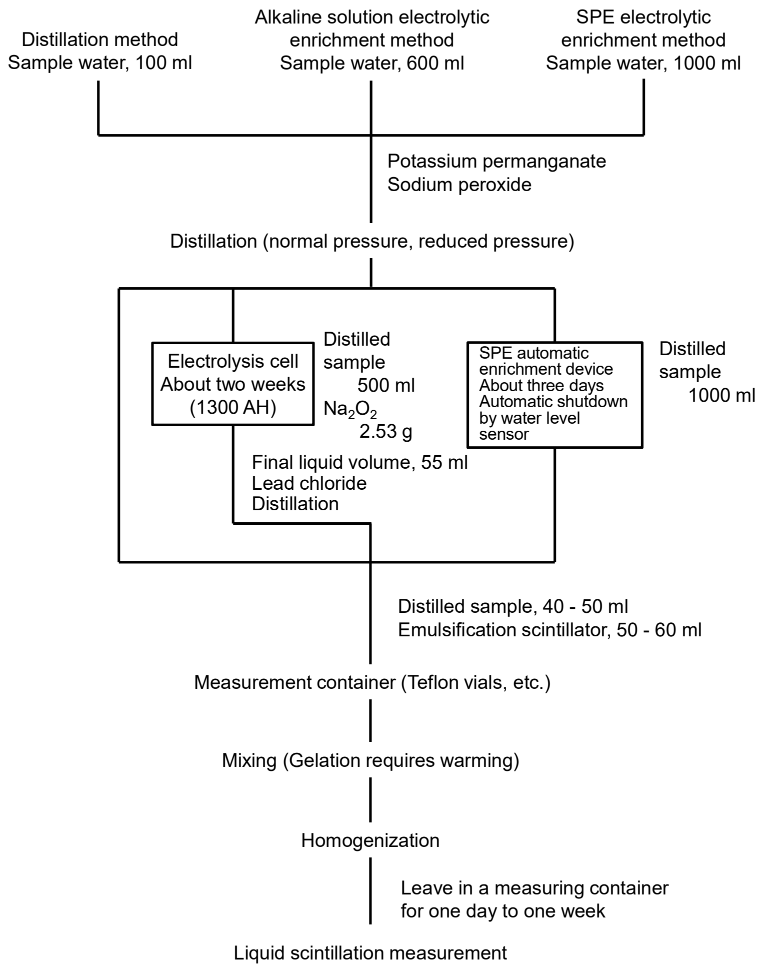
(Source : TEPCO)

Table A12-7 Analytical method used for secondary treatment testing[191]

Nuclides	Analysis method	Target lower detection limit (Bq/L)
γ-ray emitting nuclide	Collect a sample in a 5-liter marinelli beaker, and measure with a Ge semiconductor detector.	0.07 (Cs-137)[*1]
Sr-90, Sr-89	After purifying Sr with Sr resin, measure the substance precipitated and collected as carbonate with a beta spectrum analyzer.	0.04 (Sr-90)[*2]
I-129	Measure using ICP-MS after adding hypochlorous acid to a sample and adjusting it to iodate ion.	0.2
H-3	Measure with a liquid scintillation counter after mixing a sample, from which impurities have been removed by distillation, with a scintillator.	30
C-14	Add concentrated nitric acid and potassium persulfate to a sample, heat it, collect the generated CO_2 with adsorbent, mix it with a scintillator, and then measure with a liquid scintillation counter.	10
Tc-99	Dilute a sample with nitric acid, and then measure it using ICP-MS.	2
Total-α activity	Coprecipitate α-nuclides with iron hydroxide,remove the iron by extraction, evaporation to dryness and baking on a stainless steel dish, and then measure with a ZnS scintillation counter.	0.04
Cd-113m	Purify and collect Cd by ion exchange, mix with a scintillator, and measure with a liquid scintillation counter.	0.2
Ni-63	Purify and collect Ni by Ni resin, mix with a scintillator, and measure with a liquid scintillation counter.	20

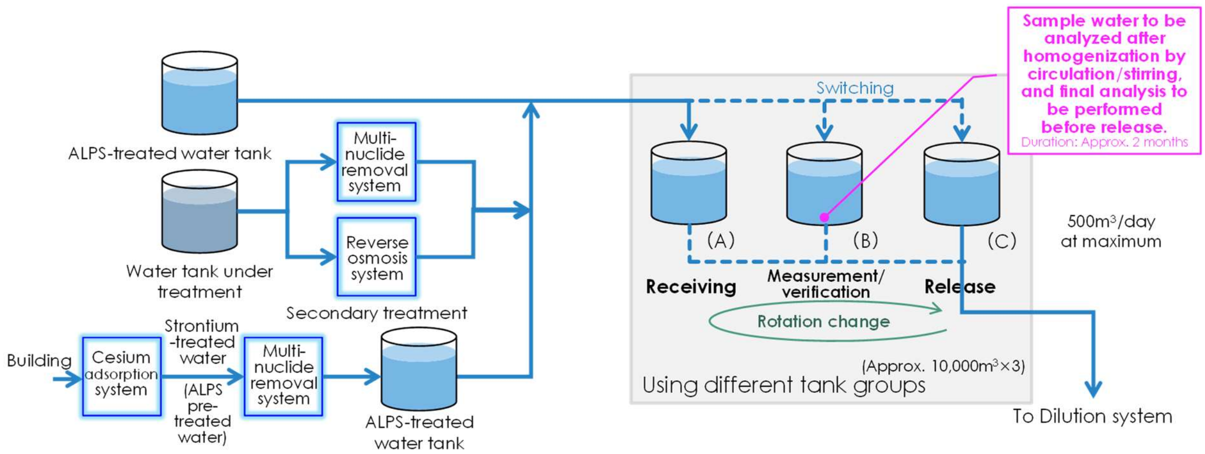
*1: The target lower detection limit of other nuclides varies depending on the baseline, interfering nuclides, background, and γ-ray emission rates.
*2: The target lower detection limit of Sr-89 varies depending on the Sr-90 concentration.

(Source : TEPCO)



(Source : MEXT)

Fig. A12-10 General analytical flow for tritium concentrations in water samples[192]



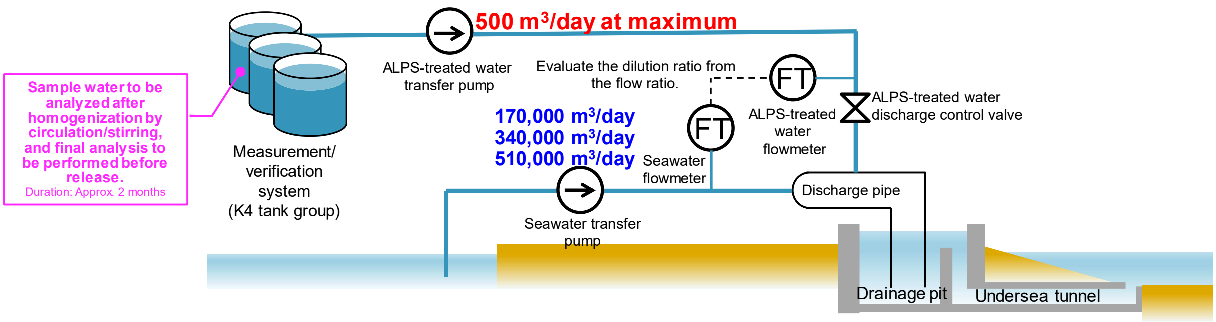
(Source : TEPCO)

Fig. A12-11 Tank management during analysis[194]

[194] The 91st meeting, Study group on monitoring and assessment of specified nuclear facilities, Material 2, Status of examining the discharge system for offshore discharge concerning treated water by multi-nuclide removal equipment, June 7, 2021

c. Dilution and discharge

After analysis and concentration verification, the ALPS-treated water is diluted with seawater and then discharged. Fig. A12-12 shows the method of diluting ALPS-treated water with seawater. For discharge, ALPS-treated water with an upper limit of 500 m^3/day is mixed and diluted in the discharge pipe (header pipe) with seawater transferred from the seawater transfer pump (up to three pumps at 170,000 m^3/day per pump) so that the tritium concentration after dilution is less than 1,500 Bq/L. Then, the diluted water is discharged into the ocean through a drainage pit and an underwater tunnel about 1 km long. Therefore, the dilution ratio is about 340 times even in the case of the minimum ratio (assuming an injection volume of ALPS-treated water of 500 m^3/day and operation of one seawater transfer pump). After this dilution process, the sum of concentration ratios required by law, except for tritium, becomes less than 0.003, resulting in a smaller effect on the human population and the natural environment.



(TEPCO material edited by NDF)

Fig. A12-12 Diluting method of ALPS-treated water with seawater[190]

As described above, since it takes from half a day to a day to measure the tritium concentration, it is difficult to measure and verify the results in real time at discharge. Therefore, the tritium concentration after seawater dilution will be checked as follows.

(1) Evaluation by the tritium concentration and flow rate of ALPS-treated water

TEPCO intends to guarantee that the tritium concentration after seawater dilution is below 1,500 Bq/L based on the tritium concentration obtained by analysis in the measurement/verification system, and the flow ratio between ALPS-treated water and seawater. The specific concentration can be calculated using the following equation:

$$\text{Tritium concentration after dilution with seawater} = \frac{\text{Tritium concentration of ALPS-treated water} \times \text{ALPS-treated water flow rate (control by discharge control valves)}}{\text{ALPS-treated water flow rate (control by discharge control valves)} + \text{seawater flow rate}}$$

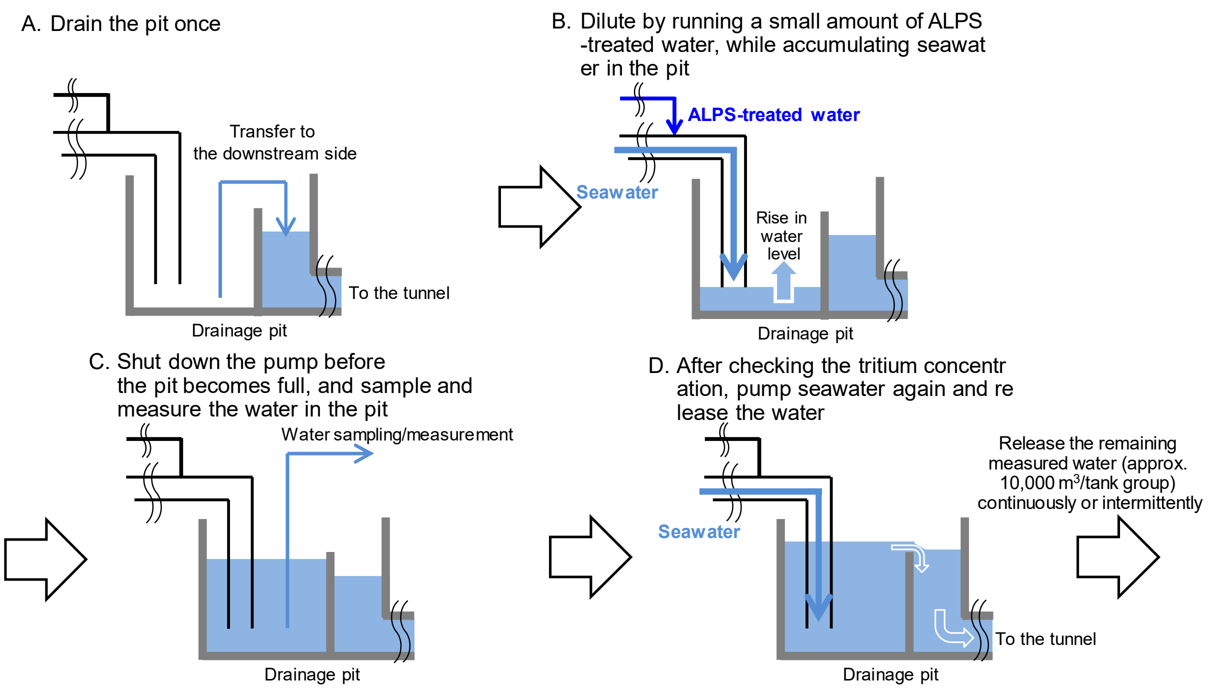
In addition to the above, in order to confirm that the tritium concentration at the end of a discharge outlet is below 1,500 Bq/L after mixing and diluting as designed, the following two measurement methods (2) and (3) are planned.

(2) Daily measurement of tritium concentration in a drainage pit

As part of the marine monitoring in the port (see Table A12-8 below), the water in a drainage pit is sampled daily to check the tritium concentration. Although the analysis results will be disclosed promptly, a time lag of about one day is expected between sampling and announcement of the results.

(3) Immediate pre- discharge measurement of tritium concentration after dilution using a drainage pit

The drainage procedure is shown in Fig. A12-13. Once the pit (approx. 2,000 m^3) is drained, dilution is performed by running a small amount of ALPS-treated water (20 m^3 or less) while operating one seawater transfer pump for about 10 minutes. Then, the pump is shut down before the pit becomes full, and the water in the pit is sampled in order to confirm that the calculated tritium concentration and the measured tritium concentration are equivalent or less than 1,500 Bq/L (about 2 days). After verification, the water in the pit and the remaining water (approx. 10,000 m^3/tank group) measured in the measurement/ verification system are discharged constantly or intermittently.



(Source : TEPCO)

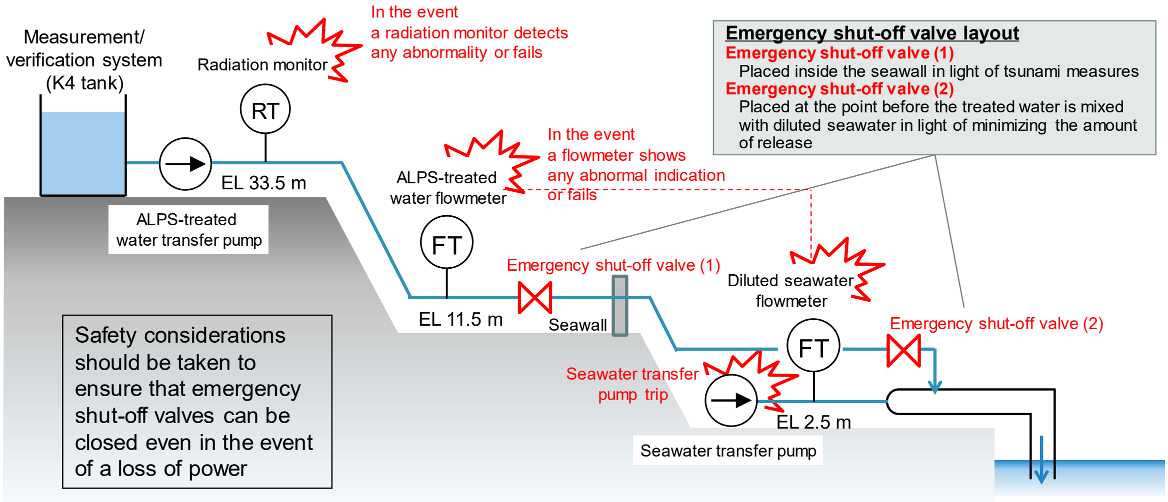
Fig. A12-13 Method for verifying tritium concentration using a drainage pit after dilution with seawater[190]

Discharge will be carried out in the manner described above. The start of discharge will be carefully initiated from a small amount. After that, the amount of tritium will be evaluated for each water sample (approx. 10,000 m^3/tank group) measured in the measurement/verification system to control cumulative values and confirm that the tritium level is below 22 trillion Bq per year.

d. Emergency measures

If there is a possibility that the dilution rate of the ALPS-treated water will become abnormal due to system failure (seawater pump shutdown, decrease of seawater flow rate, increase of treated water flow rate, failure of flowmeter), or if the properties of the ALPS-treated water are abnormal (radiation monitor operation or failure), the plan is to immediately close the two emergency shut-off valves, shut down the ALPS-treated water transfer pump, and suspend discharge. Fig. A12-14 shows an emergency shut-off valve layout. It is planned that one emergency shut-off valve will be installed in the vicinity of the seawater transfer piping to minimize the amount of ALPS-treated water discharge in the event of an abnormality, and the other emergency shut-off valve will be installed inside the tide embankment in anticipation of submergence due to a tsunami.

If any abnormal value is confirmed by marine monitoring, discharge will also be suspended. As described later, however, the marine monitoring results will be verified after discharge.



(Source : TEPCO)

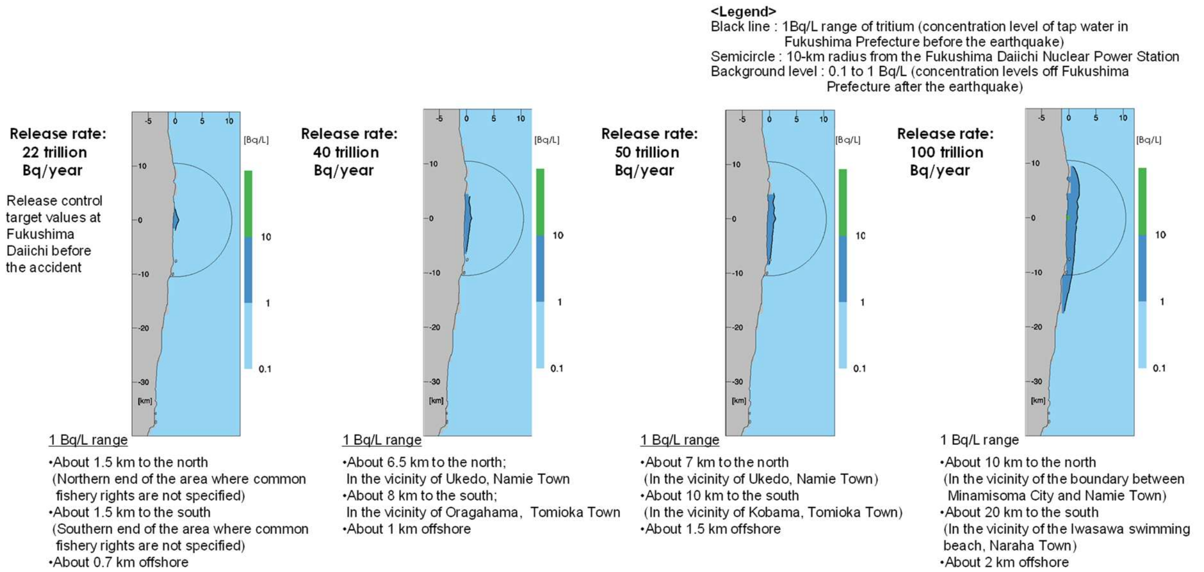
Fig. A12-14 Emergency shut-off valve layout and emergency measures[190]

(3) Marine monitoring

a. Simulation of tritium diffusion in the ocean

Fig. A12-15 shows the simulation results of tritium diffusion in the ocean performed by TEPCO using the model verified by the measured data of cesium 137 after the accident. The simulation conditions are as follows:

- Target sea area: About 500 km north-south and about 600 km offshore centered in Fukushima Prefecture
- Discharge method: Planned to discharge ALPS-treated water evenly within a 1 km x 1 km area in a horizontal direction adjacent to the 1F site.
- Resolution: 1 km mesh horizontally, 30 layers vertically against water depth (to a depth of 1 km)
- Weather conditions, etc.: Wind velocity, atmospheric pressure, temperature, humidity, and precipitation from January to December 2014 are used (including flow conditions (Black Current, mesoscale vortex) off the coast of Fukushima Prefecture)



(Source : TEPCO)

Fig. A12-15 Range of tritium diffusion[195]

The areas where the tritium concentration exceeds the background level (0.1 - 1 Bq/L) are limited within the area near the Fukushima Daiichi NPS for which the common fishery right is not specified. The result is sufficiently small compared with the WHO drinking water standard (10,000 Bq/L).

Although there are limitations in the prediction accuracy of these results, general models and appropriate publicly available weather and ocean current data for input were used. The results are assessed that they are consistent with the trend of observation values and simulation results already disclosed in published articles[196].

However, considering that the simulation results of diffusion in the ocean include uncertainties due to differences in models, external forces, and ocean fluctuations[197], it is recommended that simulations that incorporate specific discharge conditions of ALPS-treated water currently planned are conducted, and that the effects on the human population and the environment are evaluated.

b. Marine monitoring plan

According to the simulation results described above, although the range of 1 Bq/L[198] or higher is limited, TEPCO has proposed measures to strengthen the marine monitoring in order to understand the diffusion situation (Table A12-8). As mentioned in (2) c. Dilution and discharge, daily measurements of tritium concentration and increasing sampling points in a drainage pit are planned.

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[195] TEPCO Draft Study Responding to the Subcommittee Report on Handling ALPS-Treated Water, press release, March 24, 2020

[196] e-GOV portal, Public Comment, Results of a written request for public comments on the handling of treated water by multi-nuclide removal equipment, page 8, April 13, 2021

[197] Yukio Masumoto, et al.,: Near Coastal Ocean Variability Affecting Radionuclide Dispersion Simulation and Their Reproducibility in Numerical Models, Bulletin on Coastal Oceanography, Vol. 54, No. 2, 151 - 157, 2017

[198] The tritium concentration of tap water in Fukushima Prefecture is about 1 Bq/L. The WHO drinking water standard is 10,000 Bq/L.

This marine monitoring will start about one year before the scheduled discharge (around from the spring of 2022), and TEPCO plans to ask agricultural, forestry and fishery operators, local government officials, etc. to participate in monitoring and inspection when collecting samples and measuring radioactivity. In case any abnormal value is detected by marine monitoring after discharging, discharge will be suspended and a condition survey will be conducted.

In addition, the task force on monitoring and measuring the marine environment has been held under the government-sponsored Monitoring Coordination Council since June 2021. The Ministry of the Environment, NRA, The Agency for Natural Resources and Energy Agency, Fukushima Prefecture, and TEPCO are participating in the study on strengthening and expanding the monitoring[199]. In the future, related organizations will collaborate to ensure the implementation of the marine monitoring.

Table A12-8 Summary of marine monitoring plan[190]

Subject	Area sampled		Subject of measurement	Current frequency	After change (draft)	Remarks
Seawater	Inside the harbor	10 locations	Cesium 134,137 Tritium	Cesium : Daily Tritium : Weekly	Cesium : Daily Tritium : Weekly	**Perform daily for discharge vertical shaft (discharge end)**
	Within 2km (and the vicinity)	7 locations		Cesium : Weekly Tritium : Weekly	Cesium : Weekly Tritium : Weekly	**Added three sampling areas (10 areas in total)**
	Within 20km	6 locations		Cesium : Weekly Tritium : Every two weeks	Cesium : Weekly **Tritium : Weekly**	**Doubled the analysis frequency of tritium**
	Outside 20km (off the coast of Fukushima)	9 locations		Cesium : Monthly Tritium : 0 times	Cesium : Monthly **Tritium : Monthly**	**Added tritium**
Fish	Within 20km		Cesium134,137 Strontium Tritium	Cesium : Monthly (11 locations) Strontium : Quarterly (Top five samples for cesium concentration) Tritium : Monthly (one location)	Cesium : Monthly (11 locations) Strontium : Quarterly (Top five samples for cesium concentration) **Tritium : Monthly (11 locations)**	Fish are currently sampled at 11 locations to analyze cesium and tritium is analyzed in one of those locations. After the change, **tritium analysis is conducted in the remaining ten locations**
Seaweed	Inside the harbor		Cesium134,137	Cesium : Three times annually (one location)	Cesium : Three times annually (one location)	Conducted three times annually in March, May and July
	Outside the harbor		Cesium134,137 Iodine 129 Tritium	Cesium : 0 times Iodine : 0 times Tritium : 0 times	**Cesium : Three times annually (two locations)** **Iodine : Three times annually (two locations)** **Tritium : Three times annually (two locations)**	**Added two locations outside the harbor** Conducted three times annually in March, May and July (Review based on survey of habitat)

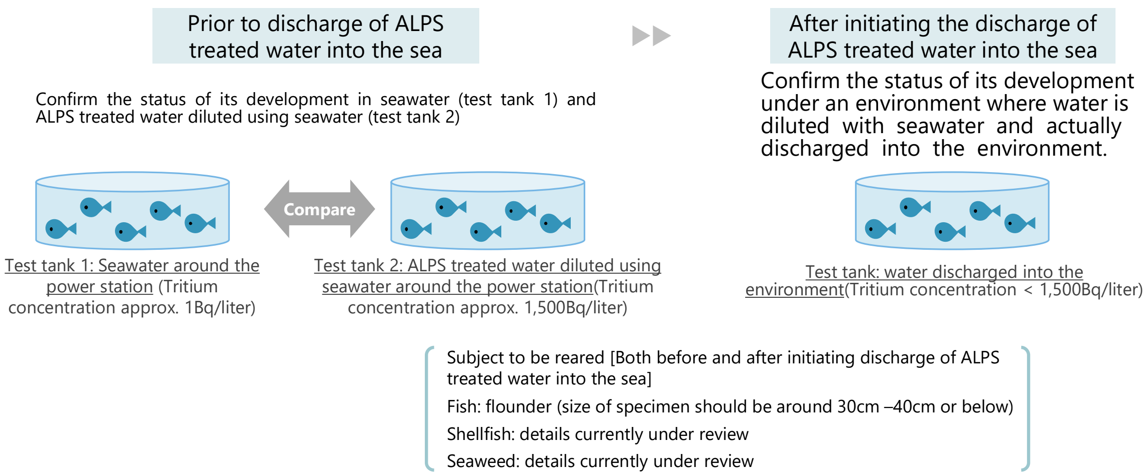
(Source : TEPCO)

(4) Marine organism feeding trial

In order to foster understanding of the discharge of ALPS-treated water into the ocean and to reduce reputational damage, a marine organism feeding trial is planned as shown in Fig. A12-16. Before offshore discharge of ALPS-treated water, this trial intends to observe the growth of marine organisms in both environments, in seawater and in ALPS-treated water diluted with seawater. After the start of offshore discharge, their growth will be observed in the water diluted with seawater and actually discharged to the ocean. These trials will see any health abnormalities in fish, etc., compare the concentration of radioactivity including tritium in breeding water and in vivo, and

[199] The 13th Monitoring Coordination Council, Reference 5, Establishment of the Task Force of Monitoring and Measuring the Marine Environment, April 27, 2021
https://www.env.go.jp/water/post_110.html

observe hatching rate of eggs, survival rate of adult fish, etc., while streaming of breeding conditions is planned.



(Source : TEPCO)

Fig. A12-16 Summary of marine organism feeding trial[190]

(5) Technical study on tritium separation

TEPCO plans to publicly solicit technical proposals from third-party organizations to identify new trends in tritium separation technologies. They will solicit technologies that are expected to reduce tritium concentration to 1/1,000 or less and to expand operating capacity to 50 - 500 m^3/day in the future. Primary evaluation of the technologies sought from the public will be conducted by a third-party organization, followed by a secondary evaluation by TEPCO. As a result, if it is confirmed that a technology is realistically practicable for ALPS-treated water, etc., it is planned that a specific design will be examined and technology demonstration testing will be conducted to establish the technology.

(6) Simple estimation of the volume reduction of tritium due to the discharge of ALPS-treated water into the ocean

As mentioned above, the discharge system used for offshore discharge consists of the process from [purification] → [analysis/verification] → [dilution] → [discharge]. Technology and systems with past results are used, and thus strict compliance with the regulatory standards and operation procedures enables safe discharge of the ALPS-treated water.

Fig. A12-17 shows the results of a simple estimation of the change in the amount of tritium stored in tanks if the ALPS-treated water is discharged into the ocean. In this estimation, it was assumed that as of 2020, 780 trillion Bq existed in the tanks, and that 300,000 Bq/L (based on the recent tritium concentration shown in Fig. A12-7) of stagnant water in buildings would be newly generated every year by 51,000 m^3 (140 m^3/day x 365 days). Tritium is assumed to decay spontaneously (half-life of 12.3 years), and the offshore discharge is assumed to start in 2022.

In this estimation, the amount of tritium stored in tanks became zero after 2045 by discharging 22 trillion Bq of tritium per year into the ocean.

A plan has been developed to maintain the total amount of tritium to be discharged per year below the discharge control target (22 trillion Bq per year) before the accident at 1F, and the plan is to be revised as necessary according to the progress of decommissioning.

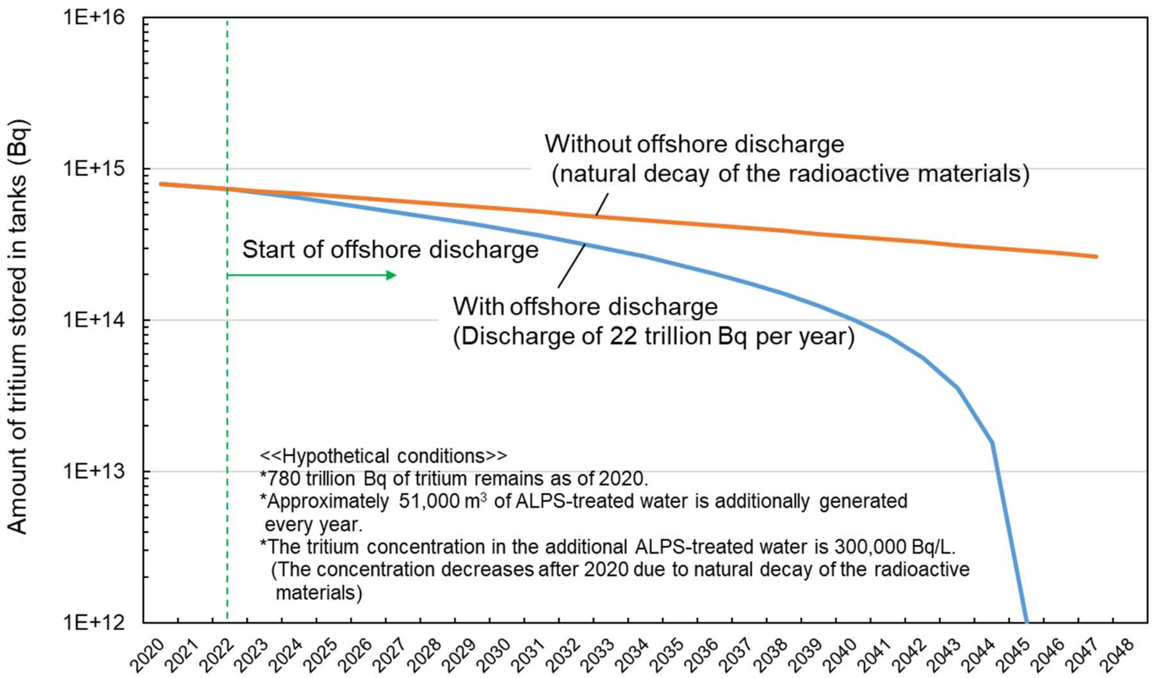


Fig. A12-17 Simple estimation results of the amount of tritium stored in tanks

4. Future efforts

TEPCO's planned discharging system, if operated reliably in accordance with the implementation plan approved in the review by the NRA, will have no adverse effects on humans and the environment, including other radionuclides, and therefore it is an important issue to operate the system "reliably" “as planned”.

Going forward, TEPCO will need to proceed with the following preparations in order to realize the discharge of the ALPS-treated water into the ocean in a safe way. During actual operation, it is necessary to ensure the implementation of the plan (system, operation, information distribution, etc.) established in the preparation stage, to perform checks and reviews, and to review and expand the plan as needed.

- In the operational phase, develop a series of operation plans including system operation, analysis of ALPS-treated water, flow control of the treated/diluted water, marine monitoring, maintenance, and troubleshooting, and then develop a system plan that minimizes risks and eliminates social concerns
- Perform a radiological impact assessment on the human population and the natural environment, and disclose evaluation results based on the specific discharge discharge plan
- Verify safety by experts from the International Atomic Energy Agency (IAEA) and other agencies
- Develop a plan to strengthen marine monitoring, and perform marine monitoring before the discharge
- Education and training on system operation and analysis, etc., for parties concerned including contractors (TEPCO)
- Development of strategies to provide accurate and understandable information domestically and internationally without causing anxiety from a social perspective, and timely dissemination of the status of preparations
- Ensuring implementation of measures to prevent reputational damage as set forth in the Government’s basic policy announced in April 2021

NDF will provide technical and professional support for TEPCO in designing the discharge system and considering discharging methods, while promoting distribution of accurate information and increasing understanding through various opportunities in Japan and abroad in line with the interests of those who will receive the information. NDF will also ensure that TEPCO implements measures to minimize reputational damage, and that TEPCO takes action with adequate and sufficient compensation in the event of reputational damage.

Attachment 13 Basic direction of 6 Essential R&D Themes

December 12, 2017
Nuclear Damage Compensation and Decommissioning Facilitation Corporation (NDF)

Basic direction of 6 Essential R&D Themes

The Mid-and-Long-Term Roadmap towards the Decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Station (September 26, 2017) specifies enhancement of the activities for matching the R&D required for decommissioning (Needs) with the basic and fundamental R&D (Seeds) and for human resource development. It also specifies enhancement of the functions of Japan Atomic Energy Agency's Collaborative Laboratories for Advanced Decommissioning Science (JAEA/CLADS) and promotion of joint researches with domestic and international universities and researching institutions to establish the international decommissioning research center with concentrated wisdom.

Following the above, the Ministry of Education, Culture, Sports, Science and Technology (MEXT) states in its budget request for FY2018 that they will reform the Center of World Intelligence Project for Nuclear Science and Technology and Human Resource Development ("World Intelligence Project") into a subsidy program intended for JAEA/CLADS and the program will be implemented under the system centered by JAEA/CLADS from the newly adopted proposals from FY2018.

On the occasion of this reform, MEXT showed NDF their intention to discuss how to proceed with the future R&D of the World Intelligence Project. This includes selection of the theme for the call for proposal considering the Essential R&D selected by the task force from the viewpoint of promoting basic and fundamental researches with satisfactory understanding of the Needs.

Therefore, concerning the 6 Essential R&D Themes which are described in the interim report of the Task Force on Research Collaboration (November 30, 2016), the Basic Direction of the 6 Essential R&D Themes was compiled including the background of the issues, the problem consciousness, and the expected research image, with consulting the discussions in the working group for each theme.

Theme	(1) To identity process of characteristic change in fuel debris over time
"Descriptions / Background issues" on the interim report	The fuel debris retrieval is scheduled for 2021 onward, 10 years after the fuel debris production. And since it is anticipated that the retrieval will require a long period of time, the fuel debris will remain inside the reactors over 10 years. In addition, the retrieved debris must be stored safely. Choosing the best possible methods of retrieve/transmission/storage of fuel debris requires predictions of characteristic changes of fuel debris over time.
Basic direction	In relation to the accident of Chernobyl nuclear power plant, detection of particulates of the micron order which contain fuel components from around the fuel debris has been reported and the national report of the Ukrainian government showed concerns about the increase risk of radioactive dust emergence over time through self-decay. One of the possible reasons is that the fuel debris with high radioactivity exposed to rich humidity caused rapid progress of aging which quite slowly proceeds with normal uranium mineral in the geological environment. It results in oxidation activated by radioactive dissolution and generate hexavalent uranium compound. On the other hand, because the PCV of the Fukushima Daiichi NPS (hereinafter referred to as "1F") is currently in the nitrogen atmosphere under subtle positive pressure, oxidation is unlikely to proceed immediately. In the future, such an event similar to the above may occur because the air that contains oxygen may flow into the PCV when a negative pressure control is applied to retrieve the fuel debris. Since the radiation level is about one order higher in 1F than that of the fuel debris in the case of the Three Mile Island Unit 2 (TMI-2) accident (occurred in a short time after the operation started), it is under unexperienced condition. In addition, it should be noted that it will take a longer period to complete the retrieval of the fuel debris from the time of accident than in the case of TMI-2. Various factors are involved in the aging of such fuel debris in addition to the oxidation described above. Roughly classifying, those factors may include the chemical mechanism (oxidation-reduction, leaching of included components, changes in the chemical form or the phase state, etc.), physical mechanism (structural or characteristic changes by heat cycle etc., irradiation damage by α-ray), and coupled actions of these factors. Since decay or leaching of fuel debris due to aging lead to emission of FP particles or gas, or effluent of particles that contain α-nuclides that are confined in the fuel debris, they have significant impact on the system design and procedures. It includes the retrieval mechanism, the cooling and circulating system, the containment function, the criticality monitoring system, the PCV gas control system, exposure evaluation, containing, transferring and storing, and processing and disposal. In particular, as for the Mid-and-Long-term Roadmap, while the processing and disposal method of the fuel debris will be decided in the third period (from 2022) after the retrieval of the fuel debris is started,

	obtaining the aging information of the fuel debris is an urgent issue. While taking into consideration the permission and authorization about the safety regulation, to provide sufficient prediction and explanation of the risk changes resulted from the aging of the fuel debris, it is required to clarify the real situation of what are expected to have critical impacts on the decommissioning works preferentially. Therefore, it is necessary to build a fundamental theory of the aging model by clarifying the aging process while using the current knowledge of the actinoid chemistry. To do so, demonstration test should be performed using real uranium according to the matrix pattern of parameters (temperature, pH, etc.) to collect basic data, and it needs to establish the prediction method of aging. In this case, it is important to maintain the foundations for advancement of the actinoid chemistry, which provides the basis for examination of the physical property of fuel debris. In addition, heat analysis for 1F should be included in the basis of investigation since the temperature distribution of the fuel debris has to be understood by calculating the heat distribution and the impact of regional temperature rise due to the decay heat should also be required to be examined.

Theme	(2) To elucidate corrosion mechanisms under unusual/extreme circumstances
"Descriptions / Background issues" on the interim report	It is required to collect data on corrosion under a variety of circumstances with consideration of the circumstances specific to 1F decommissioning such as high radiation levels and unsteady routes of cooling water in order to prepare for potential corrosion during decommissioning.
Basic direction	A boiling water reactor (BWR) consists of various metallic material. While stainless steel, which is corrosion resistant, is used inside reactor where it is high temperature and high oxidizing environment, carbon steel, which is not corrosion resistant, is used for the PCV that is the confinement boundary and is assumed to be used in the normal atmosphere. On the other hand, substantial knowledge has been collected so far about corrosion of structure and piping for commercial electric generation reactors, and especially, data has been collected being focused on the corrosion data in the environment of high radiation, high temperature, and deionized water for the operation of BWR. However, after the accident, 1F has been in a special environment with high radiation, room temperature, suspended solids and deposited materials. The knowledge about such environment is not sufficient. Since water has been injected into PCV to cool the fuel debris, carbon steel is dipped in the water. In addition, it is known that chemical species of oxidation nature such as hydrogen peroxide and various radical species have been generated through radiolysis of water. Currently, since nitrogen has been injected into PCV to prevent hydrogen explosion and the oxygen density has been decreased, the densities of the oxygen and the hydrogen peroxide are considered to be decreased in the water and corrosion of PCV is also considered to be suppressed in some degree. In the future, since the air containing oxygen flows into PCV when a negative pressure control is applied to retrieve the fuel debris, it is important to maintain soundness of the structure and pipes that form the boundary for confinement of radioactive materials and it is also important to prepare countermeasures based on the knowledge on corrosion in such environment. Since the corrosion is essentially a kind of battery reaction, it is likely to happen if the electric conductivity of water raises, pH falls, and the electric potential raises under the condition of declining of surrounding water quality. Although corrosion has been suppressed by nitrogen injection in general, it is still under the condition of potential corrosion. Regional changes of the environmental condition may lead to an increase of corrosion speed at the part. It is quite a special environment surrounded by various factors that promotes corrosion such as a formation of liquid film of dew, a humid environment that repeats wet and dry conditions near the water surface, an irregular flow of cooling water, convection flow, or backwater due to irregular paths created by gaps between various shapes of fallen objects or deposited materials, a progress of corrosion on the anode side between different kinds of metals touching each other, a progress of acid-base reaction by microbes, or any other potential factors. In the future, further changes may occur in the internal environment when the air that contains oxygen flows into the PCV when a negative pressure control is applied to retrieve the fuel debris. Since the corrosion progresses over time during the decommissioning works under the special environmental conditions, estimation of corrosion phenomenon and investigation of countermeasures are required based on the consideration on the environmental changes resulted from the progress of the decommissioning process. Therefore, it is necessary to collect basic data related to the progress of corrosion phenomena and systematically clarify and understand the phenomena in order to provide satisfactory prediction and explanation of the risk changes that follow corrosion of

	structures taking the permission and authorization of the safety regulation into account with giving priority to the factors of higher needs that may have critical impacts on the decommissioning works from the viewpoint of the probability of occurrence such as the factors above and the impacts on the functionality (parts and severity), the scale and the timing. In this case, in order to examine various approaches including not only the use of existing anti-corrosive agents but also electric protection, it is important to accumulate and maintain the knowledge related to the corrosion phenomena in addition to the electronic state of materials in a special environment through principle analysis and clarification of the corrosion progression mechanism.

Theme	(3) Radiation measurement technologies adopting innovative approaches
“Descriptions / Background issues” on the interim report	The radiation levels are still extremely high inside the 1F reactors/buildings due to the accident and the existing measurement devices do not meet the capability/functional requirements to provide accurate figures. It is vital to develop an innovational device adopting brand-new ideas/principles based on 1F needs.
Basic direction	Currently, radiation measurement can be performed following a predefined operation procedure without detail knowledge of measurement since a number of radiation measurement products using various principles or materials including ionization chambers, counting tubes, semiconductor detectors, and scintillation detectors are already offered. However, it is very important to develop the human resources for measurement since it is necessary to understand the principles of the equipment in order to interpret the measurement data and address possible troubles such as the case of the disorder (inversion of data value) between the all β-radioactivity value and the value of Sr-90 because miss-counting has not been taken into account for the resolution time in the analysis of the sampled water at the under-ground water observation hole on 1F. In addition, general radiation measurement products are not able to offer satisfactory performance and functionality to inspect the conditions inside reactors and buildings at the decommissioning site of 1F. The decommissioning works on 1F must be performed by remote operation since the radiation level is extremely higher than the one in the work environment of existing nuclear facilities. It is necessary to develop highly radiation resistant and small sized measurement sensor, electronic circuits, and systems in order to be remotely operated. In addition, it is necessary to research on the basic mechanism related to radiation damages of materials in order to develop highly radiation resistant sensors and circuits. As for the specific examples of sensor development, it is necessary to develop measurement devices of neutron from the viewpoint of criticality prevention, real time measurement of α-ray from the viewpoint of identification of the fuel debris, and γ-ray measurement with high energy resolution for nuclide estimation under the background of high gamma radiation, those what satisfy various needs: radiation resistance, noise resistance, size (small size), counting rate and responsiveness, high radiation resistance, energy discrimination, space resolution (identification of radiation source position), ease of operation, and maintainability. As for the composition of the measurement targets, development of the technology so-called “on-site analysis” is required since there are needs of functions that can be used to analyze the target without transferring a sample to other facility or equipment and obtain rough results used to promptly judge if the target is debris or not, and if the target is debris, the function to judge co-existence of reactor internals and neutron poison is required. In addition, effective support tools for the decommissioning can be provided by developing the technology to visualize the radiation field and the contamination situation and clarify the profile of the fuel debris based on the information of the strength and the direction of the radioactive sources. It is necessary to develop the generic technologies for innovative measurement of radiation using new ideas and principles by considering on-site measurement requirements.

Theme	(4) To clarify behavior of radioactive airborne particle generated during decommissioning (incl. α-dust treatment)
“Descriptions / Background issues” on the interim report	As thermal cutting of the fuel debris via machine or Laser may produce a large amount of α-dust, it requires safety measures and dust confinement solutions. It is necessary to understand physical/chemical properties of α-dust, to predict the amount of dust to be produced for each method, and to consider how to seal the dust according to the results in order to make sure the retrieval will be conducted in a safe and effective manner.

Basic direction	As the fuel debris retrieval work will start, cutting the fuel debris will create a lot of radioactive airborne particles (α-dust) that contain α-nuclide and they will be dispersed within the boundary. When retrieving the fuel debris, since the work will be performed in the confinement boundaries which are the broken buildings, it is important to understand the property of the α-dust for studying how to secure the confinement capability, designing the filtering system, and performing the exposure evaluation of the surrounding environment and workers including the time of accident. With regard to the data about the scattering rate when α-dusts are generated, there is the data obtained at the decommissioning of JAEA's JPDR and the dismantling of glove box of JAEA's Nuclear Fuel Cycle Engineering Laboratories. However, the data has not been collected just for the nuclear fuels but collected for the objects polluted by the nuclear fuels, and the data has been collected for the amount of the radioactive materials and their densities from the viewpoint of radiation exposure control and it is not systematically organized. On the other hand, the radioactive airborne particles in 1F will be generated directly from the fuel debris when retrieving the fuel debris and from the polluted objects in the decommissioning process. The types of radioactive materials are α-nuclides and β (ϒ)-nuclides. While the α-nuclide of which typical element is plutonium is important from the viewpoint of internal exposure, the β(ϒ)-nuclide such as cesium should be well considered as well from the viewpoint of the total exposure evaluation. In order to study on collection of radioactive airborne particles, efficient filtering and purification, criticality prevention, etc., it is necessary to grasp the amount of generated particles, distribution of particle diameters, radioactive diameters, and the physical and chemical property of particles according to the differences of the target objects and the method of cutting. It is also important to understand the behavior of the generated particles in the gas phase, at the air-liquid interface, and in the liquid phase during transportation or transition. For example, it is important to understand the growth of particles through coagulation in the gas phase, evaluation of mist generation from the air-liquid interface, leaching behavior of the components into the water of the liquid phase, transportation behavior such as settling of particles in the water or filtering, etc. With regard to the exposure evaluation of radioactive airborne particles, it is important to evaluate the impact of exposure to radioactive materials derived from the fuel debris, especially the one of α-nuclides. In this case, it is important to decide whether the conventional exposure evaluation methods can be applied by judging if the chemical form and the particle diameter of the radioactive airborne particles represented by plutonium is consistent with the ones that have been used as the criteria of internal exposure evaluation for plutonium.

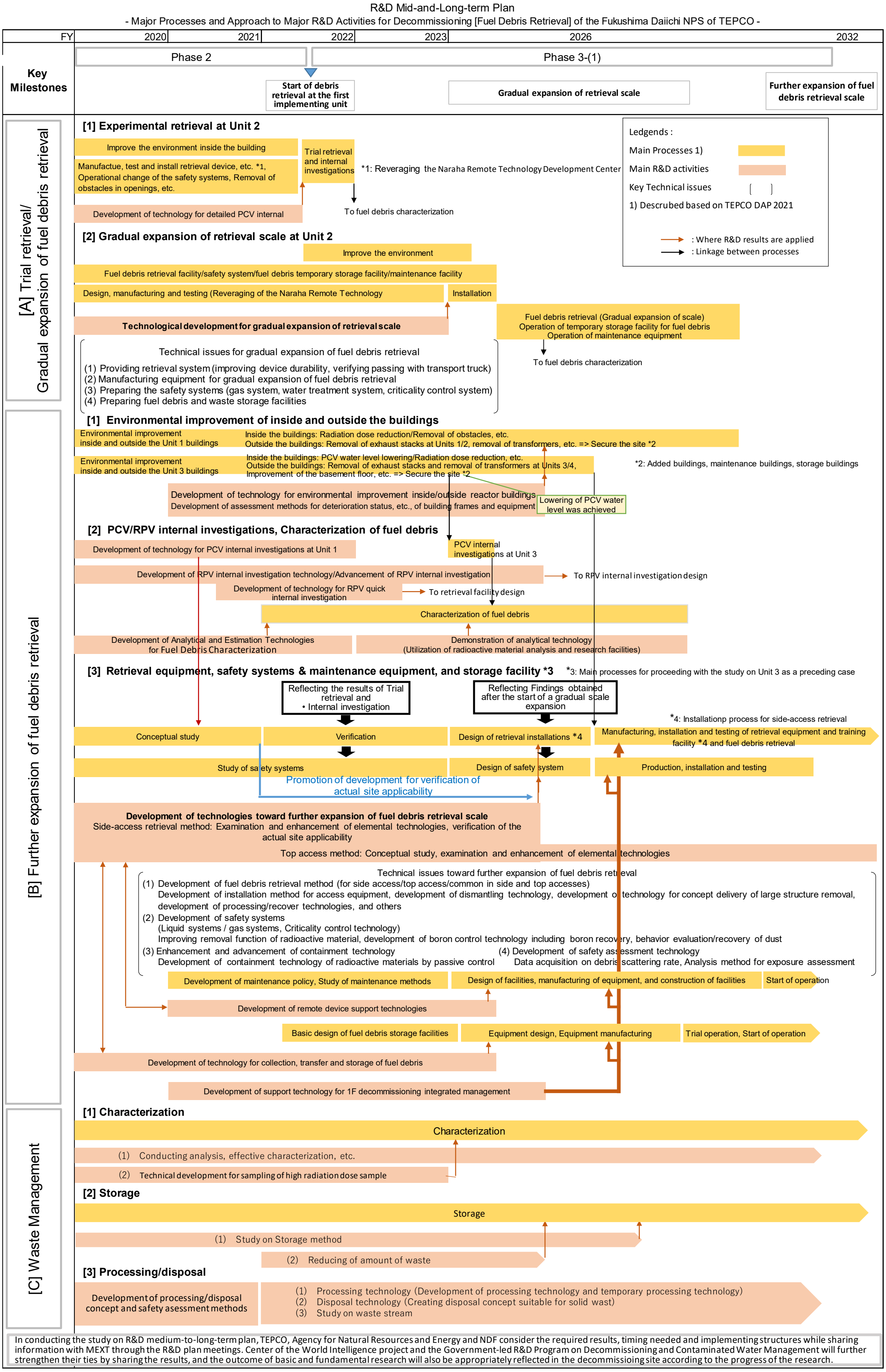
Theme	(5) To understand fundamentally mechanisms of radioactive contamination
"Descriptions / Background issues" on the interim report	To figure out the mechanism of radioactive contamination towards effective decontamination; it is critical to implement effective approaches of decontamination based on the mechanism of the contamination to radiation sources, and to decrease the volume of radioactive wastes as well.
Basic direction	With regard to reduction of the radiation in the buildings, the target object of decontamination are pipes and ducts, metals in the equipment, resins in cables and so on, paints, and concretes of the walls or floors. The contamination source includes molten high temperature fuels at the time of accident, steams that contain radioactive materials such as Cs leaked in consequence of hydrogen explosion and so on, dusts and contaminated water that contain radioactive materials. Currently, as for radiation reduction in the buildings on 1F, decontamination of floors and walls has only limited effects since there are other contamination sources such as objects remained in pipes, and hidden side behind the pipes located at high inaccessible positions. In the future, when considering each step in the long decommissioning process, it is expected that many situations that require decontamination will arise, therefore, effective and efficient decontamination is considered highly necessary. With regard to decontamination, not only reduction of radiation but also reduction of wastes should be taken into consideration. For decontamination, while engineering approaches are required, including physical methods such as dry ice blast, chemical methods such as chemical decontamination using chemicals such as acid or alkali, and decontamination methods using parting agents, it is indispensable to understand the contamination mechanism of target objects in order to perform such decontamination activities effectively. In the field of researches for clarifying the contamination mechanism, there are a sufficient number of existing researches on the metal materials, which are used in pipes,

	tanks and so on to confine radioactive materials. However, there is almost no research on the concrete of structures or radiation shields that does not directly touch radioactive materials. The inside the buildings of 1F is widely contaminated by the radioactive materials emitted by the accident. Since most parts of the buildings consist of concrete, it is important to clarify the contamination mechanism of both concrete and radioactive materials in principle in order to reasonably and effectively manage concrete wastes resulting from the decontamination and the process of decommissioning. Therefore, the contamination mechanism must be clarified in principle on the concrete exposed to the accident and the subsequent environment and the process of decommissioning by obtaining the basic data about sorption, penetration, and leaching of the typical nuclide (Cs, Sr, U, Pu, etc.) that should be well considered. In addition, from the mid-and-long-term viewpoint, it is necessary to establish the evaluation method based on the understanding of the contamination mechanism including the changes over time in the contamination state and the penetration behavior in the concrete. Even though a number of researches have been conducted on removal of contamination source in pipes during nuclear fuel reprocessing regarding the contamination mechanism of the metal of the piping and equipment by radioactive materials, few number of researches are found on the contamination mechanism of the metal of the piping and equipment in the environment of the 1F. While it is considered necessary to clarify the contamination mechanism inside PCV or RPV exposed to the high temperature environment at the time of accident, it is not necessary to take into consideration a special contamination mechanism such as penetrate into metals outside PCV. As for the decontamination mechanism of resins and paints of cables, it is considered that it not necessary to conduct a special research on decontamination since they can be replaced or removed.

Theme	(6) Environmental fate studies of radioactive materials generated during decommissioning
"Descriptions / Background issues" on the interim report	It is essential to clarify the behavior of radioactive materials such as sorption, dispersion, moving along with groundwater flow in shallow underground in order to conduct environmental fate studies to ensure they will not affect the environment.
Basic direction	In order to properly evaluate and reduce the risk of future environmental impact caused by radioactive materials in the Fukushima Daiichi NPS site, it is necessary to provide proper evaluation and estimation of the environmental fate of radioactive materials around the site via the shallow underground water and the surface water, or the ports, the marine, and the air, and to provide appropriate environmental countermeasures. Targeted radioactive materials are (1) the radioactive materials that exist in the ground or on the surface of the ground through the contaminated water leaked just after the accident (^{137}Cs, ^{90}Sr, ^{3}H, etc.), (2) the radioactive materials that poured into the ports in past and accumulated on the bottom of the sea (^{137}Cs, ^{90}Sr, etc.), and (3) the radioactive materials that are contained in the contaminated water that will be generated as the result of retrieval of the fuel debris or decommissioning and dismantling of the buildings (including ion such as actinide and suspended solids) that can be the future source term impacting environment. In order to estimate the impact of radioactive materials on the surrounding environment, it is indispensable to understand the existence form and the transport behavior of the radioactive materials as the required basic knowledge. Specifically, the targets include the existence form of the radioactive materials in the underground water, the distribution in the soil, the advection and diffusion behavior in the underground water, the existence form and the advection and diffusion in the surface layer, the existence form and the molten and diffusion behavior of the radioactive materials in the seawater in the port and on the bottom of the sea, and the transportation behavior to the surrounding environment through marine or air. Although all of those depend on the characteristic of the intermedium such as the property of soil and the geological condition, since the measurement work on 1F is limited, it is necessary to aim at establishing the evaluation method in a similar environment. In addition, in order to provide the accurate future estimation of the environmental fate, it is also necessary to develop the monitoring technology to identify the accurate contamination condition and the analysis technology to simulate the transportation behavior of the radioactive materials. As for the monitoring technology, the technology for long term and continuous remote measurement and the mapping and behavior identification technology using the big data are expected. As for the simulation technology, the creation model that can be used to analyze the behavior (influence of

	unsaturated layer, kinetic evidence, etc.) specific to shallow underground and the estimation technology using the code are expected. It is also important to aim at reducing the risk of the radioactive materials as environmental countermeasure. While a number of technologies can be developed including control of the amount of underwater, soil improvement, stabilizing agent, sorbent for purification of contaminated materials, and the permeable reaction wall, it is necessary to examine the factors that have priority since they may have critical influence on the decommissioning works. In order to provide reasonable environmental fate studies for the radioactive materials, it is important to proceed with considerations on the risk of environmental influence so the development of the evaluation method related to the risk of environmental influence has to be taken into account from this viewpoint.

Attachment 14 R&D medium-to-long-term plan



Attachment 15 Current progress of the Project of Decommissioning and Contaminated Water Management[200]

1. Grasping state inside reactor, characterizing of fuel debris, and internal investigation

1-(1) Advancement of comprehensive grasping of state inside reactor (FY 2016 - 2017)

(Related projects) Advancement of accident progression analysis technology for assessing conditions inside reactor (FY 2011)
Assessing conditions inside reactor by advancement of accident progression analysis technology (FY 2012 - 2013)
Assessing conditions inside reactor through application of severe accident analysis code (FY 2014)
Advancement of grasping conditions inside reactor by accident progression analysis and actual data, etc. (FY 2015)

1-(2) Development of analytical and estimation techniques for characterization of fuel debris (FY 2019 - 2021), Development of analytical and estimation techniques for characterization of fuel debris (Development of estimation technologies for variation character across the ages) (FY 2019 - 2021), Development of analytical and estimation techniques for characterization of fuel debris (Development of technologies for improving analytical accuracy and estimation of thermal behavior of fuel debris) (FY2020 – FY2021)

(Related projects) Characterization of fuel debris using mock-up debris and development of fuel debris processing technologies (FY 2011 - 2014)
Construction of material accountancy method related to fuel debris (FY 2011 - 2013)
Property analysis of actual debris (FY 2014)
Grasping properties of fuel debris (FY 2015 - 2016)
Development of techniques for characterizing and analyzing fuel debris (FY 2017 - 2018)

1-(3) Development of technologies for in-depth investigation of PCV inside (FY 2019 - 2021)

(Related projects) Development of investigation technologies of inside of PCV (FY 2011 - 2013)
Development of investigation technologies of inside of PCV (FY 2014 - 2015)
Development of investigation technologies of inside of PCV (FY 2016 - 2017)
Development of technologies for in-depth investigation of PCV inside (FY 2017 - 2018)

1-(4) Development of investigation technology inside RPV (FY 2016 - 2021)

(Related projects) Development of investigation technologies inside RPV (FY 2013 - 2015)

1-(5) Development of technologies for the detection of fuel debris inside reactors (using muon) (FY 2014 - 2015)

[200] Information Portal for the Research and Development for the Fukushima Daiichi Decommissioning (http://www.drd-portal.jp/)

2. Retrieval of fuel debris

2-(1) Development of technologies for retrieving fuel debris to be gradually expanded in scale (FY 2019 - 2021)

(Related projects) Development of sampling technologies for retrieving fuel debris and internal structures (FY 2017 – 2019)

*Change in the project name: With the publication of "Decommissioning R&D Plan 2020 of The 75th meeting of the Team for Countermeasures for Decommissioning and Contaminated Water Treatment/Secretariat held on February 27, 2020, Material 4: Others

2-(2) Development of retrieval method of fuel debris (FY 2019 - 2021), and safety system (FY 2019 - 2021)

(Related projects) Development of technologies for retrieving fuel debris and internal structures (FY 2014)

Advancement of retrieval method and system of fuel debris and internal structures (FY 2015 - 2018)

Development of technologies toward further expansion in scale for retrieving fuel debris and internal structures (development of technologies for dust collecting system) (FY2019 - 2020)

*Change in the project name: With the publication of "Decommissioning R&D Plan 2020 of The 75th meeting of the Team for Countermeasures for Decommissioning and Contaminated Water Treatment/Secretariat held on February 27, 2020, Material 4: Others

2-(3) Advancement of fundamental technologies for retrieving fuel debris and internal structures (FY 2017 - 2018)

(Related projects) Development of fundamental technologies for retrieving fuel debris and internal structure (FY 2015 - 2016)

2-(4) Development of PCV closed water circulation systems (FY 2018 - 2019)

2-(5)-1 Development of repair methods for leak spots in PCV (FY 2016 - 2017)

(Related projects) Development of identification technology of leaks in PCVs (FY 2011 - 2013)

Development of repair method for PCVs (FY 2011 - 2013)

Development of repair (water stoppage) technology toward water filling in PCV (FY 2014 - 2015)

2-(5)-2 Full-scale test of repair methods for leak spots in PCV (FY 2016 - 2017)

(Related projects) Full-scale test of repair methods for leak spots in PCV (FY 2014 - 2015)

2-(6) Development of evaluation methods of seismic performances of RPV and PCV and the impacts of the damages (FY 2016 - 2017)

(Related projects) Development of evaluation methods for the structural integrity of RPV and PCV (FY 2011 - 2013)

Development of evaluation methods for the structural integrity of RPV and PCV (FY 2014 - 2015)

2-(7) Development of corrosion inhibition technology for RPV and PCV (FY 2016)

(Related projects) Full-scale test of repair methods for leak spots in PCV (FY 2014 - 2015)

2-(8) Development of criticality control technologies of fuel debris (FY 2012 - 2016)

2-(9) Development of technologies for non-destructive detection of radioactive material deposited in S/C, etc. (FY 2014)

2-(10) Development of remote decontamination technology in the reactor building (FY 2014 - 2015)

(Related projects) Development of remote decontamination technology in the reactor building (FY 2011 - 2013)

2-(11) Formulation of comprehensive radiation dose reduction plan (FY 2012 - 2013)

2-(12) Development of technologies for containing, transferring, and storing fuel debris (FY 2016 - 2021)

(Related projects) Development of containing, transferring, and storing technologies of fuel debris (FY 2014 - 2015)

2-(13) Development of technologies for environmental improvement inside reactor buildings (FY2020 - 2021)

2-(14) Development of technologies for maintaining remote equipment (FY2021)

2-(15) Development of support technologies for integrated management of Fukushima Daiichi NPS decommissioning (Development of continuous monitoring system inside PCV) (FY2021)

3. Waste management

3. Research and development of processing and disposal of solid waste (FY 2019 - 2021)

(Related projects) Development of technologies for processing/disposal of secondary waste by treatment of contaminated water (FY 2012)

Development of technologies for processing/disposal of radioactive waste (FY 2012)

Research and development of processing/disposal of solid waste (FY 2013 - 2014)

Research and development of processing/disposal of solid waste (FY 2015 - 2016)

Research and development of processing/disposal of solid waste (FY 2017 - 2018)

Research and development of preceding processing methods and analytical techniques (FY 2018)

4. Spent fuel management

4-(1) Evaluation of long-term integrity of fuel assembly removed from SFPs (FY 2015- 2016)

(Related projects) Evaluation of long-term structural integrity of fuel assemblies removed from SFPs (FY 2012 - 2014)

4-(2) Investigation of method for processing damaged fuel, etc. removed from SFPs (FY 2013 - 2014)

5. Contaminated water management

5-(1) Verification tests of tritium separation technologies (FY 2014 - 2015)

5-(2) Verification of technologies for contaminated water treatment (FY 2014)

5-(3) Large-scale verification of impermeable walls (frozen wall) (FY 2014)

5-(4) Development and verification of high-performance multi-nuclide removal equipment (high-performance ALPS) (FY 2014)

Attachment 16 Major activities related to enhancing international collaboration

Table A16- 1 Intergovernmental Framework between Japan and other countries

Framework	Descriptions
Annual Japan-UK Nuclear Dialogue	This dialogue is held based on the appendix to the joint statement of the Japan-UK top level meeting in April 2012, “Japan-UK Framework on Civil Nuclear Energy Cooperation” (Since February 2012).
Japan-France Nuclear Energy Committee	It was established under the joint statement of Japan–France top-level meeting in October 2012 (Since February 2012).
Japan-US Decommissioning and Environmental Management Working Group	After the Fukushima Daiichi NPS accident in March 2011, the establishment of the US-Japan Bilateral Commission on Civil Nuclear Cooperation (the Bilateral Commission) was announced in April 2012 based on the relationship between Japan and the US to further reinforce bilateral cooperation. Under this commission, “the Decommissioning and Environmental Management Working Group (DEMWG)” was established (Since December 2012).
Japan-Russia Nuclear Working Group	The Nuclear Working Group was established after confirming that Energy is one of the eight areas of cooperation plan approved at the Japan-Russia top-level meeting in September 2016, (Since September 2016).

Table A16- 2 Inter-organizational Cooperation Agreement

Domestic	International	Descriptions
NDF	NDA	Exchange of information for various technical knowledge on decommissioning, etc. and personal exchange are provided. (Concluded in February 2015)
NDF	CEA	Exchange of information for various technical knowledge on decommissioning, etc. and personal exchange is provided. (Concluded in February 2015)
TEPCO	DOE	Umbrella Contract was made and information is exchanged as needed. (Concluded in September 2013)
TEPCO	Sellafield, Ltd.	Information Exchange Agreement for site’s operation, etc. was concluded. (September 2014)
TEPCO	CEA	Information Exchange Agreement on for decommissioning was concluded. (September 2015)
JAEA	NNL	Comprehensive Agreement for advanced technology on nuclear R&D, advanced fuel cycles, fast reactor, radioactive waste
JAEA	CEA	Cooperation Agreement for specific technical issues on molten core-concrete interaction, etc.

JAEA	Belgium Nuclear Research Center	Agreement of Cooperation for Nuclear R&D and Research on the accident of the Fukushima Daiichi
JAEA	Nuclear Safety Research Center (Ukraine)	Memorandum for decommissioning research, etc. of the Fukushima Daiichi NPS and Chernobyl was concluded.
JAEA	IAEA	Research Agreement on characterization of fuel debris

Table A16- 3 Dissemination of information to the world (Holding or attending International Conference (from April 2020 to August 2021))

Conference Name	Period	Organization
The 64th IAEA Conference Side event	September, 2020	NDF METI TEPCO
ICRP/JAEA International Conference on Recovery After Nuclear Accidents	December, 2020	NDF METI TEPCO JAEA
Japan-UK Nuclear Dialogue	December, 2020	METI
IRPA 15, International Commission on Radiological Protection, Fukushima Special Session	January, 2020	TEPCO
OECD/NEA Fukushima Daiichi Accident 10years Report, Web seminar	March, 2021	NDF
IAEA Fukushima Webinar	March, 2021	NDF
US Waste Management 2021	March, 2021	TEPCO IRID
WTO/SPS Committee	March, 2021	METI
Japan-Russia Nuclear Working Group	March, 2021	METI
The 5th UK-Japan Nuclear Industrial Forum	June, 2021	NDF METI TEPCO IRID JAEA
The 27th International Conference on Nuclear Engineering (ICONE27)	August, 2021	TEPCO IRID
Fukushima Research Conference	Year round	JAEA

Table16- 4 Dissemination of information to the world (on web (in English)

Site	Organization
Mid-and-long-term Roadmap towards the Decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Station Units 1 to 4 （http://www.meti.go.jp/english/earthquake/nuclear/decommissioning/）	METI
Monthly report to the embassies concerning discharging and seawater monitoring from the Fukushima Daiichi NPS	METI, MOFA
Nuclear Damage Compensation and Decommissioning Facilitation Corporation's website （http://www.dd.ndf.go.jp/eindex.html）	NDF
Information Portal for the Research and Development for the Fukushima Daiichi Decommissioning (http://www.drd-portal.jp/en/)	NDF
Activities for Decommissioning （https://fukushima.jaea.go.jp/english/）	JAEA
IRID website (http://irid.or.jp/en/)	IRID
Fukushima Daiichi Decommissioning Project （http://www7.tepco.co.jp/responsibility/decommissioning/index-e.html）	TEPCO
Providing English version of Press release to foreign media	TEPCO
Management Office for the Project of Decommissioning and Contaminated Water Management(https://en.dccc-program.jp/)	MRI (Business consignee)

Table A-16 - 5 Major collaborative projects with foreign organizations

Project	Contents/Period of project	Participating Organization
IAEA Project		
DAROD	・Knowledge and experience obtained from the efforts on challenges of decommissioning and recovery of damaged nuclear power facilities (regulations, technologies, systems, and strategies) are shared among the relevant countries. ・Project period : 2015 to 2017	NDF
OECD/NEA Project		
BSAF	・Researching institutions and governmental organizations from eleven countries joined to conduct benchmark study using severe 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 FPs spread inside the reactors. Knowledge and findings related 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.	IRID JAEA TEPCO

	・Project period : 2015 to 2018	
ARC-F	・In succession to the BSAF project, researching institutions and governmental organizations from twelve countries joined to investigate the situation of the accident in more detail and utilize it for further researches to improve safety of light-water reactor ・Project period : 2019 to 2021 (scheduled)	NRA IR JAEA
PreADES	・Sharing characteristics information that helps to understand properties of fuel debris such as its phase state and composition. ・Enhancing “Fuel debris Analytical Chart” that summarizes needs and priority of fuel debris analysis. ・Maintenance of tasks after analysis and analysis facility information ・Project period : 2018 to 2020 (scheduled)	METI NRA IR JAEA IRID NDF TEPCO
TCOFF	・In reference to the accident progression of the Fukushima Daiichi NPS, (1F) advancing molten core and molten fuel models, FP migration behavior model and thermodynamic database as their basis. Based on the material scientific knowhow, evaluating details of molten core and fuel on condition of 1F accident, and characteristics of fuel debris and its producing mechanism. Then, providing material scientific knowhow and result of detail evaluations to international cooperation project including PreADES, ARC-F, TAF-ID, and domestic decommissioning project like IRID. ・Project budget was contributed from MEXT. ・Project period : 2017 to 2019	MEXT JAEA IR Tokyo Institute of Technology
EGCUL	・Discussing on characterization method for waste from unknown derivation	METI NDF JAEA TEPCO