ABSTRACTS of the Technical Poster Session

Let's talk about Fukushima Daiichi Decommissioning and the Future of the Local Community

1FD6

The 6th International Forum on the Decommissioning of the Fukushima Daiichi Nuclear Power Station

Mon, August 29, 2022

Alios Iwaki Performing Arts Center in Iwaki-city,
Fukushima-prefecture, Japan

NDF Nuclear Damage Compensation and
Decommissioning Facilitation Corporation (NDF)

ひとつ、ひとつ、実現する ふくしま

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A01

Organization Profile of IRID

Isao Imamura[1]

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Abstract

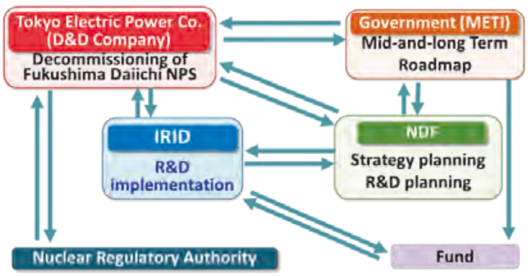
Ever since the International Research Institute for Nuclear Decommissioning (IRID) was established in August 2013, IRID has been committed to challenging an urgent issue—research and development (R&D) of technologies required for the decommissioning work of the Fukushima Daiichi Nuclear Power Station (NPS) including strengthening the foundation of decommissioning technology.

1. Roles of IRID

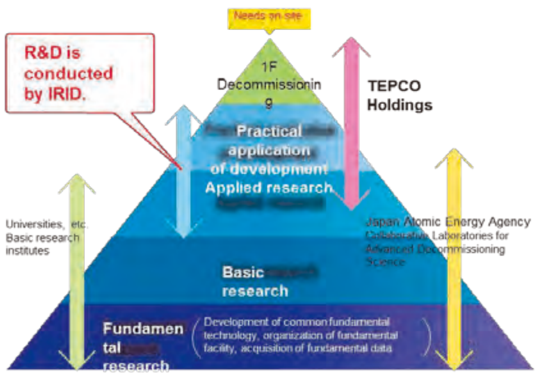
1-1.Scope of Work

- ●R&D for nuclear decommissioning
- ●Promotion of cooperation on nuclear decommissioning with relevant international and domestic organizations
- ●Human resource development for R&D

1-2. Roles of the Organizations for the Decommissioning Project of the Fukushima Daiichi NPS.

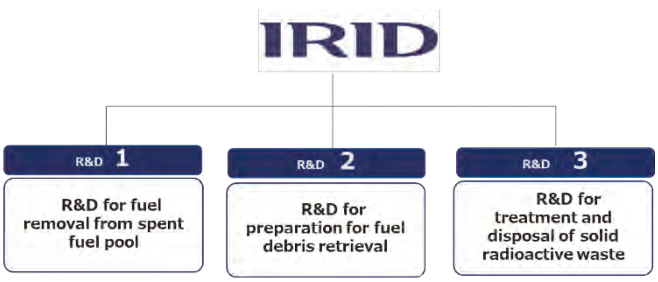


1-3.IRID R&D Scope



* The above chart was created based on the NDF Technology Strategy Plan 2017

2. IRID's Three R&D for Nuclear Decommissioning



A02

Overview of IRID R&D Projects

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International Research Institute for Nuclear Decommissioning (IRID),

Abstract

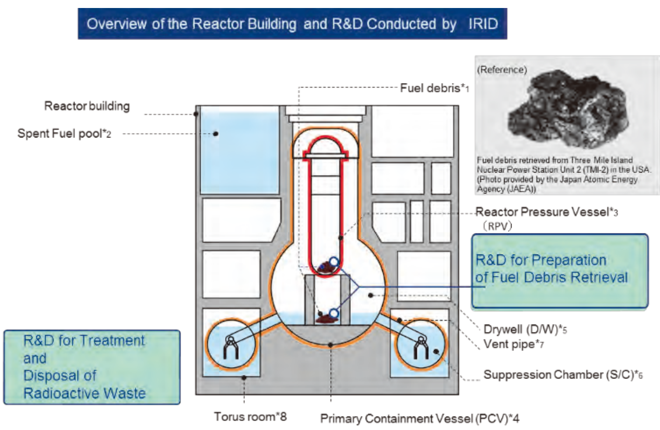
For the decommissioning of Fukushima Daiichi Nuclear Power Station (NPS), four organizations work together as one team. IRID is a complex entity consisted of eighteen organizations that play a leading role in research and development (R&D) for the decommissioning of Fukushima Daiichi NPS. IRID is conducting R&D projects including the project of "Preparation of Fuel Debris Retrieval" and "Treatment and Disposal of Radioactive Waste". Prior determination of fuel debris retrieval methods in fiscal year (FY) 2019, IRID has developed three technologies for; 1. Intensive investigation for fuel debris and the damaged conditions inside the reactor, 2. Potential risk management and verification for the nuclear safety, 3. A reliable remote operation under high radiation environments.

1. Progress of R&D

As for the preparation of fuel debris retrieval, IRID is undertaking R&D projects based on three elements. Firstly, detection technology that enables to directly access fuel debris in the PCV has been developed. In April 2015, a robot successfully entered the PCV at Unit 1. In FY 2016, a preparation for fuel debris investigation outside the pedestal started. At the same time, investigation robots for inside the pedestal, and remotely operated drilling device for the PCV penetration have been developed for reducing worker exposure at Unit 2. Additional fuel debris investigation inside the pedestal were performed by an underwater swimming robot for Unit 3, and new investigation equipment with a telescopic pipe and cameras for Unit 2. These robots remotely accessed fuel debris and successfully obtained visual data of the primary containment vessel (PCV) internals. The Severe Accident Analysis Code was upgraded to identify fuel debris inside the reactor, and investigations through the cosmic-ray muon were performed. The distribution of fuel debris in the reactor was investigated from outside the reactor building by using the muon for Unit 1. The muon investigation showed that a large amount of fuel is less likely to remain in the reactor core. The muon transmission measurement was performed for Unit 2 from March to July 2016, and for Unit 3 from May to September 2017. Essential technologies for accessing fuel debris in the RPV or the PCV are currently developed. Therefore, ensuring the safety of fuel debris retrieval are required.

2. Future Development

The purpose of IRID is to proceed with R&D for the decommissioning of the Fukushima Daiichi NPS and to acquire knowledge and expertise from around the world. Specifically, overseas technology for removal and storage of damaged fuel as well as the safety management system are required.



A03

Research and Development of the Project of Decommissioning, Contaminated Water and Treated Water Management and Connection to Preliminary Engineering

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[1]Mitsubishi Research Institute, Inc. (Management Office for the Project of Decommissioning, Contaminated Water and Treated Water Management)

Abstract

The Ministry of Economy, Trade and Industry has established the fund since FY 2013 and implemented the “Project of Decommissioning,Contaminated Water and Treated Water Management" as the subsidy program by solicitations to support R&Ds with high technical difficultiess. In this presentation, it shall be introduced the connections among the various subsidized projects with this program and the expected contributions to the decommissioning of Fukushima Daiichi NPS.

1. Introduction

In order to implement the decommissioning of the Fukushima Daiichi NPS safely and steadily, it is important to conduct R&Ds by gathering wisdom in Japan and overseas. Therefore, the Ministry of Economy, Trade and Industry has established the fund since FY 2013 and implemented the "Project of Decommissioning, Contaminated Water and Treated Water Management" as the subsidy program by solicitations to support R&Ds with high technical difficulties. Various R&Ds in the program have been managed by the Management Office for the Project of Decommissioning, Contaminated Water and Treated Water Management. Mutual coordination among the research projects are necessary to apply the results of R&Ds to the decommissioning of Fukushima Daiichi NPS.

2. Subsidized Projects of Decommissioning, Contaminated Water and Treated Water Management and Connection to Preliminary Engineering

The subsidized projects are classified into “Internal Investigation”, "Development of Retrieval Method", “Improvement of Work Environment”, and “Processing of Solid Waste, etc.”. The R&Ds of Fuel Debris Retrieval have been conducted based on the information obtained by Internal Investigation. In addition, the results of R&Ds such as Development of Fuel Debris Retrieval Method are reflected to Improvement of Work Environment. The research projects of Processing of Solid Waste are also studied in cooperation with R&Ds for Fuel debris retrieval and Improvement of Work Environment. In this way, the current projects are closely connected and then the engineering has been utilizing for wisdom in Japan and overseas.

B01 Development of criticality impact analysis method for weakly coupled systems including fuel debris particles

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Abstract

For fuel debris removal work at the Fukushima Daiichi NPS (1F), it is necessary not only to avoid a criticality by engineered safety measures but also to establish counter measures for the criticality based on impact evaluations. A space-dependent kinetic analysis code specific to weakly coupled reactors including fuel debris particles has been under development based on the integral kinetic model.

1. Introduction

It is important to evaluate the impact of a criticality that may occur in removing fuel debris from the damaged 1F reactors for ensuring safety of workers engaged in the removal operation [1]. The fuel debris located at the bottom of the 1F RCVs can be a fast-thermal weakly coupled reactor system including fuel debris particles that are possible to move (Fig. 1). So, a space-dependent kinetic analysis code specific to weakly coupled reactors, Multi-region Integral Kinetic (MIK) code, has been under development based on the integral kinetic model (IKM) [2].

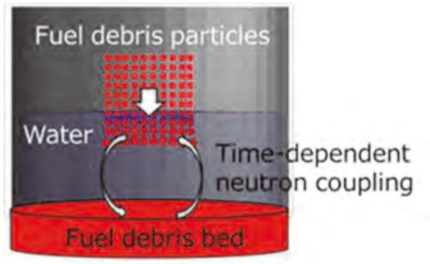

Figure 1. Example of weakly coupled system.

2. Methods

The IKM calculates the fission reaction rate in region i at the present time t ($N_i(t)$ [fissions/s]) by integrating all contributions from past fissions in source region j at past time t' to the present fissions using:

$$N_i(t) = \sum_j \left\{ \int_{-\infty}^{t} \left(\alpha_{ij}^{\mathrm{p}}(\tau) + \alpha_{ij}^{\mathrm{d}_I}(\tau) \right) N_j(t')dt' \right\}, \quad (1)$$

where $\alpha_{ij}^{\mathrm{p}}(\tau)$ and $\alpha_{ij}^{\mathrm{d}_I}(\tau)$ are the probability density functions of secondary fissions in region i at the present time t which are induced by prompt neutrons or delayed neutrons of the I$^{\mathrm{th}}$ precursor family that are generated from a source fission in region j at the past time t' with a time difference $\tau \equiv t - t'$ [secondary fissions@i/sec/source fission@j].

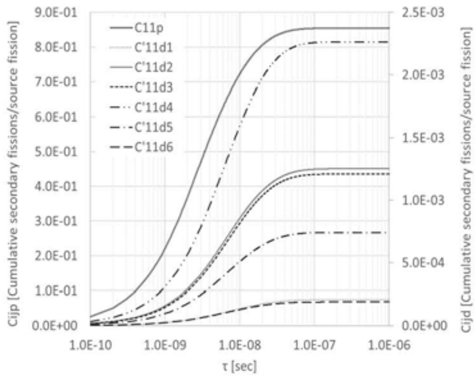

Figure 2. Example of $C_{ij}^{\mathrm{p}}(\tau)$ and $C_{ij}^{\mathrm{d}_I}(\tau)$.

3. Progress and Future Plan

Continuous energy Monte Carlo neutron transport codes, such as MVP or Serpent, will be used to calculate cumulative distribution functions $C_{ij}^{\mathrm{p}}(\tau)$ and $C_{ij}^{\mathrm{d}_I}(\tau)$, the integral form of the functions $\alpha_{ij}^{\mathrm{p}}(\tau)$ and $\alpha_{ij}^{\mathrm{d}_I}(\tau)$. Example of $C_{ij}^{\mathrm{p}}(\tau)$ and $C_{ij}^{\mathrm{d}_I}(\tau)$ functions calculated for Godiva reactor are shown in Fig. 2 and they were verified by the comparison to Godiva delayed neutron fractions. Delay of delayed neutrons emission will be introduced in Eq. (1) by the law of decay. After verification of the new MIK code, it will be coupled to an MPS code to reflect fuel debris particles movement on criticality impact analysis. This work is financially supported by the Nuclear Energy Science & Technology and Human Resource Development Project (through concentrating wisdom) from the Japan Atomic Energy Agency / Collaborative Laboratories for Advanced Decommissioning Science.

References

[1] https://clads.jaea.go.jp/jp/rd/map/2022/issues/fdr/fdr-207.html, [2] Takezawa, Tuya, Obara, NSE195(2021)1236.

B02

Remote Control Technology for Monitoring Inside PCV Pedestal during Retrieval of Fuel Debris

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Abstract

This research aims to develop human resources in the field of remote technology for the decommissioning of the Fukushima Daiichi Nuclear Power Plant (1F). We conduct research on a monitoring platform for fuel debris removal. We expect to develop research personnel through participation in projects, lectures, and facility tours.

1. Introduction

To safely and reliably remove fuel debris using a remote-controlled robotic arm, it is necessary to accurately grasp the three-dimensional situation inside the containment vessel before each operation. We propose to construct a platform for monitoring inside the pedestal. Various sensors such as cameras, gamma-ray and neutron detectors will move on this platform to perform measurements and visualization necessary for fuel debris removal.

2. Development Subjects

2-1. Monitoring platform (The Univ. of Tokyo, Fukushima Univ.)

Technology for building modular, split-type platforms and highly rigid, lightweight arms that can be retracted compactly have been developed.

2-2. Remote control interface (The Univ. of Tokyo, Kobe Univ.)

We will develop a video presentation interface for operators and a highly realistic tele-operation system.

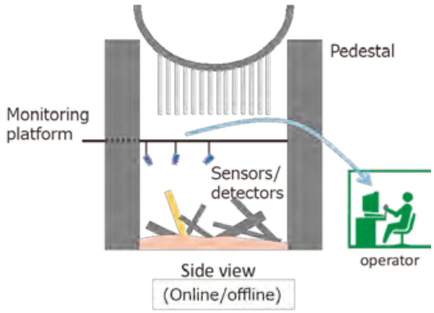

Figure 1. A monitoring platform concept

2-3. Radiation monitoring device (The Univ. of Tokyo)

Neutron/gamma-ray measurement devices that can operate under high dose rates and imaging techniques that integrate positional information has been developed.

2-4. A three-dimensional reconstruction method of environmental models (JAEA CLADS, Univ. of Aizu)

A three-dimensional restoration technique of environmental models using data obtained from cameras for understanding the in-vessel environment easily has been developed.

3. Conclusion

We have conducted research on a monitoring platform for fuel debris removal. In addition, lectures with the current status and future issues of 1F were introduced for students, and the research on master/doctor theses and conference presentations were done by younger person as a human personnel development.

The Collaborative Laboratories for Advanced Decommissioning Science (CLADS), Japan Atomic Energy Agency (JAEA), had been conducting the Nuclear Energy Science & Technology and Human Resource Development Project. This study has been conducted in the project from FY2019.

B03 Navigation and Control of a Novel Shock-resistant Mechanical Manipulator for Fuel Debris Retrieval

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[1]The Univ. of Tokyo, [2]RITECS Inc., [3]Kogakuin Univ., and [4]JAEA

Abstract

This research aims at developing a novel manipulator for retrieving fuel debris on the Fukushima Daiichi Nuclear Power Plant (1F) as shown in Figure 1. We conduct research on designing shock-resistant CVT (Continuously Variable Transmission) robot in collaboration with the University of Sussex. Our main focus is on navigation and control.

1. Introduction

Fuel debris retrieval at the bottom of the primary containment vessel (PCV) is one of the significant tasks for the decommissioning of the nuclear power plant and in particular for F1. The presence of radiation, water leakage and poor light conditions make the retrieval process challenging for conventional manipulators. We tackle those problems by the novel mechanical manipulator and its control and navigation algorithm. CVT-based actuation will enhance the robot's shock resistance. AI-based navigation algorithm aids the operator to navigate the robot and grasp the fuel debris in the cluttered environment inside the PCV.

2. Development Subjects

2-1. Optimal actuation parameters (The Univ. of Tokyo, RITECS Inc.)

We configured a simulation environment to evaluate the optimal parameters of the long-reach manipulator robot.

2-2. Navigation and Control of CVT-VIA manipulator (The Univ. of Tokyo, RITECS Inc.)

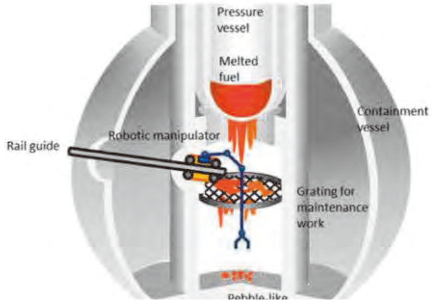

Figure 1. Robotic manipulator in PCV

We set up the machine learning environment using the dynamics simulator called "PyBullet". A human operator can command the gripper's target position by clicking on the camera output of the environment.

2-3. System evaluation and demonstration (The Univ. of Tokyo, RITECS Inc., Kogakuin Univ., JAEA CLADS)

We will conduct the evaluation and demonstration of the proposed system on a real robot.

2-4. Demonstration of use-case scenarios (The Univ. of Tokyo, RITECS Inc., Kogakuin Univ., JAEA CLADS)

We will validate the proposed robot system in the mock-up in the University of Tokyo or NARREC.

3. Conclusion

We have conducted research on a decommission robot manipulator featuring CVT-based actuation and a learning-based navigation system. Currently, component development is in progress. Future works include the development of the whole manipulator and the integration of the navigation system to real robots.

The Collaborative Laboratories for Advanced Decommissioning Science (CLADS), Japan Atomic Energy Agency (JAEA), had been conducting the Nuclear Energy Science & Technology and Human Resource Development Project. This study has been conducted in this project from FY2021.

B04

Neutron detection system without radiation protection for criticality approach monitoring based on diamond sensors and radiation-resistive integrated-circuits

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[1]High Energy Accelerator Research Organization., [2]Nagoya University, [3]Hokkaido University, [4]Japan Atomic Energy Agency, [5]National Institute of Advanced Industrial Science and Technology, [6]Kyushu University

Abstract

This report summarizes the research results of the “Technology development of diamond-base neutron sensors and radiation-resistive integrated-circuits for shielding-free criticality approach monitoring system” conducted until FY2021.

1. Introduction

The study aims to develop key components(diamond sensors and radiation-resistive integrated circuits) of a neutron detector without a radiation shield for a criticality approach monitoring system shown in Figure 1. It is required high neutron detection efficiency(a few cps/nv) under 1 kGy/h and compact-light-weight to fit constraints of the penetration size and the payload.

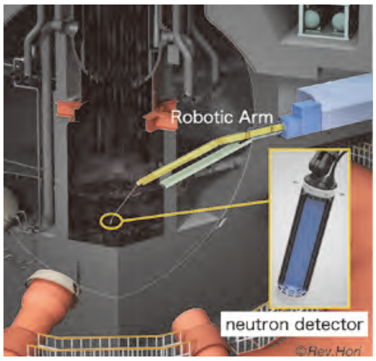


Figure 1. neutron detector for a criticality approach monitoring system

Requirements for the neutron detector are determined by numerical evaluations of subcriticality under various signal-to-noise ratios(neutron counts/γ-ray counts) by simulation studies using measurement data of reactor noise and simulation studies of neutron sensitivities for several neutron converters shown in Figure 2. One of key issues is high radiation environment. The response function of the detector have to be optimized to minimize the γ-ray counts and to maximize the neutron detection efficiency. The fabrication process of the diamond neutron sensor is newly developed for the purpose, and the developed signal-processing integrated circuits and a high speed data-transfer integrated circuit have to work up to a few MGy.

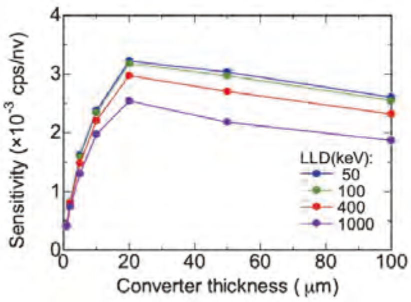


Figure 2. Sensitivity of a neutron sensor using LiF. An example of the studies.

2. Status

This feasibility study clarified that the Feynman-α method for the subcriticality measurement can measure the prompt neutron decay constant if the ratio of γ-ray counts to neutron counts is less than 1.[1] We confirmed thousands of neutron sensing devices achived a few cps/nv based on numerical simulations, and finished developments of a prototype neutron monitoring board which can be mounted 512 devices with 1 MGy radiation tolerant signal processing circuits. Diamond sensors with the signal processing circuits also work under 0.645kGy/h as show in Figure 3, after modification of production processes. We will start evaluation of the neutron detector using LiF convertor under high radiation environment in FY2022.

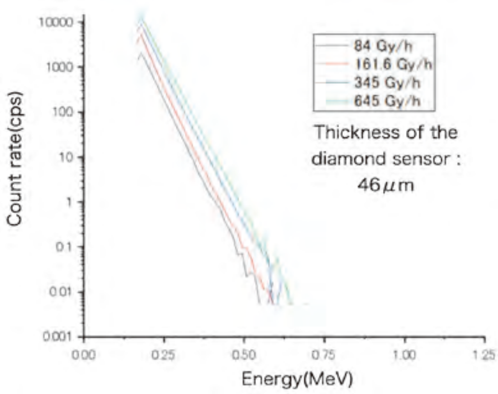


Figure 3. γ-ray noise spectrum under Co γ-ray irradiation environments.

3. Acknowledgement

This work was supported by JAEA Nuclear Energy S&T and Human Resource Development Project through concentrating wisdom Grant Number JPJA20P20336542.

References

[1] Reports of FY2021 Nuclear Energy Science & Technology and Human Resource Development Project, to be published.

B05

The world's highest radiation-resistant lubricants supporting decommissioning of the nuclear reactor

Yoshikazu Hayashi, MORESCO Corporation
Advanced Specialist of The Radiation-Resistant Lubricants

Abstract

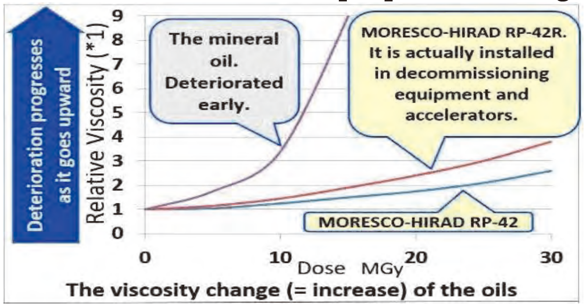
Excellent radiation-resistant lubricants are essential for equipment in high-dose areas.
The lubricants, "MORESCO-HIRAD"s showed the world's highest radiation resistance, not only against a γ ray but also a neutron-dominated mixed ray.
Accordingly, it has been comprehensively installed in the debris retrieval devices.

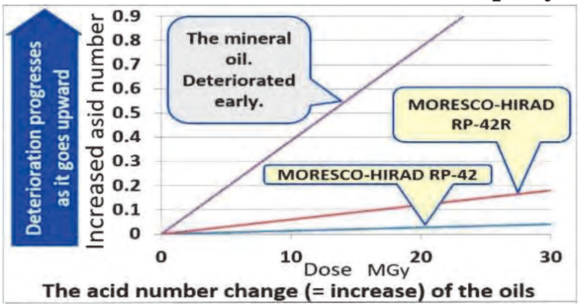
1. Introduction

It is shown below as the changes in each lublicant's properties depending on the doses that the obtained data from the concerted studies with JAEA or the European accelerator projects.
A lubricant having less change in its properties even at a high dose has an excellent resistance.

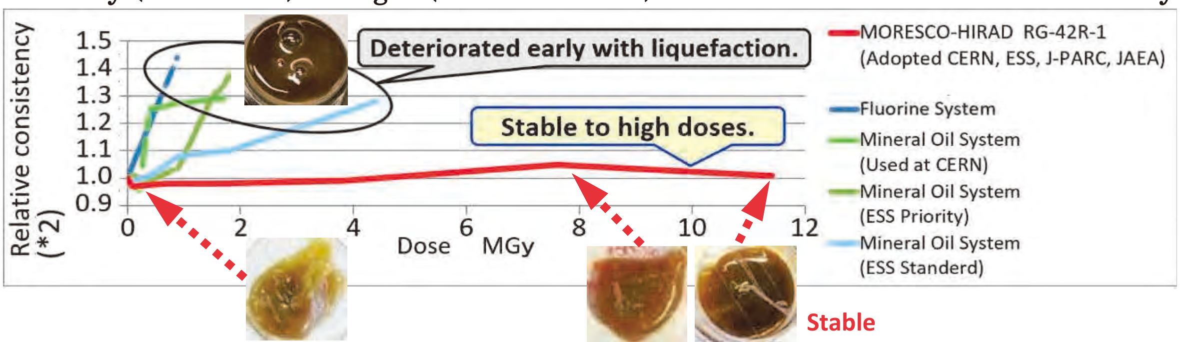
2. The irradiation-evaluations and the findings for the lubricants

2-1. With JAEA ～ The properties changes (= deterioration) of the oils with a γ-ray





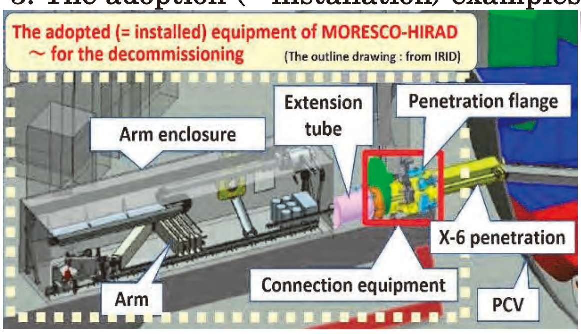
2-2. With the European accelerator projects, CERN & ESS ～ The greases' consistency (=hardness) changes (= deterioration) with a neutron-dominated mixed ray

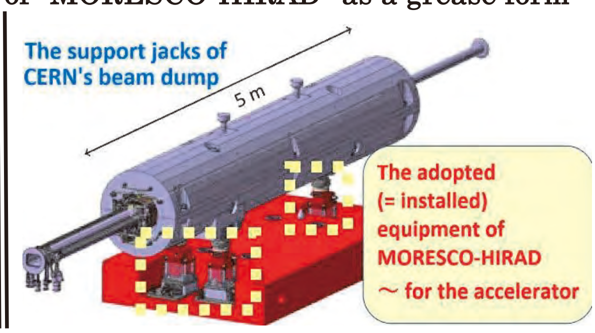


(* The appearances of RG-42R-1 for each dose)

(*1) and (*2) is calculated with each sample's respectively value at the time of non-irradiation as "1".

3. The adoption (= installation) examples of "MORESCO-HIRAD" as a grease form





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B06 Labor-saving design and construction for Fuel Removal at Fukushima Daiichi Nuclear Power Station Unit 2

Motohiro Taniyama[1], Jyunro Nakagoshi[1], Miho Miyazaki[1], Ippei Matsuo[1], Kihei Ogawa[1]
[1]Kajima Corporation

Abstract

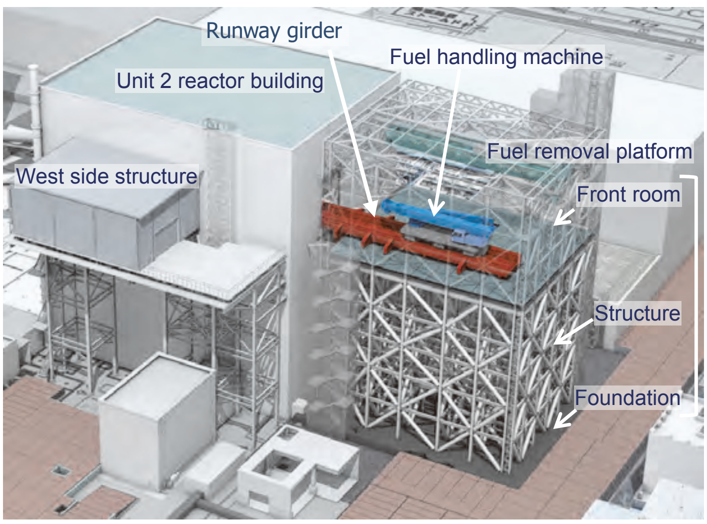
Since the reactor building of Unit 2 did not experience a hydrogen explosion and its structure frame remains, the existing frame will be utilized to the maximum extent. However, the core has been melted down and the radiation dose is very high, so it will be required to consider minimizing the exposure of the workers. This paper presents the labor-saving design and construction will be used for the fuel removal at Unit 2.

1. Labor-saving design

The fuel removal structure made of steel is a rectangular structure of 27.0 m in the east-west direction, 32.7 m in the north-south direction, and 44.75 m in height. In order to control the relative displacement to the reactor building and to suppress the deformation of the upper part of the structure, four oil dampers will be installed between the buildings. The runway girder will be fixed in the structure with straddling the reactor building through an opening in the south wall of the reactor building. Elastic bearings and oil dampers with springs will be installed between the runway girder and the floor of the reactor building. This equipment will be installed by not fixing in the existing building with high radiation dose in order to reduce radiation exposure of the workers.

2. Labor-saving construction

In order to reduce the amount of work at the site where the ambient dose is high, to assemble a part of the steel frame is planned at the off-site area where the ambient dose is low. About 70% of the entire steel frames are assembled at this assembly yard. These assembled units are transported to the site by a self-propelled multi-axle cart. The transported units are lifted, and the steel frame of the platform structure is constructed. The large units can be incorporated with ancillary facilities, which contributes to reducing the amount of work other than steel frame construction. The ground improvement machine has been also modified to allow remote operation, and the operator can check the status of the improvement work from the remote-control room.



Overview of Unit 2 fuel removal structures

C01 Evaluation of fuel debris behavior and α- and β- ray radiation effects on corrosion

Nobuaki Sato[1], Kuniki Hata[2], Tomonori Sato[2] and Chiaki Kato[2]
[1]Tohoku Univ., [2]Japan Atomic Energy Agency

Abstract

To know the behavior of fuel debris generated by severe reactor accident, chemical structure and stability of nuclear fuel debris were evaluated. Radiation effects such as radiation dose and range by of α- and β- rays from radionuclides in fuel debris were also analyzed.

1. Introduction

The special corrosive environment of inner part of 1F PCV should be considered if structural materials directly contact with α/β nuclides of nuclear fuel debris at any positions in the PCV. Though the radiation distance of these α– and β– rays are shorter than that of γ ray, the locally high energy deposition would accelerate the corrosion of materials. The radiation distance and dose are dependent on the configuration of fuel debris. Types of fuel debris are classified as oxide debris, alloy debris and so on. Oxide debris is mainly consisted of UO_2 and ZrO_2, with fission products (FP) and minor actinides (MA). Alloy debris is the mixture of UO_2 and melted alloy such as zircalloy and iron steel. The MCCI debris is the product of molten core concrete interaction when molten core dropped down into concrete part. In this paper, characterization of fuel debris and radiation effects of α- and β- rays were evaluated for α/β nuclides of fuel materials and minor actinides contained in the fuel debris.

2. Results and discussion

2-1. Oxide debris

According to the pseudo binary UO_2-ZrO_2 system, the cubic $Zr_yU_{1-y}O_{2+x}$ phase is stable in uranium rich region, while $U_yZr_{1-y}O_2$ phases (monoclinic, tetragonal, cubic) appears with increasing temperature. In each oxide phase, Pu, MA and FP elements coexist in the $(U,Zr)O_2$ forming solid solution. As for the radiation effect of these elements, two β emitters (^{90}Sr, ^{239}Np, ^{241}Pu) and four α emitters (^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Am) were considered. Radiation dose and distance were calculated using PHITS 3.14 program with cylindrical UO_2(1g) source from its surface (z=0cm) to water. Distances from debris surface were found to be 40μm and 2mm for α- and β- rays, respectively, while their surface dose rates (0.1 Gy/s) were higher than that of γ–ray.

2-2. Alloy and MCCI debris

The alloy debris is thought to be a mixture of $(U,Zr)O_2$ and Fe-Zr alloy. Most of high radioactive elements are contained in the oxide phase. In the case of homogenous mixture of oxide and alloy, the dose rate decreases with increasing alloy ratio, while the distances are the same as those of oxide debris. If the oxide phase is covered by thin alloy layer, radiation shielding effect seems to appear, while a part of α/β radiation comes out through the alloy film thinner than 0.1μm. Similar behavior of radiation effect was observed for MCCI debris.

3. Conclusion

α/β radiation effects from high radioactive nuclides in fuel debris were evaluated for different types of debris. The alloy or concrete thin layer of heterogenous debris seem to suppress radiation effect from oxide debris.

Acknowledgments

This study was supported by the JAEA Nuclear Energy S&T and Human Resource Development through concentrated wisdom [grant number JPJA20P20333127].

C02 Preparation, microstructure characterization and mechanical property evaluation of metallic solidified mixtures

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Abstract

In this study, the SUS-Zr-B_4C mixtures were prepared to simulate the metallic-dominated corium formation in 1F. The microstructure was characterized to reveal the phase constitution in these mixtures, in addition, the mechanical property of each constituent phase was evaluated. The results will be helpful for reaching a better understanding of the fundamental property of the 1F fuel debris.

1. Introduction

Achieving a better understanding of the fundamental property of fuel debris is required for the retrieval, long-term storage and processing, and aging behaviors of 1F fuel debris. For example, the mechanical properties such as hardness, Young modulus, and fracture toughness are important indicators for selecting the cutting method and/or tools for fuel debris retrieval. The fuel assembly in 1F is mainly composed of fuel, Zircaloy fuel cladding, B_4C control rod and its stainless steel (SUS) cladding, for the sake of simplicity, the research objective of this study is starting from the corium formation of non-fuel components, i.e., the solidified mixture from SUS-Zr-B_4C melt. The acquisition of the mechanical property of the constituent phases in the metallic fuel debris is the purpose of this study.

2. Experiments

The SUS-Zr-B_4C solidified mixtures were prepared by high frequency vacuum melting method. Two types of ingots were prepared with the ratio of SUS/Zr of 1:3 and 1:1 respectively, and the ratio of B_4C was 5 wt.%. The microstructure of the fabricated ingots was characterized using scanning electron microscopy-energy dispersice spectroscopy, Vickers hardness and nano-indentation tests were conducted to achieve the hardness of the bulk specimen and each constituent phases.

3. Results and discussion

Microstructural characterization results indicate SUS-75Zr-B_4C specimen consists of Zr_3Fe, Zr_2Fe, and $ZrFe_2$ type intermetallic compounds, and Zr, ZrC, ZrB_2 phases, whereas SUS-50Zr-B_4C specimen is dominated with $ZrFe_2$ phase. The phase constitution approximately follows the Zr-Fe binary phase diagram, although the volume fraction of the constituent phases was deviated from the equilibrium phase diagram. The mechanical properties of the constituent phases in these ingots, such as nano-hardness and Young modules, were successfully achieved. Results show that the hardness and Young modules of $ZrFe_2$ compound is the greatest among the intermetallic compounds, which is responsible for a much greater hardness of SUS-50Zr-B_4C ingot relative to SUS-75Zr-B_4C ingot. In addition, it is thus assumed that the Fe-rich metallic corium is more likely to exhibit a higher cracking susceptibility than the Zr-rich metallic debris.

Acknowledgement

This report includes part of the results obtained from the project “Investigation of Environment Induced Property Change and Cracking Behavior in Fuel Debris”, supported by Japan Atomic Energy Agency (JAEA) Nuclear Energy S&T and Human Resource Development Project through concentrating wisdom.

C03 Effects of beta-ray irradiation and gas-phase radiolysis on corrosion

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Abstract

To understand the local corrosion environment around fuel debris, effects of β-ray irradiation on the corrosion behaviors of carbon steel and stainless steel were investigated by electrochemical tests. Effects of gas-phase radiolysis on the corrosive environment in the liquid-phase were also investigated by γ-radiolysis experiments.

1. Introduction

The Primary containment vessels (PCVs) in Fukushima-Daiichi nuclear power station (1F) is exposed to the corrosive environment under irradiation. Effects of γ-rays on corrosion of materials in contaminated water had been investigated and "general effects" of radiation on corrosion were summarized in our previous work[1]. However, the materials nearby the fuel debris are exposed to not only γ-ray but also α- and β-rays from fuel debris. Effects of gas-phase radiolysis could also be important if large amount of gaseous species are continuously irradiated at the headspace of PCVs. This gas phase radiolysis affects corrosive environment in some isolated water, which might locally exist in the PCVs. To understand the corrosive environment in 1F PCV in detail, these "local effects" of radiation should be understood as well as the "general effects". In this study, some electrochemical tests and radiolysis experiments were carried out to estimate the "local effects" of radiation on corrosion.

2. Effects of β-ray irradiation on corrosion of steels

The electrochemical tests using a sealed β-ray source (3.0 MBq Sr-90) were performed to evaluate the effects of β-ray irradiation on corrosion behaviors of steels. The β-ray source was set at the bottom of a test vessel, and 200-times diluted artificial seawater was poured in the vessel. The test solution and the β-ray source were separated by polyimide film. A working electrode made of stainless steel covered by peek was set in the vessel with a certain distance from the β-ray source, and the corrosion potential of stainless steel were measured. The corrosion potential showed a higher value than that measured without irradiation. The result indicated that the irradiation of β-ray would enhance the localized corrosion of stainless steel. The polarization resistance of carbon steel was also measured by using the same system.

3. Effects of γ-ray irradiation to gas phase on corrosive environment

Radiolysis experiments were carried out at the γ-rays irradiation facility in QST. Liquid samples with a certain amount of headspace were irradiated, and the concentrations of nitrogen oxides (nitrite and nitrate ions) were measured. The current results show that the concentrations of nitrogen oxides tend to increase when the gas/liquid ratio of samples increased. Oxygen in the headspace also enhanced the nitrogen oxide production.

4. Conclusion

In this study, effects of β-ray irradiation and gas-phase radiolysis on the corrosive environment were investigated. These data would be valuable information for precise estimation of corrosion behavior of the structural materials in 1F PCVs.

Acknowledgments

This study was supported by the JAEA Nuclear Energy S&T and Human Resource Development through concentrated wisdom [grant number JPJA20P20333127].

Reference

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C04 Development of Estimation Technology of Aging Properties of Fuel Debris of Fukushima Daiichi Nuclear Power Stations

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Abstract

The understanding of long-term stability of fuel debris (FD) is one of the important issues for the decommissioning work. Nine types of simulated FD samples underwent water immersion tests or air exposure test to evaluate the probability of micro-particle generation due to chemical aging under the environments to which FD is exposed in the Primary Containment Vessel (PCV) interior, or during the retrieval and storage operation. The experimental results revealed the particle generation occurred on specific FD composition and the environmental condition.

Introduction

The objective of this study is to declare the probability of long-term aging of FD under the circumstance, and to estimate the aging property to evaluate the influence on the defueling operation. It is reported that part of Chernobyl fuel containing materials (FCM) becomes fragile and micro-particles generation are progressed, over 30 years has passed since the accident. The radio-active particle generation is an important aging phenomenon from the viewpoint of the risk during Fukushima Daiichi (1F) decommissioning. Therefore, based on the Chernobyl knowledge, we defined the probable factors which induce the aging of FD under 1F environment, and conducted experiments to evaluate the property of particle generation using several types of simulated FD including glassy and non glassy composition.

Experimental procedure and results

The surrounding of FD is considered to be changed with the progress of decommissioning work, that is, Nitrogen gas injection circumstance may be changed to oxygen containing atmosphere. Hence, we presumed the aging mechanism of FD under 1F environment as follows. (1) Particle generation due to the oxidation of inclusion or precipitate contained in FD. (2) Particle generation due to dissolution of FD matrix. Nine types of FD simulated samples which have aging factors mentioned above were fabricated for examinations. Air exposure tests were conducted at 323 K for 500 h, and at 383 K for 100 h in atmospheric condition. Water immersion tests were conducted at 303 K for 500 h, and at 363 K for 100 h in air saturation water.

Table 1 shows the types of FD samples and experimental results. U-containing micro-particles observed in case of Zr(O) containing (U, Zr)O_2 after air exposure, grassy FD with UO_2 or (U, Zr)O_2 inclusion or after water immersion.

Table 1 Types of FD samples and experimental results

Type of FD sample	Test	Results
(U, Zr) O_2 + (Zr, U) O_2	Air	No particle detected
	Water	No particle detected
Zr (O) + (U, Zr) O_2	Air	U-containing particles found
Glass + UO_2 inclusions	Air	No particle detected
	Water	U-containing particles found
Glass + Zr-U-O inclusions	Air	No particle detected
Glass + Fe inclusions	Air	Fe- containing particles found
	Water	Fe- containing particles found
Glass + FeO inclusions	Air	Fe- containing particles found
Glass without inclusions	Water	No particle detected
Glass containing Boron	Water	No particle detected
Glass + (U, Zr) O_2	Water	U-containing particles found

- Glass: SiO_2 containing FD which simulates molten core concrete reaction. - Air: Air exposure test. - Water: Water immersion test.

Conclusion

It is found that particle generation of FD occurred on specific FD composition and the environmental condition.

ACKNOWLEGEMENT

This study is based on the results of Subsidy Program “Project of Decommissioning and Contaminated Water Management”.

D01 Development of Real-Time Dose Fiber-Monitor with Red-Scintillator

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Abstract
We have novel red-emission scintillator with a high light output of over 60,000 photons/MeV compared to other conventional scintillators, and we demonstrated real-time monitoring system for high-dose-rate with wide dynamic dose range .

1. Introduction

Mapping of high-dose-rate distribution in Fukushima Daiichi Nuclear Power Station is the one of the first steps for decommissioning, while conventional techniques are hard to monitor the rate due to the such high-dose rate condition of over 1 Sv/h. We have developed a fiber-type monitor with red-emission scintillator, because large mount of noises is generated in a photo detector excited by radiations. Scintillation photons are read with a photo-detector in lower rate thorough an optical fiber. On the other hand, fiber made by glass itself are also generated several noises excited by radiations: Cherenkov and scintillation photons in the glass. These noises have generally emission wavelengths of blow 550 nm, and to discriminate the noise, we have developed red-emission scintillator, Cs_2HfI_6 with an emission band of over 600 nm.

2. Methods

As the starting materials 99.9%-pure HfI_4 and 99.999%-pure CsCl beads were mixed inside the glove box filled with the inert Ar gas. The ampoule was evacuated and sealed-off, and then the crystal growth was performed using the vertical Bridgman-Stockbarger furnace. The X-ray excited RL spectrum was measured with a CCD camera, DU420-OE (ANDOR). The Cu-Kα X-ray was used for irradiation from a Mini-X (AMPTEK) with an accelerating voltage of 30 kV and beam current of 100 mA. The intensities were corrected by a CCD sensitivity, grating efficiency and reflection efficiency of the mirror in the spectrometer.

3. Results

Cs_2HfI_6 had an emission wavelength of around 700 nm excited by X-ray, and this scintillator was estimated to have light output of ~63,000 photons/MeV. Also, The FWHM energy resolution at 662 keV was calculated to be ~4.4%. Using this scintillator, we have developed the fiber-reading radiation dose monitor system with a 20-m-long optical fiber and a CCD spectrometer. This results shows we performed the real-time radiation detection test using a ~100 TBq ^{60}Co gamma-ray source, maximum dose ratio of 1.1 kSv/h. The scintillation signal intensities were measured as a function of the dose ratio, then the monitorable dynamic range was determined as the order of 10^{-3} up to 100 kSv/h [1].

4. Conclusion

This novel scintillator enables us to monitor high-dose-rate condition with optical fiber, and we succeeded in demonstration with this monitoring system with wide dynamic dose range .

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D02 Development of Rapid and Sensitive Radionuclide Analysis Method by Simultaneous Analysis of β, γ, and X-rays

Hirofumi Shinohara[1], Masumi Oshima[1], Yuichi Sano[1], Katsuyuki Suzuki[1], Haifeng Shen[1], Jun Goto[2], Tadahiro Kin[3], Takehito Hayakawa[4], Masahiro Taniguchi[5], Tomoko Haraga[6], Masato Asai[6]

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Abstract

We aim to realize the rapid analysis of radioactive nuclei in fuel debris and waste. Thus, we are developing a new spectrum analysis method (SDM) to apply to the latest measurement system such as the multiple γ-ray detection system, and to integrate the research of β(+X)-ray, γ-ray, and multiple γ-ray spectrum.

1. Introduction

In the future, trial removal of fuel debris in Fukushima Daiichi nuclear reactors will begin. It is essential to understand the characteristics of the fuel debris to remove the fuels safely and smoothly from the reactors, and thus various tests will be conducted. When analyzing radioactive nuclei in fuel debris, the conventional radioactive analysis method requires individual measurements of radioisotopes after complicated chemical separation for each element, which takes a great deal of time and effort. In addition, since radiation protection during work and training of engineers are issues, it is necessary to develop a simple and rapid method for analyzing radioactive nuclei.

2. Experiment

While the conventional analysis method (Total Peak Area Method) uses only the full energy peak for quantification, we have developed the SDM for γ-ray determination that uses the entire spectrum, including peaks and continuum components [1]. Compared to conventional methods, the SDM has high sensitivity because of using more number counts for quantification. Furthermore, the SDM has high accuracy because of directly derives radioactivity values without using peak analysis or nuclear data analysis.

3. Simulation and analysis

To develop the SDM, we have created standard spectra of 40 nuclides. When measuring a spectrum from standard radiation sources is difficult, we create it using the Geant4 simulation code. In addition, a machine learning model was built to identify simultaneously multiple nuclides such as Co-60, Cs-134, Cs-137, Eu-152, etc., and a convolutional neural network (CNN) was used to improve the accuracy of the SDM**.** We also examined the optimization of chemical separation suitable to the measurement conditions of the SDM.

4. Conclusion

The SDM is a reliable γ-ray determination with a simple procedure demonstrated in our study. Thus, we can expect to shorten the chemical separation process by simultaneous quantification of several nuclides with the SDM to reduce worker exposure and to obtain accurate results without advanced chemical separation skills.

Acknowledgment

This work was supported by JAEA Nuclear Energy S&T and Human Resource Development Project through concentrating wisdom Grant Number JPJA20P20333366.

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D03

Progress in the development of the continuous monitoring of tritium water by mid-infrared laser spectroscopy (Part I)

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Abstract

The present study aims to demonstrate the principle of short-time measurement of tritiated water using a cavity ring-down spectroscopic measurement system with a mid-infrared laser. A specially constructed optical cavity for mid-infrared CRDS was constructed. We have also demonstrated a mid-infrared laser that has a measurement advantage with broadband wavelength tunability. In addition, stable isotope heavy water samples were prepared as standard samples.

Our group has been developing a detection technique for hydrogen isotopes using mid-infrared lasers. The mid-infrared region has very large absorption lines of vibrations of hydroxyl groups, which can be used for highly sensitive detection. Recently, we have demonstrated the H_2O and D_2O in the liquid phase were measured using mid-infrared lasers with wavelengths of 2.9 μm and 3.9 μm, respectively[1]. The present study aims to demonstrate the principle of short-time measurement of tritiated water at the "60 Bq/cc level" using a cavity ring-down spectroscopic (CRDS) measurement system with a mid-infrared laser. In order to achieve the above goal, (1) research on the CRDS system and (2) evaluation of hydrogen isotope composition under environmental conditions and preparation of standard samples (subcontractor: Hirosaki University) were conducted this fiscal year. In (1), a mid-infrared CRDS test was conducted. An optical bench (3m x 1.2m) was set up in the laboratory, and a designed optical cavity was constructed on the optical bench. Next, a laser source essential for the CRDS measurement was developed. We have constructed a 4 μm wavelength Fe:ZnSe laser using a 2.9 μm wavelength Er:YAP laser, which was originally developed by our group, as the pump source. By assembling a Littrow configuration cavity with brazed gratings, we succeeded in developing a laser with a tunable wavelength range from 4.34 μm to 4.72 μm, a maximum power of 22 mW, and a beam quality of M^2=1.1. In (2), we prepared hydrogen isotope standard solutions using commercially available heavy water standard solutions from several reagent companies to prepare stable isotope heavy water samples as standard samples. We also purchased commercially available heavy water reagents and prepared a standard sample of approximately 100 Bq/L. In addition, preparations were made for isotope ratio measurements both indoors and outdoors. The performance of a low-background liquid scintillation counter was evaluated, and the lower detection limit was confirmed to be approximately 0.6 Bq/L after 2,400 minutes of measurement with a sample volume of 10 mL.

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D04 Development of the Sample-return Technique for Fuel Debris Using the Unmanned Underwater Vehicle

So Kamada[1], Kazuya Nishimura[1], Tetsuichi Kishishita[2], Takahiro Oyama[2], Keisuke Okumura[3], and Kennichi Terashima[3], and Masayuki Hagiwara[4]

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Abstract

To develop a fuel debris sampling system comprising a high radiation tolerance neutron detector, a camera, an underwater Lidar, and end-effector with powerful cutting and collection capabilities, and a manipulator under the Japan-UK joint research team. These are incorporated into an unmanned underwater vehicle (UUV). Also, we will develop a system that supports remote control utilizing a VR system with multiple sensors.

1. Introduction

To retrieve the fuel debris, understanding its distribution and its properties of fuel debris is a critical issue. Therefore, we will develop a technique to collect a small amount of fuel debris and detect a small number of neutrons from the fuel debris under a high gamma-ray dose. Therefore, under the bilateral joint research system between Japan and the UK, we will develop an unmanned vehicle-based debris sampling technology equipped with a debris mapping tool that combines a highly radiation-resistant neutron sensor, lidar, and a camera.

2. Devices

2-1. Neutron sensor

We adopted Micro-Structured Neutron Detector (MSND) from RDT of the United States as a neutron sensor, designed a signal processing circuit suitable for measuring weak MSND signals, and conducted characteristic tests using a neutron source. In addition, using a water tank, the difference in the detector response to neutrons due to the difference in water thickness was confirmed in the test.

2-2. UUV platform

We discussed the operation method of the neutron sensor to be developed, considering the interlocking with the underwater camera and sonar. Subsequently, we designed the communication/control circuit and interface. In addition, we examined tools for projecting image data from underwater cameras and sonar onto VR

2-3. Radiation environment simulations in PCV

Based on the internal investigation of Unit 3, we created a radiation source model assuming the fuel debris of Unit 3 by using the PHITS code. Subsequently, a gamma ray transport calculation was performed, and it was confirmed that the gamma-ray characteristics in the pedestal could be obtained.

2-4. Sampling tool

The Stewart platform required for testing manipulator development and UUV (Blue ROV2) were procured and assembled. These were used to integrate the manipulator and control system. As a result of designing the manipulator and developing and evaluating several conceptual designs, The UK team created a design drawing for preliminary actuators, component selection, manufacturing, and essential design specifications.

3. Japan-UK collaboration

We worked closely with CLADS to carry out the research. In addition, meetings were held to promote research implementation plans, including meetings with UK team.

D05

Technologies for measurement and management of radioactive materials and wastes

Junichi Hikosaka [1], Chris Carter [2] and David John [2]

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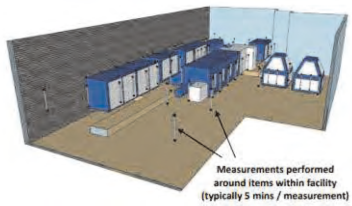
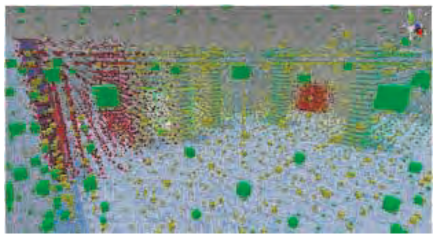
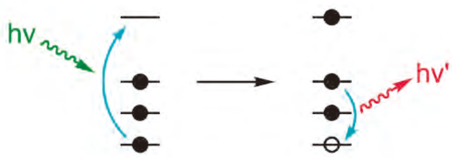
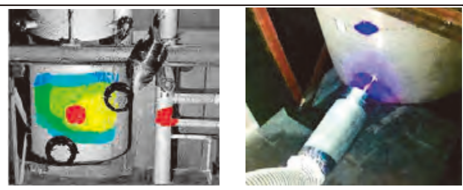
Abstract

As decommissioning and waste management challenges grow in complexity, technology and innovation is constantly evolving to provide remotely operated technical solutions that move humans away from harm and to provide data that can be used to optimise waste retrieval methods and waste management strategies.

1. Introduction

The poster shows examples of recent technologies that have been developed by Cavendish Nuclear for use in waste retrieval and decommissioning projects.

2. Technologies for measurement and management of radioactive materials and wastes

PHUMS and AmCAM – a fast neutron detection technology able to accurately assess the plutonium holdup in single plant items such as gloveboxes as well as entire facilities and cells. PHUMS and AmCAM measurements have been successfully undertaken in a number of alpha facilities in the UK	 **PHUMS measurements**
Particle Swarm Imaging or PSIM is an innovative computational method developed by Cavendish Nuclear that creates an image of the source term using the information obtained from multiple measurements at different positions within a facility. The type of measurement systems include gamma spectrometry, dose probes, gamma imagers and neutron measurement.	 **Dose rate map overlayed on 3D model**
OptiSort is a used for fully automated radioactive waste sorting and segregation. This technology includes a non-intrusive method for determining the chemical composition of the surface of solid waste items using X-ray fluorescence (XRF) and Raman Spectroscopy techniques to allow remote non-destructive identification of the chemical composition of the surface of waste materials.	 **Schematic representation of X-ray fluorescence**
In-Cell Decommissioning System (IDS) - The IDS uses its spatial and radiometric scanning technologies that are integrated with remote operation and virtual reality for safe and efficient decommissioning operations. IDS is used to map and characterize the contaminated cells using 3D LIDAR scan and gamma spectrometry to create an accurate virtual environment and to conduct automated size reduction decommissioning operations via remote VR control.	 **Image of cell with radiological overlay and laser cutter system.**

D06

Local Identification of Melted Multicompnent Materials and Mapping of Distributed the Original-material using Laser-Induced Breakdown Spectroscopic Multi-Spectra via Multivariate Analysis

Tomohiko Kawakami[1], Sakiko Nagayama1,Koudai Okazaki[1], Yuta Abe[2], Etsuyo Makuuchi[3], Masahiko Ohtaka[2] and Yoshitaka Takagai[4]

[1]Kaken., [2]JAEA, [3]Art KAGAKU, [4]Fukushima Univ.

Abstract

To understand the hardness of fuel debris which is essential information for cutting and removing, a laser-induced breakdown spectroscopy was investigated as an estimating means of the hardness of materials. Based on the analytical results of elements measured by LIBS, multivariate analysis was conducted estimating the materials and the hardness of the laser-shot point of the samples.

1. Introduction

In the decommissioning of the Fukushima Daiichi Nuclear Power Plant, where boron carbide is used as a control material, the cutting and removing fuel debris, which is a mixture of materials such as metals, oxides, and borides that are approximately twice as hard as oxides, is a challenge [1]. Laser induced breakdown spectroscopy (LIBS) is an analytical technique that can simultaneously measure elements from light (especially boron and oxygen) to heavy elements (metals), and is expected to discriminate the analytical target based on elemental composition ratio of the sample material. In this study, we present that the applicability of the compositional distribution by multivariate analysis based on the analytical results measured by LIBS.

2. Method

A simulated fuel assembly that mimics the core of a boiling water reactor (BWR) containing borides and oxides, was fabricated, and the test piece of the mimic fuel debris was then prepared by plasma heating (CMMR test piece)[2,3] (Figure 1(a)). The measurement was conducted by original system which was combined devices such as LIBS, a displacement gage, and a X-Y automation stage. The data obtained by those devices was finally correlated and merged. After the measurement, a multivariate analysis was conducted using special software TIBCO Spotfire.

3. Result

Figure 1(b) exhibits the compositional distribution of the CMMR test pieces constructed by multivariate analysis. Each elemental area such as oxygen to zirconium, boron carbide or control blade-derived boron or metal, and oxygen was discriminated by the multivariate analysis in the local points of samples (ex. the simulated fuel pellets (ZrO_2), cladding (Zr), and melted materials including Zr-O and Zr-B). The observed mapping was agreed with the results of mapping using an electron probe analyzer.

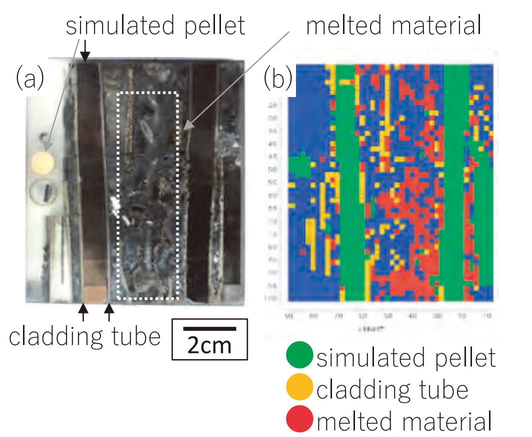


Figure1 The results of multivariate analysis in focusing on Zr-measurement points
(a) photograph, (b) materials distribution

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D07 Contamination of SGTS filter trains of Units 1 to 4 caused by the backflow of vent gas

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Abstract
In order to clarify the cause of the contamination of the Standby Gas Treatment System (SGTS) filter trains of Units 1 to 4, TEPCO investigated the SGTS rooms, and obtained information on dose rates and contamination around filter trains, vent lines, and rupture discs.

1. Introduction

The SGTS piping and filter train are connected to the vent line (Fig.1), but are basically not contaminated by the venting process. However, the filter trains of Units 1 to 4 were confirmed to be contaminated in previous investigations. Since Units 1 and 2 and Units 3 and 4 each share the stacks and Units 1 and 3 were estimated to be vented successfully, we assume that this contamination caused by the backflow of vent gas from Units 1 and 3. Therefore, the investigation of SGTS rooms of Units 1 to 4 was conducted to obtain information to support that assumption.

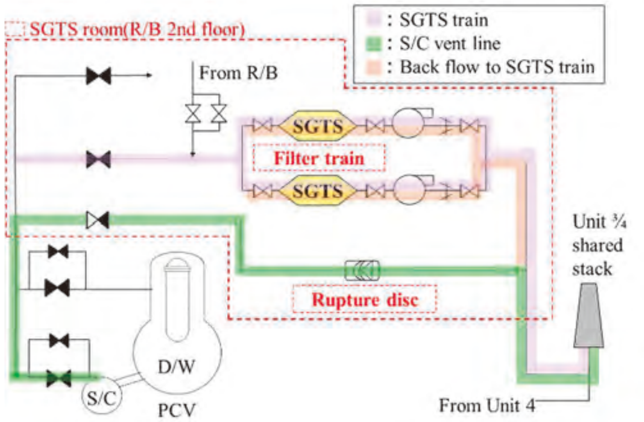

Fig.1 Schematic diagram of SGTS and vent line in Unit 3

2. Result of the investigation

In Unit 1, high dose rates was confirmed around the filter trains, with higher dose rates on the downstream sides. In Unit 2, contamination downstream of the filter train was confirmed (Fig.2). In addition, almost no contamination was confirmed around the rupture disc in Unit 2. In Unit 3, contamination was confirmed at the piping back flowing from vent line to the filter train and around downstream of the filter train (Fig.3). In Unit 4, contamination was confirmed downstream of the filter train (Fig.4). In addition, filter trains were opened in Units 3 and 4, and dosimetry and smear sampling were conducted on the filter surfaces.

3. Conclusion

This investigation clearly showed that the contamination of the filter trains of Units 1 to 4 was caused by the backflow of vent gas from Unit 1 or 3.

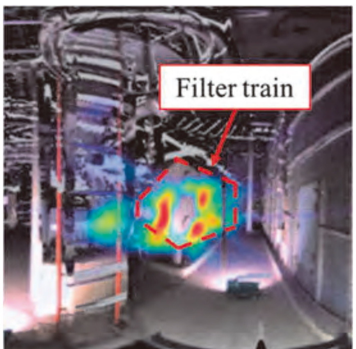

Fig.2 Filter train of Unit 2

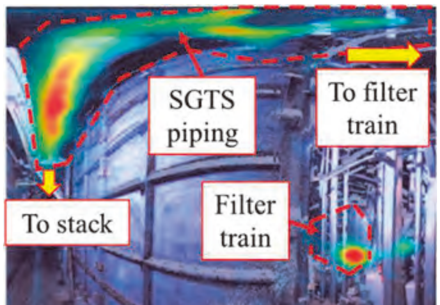

Fig.3 SGTS piping and Filter train of Unit 3

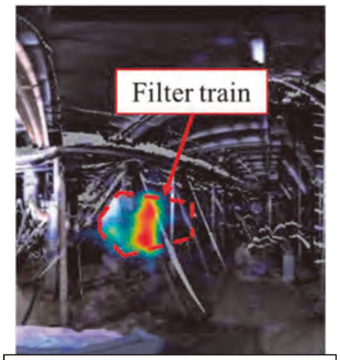

Fig.4 Filter train of Unit 4

W01

Study on Rational Treatment/Disposal of Contaminated Concrete Waste Considering Leaching Alteration (1) Overview

Tamotsu Kozaki[1], Takafumi Sugiyama[1], Katsufumi Hashimoto[1], Naoko Watanabe[1], Shinichiro Uematsu[1], Daisuke Kawasaki[2], Satoshi Yanagihara[2], Daisuke Minato[3], Toru Nagaoka[3], Yuka Morinaga[3], Naofumi Kozaki[4], Yoshikazu Koma[4]

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Abstract

A series of investigations was carried out in order to discuss rational approaches to treat and/or dispose of contaminated concrete waste with a focus on underground concrete structures of Fukushima Daiich Nuclear Power Station (FDNPS) in contact with contaminated water.

1. Introduction

Radioactive waste management is one of the integral issues in decommissioning of FDNPS. Underground concrete structure, which account for a significant fraction of total concrete, is in contact with contaminated water and when leaching alteration occurs migration behavior of radionuclides may change. The purpose of this study is to investigate migration behaviors of radionuclides in relation to the properties of concrete altered by leaching, to develop a model to calculate radioactivity concentration profiles and to estimate the amount of radioactive waste in each class, and to analyze and evaluate multiple waste management scenarios.

2. Overview of the project

Hardened cement paste (HCP) specimens, which underwent simulated leaching, were used for sorption and diffusion experiments with radioisotopes such as ^{125}I, ^{137}Cs, U and ^{14}C to reveal that migration behaviors depend on radionuclides, their chemical species and degree of alteration. Leaching behavior of concrete was visualized by non-destructive integrated CT-XRD method, and was quantitatively simulated by an improved simultaneous transport model. Calcium Silicate Hydrate (C-S-H) and Calcium Aluminum Silicate Hydrate (C-A-S-H), which are components of cementitious material, were synthesized, simulating different degrees of degradation, and were analyzed for microstructure using solid-state NMR. A method for statistically estimating radionuclide concentrations of the concrete debris stored in solid waste storage facilities was developed using radiation dose rate data. A simulation model for migration of radionuclides in concrete submerged in water was developed using particle tracking method in 3D space to obtain radioactivity concentration profiles of multiple radionuclides. Transport of Pu via the contaminated water, transfer models of ^{129}I into contaminated water, and the source term of ^{14}C were considered. In addition, several waste management scenarios were formulated for radioactive concrete waste and the application of Safety and Environmental Detriment (SED) index was considered to evaluate and analyze potential radiological risks arising from different waste management processes.

3. Summary

Migration behaviors and properties of altered concrete is examined and using the acquired data, a model is developed to predict radioactivity concentrations, and waste management scenarios are evaluated.

Acknowledgements

This study was a part of the outcomes of the “Nuclear science and technology and human resource development program”, “Study on rational treatment/disposal of contaminated concrete waste considering leaching alteration” (2020−2022). Part of this work has been performed in the facilities of the Central Institute of Isotope Science, Hokkaido University.

W02

Study on rational treatment/disposal of contaminated concrete waste considering leaching alteration: (2) Effect of leaching alteration on ^{137}Cs migration in cement paste

Shinichiro Uematsu[1], Kyoya Watanabe[1], Keisuke Matsumoto[1], Naoko Watanabe[1], Tamotsu Kozaki[1], Yuka Morinaga[2], Daisuke Minato[2] and Toru Nagaoka[2]

[1]Hokkaido Univ., [2]CRIEPI

Abstract

Sorption and diffusion behavior of ^{137}Cs in hardened cement paste altered by leaching was studied. Effects of the leaching alteration of cement paste were found for the sorption coefficient, diffusion coefficient, and the activation energy for diffusion.

1. Introduction

Radioactive waste management of concrete materials is an important issue in decommissioning of the Fukushima Daiichi Nuclear Power Station (FDNPS). Concrete materials in contact with water in the FDNPS may have been altered by leaching. In this study, sorption and diffusion coefficients of ^{137}Cs were experimentally determined for hardened cement paste (HCP) with and without alteration by leaching treatment to clarify the effect of leaching on migration behavior of ^{137}Cs in HCP.

2. Materials and Methods

Cement paste prepared from ordinary portland cement with the water-to-cement ratio of 0.36 was cured in an equilibrated cement solution for 28 d at 323 K. The leaching of the HCP was conducted by immersing the HCP in 6 M NH_4NO_3 solution for 5 d. Apparent diffusion coefficients (D_a) of ^{137}Cs were determined from one dimensional non-steady diffusion experiments at contrasting temperatures between 288−323 K. The apparent sorption coefficients (K_d) of ^{137}Cs were determined for ground HCP samples by a batch method.

Table 1. Sorption coefficients (K_d) of ^{137}Cs on HCP samples.

HCP sample	K_d (L kg^{-1})
Unaltered	1.0±0.1
Altered (6 M NH_4NO_3)	17±3

3. Results and Discussion

The K_d value was higher for altered HCP than for unaltered HCP as indicated in Table 1. Fig. 1 shows the temperature dependence of the D_a of ^{137}Cs in unaltered and altered HCP samples. The D_a values were different between the unaltered and altered samples at each temperature except at 313 K. In addition, the activation energy for diffusion (E_a) of ^{137}Cs for the altered HCP sample was *ca.* 60 kJ mol^{-1}, which was higher than that reported for unaltered HCP samples (37 ± 2 kJ mol^{-1}) [1]. Since E_a was determined from the temperature dependence of D_a, it may include the effect of the sorption enthalpy. Therefore, our result of the increase of E_a is likely to be caused by (1) changes in sorption behavior by alteration of HCP, (2) changes in the dominant diffusion process as a consequence of alteration, or both.

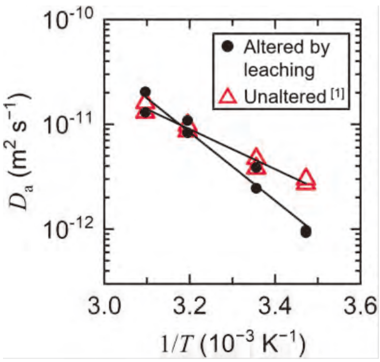

Fig. 1. Temperature dependence of the apparent diffusion coefficients (D_a) of ^{137}Cs.

References [1] Morishita *et al.*, ICONE26-82570, London (2018).

Acknowledgements This study was a part of the outcomes of the "Nuclear science and technology and human resource development program", "Study on rational treatment/disposal of contaminated concrete waste considering leaching alteration" (2020−2022) funded by the MEXT. A part of this work has been performed in the facilities of the Central Institute of Isotope Science, Hokkaido University.

W03

Quantitative Evaluation of Long-Term State Changes of Contaminated Reinforced Concrete Considering the Actual Environments for Rational Disposal

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[1]Univ. of Tokyo, [2]National Institute for Environmental Studies, [3]Taiheiyo Consultant Co. Ltd., [4]Taiheiyo Cement Corp., [5]Nagoya Univ., [6]Hokkaido Univ., [7]Japan Atomic Energy Agency

Abstract

In the decommissioning of concrete structures at the Fukushima Daiichi NPP, it is important to estimate the amount and concentration of waste. In this study, a prediction method of radionuclide penetration behavior considering the actual environment is investigated for quantitative prediction of contamination concentration distribution in concrete components. In this report, the outline of this study is described.

1. Overview of the study

1-1. Examination of condition setting of concrete members assumed in the actual environment

In order to set the condition of concrete members as expected in a real environment, the mesoscale concept have applied to mechanically evaluate the amount, width, distribution, and depth of cracks in concrete members. To evaluate the behavior of cracks, data on deformation and moisture movement due to drying and reabsorption of mortar have been obtained. In parallel, a rigid-body spring model (RBSM) was used to develop a program that can take into account changes in concrete age and temperature, water, and stress conditions.

1-2. Examination of penetration behavior analysis method considering the condition of concrete members

In order to evaluate the long-term penetration behavior of radionuclides into actual concrete matrices, data on sorption were obtained and a mathematical model was developed. And to evaluate the penetration behavior of radionuclides into concrete through cracks, the penetration of Cs and Sr into mortar specimens with different crack widths at concentrations of various ions equivalent to those of cooling water released immediately after the accident was evaluated by autoradiography.

1-3. Experimental evaluation of penetration behavior of α-radionuclides on concrete members

In order to investigate the penetration behavior of α-radionuclides into the concrete matrix, long-term immersion tests of cement paste specimens were initiated, and distribution ratios of α-radionuclides to aggregate, paint, and steel bars were measured.

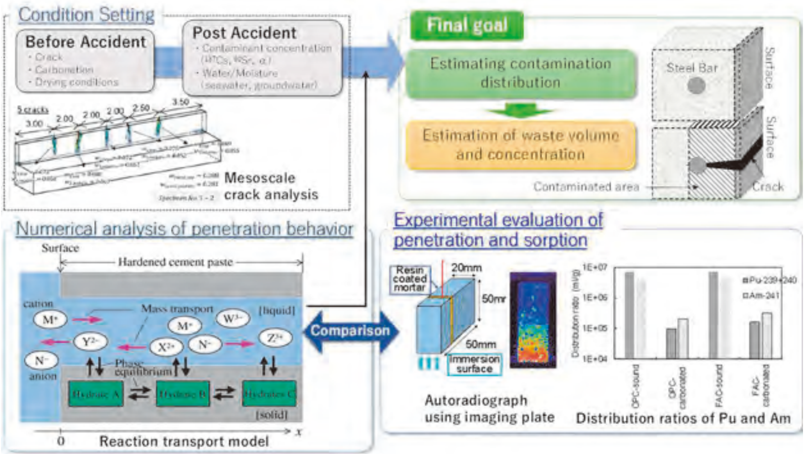


Figure 1. Framework of the study

2. Future Action

Based on the results of the above studies, we will estimate the contamination distribution and attempt to quantitatively predict the volume of waste and contamination concentration.

Acknowledgement

This work was supported by JAEA Nuclear Energy S&T and Human Resource Development Project through concentrating wisdom Grant Number JPJA20P20333545

W04 Challenge of Novel Hybrid-waste-solidification of Mobile Nuclei Generated in Fukushima Nuclear Power Station and Establishment of Rational Disposal Concept and its Safety Assessment

Masahiko Nakase[1], Miki Harigai[1], Shinta Watanabe[1], Ryosuke Maki[2], Hidetoshi Kikunaga[3], Tohru Kobayashi[4], Tomofumi Sakuragi[5], Ryo Hamada[5] and Hidekazu Asano[5]

[1]Tokyo Institute of Technology, [2]Okayama University of Science, [3]Tohoku University., [4]Japan Atomic Energy Agency, [5]Radioactive Waste Management Funding and Research Center

Abstract

The present study aims to establish the rational waste disposal concept of various wastes generated in the Fukushima Nuclear Power Station by the novel hybrid-waste-solidification concept. The radioisotope in synthesized ALPS sediment wastes, and AREVA sludge wastes, which were secondary wastes generated by water decontamination, were immobilized in phosphate waste form. They were processed to the stabilization treatment such as Spark Plasma Sintering (**SPS**) or Hot Isostatic Pressing (**HIP**) with well-characterized materials such as SUS and Zircalloy, which make the long-term stability evaluation and safety assessment possible.

1. Introduction The decommissioning of the 1F NPP station is in progress, and the retrieval of fuel debris will also be started The next issue is the disposal of various wastes, including some of the important elements such as Iodine, which is a mobile element and toxic alpha emitters of Actinide. Not only developing the waste form but also the compatibility with the actual disposal s essential. Therefore, we started the new project on hybrid-waste-solidification to connect the waste form study to the future disposal scenario.

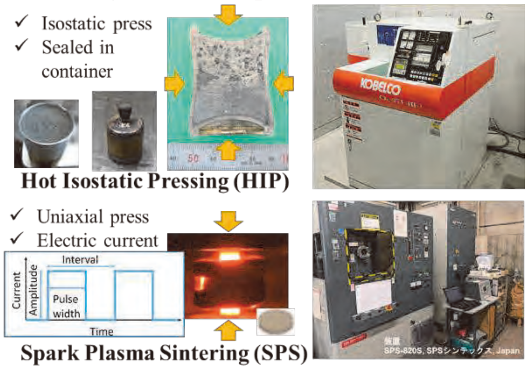


Fig.1 Schematics of HIP and SPS treatment

2. Outline of project and goals In the project, the research was divided into 5 themes as shown in Fig 2. In theme 1, some of the wastes such as ALPS, AREVA sediment wastes, and their stabilized waste forms are synthesized as well as a mobile and difficult element of iodine such as iodine apatite. In theme 2, structural study on wastes as well as leaching and α, β and γ irradiation behaviors are studied. Theme 2 includes both microscopic and macroscopic analyses such as SEM, TEM, XAFS, etc. In theme 3, first principle and molecular dynamics calculations were implemented to elucidate the mechanism of HIP and SPS treatments. In theme 4, synthesis of hybrid waste form by SPS and HIP and characterization are ongoing by varying the matrix materials. In theme 5, a scenario study on the disposal of hybrid waste form based on the inventory study is ongoing, and a safety assessment based on the corrosion behavior of the matrix material, and migration calculation are iimplementing.

1. Synthesis of primary waste form
2. Structure and characterization study
3. Chemical simulation approach
4. Synthesis of hybrid waste form
5. Scenario and safety assessment studies of disposal

Fig.2 Research themes of the project

3. Conclusion A new research project on hybrid-waste-solidification and safety assessment was started last year. The outline of the project and the scope, and some of the latest results will be shown in the poster presentation.

4. Acknowledgment This work is funded by the JAEA Nuclear Energy S&T and Human Resource Development Project through concentrating wisdom Grant Number JPJA21F21460873.

W05

SIAL® : Geopolymer solidification technology approved by Slovak / Czech Nuclear Authority

Milena Prazska and Marcela Blazsekova , Jacobs (Slovakia)
Hisashi Mikami , Isamu Kudo and Nobuyuki Sekine , Fuji Electric Co.,Ltd.

Abstract

We present the features and the performance records of SIAL® geopolymer solidification technology is licensed by both the Slovak (ÚJD SR) and Czech Nuclear (SUJB) regulators, and the technology has been used successfully for 20 years. More recently, geopolymers have been noted as an immobilization technology and which shows potential of immobilizing resins generated by treatments of contaminated water at Fukushima Daiichi Accident. Reference will also be made to some of the activities being undertaken in Japan to demonstrate its performance.

1. Introduction

The Nuclear Power Plant (hereinafter called NPP) Unit A1 located in Jaslovske Bohunice, which was completed in 1972 and had been operated for 5 years until two accidents happened in 1976 and 1977. After the second accident (INES level 4), NPP Unit A1 was permanently shut-down for decommissioning. Damaged fuel assemblies and claddings in the accidents caused contamination of strontium-90, caesium-137 and transuranic. As a result of a long-term corrosion of barrier’s materials, highly contaminated sludge were accumulated, and the waste could not be effectively immobilized with using conventional methods such as Cementation or Bitumen treatment due to negative impact on physical-chemical properties and high specific activity (caesium-137) of the waste. This challenge led to developing SIAL® solidification technology. Today, SIAL® is proven and widely used for on-site solidification of radioactive waste streams such as sludge, resins, sorbents and organic liquids. This is directly applicable to the conditions at Fukushima Daiichi as well as other Japanese NPPs where the waste streams are not well understood.

2. Feature

SIAL® matrix can provide efficient and practical on-site treatment of radioactive waste streams at room temperature, and can incorporate times as much wastes as Cement matrix equivalent on average.

The equipment used to deploy SIAL® solidification technology is also modular, flexible and versatile. It can encapsulate waste streams quicker than Cementation, and can be applied under water.

SIAL® solidification technology can realize higher compressive strength and lower leachability compared to Cementation, and posing a low fire risk and excellent physical stability in the presence of frost and water (no distortion or cracking).

Figure 1 SIAL® solidified samples

Figure 2 Sludge and slurry waste streams cross section observation

3. Performance Record Example

About 3×10^6 kg of radioactive waste streams (resins, sludge and crystalline borates) is successfully immobilized using SIAL®. This comprehensive scope of works started with licensing processes, solidification, and then were followed by decontamination and cleaning of the workplace post cleanup and transports of all equipment to off-site.

W06

Waste Management Symposia: The Annual Phoenix Radioactive Waste Conference Exchanging Knowledge from Around the World

Kazuhiro Suzuki[1], Gary Benda[2] and Akira Ono[3]

[1] WM Symposia Board of Director, [2] WM Symposia Deputy Managing Director/ Program Advisory Committee (PAC) Chair and [3] WM Symposia PAC Member

Abstract

WM Symposia's annual Waste Management (WM) Conference attracts thousands of registrants from around the world and is widely regarded as the premier international conference for the management of radioactive waste and related topics.

1. Introduction

The WM2023 Conference, the 49th version, will be held February 26 - March 2, 2023 at the Phoenix Convention Center in Phoenix, Arizona. Conference theme is "Planning for the Future: Innovation, Transformation, Sustainability" with a strong Conference focus on promoting the next generation of students and radwaste management professionals, encouraging exchange of expertise to young professionals around the world. WM2023 will feature over 500 papers and more than 80 panel discussions in over 160 technical sessions, complemented by nearly 200 exhibiting companies, the industry's largest.

2. Poster

The poster provides Conference details and describes the Technical Panel, Poster and Oral Sessions, Exhibitor, Student and Sponsorship program, as well as the opportunity to network with over 2,000 industry specialists and managers from more than 35 countries and learn of trends and developments from the most senior industry managers around the world. WM2023 has special programs aimed at encouraging participation of the world's leading companies in the exhibit hall, including a program for live product demonstration.

3. Conclusion

The Conference promotes, among Japanese and professionals from around the world, a broad exchange of knowledge in 12 Technical Tracks, including technologies, operations, safety, security & safeguards, waste management issues, decommissioning and dismantling, environmental remediation, STEM education and advanced nuclear reactors. The deadline for submittal of Abstracts for WM2023 is August 26, 2022. Details are shown on the poster.

Reference: www.wmsym.org.

W07

Integrated Waste Management and its Application to Nuclear Decommissioning and Dismantling Projects

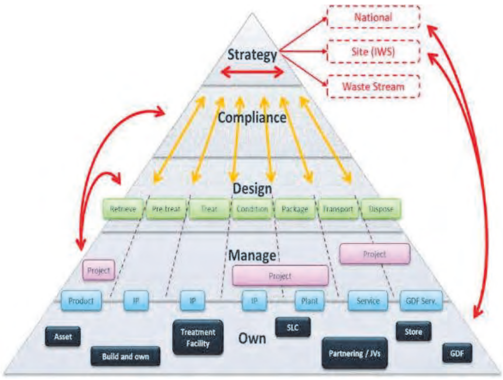
Michelle Dickinson, Antonio Guida and Bill Miller
Jacobs UK Ltd., Decommissioning and Regeneration Solutions

Abstract

Integrated Waste Management involves making waste management an integral part of the decision making, planning and execution processes for decommissioning and dismantling (D&D) projects. Waste is the main product of any of these projects, and waste management is often a limiting factor in project schedules. The success of any D&D project is therefore significantly impacted by the way in which waste management is planned and performed.

Introduction

Waste management has often been an afterthought in D&D projects, and this can result in waste disposition delaying the overall project completion or significantly increasing cost. Problems include wastes produced without appropriate characterisation; inadequate sorting and segregation; not meeting waste acceptance criteria (WAC) etc. With Integrated Waste Management, waste management experts are involved early and throughout the whole D&D project. Aspects to be considered as part of "integration" can include, depending on the circumstances:



- Integrating decommissioning and waste management plans to avoid generating wastes without a treatment route
- Integrating strategic planning with regulatory compliance and the design / operation of treatment facilities
- Integrating decision-making across all radioactive waste streams at national and site to achieve economies of scale
- Integrating treatment and decontamination methods to reduce waste volumes and maximise recycling
- Integrating national and local infrastructure to maximise efficiency and avoid duplication of facilities
- Integrating national and private companies to incentivise improved and cheaper waste treatment methods

Benefits include reduced lifetime costs, regulatory compliance and improved safety of the workforce.

Experience

At Jacobs we apply Integrated Waste Management in D&D projects in the UK, US and other countries. For example, Jacobs works with every nuclear site in the UK and across the entire lifecycle of radioactive waste:

- *Planning*, including decision making on Best Available Techniques (BAT) for waste treatment and disposal
- *Characterisation*, including performing statistical sampling and surveys of buildings and bulk wastes
- *Waste retrieval and treatment*, including high active tank wastes treated using cement or geopolymer
- *Waste packaging and transport*, including packaging of wastes to meet formal WAC for disposal
- *Interim storage and waste disposal*, including preparing operational and disposal safety cases for licensing.

Conclusion

We welcome the opportunity to demonstrate our experience and to explain the benefits of Integrated Waste Management to D&D projects in Japan that are limited by waste management plans and disposal infrastructure.

E01

Prototype of Differential Amplifier Circuits Based on Radiation hardened H-diamond MOSFET (RADDFET)

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[1]Hokkaido Univ., [2]KEK, [3]AIST

Abstract

A prototype differential amplifier circuit using hydrogen-terminated diamond MOSFETs (RADDFETs), which have excellent resistance to high temperatures and radiation, was fabricated by creating an Ltspice model based on the characteristics of the diamond FETs that make up the RADDFET. A device with 14 MOSFETs mounted on a 4 mm square diamond substrate was used. The actual circuit achieved 4.7 times amplification compared to 4.4 times in the simulation.

1. Introduction

After the accident at the Fukushima nuclear power plant, the performance required for equipment used in containment vessels to cope with severe accidents, such as operating temperature: 300°C (72 hours), integrated dose: 5 MGy, etc., has been increasing. In this study, as part of the development of electronic equipment for severe accident response, we designed and fabricated a differential amplifier circuit using diamond MOSFETs (RADDFET[1]), which are required in preamplifiers, and measured its characteristics.

2. Experiment

2-1. Preparation of RADDFET

Intrinsic layers were homo-epitaxially grown on a 4 mm square diamond single-crystal substrate by microwave CVD and the surface was hydrogen-terminated. A 40 nm oxide insulating film (Al_2O_3) was deposited as a passivation layer, and 14 diamond MOSFETs (called RADDFETs) were fabricated. RADDFETs have a proven track record of operation at 3 MGy integrated dose and 300°C. The production yield of the RADDFETs was 84%.

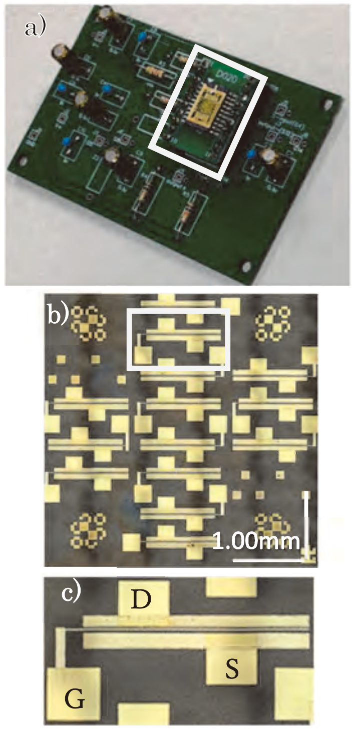

Figure1:
a) Appearance of prototype circuit
b) 14 FETs fabricated on diamond substrate
c) Enlarged view of FET device

2-2. Circuit Simulation

An LTspice model was created from the I_d-V_d characteristics of RADDFETs. A differential amplifier circuit was designed, the external supply voltage was specified so that all FETs enter the saturation region, and the input-output characteristics were investigated by simulation.

2-3. Circuitry

The prototype RADDFETs were mounted on a PCB board to create the prototype circuit shown in Figure 1. In this circuit, RADDFETs with similar characteristics are combined to construct a differential amplifier circuit.

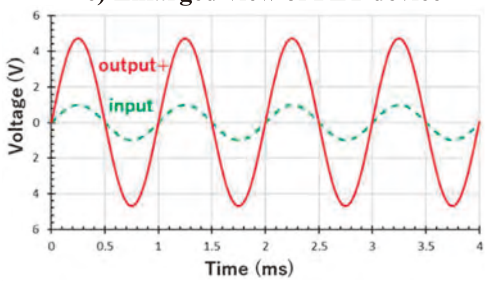

Figure2: Example of input/output characteristic measurement (Real circuit, one-sided amplification)

3. Result

Based on simulations, a voltage amplification of about 4.4 times was designed, but in the actual circuit, non-inverting amplification with an amplification factor of 4.7 times could be obtained by adjusting the external voltage and inverting and non-inverting outputs could be obtained simultaneously by further adjusting the voltage. This achievement is a major step forward toward the practical application of radiation-resistant, high-temperature-operating preamplifiers.

Acknowledgements

Part of this work was supported by the "SCORE" project of the Japan Science and Technology Agency. A part of this work was also supported by the Nuclear Energy Systems Research and Development Program (MEXT).

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[1] T. Yamaguchi et al., Appl. Phys. Lett.118, 162105

E02

Optically reconfigurable gate array VLSI without any common signal

Sae Goto, Minoru Watanabe and Nobuya Watanabe
Okayama University

Abstract

Very large-scale integrations (VLSIs) are vulnerable to radiation. VLSIs used in strong radiation environments such as space environment and nuclear power plants are always affected by radiation. Soft-errors and permanent failures on current VLSIs are caused by radiation. Therefore, we've been developing radiation-hardened optically reconfigurable gate arrays. This paper presents a proposal of a new optically reconfigurable gate array VLSI without any common signal.

1. Optically reconfigurable gate array

Optically reconfigurable gate array consists of three components: a programmable gate array VLSI, a holographic memory, and a laser array. The programmable gate array includes logic blocks, switching matrices, I/O blocks [1]. Although the programmable gate array structure is same as that of field programmable gate arrays (FPGAs), the configuration procedure is different from each other. The configuration of optically reconfigurable gate arrays can be executed in a perfectly parallel. As a result, even if a part of the programmable gate array and/or a part of the configuration circuit are broken by radiation, the operation on the programmable gate array can be executed continuously by reconfiguring the programmable gate array. In order to support such parallel configuration, optically reconfigurable gate array VLSIs have many photodiode circuits to detect optically-applied configuration contexts generated from the holographic memory. However, such optical configuration procedure requires two common signals: a refresh (nREF) signal to charge the junction capacitance of photodiodes and a configuration clock (CCLK) signal for flip-flops. The common signals were weak points for radiation.

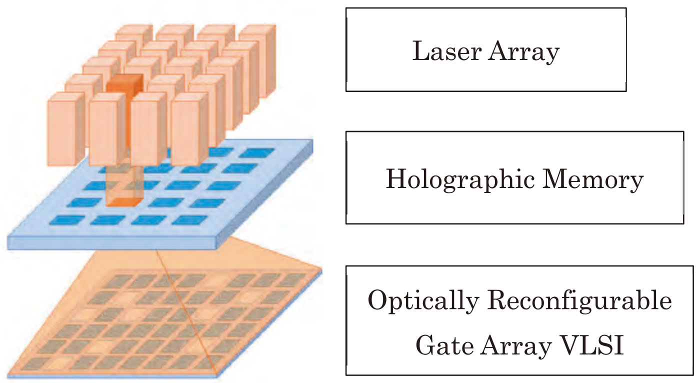

Figure 1. Structure of optically reconfigurable gate array

2. Optically reconfigurable gate array VLSI without any common signal

We have developed a new optically reconfigurable gate array VLSI not using such common signals. The configuration circuit never uses flip-flops so that the configuration clock (CCLK) signal is perfectly removed. Also, the refresh (nREF) signal is adjusted constantly to an intermediate value which is provided only on a metal wire without any buffer.

Reference

[1] Y. Takaki, M. Watanabe, "Optical multi-context blind scrubbing for field programmable gate arrays," IEEE Photonics Journal, Vol. 12, Issue 6, 7801411, Dec., 2020.

E03 Total-ionizing-dose tolerance of an optically reconfigurable gate array VLSI

Kaho Yamada, Takeshi Okazaki, Minoru Watanabe and Nobuya Watanabe
Okayama University

Abstract

Currently available Field Programmable Gate Arrays (FPGAs) use a serial configuration circuit. The serial configuration circuit is too vulnerable to radiation. Therefore, we have been developing an optically reconfigurable gate array VLSI with an optical configuration function. This paper presents the total-ionizing-dose tolerance of an optically reconfigurable gate array VLSI.

1. Introduction

At the Fukushima Daiichi Nuclear Power Station, a decommissioning work is currently being executed. However, when using a robot in such environment, there is a problem that integrated circuits inside the robot are easily broken by radiation. Therefore, we have been developing a radiation-hardened Optically Reconfigurable Gate Array (ORGA)[1][2]. In the ORGA, circuit is configured in parallel by light, so even if a part of the configuration circuit is broken by radiation, the configuration of the remaining region can be executed correctly.

2. Total-ionizing-dose tolerance of an ORGA-VLSI

An ORGA consists of a laser array, a holographic memory, and an ORGA-VLSI. Since the semiconductor devices are vulnerable to radiation, in the ORGA, we use a repairable VLSI concept. If we can use a programmable architecture for VLSIs, even if a part of a programmable gate array is broken by radiation, the non-damaged area of the VLSI can be used continuously so that the radiation-tolerance of the VLSI can be increased. To realize it, we have introduced parallel configuration architecture onto the ORGA instead of serial configuration architecture. Since laser arrays and holographic memories are relatively strong for radiation, ORGAs can achieve high-radiation tolerance. In this test, an ORGA-VLSI was exposed to 290 Mrad gamma radiation. Under the radiation environment of the Fukushima Daiichi Nuclear Power Station, although the existing integrated circuits will not operate normally in 10 hours, the ORGA can normally operate for 120 days.

3. Conclusion

In this test, only a 2.4 % performance degradation was confirmed at an absorbed dose of 290 Mrad. The small degradation can be out of consideration.

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G01 Development of technology to reduce environmental problems via innovative water purification agents

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Abstract

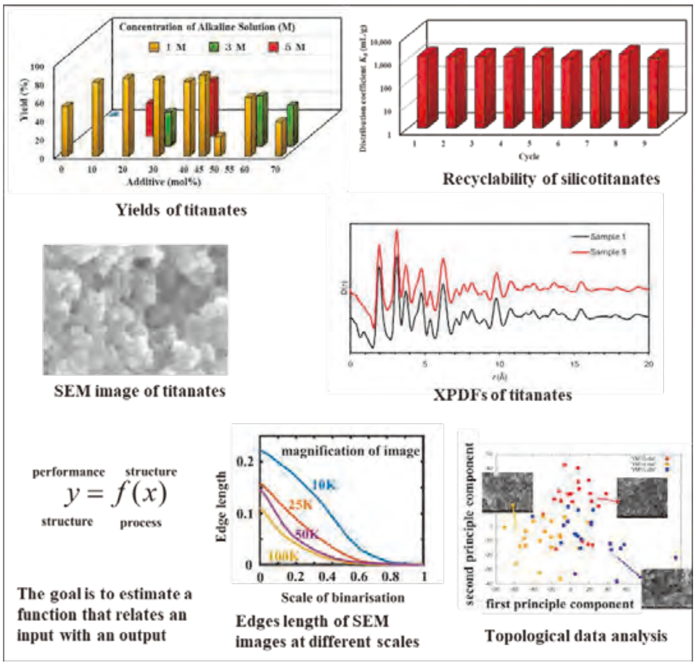
An efficient fabrication method to produce granular adsorbents for strontium without granulation process was developed, and titanates and silicotitanates were obtained as grains with suitable size for column operation in the decontamination process. They exhibited remarkable ion-exchange properties for removal of strontium ions from simulant contaminated water and behaved as reusable adsorbents without any significant loss of activity.

1. Introduction

^{90}Sr is one of the dangerous radioisotopes because its chemical similarity to Ca promotes the accumulation in bones and teeth of the human body, which causes long-term internal exposure. Although the decontamination work at Fukushima continues using various adsorbents, it is still desirable to develop highly efficient and inexpensive adsorbents. Furthermore, an adsorption process inevitably generates used adsorbents as secondary pollutants, which are required to be safely stored under strict control for a long time. Therefore, the suppression of generation of used adsorbents has emerged as an urgent issue. To overcome the above-mentioned problems, the present interdisciplinary collaboration research has been launched.

2. Results and discussion

The fabricated titanates and silicotitanates were analyzed by SEM, XPS, and PDF, and these data together with adsorption results were used for data analysis and theoretical calculations, leading to accurate prediction of the improved fabrication method of adsorbents. The resulting titanates and silicotitanates exhibited remarkable ion-exchange properties for removal of strontium ions from simulant contaminated water. Furthermore, they could be reused repeatedly without any significant loss of activity.



3. Conclusion

Granular adsorbents having high Sr uptake ability were fabricated without granulation process. Further studies to elucidate the adsorption mechanism is in progress. Additional studies to produce suitable waste forms to contain the Sr species produced during recycling and for the final spent absorbent are also in progress.

G02 Fukushima Forests: Decontamination and Restoration

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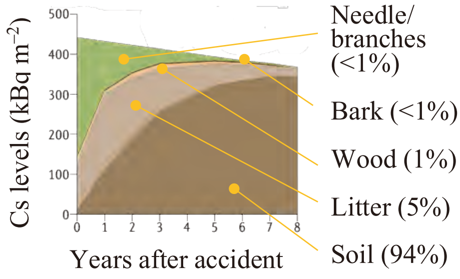
[1]NeuroSyntek, [2]University of Tokyo, [3]Casto Group

Abstract

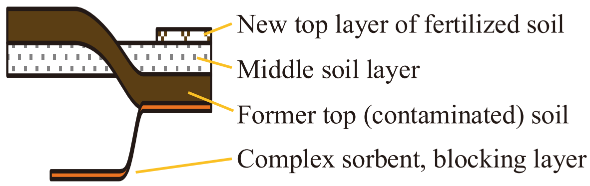
NeuroSyntek [1] and partners are actively involved into the Fukushima region revitalization – leveraging our experiences from Hanford (USA), Chernobyl, Daiichi, and Silicon Valley (USA). There are many challenges to decontaminate/clean-up Fukushima forests. Corresponding opportunities are enormous.



^{137}Cs dynamics in Fukushima forests [2]

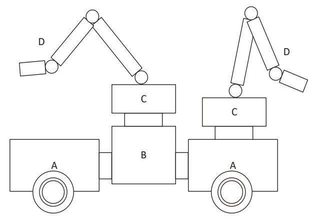
From Problems to Opportunities

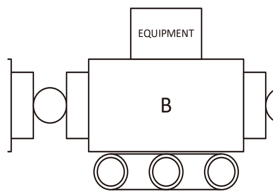
Two thirds of Japan land is forest. More than 40% are planted forests, such as cedar and cypress; most with mountain terrain. With aging population forestry it is hard to sustain with traditional human-centric technologies. Significant cesium contamination from Daiichi to northwest causes very serious concerns about groundwater as cesium penetrates into the soil and downstream.



Forest soil recovery: layers replacement

Solutions & Products: We suggest *Autonomous and Semi-Autonomous Forestry Robotic Systems with core AI concept* – with unique Intellectual Property set (IP). IP examples: Robotic Machine & Method; Robotic Off-Road Vehicle; System & Method of Forest Logging; Cesium sorbent & placement method. We also envision Training and Gaming capabilities engaging Fukushima RTF (Robotics Test Field), development of Game and Media Ecosystem for Fukushima Reconstruction, including Forest Clean-up Reality Game. **Software Segment & Robotic Contest:** A competitive environment is proposed to start development of a remote-controlled autonomous technology for equipment used in agriculture and forestry. First stage deals with pure software developments of most critical elements of robotic technologies. At the second stage the developed technologies are deployed in-the-field on dedicated test robotic vehicles.





IP examples. Modular robotic off-road and forest vehicles

Acknowledgments: Hajimu Yamana, Yoshimi Ota, Paul Dickman, Haruaki Matsuura, Reconstruction Agency Team

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J01

JAEA's Efforts for Human Resource Development and Regional Commitment in Fukushima

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[1]Japan Atomic Energy Agency

Abstract

Sector of Fukushima Research and Development, Japan Atomic Energy Agency (JAEA) conducts human resource development and regional contribution activities along with research and development.

1. Introduction

Research and development for decommissioning of Fukushima Daiichi NPS and environmental restoration of Fukushima Pref. are the main missions that JAEA is working on. Considering that these activities will be conducted over the medium to long term, the following points are important: a) securing and fostering human resources, b) participation of local companies, and c) technology transfer to the local businesses and field implementation.

2. Efforts for Human Resource Development

JAEA is working to develop human resources through cooperative agreements with KOSEN and universities, and through the implementation of the Nuclear Energy Science & Technology and Human Resource Development Project (Project name: Mission H), which includes giving lectures to students, accepting research students, and holding the Creating Robot Contest for Decommissioning. Through these efforts, the promotion for understanding of decommissioning and environmental restoration and the development for human resources who will lead the future together with some educational institutions are being implemented in JAEA.

3. Technology development with local industries in Fukushima

In collaboration with local companies in Fukushima, we are advancing analytical equipment and monitoring technologies for on-site implementation. By transferring these technologies to the local companies for practical use, we will contribute to the commercialization of the developed technologies and the industrial recovery of Fukushima.

J02 Summary of JAEA's efforts for fuel debris retrieval

Shin-ichi Koyama, Naoya Kaji, Satomi Kakutani
Japan Atomic Energy Agency

Abstract

Japan Atomic Energy Agency (JAEA) will build a knowledge base that enables safety and risk assessment through basic research as a sherpa for TEPCO HD and NDF for the retrieval of fuel debris from TEPCO's Fukushima Daiichi Nuclear Power Station (1F).

Outline

The efforts implemented by JAEA are classified as follows. Research and development by utilizing new constructed facilities for fuel debris retrieval work, such as demonstration of fuel debris retrieval device and dose distribution evaluation for reduction of radiation exposure during the work (①, ②, ③, ④). Facility maintenance and technological development related to analysis of fuel debris (④, ⑤, ⑥). Basic research related to core status estimation and safety / risk assessment inside the containment vessel / reactor building (③, ⑦, ⑧, ⑨).

① **Support for IRID full-scale mockup test at Naraha Remote Technology Development Center.**

Mockup test and operation training for fuel debris retrieval from Unit 2.

② **Visualization of environment, dose and radiation source distribution for reducing exposure.**

Development of a system that can study the reduction of radiation exposure of workers under high dose without exposure for the preparation work for seeking an access route in R / B.

③ **Database / sharing of radioactive source / dose distribution evaluation and core status in PCV.**

Comprehensively evaluate the radiation source and dose rate distribution in the PCV and inside the building, and provide and share information to related organizations. Provide and share the knowledge of the inside of the core and debris characteristics as a database "debris Wiki".

④ **Development of analysis and research facilities for radioactive waste and fuel debris at the Okuma Analysis and Research Center.**

Under construction of a radioactive material analysis / research facility for analysis and research to understand the properties of radioactive waste and fuel debris on the adjacent to 1F.

⑤ **Development of remote, quick and simple analysis technology for fuel debris.**

⑥ **Development of α-ray imaging detector for evaluation of exposure.**

Development of a portable laser-induced emission analysis (LIBS) device with high radiation resistance that can be applied to the 1F and succeeded in analyzing the sample collected from Unit 2 on the 1F site. Development an α-ray imaging detector to quickly evaluate the concentration and particle size of α-ray dust.

⑦ **Verification of fuel debris characteristics by simulated experiments and comprehensive evaluation utilizing 1F site information.**

Evaluation of metal debris generation behavior by manufacturing large scale equipment and testing to reproduce accident.

⑧ **Elucidation of basic fuel debris characterization and aging degradation.**

⑨ **Confirmation of cooling status in the reactor core by thermal analysis of fuel debris.**

Collection of knowledge necessary for long-term storage safety evaluation. Evaluation of the temperature distribution of fuel debris and its surroundings when water is injected and stopped for fuel debris retrieval.

J03 R&Ds on Waste Management Technologies for Fukushima Daiichi NPS with National and International Collaboration

Yoshikazu KOMA, Naoya KAJI [1]

[1] Japan Atomic Energy Agency (JAEA)

Abstract

Japan Atomic Energy Agency has been conducting research and development on waste management technologies for Fukushima Daiichi Nuclear Power Station in the field of characterization/analysis, safe storage and processing/conditioning with help of national and international collaboration.

1. Introduction

Decommissioning and its waste management requires R&D for characterization, safe storage, processing and disposal in parallel to provide promising technical options for future execution. In order to investigate such technologies for the unconventionally generated waste, knowledge and experiences should be assembled from national and international organizations and utilized. JAEA is conducting R&Ds with wide collaboration.

2. Characterization/Analysis

Radioactive nuclides of long half life are important target for analysis due to its effect to disposal safety. The analysis procedure is complex and time consuming, thus, JAEA collaborate with laboratories of private companies to provide data for related decommissioning activities. Furthermore, Radioactive Material Analysis and Research Facility Laboratory – 1 has been completed at the Fukushima Daiichi site to start analysis in the near future. Essential methodologies including analysis planning, which is based on Data Quality Objectives process and Bayesian statistics [1], as well as inventory estimation [2] are the result of collaboration. International knowledge is helpful to accelerate R&Ds above and JAEA participated in NEA expert group [3].

3. Safe Storage and Processing/Conditioning

The secondary waste from water decontamination is designated as the target, which hazard should be decreased. To investigate safe storage of such a waste including adsorbent with highly concentrated radioactive cesium, collaboration with the equipment provider is indispensable and the result is already utilized [4]. For conditioning such waste, solidification with cement and alkaline activated material (AAM, or also known as geopolymer) is investigated in the view of highly radioactive and water-bearing waste [5].

4. Conclusion

Waste management for decommissioning should be efficiently conducted, and JAEA promotes national/international collaboration for establishing potential technologies.

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J04 Experimental research related to formation of debris using large-scaled test facility in CLADS

Yuji NAGAE
Japan Atomic Energy Agency, Collaborative Laboratories for Advanced Decommissioning Science

Abstract
JAEA/CLADS has conducted two types of large-scaled experiments using the LEISAN (Large-scale Equipment for Investigation of Severe Accidents in Nuclear reactors) facility at Tomioka-site . In-situ and post investigation provide all of us updates of understanding for debris formation inside and outside reactor pressure vessel, combined severe accident analysis.

1. Introduction
Investigation of the accident study and recovery actions for TMI-2 and a lot of tests have given us accident progression for Pressurized Water Reactor. There is a lack of knowledge for Boiling Water Reactor (BWR), in which a large amount of metal be used in control blades, control rod drive systems etc. A set of control blades included rod would be possibly degraded by reaction between stainless steel of control blade and at the early stage of accident. Furthermore, these metallic debris would possibly relocate to lower plenum in accident progression, and by reaction between the metallic debris and structural materials or weldments. We would like to introduce outline of two kind of tests;

1. Experiments for metallic debris formation (Control blade degradation behavior)
2. Lower head of RPV failure by interaction between metallic debris and structural materials included weldments (RPV failure behavior)

2. Outline
2.1 Control blade degradation behavior (CLADS-MADE test、Mock-up Assembly Degradation test)
We have done tests under several steam conditions using test pieces with a size of approximately 1.2m×0.1m×0.1m According to experimental result, control blade including B_4C started melting at approximately 1200 degree C, but the melt did not completely melt up to around 1500 degree C. SEM/EDX and Raman spectroscopy have shown that Cr-rich borides, which formed on the surface of control blade's rods by reaction between stainless steel of control blade and B_4C of absorber, would protect evaporation of Boron.

2.2 RPV failure behavior (ELSA test、Experiment on Late In-vessel Severe Accident Phenomena)
RPV Lower region of BWR has complex structure, for instance a lot of pipes for control rod drive system and weldments between several parts. According to sever accident analysis, main three RPV failure modes have been given as follows; Thermal damage, Mechanical damage, Eutectic reaction damage. We would like to explain eutectic reaction, melting and relocation phenomenology. Assuming the conditions for the accident progress of Unit 2, RPV lower head failure tests have been conducted under inert gas. According to in-site observation and post analysis, solid metals of mixture of stainless steel and Zr alloy were melted and molten pool was formed around 1000 degree C. After that, the molten material and weldment was reacted and melted at the temperature of 1100-1250 degree C, and the melts penetrated the pipe of control rod drive system.

J05 Advanced application method of optical fiber radiation sensor towards the decommissioning of Fukushima Daiichi NPS

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Abstract

We have developed the advanced application method of optical fiber towards the radiation distribution measurement inside Fukushima Daiichi Nuclear Power Station (FDNPS). Position-sensitive optical fiber radiation sensors based on the photon time-of-flight and wavelength resolving analysis were introduced, and the advantage of both methods for the nuclear decommissioning application was discussed.

1. Introduction

For the decommissioning operation of the Fukushima Daiichi NPS, measurement of radiation distribution is indispensable for the radiation protection of workers. The optical fiber radiation sensor, which has a nature of high sensor flexibility and long-range measurement performance, is one of the promising sensors for radiation distribution measurement for nuclear decommissioning. In this study, new application methods of optical fiber radiation sensor were proposed, which has the characteristic of selective measurement of surface contamination and wide dose rate range applicability.

2. Advanced application of optical fiber radiation sensor

2-1. Time-of-flight method

In this study, quartz and liquid core type optical fibers were newly applied to the conventional time-of-flight method (Fig. 1). The TOF method determines the incident position of radiation and optical fiber from the TOF analysis of radiation-induced photons. Here, since the low-energy gamma rays do not produce Cerenkov photons, quartz and liquid core-type optical fibers are useful for selective measurement of $^{90}Sr/^{90}Y$ distribution on the contaminated surface under the scattered-gamma-ray dominant environment such as inside the FDNPS reactor buildings [1].

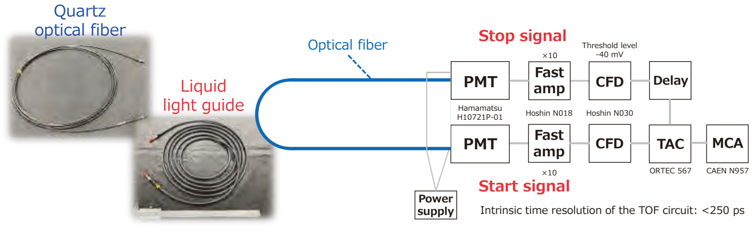

Fig. 1 Schematic of a TOF method.

2-2. Wavelength-resolving method

Since the TOF method requires signal readout from both ends of an optical fiber, there may have some restrictions on installation at FDNPS. Therefore, we have developed a "wavelength-resolving method" based on wavelength analysis of the emitted light from the optical fiber end, which enables the measurement of radiation distribution by the single-end readout. As shown in Fig. 2, this method is promising because it can be applied to a wide dose rate range from several tens of μSv/h to several Sv/h by optimizing the photodetector [2,3].

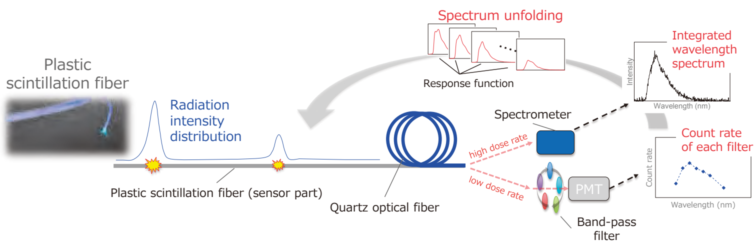

Fig. 2 Schematic of a wavelength-resolving method.

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J06

Overview of Radioactive Material Analysis and Research Facility Laboratory-1

Yuki Sugaya, Hideyuki Akutsu, Hiroshi Shibata, Koji Ichitsubo
Japan Atomic Energy Agency (JAEA)

Abstract

The construction of a new facility named "Laboratory-1" at Radioactive Material Analysis and Research Facility (hereinafter referred to Laboratory-1) was completed in June 2022. Laboratory-1 has a key role in obtaining analytical data needed for treatment and disposal of radioactive wastes generated from Fukushima Daiichi Nuclear Power Station (1F) of TEPCO. This presentation provides an overview of Laboratory-1, including basic designs, activities and operational plans.

1. Features of Laboratory-1

1-1. Object of the analysis

Laboratory-1 will receive low level and medium level radioactive wastes. The surface dose rates of them are up to 1 mSv/h and 1 Sv/h, respectively. The objects of the analysis are mainly rubble which is scattered in 1F site, incineration ash, secondary waste from water processing and ALPS treated water.

1-2. Equipment

Laboratory-1 has steel-shielded hot cells, glove boxes and fume hoods as shown in figure 1, which are used for pretreating samples. The pretreated samples are analyzed by analytical instruments such as Ge semiconductor detector, liquid scintillation counter, ICP-AES and ICP-MS.

1-3. Target analytes

Target analytes of rubble, incineration ash and secondary waste from water processing are radioactive nuclides which are needed for understanding the characteristics and for safe disposal of radioactive wastes. In addition, Laboratory-1 will obtain the analytical data of ALPS treated water as a third party.

Figure 1. Overview of Laboratory-1

2. Operational Plans

Laboratory-1 is currently operated without radioactive substances in preparation for maintenance and analysis towards practical use. In September of this year, we will designate a radiation controlled area at Laboratory-1. After that, Laboratory-1 will begin the analysis with radioactive materials, including validation of analytical methods, research and development.

J07 Development of Exposure Reduction Technologies by Digitalization of Environment and Radioactive Source Distribution -Current Status of Development Project-

Masahiro Suzuki, Takashi Yamaguchi, Masahiko Machida, Kuniaki Kawabata, Rintaro Ito, Koji Okamoto
Japan Atomic Energy Agency

Abstract

In the decommissioning of the Fukushima Daiichi NPS (1F), prior to the full-scale implementation of fuel debris removal operations, it is necessary to improve the on-site environment in order to safely and efficiently construct access routes in the reactor building (R/B). Then, JAEA is now developing a prototype system that estimates radioactive source distributions by using on-site dose rate data and examines countermeasure effects of such as decontamination, shielding with estimated contamination (radiation sources) on the estimated sources in cyberspace. This report presents overview of the development project and its current status.

1. Introduction

In order to improve the environment at the R/B, it is important to identify the distribution of contamination based on the on-site structural data and dose rate data, and to evaluate the worker's exposure by decontaminating and shielding the contaminated radioactive sources.

To evaluate the effect of exposure reduction, therefore, JAEA is developing the prototype system to identify the intensity and location of high-dose radiation sources based on the structural data and spatial dose rate data in the R/B, and to simulate the change of dose rate by decontamination of radiation sources and installation of shields in cyberspace. In order to confirm the effectiveness of the system, analysis and evaluation are carrying out using the data of JMTR, where the radiation sources (source locations) have already been known.

2. Overview of the development project

The development project is planned for two years since Jun. 2021, and is proceeding by three themes shown below.

a. Development of technologies for inverse estimation of radiation source: (a) source inversion estimation, (b) spatial dose rate estimation, and (c) development of interactive visualization technologies.

b. Development of basic technologies for digitizing real environmental data: (a) development of technologies for building an environment of data preparing, storing and utilizing for source inverse estimation, and (b) research and development of basic technologies for measuring real environmental data.

c. Development of technologies for field application: (a) verification tests for prototype, (b) research and development of data update technology based environmental change, and (c) basic study of connectivity and expandability with other systems at 1F.

3. Main developments to date

This section introduces the status of the implementation of the key technology in the project, that is the "Development of Technologies for Inversion Estimation of Radiation Source". The validity assessment of source estimation is carried out by the following steps.

(1) The structural data of the site for the estimation is set (simultaneously, mesh model of the structure is created).

(2) Using the "Observation Point Indication Tool" developed by the project, the number of observation points and locations of the points are calculated for the success of the source inverse estimation, and measurements are carried out based on their data.

(3) Assuming that there is a source at each mesh, the radiation contribution from each mesh to the observation point is calculated by PHITS (Monte Carlo transport calculation code), and from the radiation contribution (matrix) and the measured data, inverse estimation of the source based on LASSO regression is performed.

(4) The validity of the method is examined by comparing the calculated dose rate calculated from the source distributions obtained by inverse estimation using PHITS with the measured data at the arbitrary observation points.

Through successive operations from the inverse estimation to the calculations of air dose rate distributions with using JMTR data, the key technology in the project can be evaluated.

Acknowledgements

This work was partially carried out in a subsidy program of "Project of Decommissioning and Contaminated Water Management", entitled "Development of Technologies for Work Environmental Improvement in Reactor Building (Development of Exposure Reduction Technologies by Digitalization of Environment and Radioactive Source Distribution)".

F01 Overview of current CEA developments for characterization techniques

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Abstract

The prerequisite to any definition of waste management routes is to characterize physically, chemically and radiologically the waste to be packaged. It involves taking samples, which can be complex depending on the operational environment, and carrying out radiochemical and/or physico-chemical analyses, or even specific characterizations if necessary. The strategy adopted aims to establish standard spectra of waste and to carry out non-destructive/destructive nuclear measurements, chemical analyses, etc.

Today the characterization technologies have reached a satisfactory level of maturity (detection/quantification limits, evaluation of uncertainties). However, further developments are needed to meet the following two issues.

Issue 1. Making advanced technologies accessible to the nuclear industrialists: by conveying the characterization devices on site, making them easy to use (adapted software, training, etc.);

Active Neutron Interrogation (INA) is the most effective technology for the quantification of fissile mass in radioactive packages, an inseparable objective of waste management optimization. X-ray tomography is an imaging technique that allows non-destructive characterization of the contents of radioactive waste containers to support knowledge of the internal constitution of these containers.

The CEA TOMIS (in situ Tomography) (2017-2023) and ANAIS (in situ neutron activation interrogation) (2022-2026) projects supported by the French government *"Booster Economy Initiative"*, innovate the nuclear measurement field by developing transportable and adaptable technologies. Thus, these technologies can be relocated as close as possible to the production or interim-storage sites of waste: They authorizes the pooling of costs, increases the non-destructive characterization rates without having to transport wastes over long distances to a dedicated facility.

The project structuration consists in the design and realization of a prototype, as well as radiation protection studies to design the modular parts that can be deployed on site.
Each subset will be developed almost independently, the system being integrated in the final phase of the project on a pilot site before being tested

TOMIS and ANAIS propose similar concepts based on complementary technologies. In the longer term, a mobile measurement platform combining both systems (ANATOMIS) would be possible for a complete characterization of the package contents.

Issue 2. Developing breakthrough characterization solutions for large volume packages (500/870 Liter) to measure and quantify the fissile mass and detect the presence of materials prohibited for disposal;

To date, there is no non-destructive technique to probe the center of large packages (500/870 Liter). To achieve this, the CEA started to invest in a 15 MeV accelerator and, now, proposes a dedicated R&D program to quantify fissile material and detect the presence of materials prohibited for disposal (toxic compounds, pyrophoric materials, etc.). This technology will not only meet the CEA operational needs, but will also offer to all nuclear licensees who will have to manage their historical packages a technological solution that does not exist today.

F02 Cement-based irradiation-resistant materials

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Abstract

The production of hydrogen from the irradiation effect of cementitious materials leads to constraints for the conditioning of radioactive waste with conventional cements (Portland-type). The objective of the MATRICE project, completed in 2021, was to allow for the incorporation of significantly higher activities (typically a factor of 10), while maintaining acceptable radiolysis gas releases with regard to current limitations (interim or final-storage environment) and even decreasing them.

Three experimental tracks were assessed:

- The minimization of the water content in conventional cements (silico-calcium cements) by adding specific compounds (superplasticizers/water reducers for example);
- The use of “alternative” cements, compared to conventional cements (Portland), such as sulpho-aluminous and phospho-magnesian cements; these cements have higher chemical demands for water than conventional cements (thus less "radiolysable" residual water, which can produce hydrogen);
- The implementation of processes, such as vibro-compaction, to facilitate the shaping of low-water concrete.

The major MATRICE project results are:

- First, the decrease of the H_2 rate production by factor of 2 with the using of superplasticizers compounds to reduce the amount of water;
- Second, the decrease of the H_2 production by factor of 3 to 10 by addition of $NaNO_3$ to the cement formulation with a low exothermicity of this formulation allowing its implementation on large size packages;
- Third, the decrease of the H_2 production by factor of 2 with phospho-magnesium cement formulation and satisfactory implementation of these formulations up to the 200 L container-scale.
- Then, the implementation of vibro-compaction process is especially interessing for the conditioning of small porous solid waste hydrated with little water.

From these reference results for the formulation of cement-based irradiation-resistant materials, the aim will be to develop H_2 predictive modeling for cemented-waste packages.

F03

PACH3 (PACkage for H3-waste) project

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Abstract

PACH3 (PACkage for H3-waste) project (2022-2026) aims to develop a low degassing package for tritiated wastes with no available disposal option. It is based on a two-pronged approach:

- The qualification of a composite cement matrix, consisting of a cement mortar (the nature of which is selected so as to minimize chemical interactions with the waste) and a mineral tritium trap type -MnO2/Ag2O;
- The development of a high sealing joint for metallic containers.

The technical challenges are (i) to demonstrate the irreversibility of the chemical trapping HT/T_2, (ii) the time evolution of the trap, (iii) the evolution of the properties of the metallic seal to guarantee a high sealing of the container in time.

The "major" milestones of the PACH3 project are the demonstration of the irreversibility of tritium trapping, the durability studies of the materials and the realization of 50 L package mock-ups.
By developing low-degassing packages, this work will ultimately reduce the storage time of tritiated waste, thus reducing capital expenditure, operating costs, and reducing tritium releases to the environment.

F04

Remediation of contaminated soils in post-nuclear accident situations: the DEMETERRES MOUSSE project

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Abstract

DEMETERRES MOUSSE (2022-25) aims to industrialize the Flotation Foam Technology. Such technology enables to isolate the most contaminated soil particles, allowing waste volume reduction and the reuse of decontaminated soil. The project gathers CEA (French Alternative Energies and Atomic Energy Commission), Orano, Veolia, and IRSN (French public expert in nuclear and radiological risks). DEMETERRES MOUSSE has been selected as a “main impact project” in the “Booster Economy Initiative” of the French government.

1. Presentation of the Flotation Foam Technology

The Flotation Foam Technology is a non-intrusive technology for contaminated soils remediation; it is selective for radionuclides attached to clay particles, mainly cesium. The use of Flotation Foam Technology enables to reuse decontaminated soils while preserving their fertility and agricultural properties. The first tests carried out in Japan in 2017 (Figure 1) with soils demonstrated by the Fukushima nuclear accident attested the proper operation of the technology in real conditions. The performances obtained were a reduction of waste volume by a factor of 3.6 to 7.4 and a decontamination factor of 1.5 to 3.5. Laboratory-scale tests have shown that these values can be significantly improved.

Figure 1: First tests of the Flotation Foam Technology in Okuma in 2017 (©CEA/J.-L. Sida)

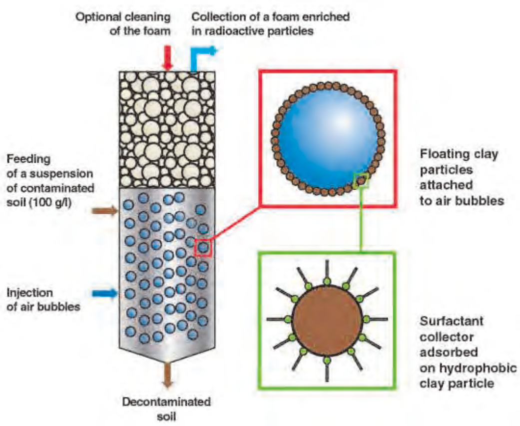

Figure 2: Principles of Flotation Foam Technology

2. Goals of the DEMETERRES MOUSSE project

The industrial optimization of the Flotation Foam Technology remains to be achieved: this is the technological goal of the DEMETERRES MOUSSE project. Indeed, the capture of clay particles by foam bubbles (Figure 2) is a complex phenomenon governed by a succession of processes involving different types of physico-chemical interactions which have to accommodate composition fluctuations related to the soils origin. Thereby, DEMETERRES MOUSSE will implement a new pilot system: improvements will be made to enable optimal screening and dispersion of soils at the process inlet. Moreover, an industrial sizing and design of the flotation column will enable more refined particle selectivity. Finally, the efficiency of the separation between soils and water at the bottom of the flotation column will be improved.

Experimental data from the DEMETERRES MOUSSE project will be used to improve models and Decision Supporting Tools developed for the French authorities to react effectively in the event of a radiological accident.

F05 French facilities in support of Fukushima Daiichi fuel debris cutting and retrieval R&D

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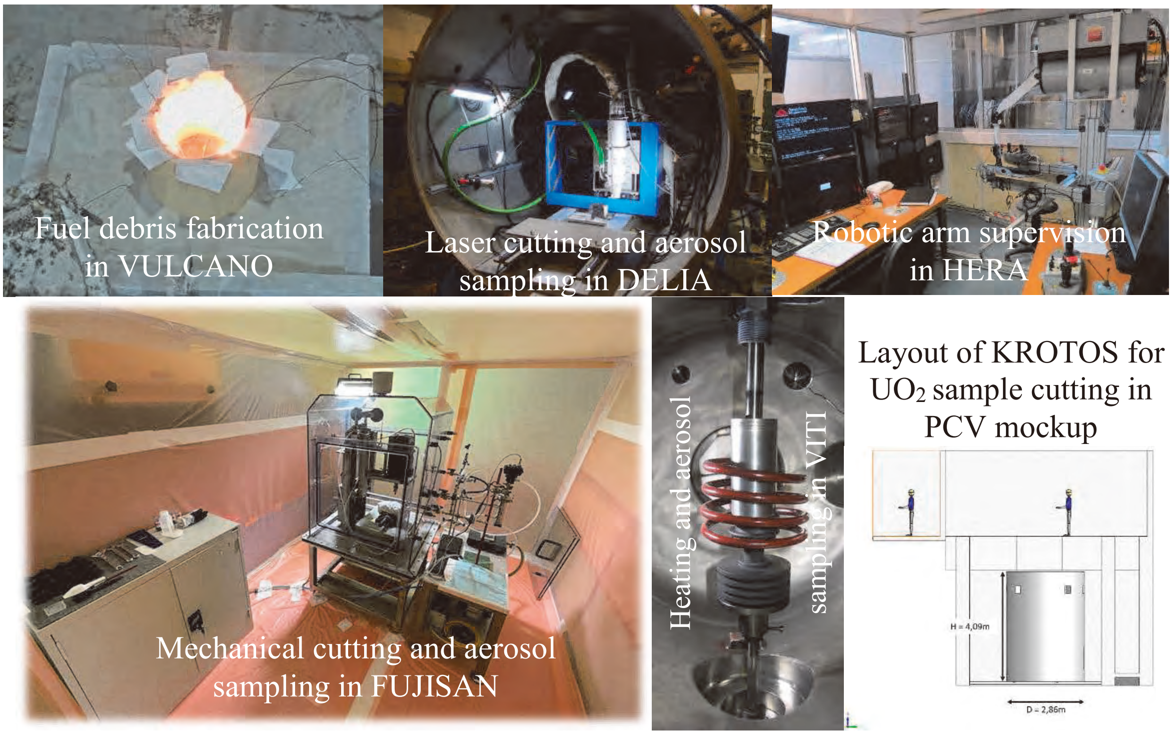
A French consortium is being contracted by Japanese partners since 2014 to study several aspects of Fukushima Daiichi fuel debris cutting and related aerosols. This R&D is based on a unique set of experimental facilities that have been adapted to these issues: fabrication of prototypic samples (VULCANO), laser cutting (DELIA), mechanical cutting (FUJISAN, CAPIMIF), thermal release of aerosols (VITI), dust mitigation (TOSQAN), supervision and testing of robotic dismantling tasks (HERA). KROTOS facility could also be adapted for laser and mechanical cutting in 1:3 scale pedestal geometry.



F06 Mitigation of radioactive aerosols dispersion during laser cutting

Damien Roulet[1], Rémi Delalez[1], Thomas Da Silva[1], Emmanuel Porcheron[2], Thomas Gelain[2], Yohan Leblois[2], Ioana Doyen[3], Christophe Journeau[4], Christophe Suteau[4]

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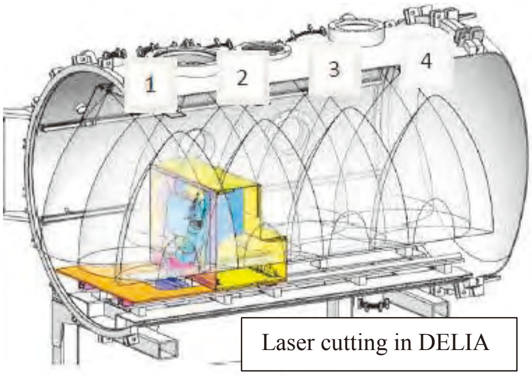
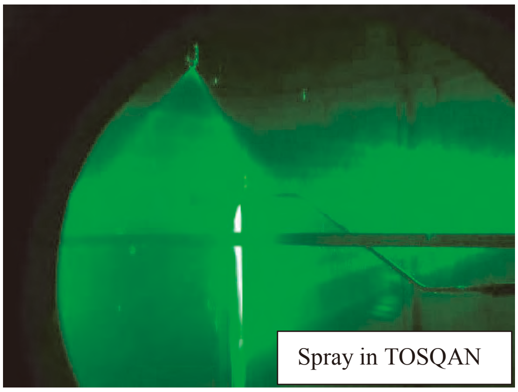
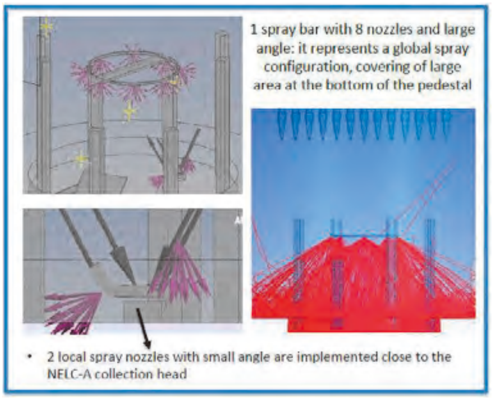
This study is related to the demonstration of feasibility for the use of laser cutting techniques for the fuel debris retrieval in the damaged reactors of Fukushima Daiichi. Besides laser cutting techniques development, ONET Technologies has been leading a team with IRSN and CEA to develop new technologies for dust mitigation with local collection and spray scrubbing systems of aerosols generated during laser cutting and mechanical cutting operations.

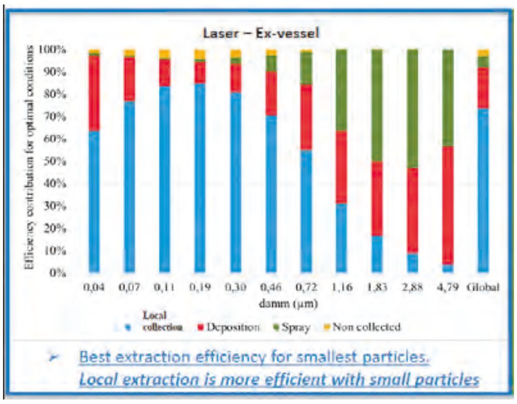


Laser cutting in DELIA


Spray in TOSQAN


Core boring on a block of ex-vessel fuel debris simulant





F07 French Expertise and Technologies applied to Fukushima Daiichi Fuel Debris Sorting

Damien ROULET[1], Yvan LALLOT[1], Thomas GRUNENBERGER[1], Daphné OGAWA[2], Nicolas DEWYSE[2], Pierre STROCK[2], Benoit TASTET[2], Frédérick CARREL[3], Adrien SARI[3], Roberto DE STEFANO[3], Aly ELAYEB[3]

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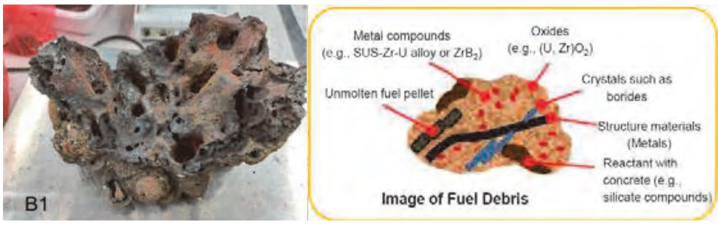
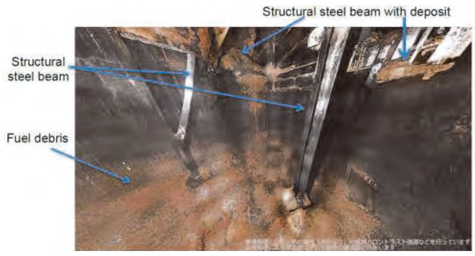
One major challenge of Fukushima Daiichi decommissioning is the safe management of Fuel Debris (FD) from retrieval to storage. Several types of waste will be generated during the large scale retrieval operations and using the best management routes will have a decisive impact on the final cost for storage and disposal. To overcome this challenge in the optimal way, a key step is waste characterization and sorting. Several criteria are necessary, one of them being the measurement of the fissile material contained in retrieved FD, which is deemed extremely complex due to the variability of the FD in its physicochemical and radiological content.

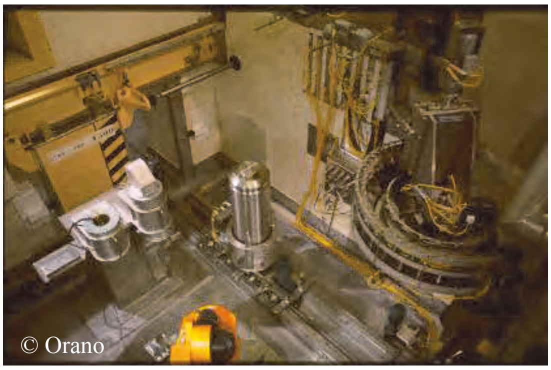
ONET, ORANO and CEA have started undertaking the above challenge, on the basis of existing technologies and French expertise as well as operational feedback throughout the whole chain for the management of nuclear waste from the retrieval to the final disposal.



FD simulant produced by CEA (left) and representation of FD showing complexity (right)



Example of types of waste identified inside the PCV on TEPCO picture



Orano La Hague ACC facility: solid waste canisters (CSD-C) are measured and controlled in a fully remote hot cell, prior to being processed (drying and compaction in this case).

ECC is a fully remote and automated facility in Orano La Hague where hundreds CSD-C are temporally stored in a safe and organized way, until final disposal site is available.

F08 Orano expertise to propose a comprehensive solution for Large-Scale Retrieval and Interim Storage of Fuel Debris at 1F

Daphné Ogawa[1], Nicolas Dewyse[1], Vincent Janin[1], Arnaud Rollet[1], William Rocher[1]
[1] Orano

Abstract
Orano 50-year experience for the design, operation & maintenance, decommissioning & waste management of high active facilities related to nuclear fuel cycle is unequalled worldwide. Orano gathered experts from in engineering of Hot Cells, robotics, decommissioning scenarios, reactor dismantling, nuclear measurement, nuclear packages, and project management to propose a conceptual study for Large-Scale Retrieval and Interim Storage of Fuel Debris for 1F Unit 3.

1. Introduction

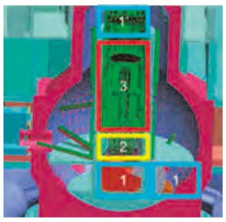
In 2020, TEPCO entrusted Orano to perform a study on a comprehensive solution to perform safe retrieval of Fuel Debris (FD) from 1F Unit 3, from the retrieval of the FD located inside the PCV through to its interim storage on-site. The Fuel Debris Retrieval and Storage scenario was divided into 6 processes:

- Developing access routes into the PCV
- In-PCV processes to retrieve FD
- In-cell processes to treat FD
- On-site transport
- Interim storage on-site
- Dismantling and removal of FD retrieval facility and equipment at project end

2. Outline of the proposed scenario

2.1 Retrieval sequence

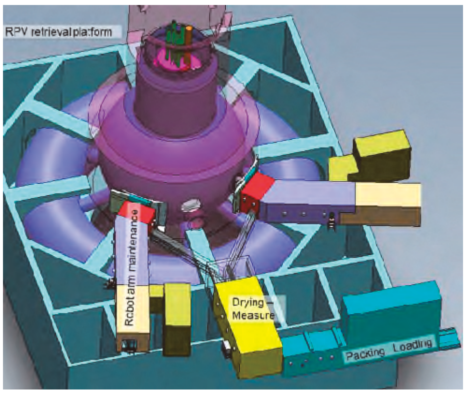
(1) FD inside and outside pedestal are retrieved through a side-access while (1') top-access is prepared in parallel. (2) Removal of other internal equipment accessible from side-access and at last (3) FD inside RPV upper part is removed by the top-access retrieval.



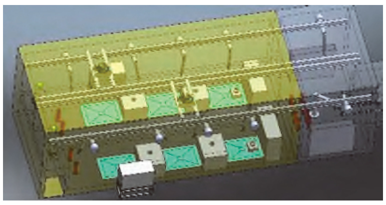
Retrieval sequence numbered

2.2 Main technological solutions proposed

SIDE ACCESS



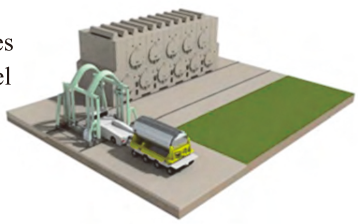
FD treatment cells are designed based on Orano high activity engineering principles to ensure safe operation, maintainability and later dismantling.



Retrieval is done through X6 and X1-B penetrations with robotic means.

STORAGE

NUHOMS® canisters and concrete modules are licensed storage solutions for Spent Fuel and Rad Waste, used to store TMI fuel debris. For 1F, the MATRIX two-stories system is proposed for a reduced footprint.



TOP ACCESS

A new rotating platform is installed on-top of the RPV. Cutting and handling tools are set on the platform.

3. Conclusion

The concept proposed by Orano is: based on high TRL solutions, cost competitive, modular and designed to be operated and maintained remotely.

The study was performed under given conditions (ex. in-air) with little knowledge of the actual plant conditions. Further studies are required to precise the best scenario and associated solutions.

F09

Innovative solutions for Fuel Debris Retrieval (FDR)
New Lateral Opening for RPV access and cells conceptual design for Large Scale FDR through PCV side access

Laurent David[1], Daphné Ogawa[1], Vincent Bessiron[2] and Pascal De Vito[2]

[1]Orano, [2]Framatome

Abstract

On the one hand, a lateral opening toward the Reactor Pressure Vessel is made possible by integrating water jet cutting and demolition tools combined with a ferro-concrete debris suction system. On the other hand, cells conceptual design for Large Scale FDR through PCV side access is completed. Both solutions are designed for 1F needs based on French technology.

1. New Lateral Opening for RPV access

A new access path from the outside of the reactor building toward the inside of the Reactor Pressure Vessel is performed by a tool head that combines several technologies such as Abrasive Water Jet Cutting (AWJC), Water Jet Demolition (WJD) and a ferro concrete debris suction system (see figure 1). The access allows internal investigation and possibly sampling of Fuel Debris laying inside the RPV.

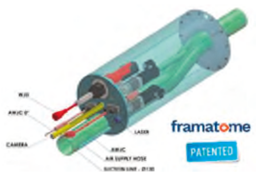


Fig 1. Framatome combined tool

The combination of the three technologies in one tool allows to avoid changing the tool each time the material to be cut changes (from concrete to steel or from steel to concrete), to minimize the quantity of abrasive to inject (only water is used for concrete demolition), to recover most contaminated debris, abrasive, and water, for treatment and recycling. This tool head is positioned at the end of a mechanical system to guide the tool inside the reactor building (see figure 2).

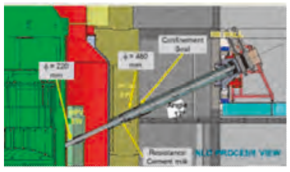


Fig 2. Orano-Framatome facility

Tests were conducted to confirm the feasibility of the system.

By optimizing the functions, a miniaturization of the system could be achieved, making the site implementation easier.

2. Cells conceptual design for Large Scale FDR through PCV side

The conceptual design includes cells (see figure 3) ensuring Full Scale FDR robot arm access to PCV, maintenance of the robot arm, extraction, sorting and packing of and Fuel Debris in cans and canisters and of Low-Level Waste in dedicated containers. Contamination control, decontamination systems, maintenance solutions including emergency situations are anticipated, based on proven solutions with operating experience feedback.

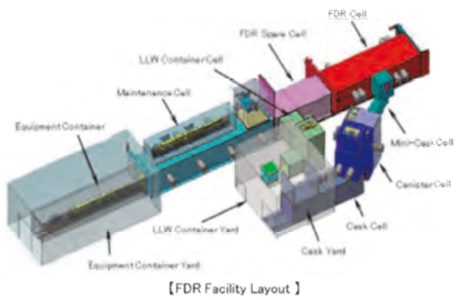


Fig 3. Fuel debris retrieval facilities overview

Conclusion

Orano and Framatome join forces to develop solutions adapted to 1F situation based on their experience in robotics, remote operations, hot cell design and waste management.

Those studies were performed in the frame of IRID "Development of Technology for Gradually Increasing Retrieval Scale of Fuel Debris" and "Development of Technology for Investigation inside Reactor Pressure Vessel (RPV)"

F10

DEM&MELT In-Can Process for Fukushima Daiichi Nuclear Power Station Water Treatment Secondary Waste

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Abstract

The DEM&MELT In-Can vitrification process is a robust, simple and versatile in-situ vitrification process. It is designed for intermediate and high-level waste and is compact enough to be implemented in an existing facility or close to the waste to be treated. It is developed to treat liquid and solid waste, to produce a small amount of secondary waste and to minimize investment and operating costs. The applicability of the DEM&MELT technology to water treatment secondary waste arisen at Fukushima Daiichi Nuclear Power Station (1F waste) was studied and tested. High rates of waste incorporation and good performances of the wasteform especially with regards to its durability were verified.

1. Introduction

The DEM&MELT In-Can vitrification technology is considered as a mature solution that could be applied for the treatment and conditioning of waste resulting from the D&D and remediation works in Fukushima Daiichi Nuclear Power Station. As a result, the DEM&MELT technology was tested From FY2018 to FY2021 on a wide range of 1F waste in the frame of the R&D funded by the Japanese Ministry of Economy, Trade and Industry as The Subsidy Program “Project of Decommissioning and Contaminated Water Management”.

2. Evaluation of the In-Can vitrification technology for 1F waste – Main results

Orano and CEA contributed to the successful achievement of the following results to demonstrate the applicability of the thermal treatment through DEM&MELT:

- High waste loadings were achieved (up to 80 wt.% oxide) for a wide range of waste (zeolites, CST and ALPS slurries), allowing significant volume reduction.
- Very low radionuclides volatilities were obtained during the treatment.
- Demonstrations of the capabilities to treat 1F waste in mixture or one-by-one were performed.
- Technological and process demonstrations were performed at pilot scale and at full-scale, with challenging waste (Figure 1).
- The wasteforms produced allow long-term immobilization of the radionuclides. The material performances such as chemical durability have been assessed and are very promising.
- Two compact facilities were conceptually designed to process all the waste from ILW to HLW. A safety study revealed the prospect that the DEM&MELT process can meet the safety requirements of the Japanese regulations.

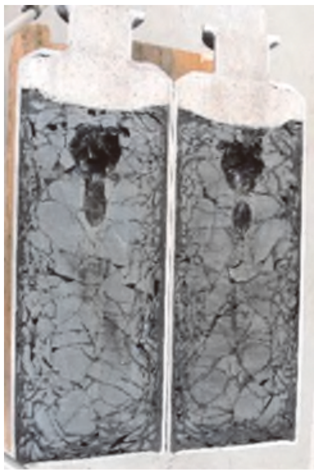

Figure 1: Vitrified ALPS slurries (270 kg at 42% WLox)

3. Conclusion

The tests carried out confirmed the process ability to treat a wide variety of waste with easy and reliable operation. The technology makes it possible to achieve high rates of waste incorporation into the matrix while guaranteeing good performances of the wasteform — especially with regards to its durability, in order to limit the release of radionuclides into the biosphere — and ensures very low volatility of radionuclides during thermal treatment.

F11 Chameleon and Anemone tools - Innovative gripping technologies capable of sampling and recovering fuel debris

D. OGAWA, K. LE FLANCHEC, V. TOULEMONDE, A. COUDRAY

ORANO

Abstract

Orano develops various tools for remote investigation and sampling, designed to be remotely operated and maintained in harsh environment. Since 2018, Orano is manufacturing two types of end-tool dedicated to the sampling and recovery of fuel debris from Fukushima-Daiichi unit 2 RPV.

- The Chameleon Tool, called as Flexible Gripper Type Sand-like debris sampling Tool, at the industrial prototype stage is ready to undergo combination tests with the robot arm.
- The Anemone Tool, which detail design is conducted during JFY2022 with the aim to manufacture the final tool by 2024.

The end-tool are bio-inspired, meaning they are inspired by nature/animal life.

1. Chameleon Sampling Tool

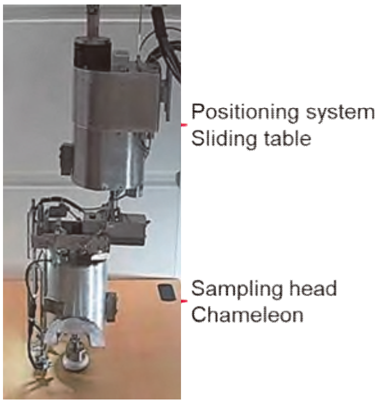

Figure 2 - Chameleon Sampling Tool parts

Figure 1 - Chameleon head

The Chameleon Sampling Tool (or CST) has been developed to catch small pieces of Fuel Debris, up to 100g. The system is based on a silicone membrane which enwrap the target piece by vacuum suction. The name "Chameleon" was given as the grasping principle was inspired from the capability of the animal's tongue to catch its target. Elementary tests were performed on active corium simulants in 2019, and since that Orano has been developing an industrial prototype. The CST consists of a sampling head (see figure 1) and a sliding table (see figure 2) allowing accurate positioning of the tool from the sampling target, to compensate the robot arm positioning uncertainties. The CST prototype will soon be tested in Europe in inactive conditions in combination with the robot arm, before being transferred to Japan for additional testing.

2. Anemone Retrieval Tool

The second gripping tool is the Anemone Retrieval Tool (ART) and is inspired from the sea anemone behavior. After the investigations conducted inside unit 2 PCV, it appeared that the size of the Fuel Debris pebbles to be collected are bigger than planned. Orano suggested the use of an alternative gripping technology, with higher gripping force. The Anemone solution has been developed for nuclear application in France. It has successfully collected graphite samples from a legacy waste pit in April 2022. For its application at Fukushima, tests were performed in 2021 with samples up to 1kg in air and underwater. The good results confirmed that for bigger Fuel Debris pieces, the Anemone head combined with the sliding table for positioning would be a good solution, and Orano is now conducting the design and manufacturing of the tool for a future use on-site.

Figure 3 - QR code to Anemone video

Figure 4 - Anemone tool retraction principle

Conclusion

Both gripping tools designed by Orano are bio-inspired, thus easy to be deployed on-site in humid or dry condition, even with poor visibility thanks to its dedicated positioning system. The heads are pneumatic-actioned thus less sensitive to radiation than other electro-mechanical tools.

This achievement includes the results of research and development which MHI, as a member of IRID (International Research Institute for Nuclear Decommissioning), has had implemented by the subsidy of the projects of METI (Ministry of Economy, Trade and Industry) on the Decommissioning and Contaminated Water Management.

F12 Tooling Solutions for Nuclear Dismantling Projects

Heiko Herbell[1], Holger Damies[1], Frank Kostroun[1], Christoph Stiepani[1], Norio Terada[2], Vincent Bessiron[2]

[1]Framatome GmbH, [2]Framatome Japan KK

Abstract

Framatome proposes high technology products and services for the D&D of reactors worldwide and is already involved in the Fukushima Dai Ichi project with engineering activities for Fuel Debris Retrieval. Additionally, decommissioning of nuclear sites demands dedicated tooling solutions ensuring reliable dismantling operations fully complying to all safety requirements. Application of specific manipulators or remotely-operated tools is mandatory especially in high-radiation areas. Beside segmentation technologies like mechanical cutting the D&D tooling also covers decontamination or water filtration systems, handling devices, support structures or dedicated manipulators. The poster provides an overview on tooling concepts applicable at the Fukushima plant.

Tooling Solutions related to Plant Decommissioning Strategy

Many nuclear dismantling projects are following a decommissioning strategy from hot to cold or from inside to the outside which enables an early removal of the radiological inventory. The overall dismantling sequence basically consists of:

- Full System Decontamination
- Radiological evaluation of the systems using Sampling and/or inline radiological measurements
- Segmentation and packing of the RPV internals pieces into waste containers
- Segmentation and packing of the RPV pieces into waste containers
- Removal of other large components (e.g., steam generator, pressurizer, …)

Decontamination prior to decommissioning can significantly reduce volume of radioactive waste, minimize personnel exposure to radiation and thereby improve safety. Full System Decontamination (FSD) is an efficient and reliable chemical decontamination process of primary cooling circuit and main auxiliary systems. The technology is easily adaptable to any NPP to

- Minimize activity inventory
- Generate metallically clean surfaces
- Optimize gamma/alpha ratio
- Reduce generation of waste

The process has been extensively tested and features a high degree of material compatibility.

For the segmentation of large components mechanical cutting techniques (e.g. sawing, nibbling) are preferred solutions from different reasons. Besides being reliable and allowing easy maintenance they are generating few aerosols, hydrosols and are therefore not linked to specific heat or secondary waste constraints.

Typically, a mechanical segmentation tool set is containing

- Sawing solutions including disc saws, band saws as well as complete sawing racks and cutting stations
- Handling tools like various grippers including connecting rods, specific couplings and hoisting devices / traverses
- Special tooling as nozzle cutter, drilling or milling devices also for contingency operations
- Auxiliary tooling as water filtration systems, shielding covers, cameras

In terms of efficient dismantling execution, a reliable sequence planning is needed based on the radiological conditions but also the specific state of the nuclear site and the main components. As an example robotics and remotely-operated tools like in-pipe manipulators can be applied for visual inspections of hardly accessible high-dose areas providing valuable information on plant conditions.

There are several such technologies that have been already applied in recent projects with a proven performance and continuous improvement. It can be adapted to be applicable at the Fukushima plant. There are also new solutions under development addressing future D&D needs.

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