ABSTRACTS of the Technical Poster Session

Let's talk about Fukushima Daiichi Decommissioning and the Future

1FDIV

The 4th International Forum on the Decommissioning of the Fukushima Daiichi Nuclear Power Station

Mon, August 5, 2019

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

NDF Nuclear Damage Compensation and Decommissioning Facilitation Corporation(NDF)

ふくしまから
はじめよう。
Future From Fukushima.

Contents

1: Japan Atomic Energy Agency (JAEA)
2: International Research Institute of Nuclear Decommissioning (IRID)

J07 Approach to analysis of hydrogen combustion inside of radioactive waste vessels using CFD softwares based on an open source "OpenFOAM"

Thwe Thwe Aung, Atsuhiko Terada and Ryutaro Hino
Japan Atomic Energy Agency

J08 Detection of Alpha Particle Emitters Originating from Nuclear Fuels of the Fukushima Daiichi Nuclear Power Station

Yuki Morishita
Japan Atomic Energy Agency (JAEA)

J09 On the hydrogen production of geopolymers under gamma irradiation

Vincent Cantarel, Makoto Arisaka and Isao Yamagishi
Japan Atomic Energy Agency

J10 Development of an Adaptive Control System for Laser Cutting of Fuel Debris and Structural Components for the Decommissioning of the Fukushima Daiichi Nuclear Power Plant

Koichi Saruta, Naomitsu Kamei, Yuji Sato and Toshiharu Muramatsu
Japan Atomic Energy Agency (JAEA)

J11 Prediction of the dose rate of various fuel debris to be sampled from Fukushima Daiichi Nuclear Power Station

Kenichi Terashima[1,2] and Keisuke Okumura[1,2]
1: International Research Institute for Nuclear Decommissioning (IRID),
2: Japan Atomic Energy Agency (JAEA)

J12 Innovative laser remote analysis for surveillance of nuclear debris in decommissioning of Fukushima Daiichi Nuclear Power Station

Ikuo Wakaida[1], Hironori Ohba[1,2], Masabumi Miyabe[1], Katsuaki Akaoka[1], Masaki Ohba[1], Koji Tamura[1,2], Morihisa Saeki[1,2], Yuji Ikeda[3] and Takunori Taira[4]
1: Japan Atomic Energy Agency (JAEA),
2: National Institutes for Quantum and Radiological Science and Technology, (QST), 3: Imagineering Inc., 4: National Institutes of Natural Sciences (NINS)

J13 Research and Development on Solidification Technology for Fukushima Daiichi NPS Accident Waste

Jun Kato[1,2], Taniguchi Takumi[1,2], Takeshi Osugi[1,2], Osamu Nakazawa[1,2], Tomoyuki Sone[1,2] and Ryoichiro Kuroki[1,2]
1: Japan Atomic Energy Agency (JAEA),
2: International Research Institute for Nuclear Decommissioning (IRID)

J14 Sensitivity analysis of core slumping and alternative water injection in Fukushima Daiichi Nuclear Power Plant Unit 3

A01

A Study of Risk Analysis and Evaluation during a Debris Removal Process

Akio Yamamoto[1,2], Takashi Takata[1,3], Satoshi Takeda[1,4], Takashi Ino[5] and Kazuki Hida[5]

[1]Risk Analysis and Evaluation subcommittee, Review Committee on Decommissioning of the Fukushima Daiichi Nuclear Power Station, Atomic Energy Society of Japan, [2]Nagoya Univ., [3]JAEA, [4]Osaka Univ., [5]NDF

Abstract

Risk analysis and evaluation play an important role in decision making of the decommissioning process and plan in the Fukushima Daiichi Nuclear Power Station. In the present poster, activities of the risk analysis and evaluation subcommittee are introduced so as to support an establishment of the risk analysis methodology.

1. Objectives

A radiological risk is one of the most important issues and it will vary during the decommissioning process. However, the risk analysis methodology has not been well considered during the decommissioning process because the plant is already damaged and thus it includes plenty of uncertainty. Furthermore, each sub-process is not designed in detail. Therefore, the risk analysis and evaluation subcommittee support to develop the methodology by reviewing each fundamental technique. A near-term goal of the development is to find important scenarios and to investigate a relative comparison among different processes of debris removing.

2. Review of Fundamental Techniques

2-1. Scenario assessment

Taking into account a complete uncertainty in the risk, an exhaustive scenario assessment, which includes an initiating event and its progress, should be carried out. Consequently, a combination of top-down and bottom-up approaches are considered. As concerns the top-down approach, a master logic diagram (MLD) is applied. A hazard and operability study (HAZOP) and a failure mode and effects analysis (FMEA) are chosen as a bottom-up approach (Figure1).

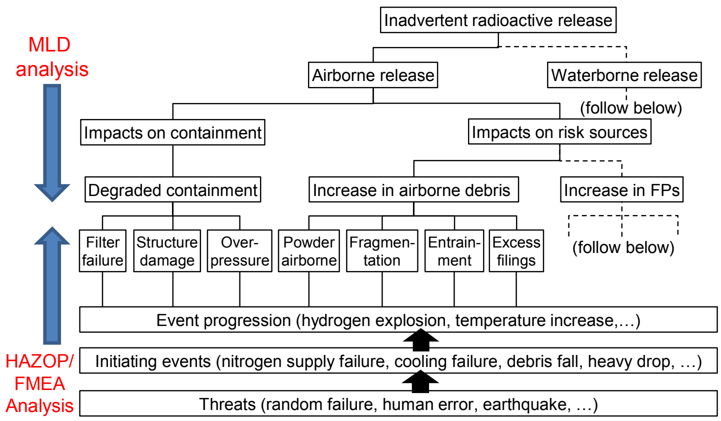


Figure 1. Scenario assessment with hierarchical tree

2-2. Expert elicitation and quantification of radioactive release

An expert elicitation will be efficient to investigate an occurrence probability and magnitude of the consequence of physical phenomena that include plenty of uncertainty. However, a variation becomes large in general. An effective manner of the elicitation and implementation of some fundamental physical parameter into the elicitation are discussed. A quantification of radioactive release is one of the key challenges in the risk analysis. As a first step investigation, a simplified method with the multiplication of factors, which is applied to an assessment of nuclear fuel cycle facility accident [1], is applied and the applicability of each factor to the decommissioning process is discussed.

References

[1] U.S. NRC, NUREG/CR-6410, 1998.

A02 Research and Development of the Project of Decommissioning and Contaminated Water Management and Connection to Preliminary Engineering

Masaki Kawai[1], Fumiya Sato[1]

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

Abstract

The Ministry of Economy, Trade and Industry established a fund since fiscal 2013 and has implemented the “Project of Decommissioning and Contaminated Water Management" as a grant-funded program by public bid, which supports research and developments with high technical difficulty. We will show the relationship between the various subsidized projects of this program and the expected contributions to the decommissioning of Fukushima Daiichi NPP.

1. Introduction

To safely and steadily carry out the decommissioning of the Fukushima Daiichi NPP, it is important to gather together wisdom in Japan and overseas, and conduct R&Ds; therefore, the Ministry of Economy, Trade and Industry established a fund since fiscal 2013 and has implemented the "Project of Decommissioning and Contaminated Water Management" as a grant-funded program by public bid to support R&Ds with high technical difficulty. Various R&Ds in the program have been managed by the Management Office for the Project of Decommissioning and Contaminated Water Management. Mutual coordination among the researches is necessary to apply the results of R&D to the decommissioning of Fukushima Daiichi NPP.

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

In accordance with the R&D plan for decommissioning[1], the subsidized projects are classified into “Internal Investigation”, "Debris Retrieval" and "Processing of Solid Waste". The R&Ds of Debris Retrieval are carried out based on the information inside the reactor vessel obtained by Internal Investigation. In addition, the results of R&D such as Development of Debris Retrieval Method including the concretization of fuel debris removal process are reflected in the preliminary engineering by the operating entities. The researches of Processing of Solid Waste are also studied in parallel with R&Ds for debris retrieval. In this way, the current projects are closely related and aimed at engineering.

References

[1] Ministry of Economy, Trade and Industry, Agency for Natural Resources and Energy, R&D plan for decommissioning in FY 2019, Decommissioning and Contaminated Water Management Team Meeting / Project Management Office Meeting (63rd) (Japanese only)

A03 Value Chain in the Field of Decommissioning and Dismantling of Nuclear Facilities

Mamoru Numata, Sadaaki Abeta, Takeshi Nitta,
Ken Kojima and Erwan Hinault
Kurion Japan K.K.,

Abstract

Veolia Nuclear Solutions (VNS) includes the most comprehensive range of technologies and services for facility restoration, decommissioning of plants, and the treatment of radioactive waste, all nurtured by our nuclear experts and backed by thousands of Veolia staff worldwide.

1. An Integrated offer within Veolia for the Nuclear Industry in Facility Clean-up

1.1. Characterization

VNS operates radiological laboratories in the U.S. and France. We have all the resources required to perform onsite operations for clients based on recognized expertise in characterization and measurements: mobile pressure and characterization units, mobile measuring equipment, sampling techniques, mapping and modeling tools.

1.2. Robotics

VNS is developing new remote handling methods in order to ensure the safety of its operators. These methods can be used to decontaminate sensitive sites, which cannot be accessed by operators due to high levels of radioactivity.

1.3. Separation and waste treatment

VNS deploys effluent treatment technologies as modular skid-based systems that allow for quick deployment, plug & play functionality, off-site commissioning and deployment of facility-based technologies to the most dynamic nuclear environments.

1.4. Stabilization

VNS offers an economical, practical way to protect the environment from the threats of nuclear and hazardous waste. GeoMelt® is ideal for solid waste and debris, while the Modular Vitrification System (MVS®) is designed for liquid waste.

A04

Remote Access Technologies for Decommissioning of Nuclear Facilities

Mamoru Numata[1], Sadaaki Abeta[1], Takeshi Nitta[1], Ken Kojima[1], Erwan Hinault[1], Mark Sharpe[2], David Loughborough[2], Scott Martin[3], Matt Cole[3], and Marc Rood[3]
[1]Kurion Japan K.K., [2]Veolia Nuclear Solutions (UK) Ltd., [3]Veolia Nuclear Solutions Inc.

Abstract

Veolia Nuclear Solutions (VNS) introduced the inspection manipulator, repair manipulator, remote guide pipe and inspection boom system for the primary containment vessel of the unit 2 reactor at the Fukushima Daiichi nuclear power plant site.

1. Introduction

VNS develops technical solutions using remote technologies from concept, to final design, fabrication, testing, and field deployment. To date, the VNS team has leveraged its engineering capabilities to deliver more than 200 innovative remote systems, robotics and tools. This skill set includes the development of robotic manipulators, inspection systems, specialty tools, waste handling systems, mobile waste treatment platforms and modular containment systems. Equipment is commonly custom-designed or modified to meet the needs of our clients for deployment in high-hazard environments where failure can result in significant safety concerns.

2. Reducing worker risk by advancing remote solutions

VNS's remote access technologies and robotics eliminate human risk in the remediation of nuclear and hazardous waste in some of the most dangerous environments.

3. Fit-for-purpose engineering solutions and an agile, creative approach

Each radioactive or hazardous environment has a unique set of challenges related to cleanup, dismantlement or decommissioning. While remote access to such unique environments requires distinct solutions, fundamental knowledge and problem-solving are often applicable from one project to another. That is why it's critical to find a team that has solved similar problems before.

4. Application of remote handling technologies to the 1F site of TEPCO

VNS is working in close collaboration with IHI/Toshiba and MHI on the design and construction of a unique, remotely controlled, robotic systems to inspect and repair the suppression chamber and vent tubes using grout (IHI/Toshiba) and also to investigate the nature and location of nuclear fuel debris in the highly contaminated environment of the No. 2 of the Fukushima Daiichi(MHI).

Investigation Boom System

Dexter Manipulator

Inspection Manipulator

Repair Manipulator

Guide Pipe System

A05 GeoMelt® Application to 1F Waste

Sadaaki Abeta[1], Erwan Hinault[1], Jeremy Whitcomb[2], Brett Campbell[3], and Kevin Funicane[3]
[1]Kurion Japan K.K., [2]Veolia Nuclear Solutions Inc., [3]VNS Federal Services Inc.,

Abstract

Veolia Nuclear Solutions' GeoMelt® technologies are a group of vitrification processes that are configured in a variety of ways to meet a wide range of radioactive and hazardous waste treatment and remediation. GeoMelt® vitrification decompose organic wastes and immobilizes radionuclides and heavy metals in a stable glass. Kurion Japan K.K. is investigating the applicability of GeoMelt® to treat and stabilize radioactive waste generated by the treatment of contaminated water at the Fukushima Daiichi nuclear power plants.

1. Introduction

GeoMelt® creates a stable glass that is typically 10 times stronger than concrete, and more durable than granite or marble. Its leach-resistance is among the highest of all materials in the world. Through vitrification technologies, we offer an economical and practical way to protect the environment from the threats of nuclear and hazardous waste. GeoMelt® is ideal for solid waste and rubble.

2. Industry Challenges and Features of GeoMelt®

Despite glass's superiority, conventional vitrification technologies have been too complicated and far too expensive to deploy widely. Such technologies also require a homogeneous waste feed and are less flexible than GeoMelt®. Unlike most other vitrification technologies, GeoMelt® is cost effective and can process various types of wastes simultaneously.

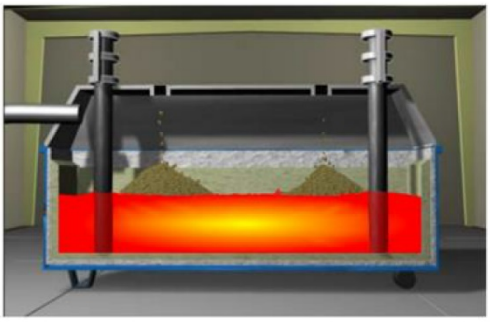
3. Development of GeoMelt®

GeoMelt® have been treating nuclear and hazardous waste since the 1990s, producing over 26,000 metric tons of glass for disposal in the U.S., UK, Australia, Japan and other countries. Initially developed by Pacific Northwest National Laboratory in the U.S., GeoMelt® has been used successfully around the world for the U.S. Department of Energy (DOE) at Hanford and at Sellafield in the UK for example.

4. Efforts for Wastes Arising from 1F

In Fiscal Year 2017 and Fiscal Year 2018, Kurion Japan and Veolia Nuclear Solutions successfully completed two phases of testing which demonstrated the suitability of its GeoMelt® In-Container Vitrification (ICV)™ process for the treatment and stabilization of Fukushima Daiichi contaminated water treatment secondary waste (i.e. zeolite absorbents, titanate absorbents, multi-nuclide removal equipment [ALPS] slurries, water decontamination device [AREVA] sludge). This testing was conducted under the research program entrusted to IRID (International Research Institute for Nuclear Decommissioning) as a METI (Ministry of Economy, Trade and Industry) subsidy program.

This year, a holistic evaluation on the implementation of GeoMelt® In-Container Vitrification ICV™ is being carried out under subsidy program of METI.



In-Container Vitrification GeoMelt® ICV™

Sellafield Central Laboratory Facility

A06 MITSUFUJI Wearable HRV Monitoring System with Garment Type Sensor Predicts Workers Health Problems in Real Time

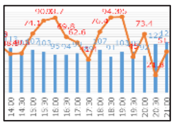
Hiromichi Osoegawa, Executive Officer Medical Project Div., MITSUFUJI Corporation

To adopt technologies for works in F1, with being realized values in Fukushima.


Over 100 people in Kawamata town are wearing hamon shirts every day to monitor ones's health by the smart phoe application.

- Trial: Feb,2019 ~ Feb, 2021
- Users max : 500 people (age 0 ~)
- Sampling data: HRV, stress, condition, sleep, activity, position, etc.

Android App screenshot



Daily reports

hamon Solutions

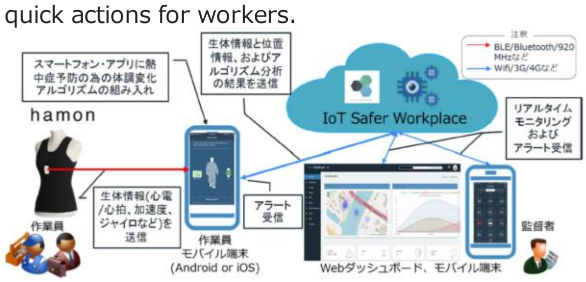
Silver-metalized conductive fibers (Agposs®)

[Transmitter]
- Size: 57mm×33mm×12mm (excluding male connectors)
- Bluetooth: BLE 4.0
- Available for 24 hours after 3 hours of charge
- Sampling rate : 250 Hz

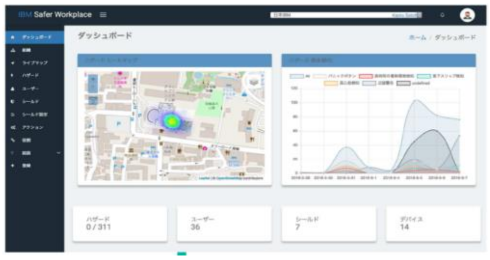
[Wearable Sensor]
- Available in LL,L,M,S size
- With Bra-tops for women
- Wash ability: up to 100～150 times

Wearing Garment type Sensor by workers and monitoring them by managers remotely

- MITSUFUJI provides Worker Insights solution by IBM to workers who are working at F1, with wearing Garment type Sensor and detecting health conditions throughout HRV and GPS data.
- Managers at site can monitor workers' condition real-time and remotely to reduce risks and take quick actions for workers.

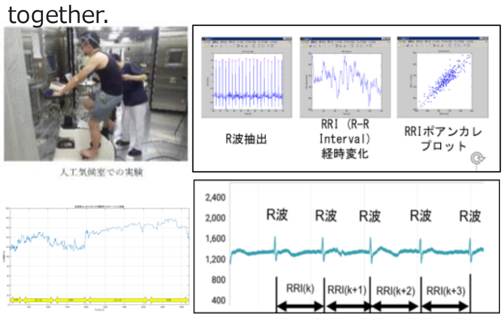


Solution image with Worker Insights Solution



Screen samples to detect works problems

- MITSUFUJI has been developing various algorisms to detect various health signals and predict heat-related health condition risk with University of Occupational and Environmental Health for a couple of years.
- MITSUFUJI has a partner which has real sites to make some trials for refining the algorisms together.



Joint study at the University

Joint trial at real construction sites

B01

The Fukushima Consortium of Robotics Research for Decommissioning and Disaster Response

Yoshiro Owadano[1]

[1] Fukushima Consortium of Robotics Research for Decommissioning and Disaster Response

Abstract

In order to promote the market entry of manufacturing companies in Fukushima Prefecture to the decommissioning, decontamination and disaster response robotics fields, Fukushima Consortium of Robotics Research for Decommissioning and Disaster Response （hereinafter called “the Consortium”） is working to establish a network among manufacturing industries in Fukushima Prefecture and related organizations, through seminars and robots demonstration events.

1. Overview of the Consortium

The Consortium was established to support the manufacturing companies in Fukushima Prefecture which intend to participate in the robotics for decommissioning of nuclear power plant, and to expand their business to disaster response robotics by utilizing their technologies developed and accumulated in the decommissioning projects. This Consortium organizes seminars, robots demonstration events and other products for networking and information exchange with related organizations.

2. Main activities of the Consortium

2-1. Robots demonstration events and exhibitions

The Consortium exhibits its products at the exhibition in Tokyo where many related people visit. In addition, the Consortium holds a robots demonstration events in JAEA to appeal new robots by demonstrating in front of potential customers.

Figure 1. Robots demonstration events

2-2. Technical seminars and matching events

The Consortium provides its members opportunities to exchange technical information useful at site of decommissioning and disaster. It also offers meetings for discussing various use of the robots under development.

2-3. Other activities

The Consortium members can be supported by expert coordinators. Market survey and site visit to related organizations, and information gathering from governmental organizations are possible. Besides, latest project information of new governments will be provided timely to our members.

Figure 2. Technology matching events

3. Unique Features of the Consortium

Because the Consortium is managed by Fukushima Technology Centre, a public research and testing organization established by Fukushima local government, the members can exchange information with companies involved in the rolated projects, and also can get the latest information about the decommissioning projects headed by the national government which is normally difficult for local companies to get.

B02

R&D of an Automatic Suppression Robot that can Pinpoint the Source of an Incipient Fire and Immediately Suppress the Fire for Nuclear Facilities

Yukio AKATSU, Shigeo KANKE, Kaoru AKATSU
Tokyo Bosai Setsubi Co., Ltd.

Abstract

It is difficult to fight a fire once a fire has developed and spread. Automatic suppression robots can extinguish the fire with a high degree of accuracy and with a minimum use of foam or another agent. When a fire occurs, people should safely evacuate. These suppression robots can detect a fire at the incipient stage and suppress the fire before manual firefighting is needed. Intelligent AFEX or flying AFEX, automatic suppression robots help you focus on other decommission work.

Automatic Suppression Robots

1. **"Intelligent AFEX (Automatic Fire Extinguisher)"** (Fig. 1) is stand-alone, portable and can be placed in locations with hazards and can be moved as decommissioning progresses. This device detects and extinguishes fires in protected areas (Fig. 2).

Figure 1.
Intelligent AFEX

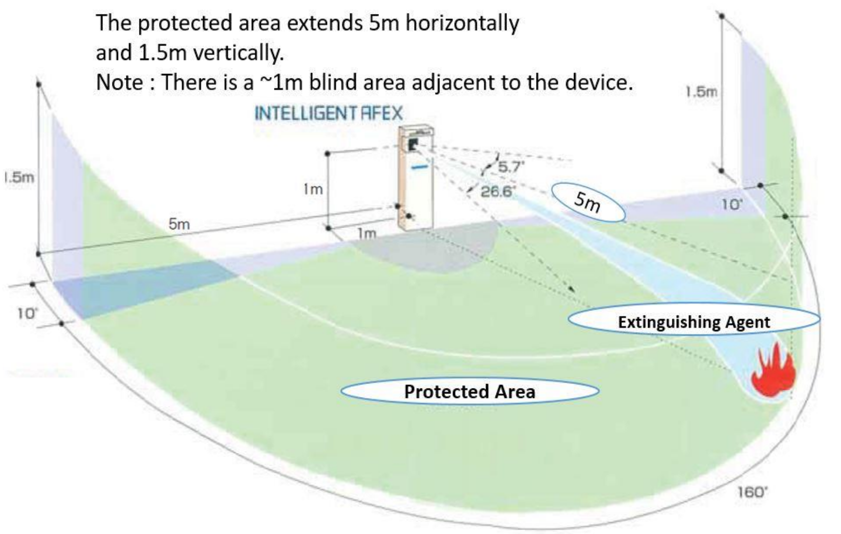


Figure 2.
Intelligent AFEX Protected Area

2. **"Flying AFEX"** (Fig. 3) is movable on the ceiling rail by receiving a fire alarm signal from fire detecters (Fig. 4).

Figure 3. Flying AFEX

Figure 4. Flying AFEX
Extinguishing Fire

B03

Development of underwater robot Named "RADHOTAR" RADiation HOsTile semiAutonomus Robot

Shigekazu Suzuki[1], Shinji Kawatsuma[1], Eiji Aoki[1], Masanori Takahashi[1], Koki Watanabe[2], Shuzo Nakano[3], Kosuke Haga[3], Hideyuki Hnawa[4]

[1]National Institute of Technology, Fukushima College,

[2]Takawa Seimitsu Co. Ltd.,

[3]East Japan Accounting Center Co. Ltd., [4]Ascend Inc.

Abstract

National Institute of Technology, Fukushima College and several loco-enterprises have been developing underwater robot "RADHOTAR". The robot is aimed to be deployed in hostile field such as Fukushima daiichi Nuclear Power Plants. The robot is to be small, radiation tolerant and high operable in order to use inside of Primary Containment Vessels.

1. Background

Fukushima daiichi Nuclear Power Plants, attacked by huge Tsunami following Tohoku Region Pacific Coast Earthquake on March 11th, 2011, fall into melt down and hydrogen explosions, which released radioactive materials inside of primary containment vessels.

In order to aid Decommissioning, the college and the enterprise started to develop under water robot "RADHOTAR".

Fig.1 Fukushima daiichi NPPs after explosion [1]

2. Features and technology under developing

The RADHOTAR robot is planned to be small, radiation tolerant and high operability. [2] A direct drive thruster mechanism newly developed to be small in order to enter the vessels through narrow tube. Semiconductors are reduced and selected as far as possible in order to increase radiation tolerance. [3] Autonomous control using image processing and Structure from Motion technology have been applied in order to increase operability.

Fig.2 RADHOTAR robot (110 mmΦ×600mmL)

3. Acknowledgement

This RADHOTAR is being developed under an assistant project of Fukushima Prefecture.

References

[1] http://photo.tepco.co.jp/library/110316/110316_1f_chijou_2.jpg

[2] Suzuki, et al, "Development of underwater robot " RADHOTAR", (1) Overall plan"

[3] Kawatsuma, et al, "Development of underwater robot "RADHOTAR", (2) Radiation Tolerance Examination"

B04

Disaster Response Robots ‘Giraffe’ Developed by the Members of Fukushima Consortium of Robotics Research for Decommissioning and Disaster Response

Noritaka Baba

Aizuk, Inc.

Abstract

One of the biggest task for disaster response robot is consider about peaceful use on a daily basis. Because of the not use on a daily basis. Sometimes robot cannot action when the disaster occurs. Therefore, we have developed a base robot for running which easily customize aiming at the peaceful use on a daily basis of disaster response robots.

1. Specifications of Disaster Response Robots ‘Giraffe’[Fig1]

Giraffe is a crawler type robot which has two main crawlers and four sub crawlers. Four sub crawlers work independently and it enable to run rough terrain.

Fig1.Giraffe

Size(L*D*H)[mm]	1060*448.7*382.1
Gland clearance[mm]	70
Driving method(main crawler)	Left and right independent crawler drive
Auxiliary run unit(sub crawler)	Four independent sub crawler system
Belt drive system	Chain and sprocket drive
Battery	Li-ion
Operating time	2hour
Maximum inclination angle	45°
Price(Minimum configuration)	¥3,000,000

Fig2.WRS

2. Joint research and development with University of Aizu.

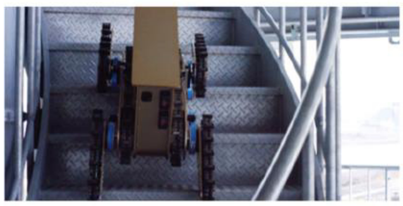
Aizuk conclude cooperation agreement with University of Aizu in 2015. We have been working on standardization of robots compliant with Open-RTM.

In 2018 we won the WRS Tunnel Disaster Response Recovery Challenge category[Fig2]

3. Cooperation with the local manufacturing industry.

Giraffe was designed by Aizuk and manufactured by SAKAE-SEISAKUSYO which in the Minami-Soma.

We want active development of Giraffe by each vendor.



B05

Organization Profile of IRID

Kazufumi Aoki[1]
[1]International Research Institute for Nuclear Decommissioning (IRID),

Abstract

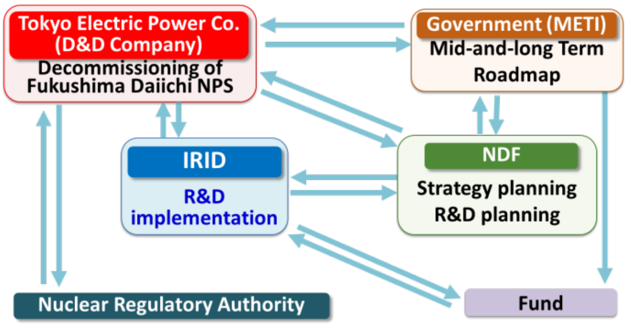
Ever since its establishment in August 2013, the International Research Institute for Nuclear Decommissioning (IRID) has been fully committed to an urgent challenge—research and development (R&D) of technologies required for the decommissioning work of the Fukushima Daiichi Nuclear Power Station (NPS)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

1. Roles of IRID

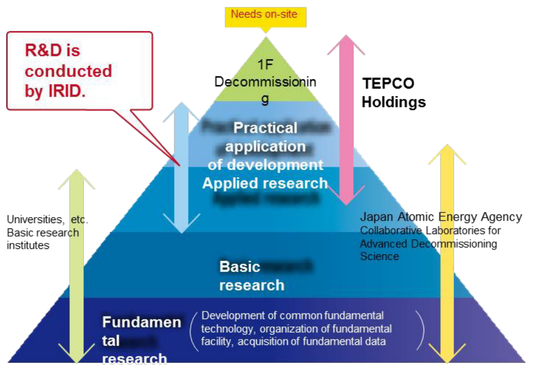
1-1.Scope of Work

- 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.

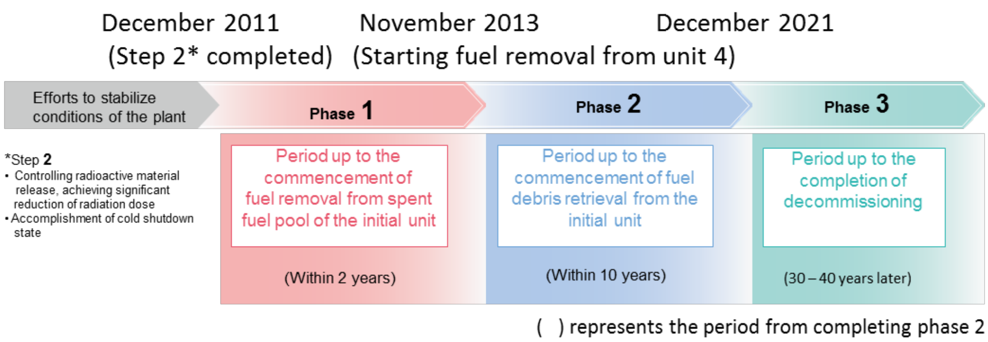


1-3.IRID R&D Scope



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

2. Overview of the Mid-and-Long-Term Roadmap



B06

Overview of IRID R&D Projects

Isao Imamura[1]

[1]International Research Institute for Nuclear Decommissioning (IRID),

Abstract

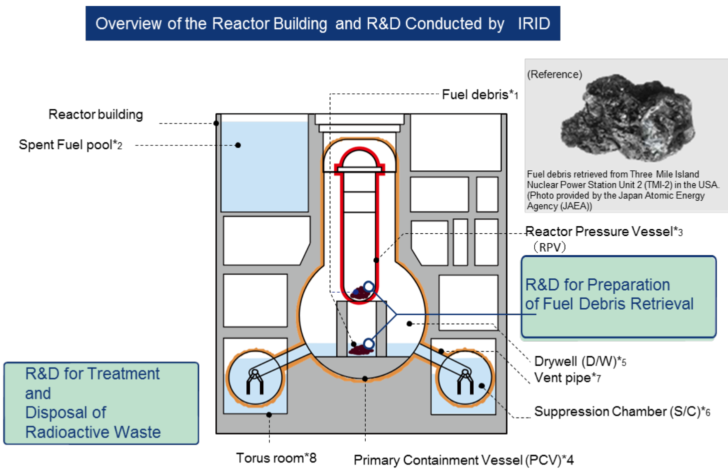
For the decommissioning of the Fukushima Daiichi Nuclear Power Station (NPS), four organizations work together as one team. IRID is a complex entity consisted of eighteen organizations that play a leading role in research and development (R&D) for the decommissioning of the Fukushima Daiichi NPS. IRID 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 a new survey device equipped 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

IRID aims 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.



B07

耐放射線軟質ＰＶＣ材料

Radiation soft PVC materials-resistant

江草　史典、山中　智之

Huminori Egusa and Tomoyuki Yamanaka

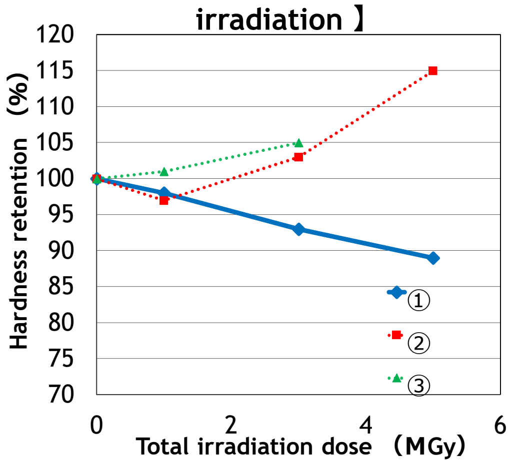
タイガースポリマー株式会社

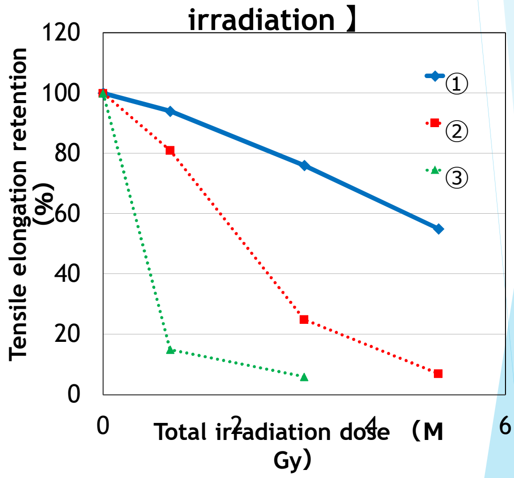
Tigers Polymer Corporation

Abstract

We developed radiation soft PVC materials with the durability of gamma beam exposure dose 5MGy-resistant.
I keep elasticity, flexibility even after irradiation, and the molding at various shapes, diameters such as hose tube seats is possible.
In addition, materials properties of matter are good and are superior in wear resistance.
Because it is PVC materials, I am superior in cost performance.

1. Gamma ray irradiation examination





Sample① : Radiation-Resistant Flexible PVC Material (Developed material)
Sample② : Soft PVC for general hose (reference data)
Sample③ : Olefinic soft material (reference data)

2.Sample

Tube

Hose

Sheet

B08 Radiation Shielding Clothing to Facilitate Decommissioning of Nuclear Reactor

Nobuhiro Shiraishi, Koji Suzuki, Yosuke Tanji
Fukushima Midori Anzen Inc.

Abstract
A new type of radiation shielding clothing designed to facilitate speedy and efficient operation and to improve the comfort and safety of the operators in order to show steady recovery of Fukushima Prefecture from the Great East Japan Earthquake.

1.Introduction

We developed a new type of radiation shielding clothing using a rubber sheet 1/4 the thickness (75% thinner) of the existing type, while providing equal or better shielding performance as the existing radiation shielding clothing for use by the operators engaged in nuclear reactor decommissioning.

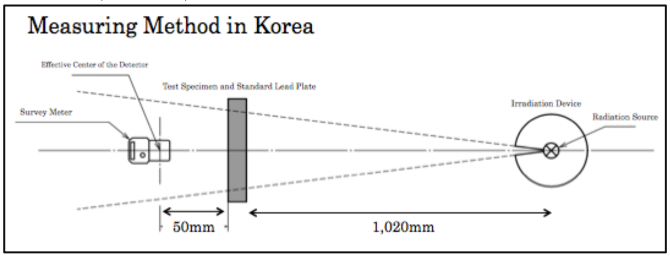
2. Four functions of the radiation shielding clothing

2-1. Thickness of the radiation shielding sheet

The radiation shield sheet of the existing products mainly used in the reactor decommissioning site is made of lead with a thickness of about 4 mm. Considering the adverse effect of lead on the human body, we developed 1 mm thick radiation shield sheet made of rubber with tungsten kneaded into the molecular level as the radiation shielding material, which results in an increase in the amount of tungsten per unit area.

2-2. Shielding effect

Shielding effect of 10% is realized with the thickness of 1 mm.
Shielding performance of the new material was measured and tested by the Korea Atomic Energy Research Institute (KAERI).



Measuring Method inKorea

Radiation shielding clothing

2-3. Feel of wearing and effect

Feel of wearing of the operators putting on this protective clothing and the burden of the radiation shield at the site where the operator must stand up and squat down frequently are greatly improved by the shielding sheet made of thinner material and integrated upper and lower pieces design. When compared with clothing designed with separate upper and lower pieces, uniform radiation exposure can be realized. The lower shielding cover is also integrated that will reduce exposure of the reproductive organs to radiation.

2-4. Multiple usage as the radiation shielding mat

It is considered that the material of the radiation shielding clothing can also be used as protective covers of precision devices and as shielding mats to create a radiation shielded space in the nuclear power plant.

2-5. Technical collaboration

This product is jointly developed in collaboration with a patent registered in Korea and a company in Fukushima, Japan, where demands are promoting the abolishment of nuclear plants.

3. Conclusion

There are 54 nuclear power generation plants on the islands of Japan located in active seismic zones where three tectonic plates meet. So, it will be too optimistic that a radiation hazard similar to the one experienced in Fukushima will never happen. In order to realize recovery of Fukushima Prefecture from the impacts of the Great East Japan Earthquake, improvements in the operating efficiency of the operators engaged in reactor decommissioning or in the transport of radioactive contaminants and commercial production, practical application, and industrialization of technical development for new tools and devices are necessary. We intend to integrate the patented technologies with other necessary technologies and to establish manufacturing and marketing facilities concentrated in the Fukushima area to contribute to the promotion of the regional economy.

B09 Monitoring technology for fuel debris removal process in Fukushima decommissioning

Akira Kobayashi Hideyuki Suzuki
HAMAMATSU PHOTONICS K.K.

Abstract

We have developed radiation tolerant image pick up tube and camera for real time monitoring in PCV and RPV in fuel debris removal process. The objective specifications of radiation tolerance are dose rate of 10 kGy/h and cumulative dose of 2 MGy.

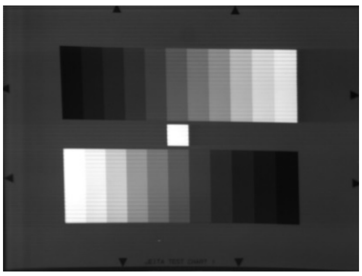
Newly developed image pick up tube and camera

Image pick up tube

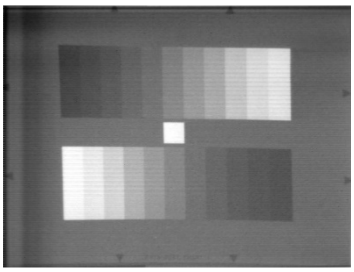
Camera system

Evaluation of radiation tolerance

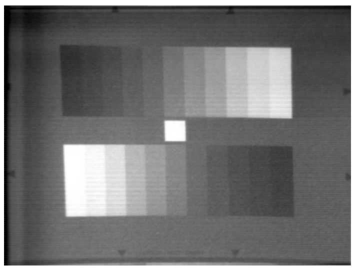
Each step on gray-scale chart can be recognized under 60Co gamma ray exposure of 10 kGy/h even after continuous operation and radiation for 200 h.



Non exposure



Start



200 h

Image samples under 60Co gamma ray exposure of 10 kGy/h

B10 Introduction of Mirion Technologies' Radiation Tolerant Camera
The World's Highest Radiation Tolerant Performance Camera

Yasuo Arai, Mikio Katsura and Toshiki Matsui
CORNES Technologies Limited

Introduction

Mirion Technologies (IST) Ltd. is an industry leader in supply of Radiation Tolerant Cameras and specialist CCTV & imaging systems for the Nuclear Industry. The Mirion IST-Rees brand of radiation tolerant cameras is recognized as the market leader in the Nuclear Industry worldwide. With a comprehensive range of products to suit all applications in every part of the Nuclear fuel cycle, Mirion is able to offer standard or customized solutions to meet customer needs and expectations.

1. HYPERION Solid-State, 1M Gy Radiation Tolerant & Robust Camera

The launch of the Hyperion™ solid-state camera follows many years of extensive research and development activity and considerable investment in new technologies. Featuring all new Radiation Tolerant Zoom lens and optics, with integrated pan and tilt, the Hyperion camera has been independently tested to 1MGy with Co-60 source. Using an FPGA platform based on SoC (System-on-Chip) technology, the Hyperion camera allows upgrades without changing hardware.

- 100Mrad / 1MGy Total Dose (Gamma)
- High performance Solid-State 1 Megapixel Sensor
- Low maintenance and long life
- Superior picture geometry
- High sensitivity and low noise
- New Integrated Pan/Tilt/Zoom Stainless Steel Outstation
- In air or underwater operation
- Compatible with existing Mirion control racks and cables

NEW

2. R93/R94x Series Nuclear Camera

The R93/R94x series are the most widely used high radiation inspection nuclear camera systems in the world, offering extreme radiation tolerance (2MGy) in a compact size with a large range of viewing attachments and options.

- Radiation Tolerance: 2MGy (2 x 10^8 rad), 1kGy/hr dose rate
- Resolution: 600 TV lines per picture height (center zone)
- R93 Mk3: Fixed NB Lens (9mm standard, 50 deg horizontal)
  R941: NB 8mm to 24mm (57.6 deg to 20.8 deg, horizontal)
  R940: NB 22mm to 90mm (23 deg to 6 deg, horizontal)
  R942: NB 12mm to 72mm (40 deg to 7 deg, horizontal)

3. R981 Compact System Camera

The R981 compact camera is a pan-tilt-zoom camera that has very high radiation tolerance (>1MGy). Mirion has combined the camera and pan and tilt into an integral unit and provided three positions for additional lights and microphone. The R981 Compact can be configured into multi camera configurations, or a single camera system can be supplied.

- Radiation Tolerance: 1MGy (10^8 rad), 1kGy/hr doze rate
- Resolution: 550 TV lines, horizontal
- 6:1 - 12mm to 72mm zoom, 40 deg Wide, 7 deg Narrow
  6:1 - 24mm to 144mm zoom, 21 deg Wide, 3.5 deg Narrow
  3:1 - 8mm to 24mm zoom, 58 deg Wide, 21 deg Narrow

4. RC911 DotCam Miniature Color Camera

The DotCam is a miniature color camera designed for use in higher radiation environments where the use of standard CCD based camera is not cost effective.

The compact size of the stainless steel camera body lends itself to a variety of deployment methods including fixed and mobile installations. Integral white LED lighting and a range of lens options increase the Dotcam HR's versatility.

- Radiation Tolerance: 1kGy (10^5 rad), 300Gy/hr doze rate
- Resolution 400 TV lines (NTSC), 450 TV lines (PAL)
- 4mm standard lens (FOV 90 deg)

C01

Development of Dosimeter for Severe Radiation Environment near Reactor Pressure Vessel

Tamotsu Okamoto[1], Yasuhito Gotoh[2], Masafumi Akiyoshi[3], Mitsuru Imaizumi[4], Tomohiro Kobayashi[5], Yasuki Okuno[6]
[1]NIT, Kisarazu Coll., [2]Kyoto Univ., [3]Osaka Pref. Univ., [4]JAXA, [5]RIKEN, [6]JAEA

Abstract

High dose-rate radiation detection is required for Fukushima decommissioning. Therefore, a compact and radiation-tolerant dosimeter without power supply using the solar cells was proposed. In this work, radiation-detection characteristics of the solar cells such as CdTe and InGaP solar cells were investigated. Radiation-induced current density due to ^{60}Co γ-ray irradiation was evaluated in the CdTe and InGaP solar cells, and it was found that radiation-induced current density increased linearly in the range up to approximately 2000 Gy/h.

High dose-rate radiation detection under severe environment such as high-level radiation, high temperature and high humidity is required for decommissioning the Fukushima Daiichi nuclear power station (FDNPS). Gamma dose-rates near reactor pressure vessel (RPV) are predicted to be very high such as over 100 Gy/h, because the nuclear fuel debris and ^{137}Cs are trapped. In addition, no power supply is desired in view of residual hydrogen. Furthermore, a compact detector is required in order to introduce through a small hole of approximately 10 cm. Therefore, we proposed a compact and radiation-tolerant dosimeter without power supply using the solar cells. In this work, we investigated the radiation-detection characteristics of the solar cells such as CdTe and InGaP solar cells.

Figure 1 shows radiation-induced current density from a CdTe and an InGaP solar cells as a function of the dose rate of ^{60}Co γ-ray irradiation. In both solar cells, radiation-induced current density increased linearly in the range up to approximately 2000 Gy/h. This result suggests that dose rate is easily measured using the solar cells.

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.

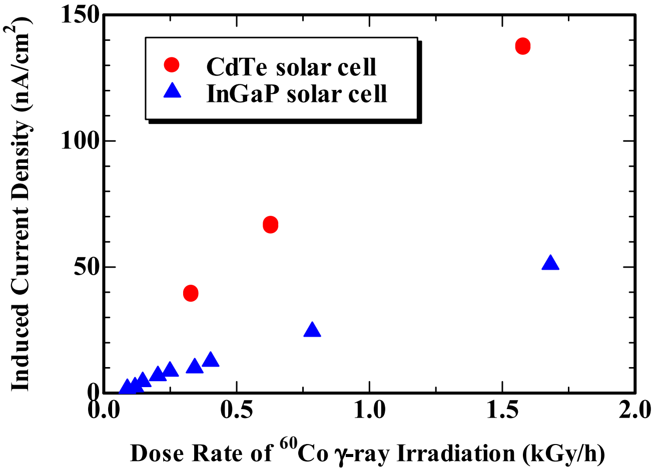


Figure 1 Radiation-induced current density from a CdTe and an InGaP solar cells as a function of the dose rate of ^{60}Co γ-ray irradiation.

C02

Progress of PYRAMID project - Piping system, risk management based on wall thinning monitoring and prediction –

Toshiyuki Takagi[1,2], Philippe Guy[3], Yutaka Watanabe[2,4], Hiroshi Abe[4], Shinji Ebara[4], Tetsuya Uchimoto[1,2], Takayuki Aoki[4], Mitsuo Hashimoto[1], Ryoichi Urayama[1], Hongjun Sun[1], Thomas Monnier[3], Jérôme Antoni[3], Bernard Normand[3], Nicolas Mary[2], Ryo Morita[5], Shun Watanabe[5], Atsushi Iwasaki[6], Hiroyuki Nakamoto[7], Christophe Reboud[8], Pierre Calmon[8], Edouard Demaldent[8], Vahan Baronian[8], Xavier Artusi[8], Sylvain Chatillon[8], Alain Lhemery[8]

[1] Institute of Fluid Science, Tohoku University, 2-1-1 Katahira, Aoba, Sendai, Japan

[2] ELyTMaX UMI 3757, CNRS – Université de Lyon – Tohoku University, Sendai, Japan

[3] Université de Lyon, INSA-Lyon, MATEIS CNRS UMR 5510, Villeurbanne F-69621, France

[4] Graduate School of Engineering, Tohoku University, 6-6-01-2 Aoba, Aramaki, Aoba, Sendai, Japan

[5] Central Research Institute of Electric Power Industry, 2-6-1 Nagasaka, Yokosuka, Kanagawa, Japan

[6] Department of Mechanical System Engineering, Gunma University, Kiryu, Japan

[7] Graduate School of System Informatics, Kobe University, Nada, Kobe, 657-8501, Japan

[8] CEA-LIST, Gif-sur-Yvette, France

1. Introduction

Cooling water circulation is an important guarantee for the safety of Fukushima Daiichi Nuclear Power Plant during decommissioning. However, when removing fuel debris, a flow with a high concentration debris of various kinds occurs in cooling water pipe. Pipe wall thinning by Slurry Flow induced Corrosion (SFC) under solid-liquid two-phase has been anticipated. This may seriously affect the safety of the cooling system. We aim at developing new tools and techniques to quantify pipe wall thinning, and provide a risk management system based on prediction-monitoring of pipe wall thinning due to SFC in piping systems.

2. Progress of the project

2.1. Modeling of SFC and prediction of wall thinning. The wall thinning evaluation model is developed based on the wall thinning rate evaluation considering the mass transfer coefficient under solid-liquid two-phase flow. Then the SFC is elucidated with experiment and numerical simulation.

2.2. Development of EMAT monitoring system. The focusing type EMAT is designed for applying to monitoring system. The designed EMAT is evaluated with experiment and numerical simulation. In addition, a new data processing method is proposed to improve the evaluation accuracy of pipeline wall thickness measurement.

2.3. Engineering risk evaluation. Based on PoF (Probability of Failure) evaluation in consideration of the various errors (inspection / damage progress and applied force) by Bayesian estimation and Reasonable plan by integrative evaluation of factors using a risk matrix, a quantitative evaluation method for engineering risks associated with SFC is proposed.

Acknowledgement

This study is the result of “Piping System, Risk Management based on Wall Thinning Monitoring and Prediction” carried out under the Center of World Intelligence Project for Nuclear S&T and Human Resource Development by the Ministry of Education, Culture, Sports, Science and Technology of Japan, and ANR of France.

C03

Development of directional gamma-ray detectors for Fukushima decommissioning

Mitsuhiro Nogami[1], Keitaro Hitomi[1], Tatsuo Torii[2], Yuki Sato[2]
Yoshihiko Tanimura[2], Kuniaki Kawabata[2], Yoshihiro Furuta[2], Hiroshi Usami[2]
Kenichi Watanabe[3], Toshiyuki Onodera[4], Keizo Ishii[1], Kenji Shimazoe[5], Hiroyuki Takahashi[5]
[1]Tohoku Univ., [2]JAEA, [3]Nagoya Univ. [4]Tohoku Inst. Tech. [5]The Univ. of Tokyo

Abstract

Our group developed directional gamma-ray detectors for Fukushima decommissioning. The directional gamma-ray detector consisted of a lead absorber and two thallium bromide(TlBr) spectrometers. The performance of the detector was evaluated using a ^{137}Cs checking source at room temperature. The directional gamma-ray detector exhibited a good directional performance suitable for Fukushima decommissioning.

1. Introduction

In promoting Fukushima decommissioning, directional gamma-ray detectors operable in intense radiation fields are indispensable for identifying radioactive materials in the contaminated areas. Thallium bromide (TlBr) is a compound semiconductor characterized by high photoelectric absorption efficiency. It has been confirmed that small-volume TlBr detector can operate in intense radiation fields. In this study, a directional gamma-ray detector consisted of a lead absorber and two TlBr spectrometers was developed for Fukushima decommissioning.

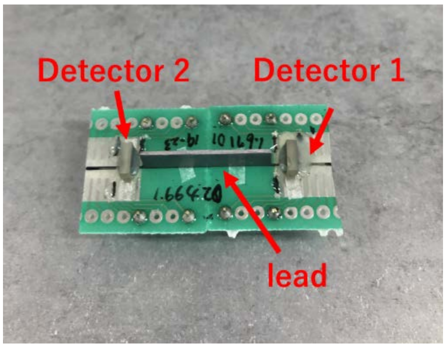


Fig. 1 Directional detector fabricated in this study.

2. Fabrication of directional radiation detector

TlBr spectrometers were fabricated from TlBr crystals. The device had a planar electrode (3.5 mm × 3.5 mm) on the cathode surface, and a pixel electrode (0.5 mm × 0.5 mm) and surrounding guard electrode on the anode surface. The directional detector has a lead absorber with a dimension of 1 mm × 5 mm × 20 mm. Fig.1 shows a directional detector fabricated in this study.

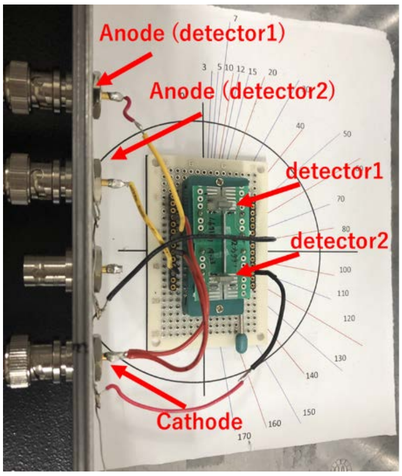


Fig. 2 Setup for directionality measurements.

3. Experiments and results

The directional detector was evaluated with a ^{137}Cs checking source at room temperature. Directional responses of the detector were recorded by changing the position of the radio isotope source around the device as shown in Fig.2.

Figure 3 shows the directional response of the detector. As can be seen from the figure, the device exhibited clear directionality. The details of the experiments will be described at the conference poster.

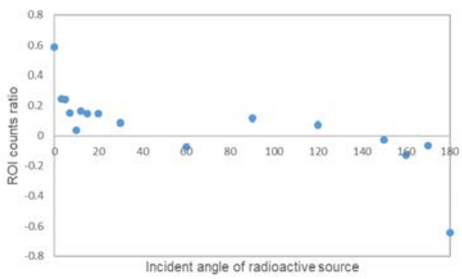


Fig. 3 Directional response of detector.

C04

Can cesium be adsorbed through the leaf and bark of trees? – Estimation of the dominant absorption route of cesium in trees-

Tomoko Ohta[1,2], Junji Torimoto[3], Yasunori Mahara[4]

[1]CRIEPI, [2]The University of Tokyo, [3]JAMSTEC, [4]Kyoto University

Abstract

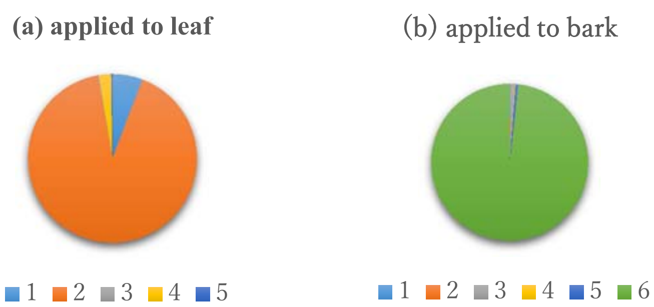
After water-soluble Cs was applied to the leaves and bark of eight pine trees, the Cs was tracked in the whole body of the tree. Although Cs on both the bark and the leaves was transferred into the xylem of tree, Cs on the bark was not taken up into the trunk more effectively than that on the leaves.

1. Introduction

Observing the radiocesium distribution in annual tree rings in fields was limited in early period after the accident. It is necessary to elucidate the dominant route of uptake of radiocesium by the multidirectional comparison methods. We are tried to tracer test to confirm the dominant route of radiocesium uptake from the leaves and bark of pine, which is a popular evergreen coniferous tree (*Pseudotsuga menziesii* var. *glauca)* in Japan, and to estimate the translocation velocity of cesium in xylem in this study.

2. Distribution of cesium in tree

A water-soluble form of cesium was applied to the leaves or bark of pine trees to elucidate the translocation route in the tree. A 10000-ppm solution of water-soluble cesium was made using cesium chloride. The cesium solution was applied to all the leaf surfaces of four trees and to bark on the trunks of five trees. The sample trees were cultivated to 30 days after the cesium solution was applied to the leaves or bark.



After cultivation, the sample trees were cut down and separated into several parts. Concentration of cesium was measured by ICP-MS.Fig. 1 shows the distribution of cesium in the tree.

Fig. 1 Distribution of cesium in tree.

(a)1: branch, 2: leaf (Applied to cesium) 3: leaf 4: trunk, 5: root , (b) 1: branch, 2: leaf 3: trunk, 4: root, 5:xylem, 6: bark (applied cesium)

The results of these tracer experiments indicated that stable cesium enters xylem from foliage and bark. However, radiocesium is quantitatively taken up into the xylem of trunk from leaves than from bark. This result supported that result of our field study the accident after 1.5 years: the amount of radiocesium in the lower trunk derived from the leaf surfaces of *Cryptomeria japonica* was markedly greater than that derived from the bark, though tree species is different.

3. Conclusion

Cesium on leaves and bark was translocated to through bark and foliage to xylem absorption within 30 days. However, in pine trees, cesium is more rapidly translocated from the leaves to the xylem of the trunk than from the bark to the xylem of the trunk. Dominant route of the radiocesium into xylem was leaves.

C05 Radiochemical analysis of actual samples collected at the Fukushima Daiichi NPS site

Masashi Ichikawa[1], Masaaki Kaneko[1], Yoshikazu Koma[2,3], Hiroshi Hinai[2,3], Ayako Nitta[2,3], Atsuhiro Shibata[2,3]

[1]Nippon Nuclear Fuel Development Co., Ltd., [2]Japan Atomic Energy Agency (JAEA), [3]International Research Institute for Nuclear Decommissioning (IRID)

Abstract

As a study on treatment and disposal of solid waste generated by the Fukushima Daiichi NPS (1F) accident, the radionuclide composition of samples collected from the bottom of the primary containment vessel (PCV) of unit 1 was acquired by radiochemical analysis. This is the first time that radionuclide composition data of actual waste samples taken from inside the reactor have been obtained, and the acquisition of these data will make it possible to estimate the origins of the radionuclide contaminants.

Figure 1. Sample from the PCV

1. Introduction

In the 1F accident, radioactive waste with a different nuclide composition from that of conventional waste was generated. To consider the treatment and disposal of this waste, it is necessary to know the radionuclide composition of actual samples. In April 2017, samples were obtained of soil-like sediments present at the PCV bottom of 1F Unit 1 [1]. Some samples were transported to the Nippon Nuclear Fuel Development Co. Ltd. for radiochemical analysis.

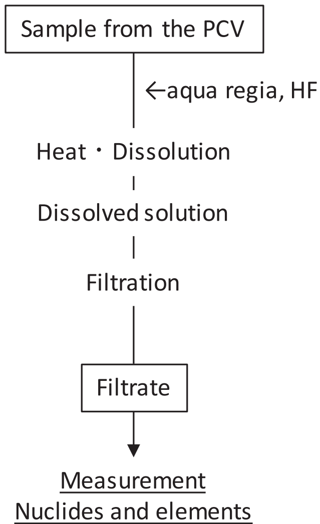


Figure 2. Sample preparation for radiochemical analysis

2. Radiochemical analysis

The photo of a contaminated soil-like sediment sample obtained from the PCV is shown in Figure 1. Figure 2 summarizes the sample preparation procedure for radiochemical analysis. A sample was dissolved in aqua regia and hydrofluoric acid with heating. After filtration of the acid solution, the filtrate was analyzed for radionuclides including those of "difficult-to-measure" and elements. Table l shows some of the measured radionuclides and elements. Actinides (U, Np Am, Cm) and fission products (such as ^{90}Sr, ^{137}Cs and ^{154}Eu) were detected. Also, activation products of Zircaloy (such as ^{93}Zr), and of stainless steel (such as ^{60}Co and ^{63}Ni) were detected. Difficult-to-measure nuclides (such as ^{55}Fe and ^{121m}Sm) were detected after carrying out chemical separation procedures. As analysis of these nuclides has become possible, an estimation of the origins of these nuclides will be possible.

Table 1. Measurement of radionuclides and elements

Origin	Radionuclide	Element
Fuel	^{90}Sr, ^{99}Tc, ^{106}Ru, ^{110m}Ag, ^{125}Sb, ^{126}Sn, ^{137}Cs, ^{154}Eu, isotope of U, ^{237}Np, isotope of Pu, Am/Cm	U
Fuel cladding (Zry-4)	^{93}Zr, ^{121m}Sn, ^{125}Sb	Zr
Structure materials (SUS etc.)	^{55}Fe, ^{60}Co, ^{63}Ni, ^{94}Nb, ^{93m}Nb	Fe

Acknowledgement

This report includes part of the results obtained from the work "Research and Development of Processing and Disposal of Solid Waste", carried out on a supplementary budget allocated for Project of Decommissioning and Contaminated Water Management.

References

[1] TEPCO, 62nd Commission on Supervision and Evaluation of the Specified Nuclear Facility (10 Aug 2018).

C06 Estimation of Aging Properties of F1 Fuel Debris

Oleg Bagryanov[1], Sergei Poglyad[2]
[1]TENEX JSC*, [2]SSC RIAR JSC*

Abstract
Rosatom successfully realized the project «Estimation of Aging Properties of Fuel Debris», main objective of which was to make a comprehensive forecast of FD aging in the range of up to 50 years and to estimate the probability of such aging depending on the initial material parameters and storage conditions.
Within the project model samples with & without Cm imitating FD of F1 were manufactured and studied. Obtained experimental data has been subjected to verification procedures based on the studies of natural geological systems, vitrified HLW and Chernobyl lavas.

1. Introduction
The application of the prediction model had the aim to assess the worst possible case of radioactive materials release into the media contacting with the FD (FD of the core & concrete melt zone periphery).

2. The following tasks have been completed within the project
- Prior research of Chernobyl fuel containing materials summarized;
- Research program for the preparation and testing of the F1 FD model samples developed;
- 14 model uranium-plutonium model samples without Cm & 22 Cm-containing samples manufactured;
- Experimental studies of such processes of FCM aging, as the change in mechanical strength, leaching of different radionuclides, the change in the phase composition performed;
- Experiments for exploring two factors of degradation – radiation factor & chemical factor fulfilled;
- Obtained experimental data has been subjected to verification procedures based on the studies of natural geological systems, vitrified HLW and Chernobyl lavas.

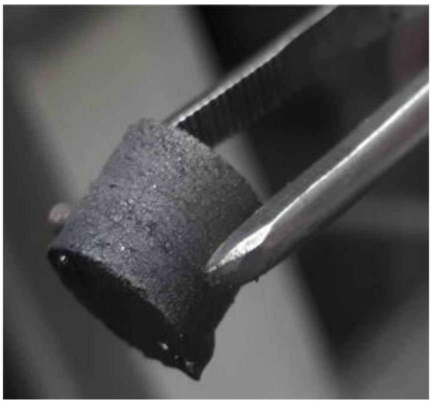


Figure 1. Metallic Phase

2-1. Fe is corrodible
The study of the obtained experimental data allowed us to formulate a hypothesis about the possible scenario of corrosion process. In the presence of the alpha radiation source there is an increase in the rate of Fe leaching, as well as the generation of hydrogen peroxide. The rapid decomposition of hydrogen peroxide leads to the oxidation of Fe, which in turn leads to the “red” suspensions formation. Such “red” suspensions can put additional stress on the safety systems that will function during the implementation of the FD retrieval activities.

3. Conclusion
- Leaching rate of controlled radionuclides of silicate samples is commensurate with leaching rates of the same radionuclides of vitrified HLW matrixes;
- Cumulative radiation-chemical damage leads to greater degradation of fuel fragments, than total radiation and chemical effect;
- Ferrous hydroxide is a good coagulant of Pu and U and other elements, except for Cs, it will lead to formation of larger quantities of small particles and hazardous fine dust. Delay in removal of fuel fragments after 20-25 years after an accident can lead to complication of dose and radiation environment, because of increased emission of plutonium, its further transit into colloidal forms and solid particles of submicrometer dimension.

Figure 2. Silicate Phase

*Companies of Rosatom group

D01 Cesium decontamination of soil by combined techniques of elution, adsorption and electromigration

Haruaki Matsuura[1], Uladzimir I. Kislyi[2], Vladimir M. Dudarchik[2],Yasuhiro Kawashima[2],
Paul G. Molchanov[2], Paul M. Bulai[2], Victor V. Boksha[2]
[1]Tokyo City Univ., [2]Nuclear water soil solutions

Abstract

Cesium decontamination of ground soil are now developing in a collaboration with Japanese and USA-European joint team. Feasibility study has been launched and at first, the preliminary experiment has been performed by using USA-European technique in Japanese facility. Two kinds of ion chelate eluents were introduced to extract radioactive cesium from the soil and eluted cesium were collected in sorbent wrapped cathodes by aid of electromigration.

1. Introduction

It is one of the critical issues of volume reduction of decontaminated soil which is widely spread due to the Fukushima Daiichi Nuclear Power Station. Our developing cesium decontamination technique for soil is as follows; 1) radioactive cesium is eluted by two kinds of eluents, 2) a direct current electromigration technique is applied to concentrate cesium in the solution, and 3) concentrated cesium is adsorbed into sheet form of sorbents located near anodes. The final decontamination target is the fertile soil, and we try to develop the decontamination technique using ecological and economical material to avoid from damage to the soil. We have launched feasibility study of combined technique which has been orginally developed for solving the problems against Chernobyl accident using Cs contaminated soil which was sampled in Fukushima prefecture.

2. Experiment

Sampled soil (ca. 20000 Bq/kg) in ammonia basic solution had been well stirring mixed with two kinds of eluents as well as without eluents for the reference at 50^{o}C for 3days. Then an electromigration under the condition of 40 V between carbon turbular anode and several cathodes wrapped by sorbent sheets have been applied to two solutions for 2 hours each. The concentration efficiency has been evaluated by gamma spectroscopy using Ge semiconductor spectrometer (SEIKO EG&G, measurement duration; 86000 s per each sample) which is focused on ^{137}Cs in sorbent, soil and solution.

3. Results and discussion

Concentration efficiency which is derived from supernatant is 6.13. Therefore two types of eluents (one is surface acitivated material and the other is natural degraded material) has clearly effective to decontamination of soil. However, concentration efficiencies which are derived by the radioactivity of sorbent devided by supernatant are 22100 for mixture with eluents and 70500 for no eluents. The reason of this fact would be as follows: affinity to eluents and cesium disturb the sorption or eluents affect transport phenomena under the electromigration. As shown in preliminary experiments, it is confirmed that eluent and sorbent works well, Further task is characterization of elution and adsorption ability both in equilibrium and kinetic point of view, and finding out optimum condition of electromigration.

D02

Adsorption Properties of Actinoids (U, Am, Np) for Various Inorganic Adsorbents

Hitoshi Mimura[1], Minoru Matsukura[1], Tomoya Kitagawa[1], Fumio Kurosaki[1], Natsuki Fujita[2], Hitoshi Kanda[2], Akira Kirishima[2], Daisuke Akiyama[2], Nobuaki Sato[2], [1]UNION SHOWA K.K., [2]Tohoku University.

Abstract

Adsorption properties of actinoids (U(VI), Am (III), Np(V)) for 19 kinds of inorganic adsorbents (zeolites, *etc*.) were evaluated under different solution conditions. The adsorption of actinoids (An) are shown by a distribution curve consisting of ion exchange, surface adsorption of hydrolysis species and sedimentation depending on equilibrium pH; the pH at the maximum is influenced by hydrolysis pH of each actinoid, and the K_d value depends strongly on pore size, Si/Al ratio and coexisting cation concentration, *etc*.

1. Introduction

The basic adsorption behavior of An on inorganic adsorbents should be clarified from the standpoint of accurate inventory estimation and safety disposal, and the decontamination of An leached from the crushed fuel debris becomes more important in near future. In this study, the practical adsorption properties of An for various inorganic adsorbents were evaluated by batch method under different solution conditions.

2. Results and Discussion

2-1. Adsorption properties of An in simple solutions (HNO_3, HCl, $HClO_4$, NaCl)

The distribution of U(VI) and Am(III) ions depended on equilibrium pH; the K_d value increased markedly with equilibrium pH and had a maximum around neutral region, and then tended to decrease in alkaline region. Zeolite A with the lowest Si/Al ratio (Si/Al=1) had a relatively large K_d value of U(VI) above 10^3 cm^3/g around pH 6~8, and zeolite L with large pore size (7.1~7.8 Å) had the largest K_d value of Am(III) ions over 10^4 cm^3/g at pH 3. On the other hand, the K_d value of Np(V) ions having a high hydrolysis pH tended to increase with equilibrium pH up to 8; the K_d value for zeolite X was estimated to be over 100 cm^3/g in the presence of 0.1 M NaCl-10^{-3} M HCl.

2-2. Adsorption properties of An in boric acid/seawater

The uptake (%) of U(VI) in 3,000 ppm boric acid/ 30% diluted seawater was above 20% for zeolites A, X and Y. The uptake (%) of Am(III) for zeolites was considerably lowered below 20%, due to the formation of Am(III) carbonate complex. Relatively large uptake (%) of Am(III) above 80% was obtained for CST (IE-911) and tin antimonate (IXE-1200G) in acidic region due to the ion exchange (Fig. 1). As for Np(V), zeolite X (13X) had a relatively large K_d value around 100 cm^3/g.

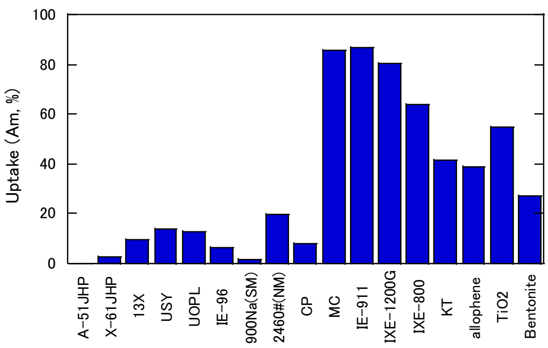


Figure 1. Uptake (%) of Am(III) in boric acid/seawater

3. Conclusion

Precise estimation of chemical species of An in the contaminated water is very important for the fuel debris treatment in Fukushima NPS, and the inorganic adsorbent for the decontamination ofAn should be selected by considering the solution conditions.

References : H. Mimura, *et al.*, Proc of GLOBAL2015, Proc. of ICONE2018, Proc. of ICAPP2019.

D03

Stable Solidification of Radioactive Wastes by Zeolites and Allophane

Hitoshi Mimura[1], Minoru Matsukura[1], Tomoya Kitagawa[1], Fumio Kurosaki[1], Yuki Ikarashi[2], Xiang-Biao Yin[2], Chun-Nan Hsu[2], Yan Wu[3]
[1]UNION SHOWA K.K., [2]Tohoku University, [3]Shanghai Jiao tong University

Abstract

Zeolites and allophane have three functional abilities effective for the treatment of radioactive wastes; nuclide adsorption, Cs trapping and self-sintering abilities. These excellent abilities are applicable to the stable solidification of insoluble ferrocyanide, Cs selective composites, high-level radioactive wastes and uranium oxide.

1. Introduction

Development of stable solidification method contributes to the advancement of decontamination system and environmental remediation at Fukushima NPP-1. The stable solidification of radioactive solid wastes using functional abilities of zeolites and allophane has been studied by UNION SHOWA K.K. and Tohoku University[1]-[4]. This paper deals with the results of stable solidification of the above radioactive solid wastes.

2. Results and Discussion

2-1. Stable solidification of Cs-insoluble ferrocyanides with zeolites

By using the immobilization ability of zeolites, *i.e.*, Cs trapping and self-sintering abilities, the discs of CsCoFC/zeolites (CP, SA-5, MC, A-51J) mixtures were sintered at higher temperatures up to 1,200ºC, and the Cs immobilization (%) for the sintered products at 1,000℃ is estimated to be above 99% (Fig.1), indicating that Cs volatilization was effectively depressed by mixing of zeolites.

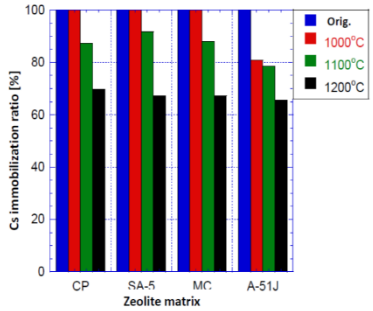


Figure 1. Cs immobilization ratio of sintered products of CsCoFC/zeolite

2-2. Stable solidification of Cs-selective composite adsorbents

Cesium selective adsorbents of insoluble ferrocyanides (NiFC, CoFC) and hetropolyacids (AMP, AWP) can be loaded on macropores of granular zeolites, porous silica gels and alumina. These composites are expected for the selective adsorption of Cs in contaminated water and wash waste water of incineration ash. As for loaded composites of zeolite, the stable solidification can be accomplished by direct sintering. The loaded composites of porous silica can be converted to the stable solid form by adding allophane.

2-3. Stable solidification of denitrated precipitates of HLLW and uranium oxide

Zeolites can be applied to the stable solidification of denitrated precipitates of HLLW. The 1:1 mixture of A type zeolite and denitrated precipitate containing Mo, Zr and La, *etc.* was converted to the stable solid form consisting of major crystal phase of Haüyne after sintering at 1,200℃ for 1 h. The 1:1 mixtures of uranium oxide (U_3O_8) and zeolite (chabazite, allophane) can be converted to the stable solid forms by press/sintering at 600℃. These sintered products had excellent mechanical strength and relatively low elution ratio of U.

3. Conclusion

The excellent immobilization abilities of zeolites and allophane are widely applicable to the stable solidification of radioactive wastes generated from the decontamination and decommissioning in Fukushima NPS.

References [1] Y. Ikarashi, et al., Proc. of WM2013, [2] H. Mimura, et al., J. Ion Exchange, 22(3) (2011)1-13,
[3] H. Mimura, et al., Ion Exchange, 23(1)(2012) 1-15, [4] H. Mimura, et.al., J. Ion Exchange, 23(2)(2012) 1-14.

E01 Major Initiatives for Water Management at the Fukushima Daiichi NPS

Tomoya Kaneda

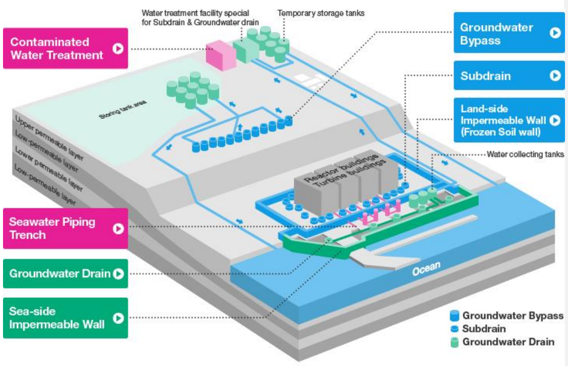
Tokyo Electric Power Company Holdings Inc.

Abstract

We have taken measures against contaminated water caused by earthquake and accident. As a result, we have reduced the rate of contaminated water converted, have made progress in removing contamination from contaminated water, and have made progress in preventing leakage of contaminated water. I will explain the status of our measures against contaminated water.

Outline of contaminated water treatment

At the Fukushima Daiichi NPS, some of the groundwater flowing from the mountain-side to the sea is entering into the nuclear reactor building, converting into newly contaminated water. For this reason, we are implementing various measures to counter the risk of contaminated water flowing into the port of the power station and the risk of contaminated water flowing out from the storage tanks. These measures are based on our three basic policies of "Eliminate contamination sources", "Isolate water from contamination", and "Prevent leakage of contaminated water".



Eliminate contamination sources

- Clean up contaminated water with Advanced liquid Processing System (ALPS)
- Removing contaminated water in trenches
- Removing contaminated water in buildings

Isolate water from contamination

- Pumping up groundwater through groundwater bypass
- Pumping up groundwater through wells near buildings
- Installation of frozen-soil impermeable wall on the land side
- Paving of site to curb permeation of rainwater into soil
- Repair damaged roof

Prevent leakage of contaminated water

- Ground improved with water glass
- Installation of impermeable walls on the sea side
- Augmentation of tanks (replacement with welded tanks, etc)

Figure 1. Main Facilities for Water Management

By implementing these measures, we have achieved the following.

- Reduce the rate of newly contaminated water converted from approximately 490 tons/day in to approximately 170 tons/day
- Complete the removal of contaminated water from the Unit 1 turbine building and radioactive waste disposal building
- Replace the assembly tank with a leak-proof welding tank.

We plan to continue these efforts and achieve the following by the end of 2020

- Reduce the rate of newly contaminated water converted to approximately 150 tons/day
- Complete the removal of contaminated water in the buildings except for the Unit 1 ~ 3 reactor building, where water is being injected to cool fuel debris

E02 Development of Laser Decontamination Method for Dismantling Flange Tank

Naoki Maeshiro[1], Tomohiro Ogawa[1], Nobushige Sasaki[1], Ryohei Takeuchi[2], Fumio Ito[2], Haruo Nagamine[2] and Hiroshi Nakamura[3]
[1]Tokyo Electric Power Company Holdings, Inc., [2]Taisei Corporation, [3]Toyo Union Co., Limited

Abstract

In this paper, the development of a method for decontaminating the inside of flange tank contaminated with radioactive materials at Fukushima Daiichi Nuclear Power Station (1F) by laser irradiation is described.

Figure-1 Dismantling flange tank status

1. Introduction

In dismantling the flange tank(Figure-1), radioactive material is fixed by applying painting on the inner surface in advance. However, we must remove it again during decontaminating dismantled piece of flange tank. In order to reduce dust scattering during dismantling, reduce the exposure of workers and improve the efficiency of work, we developed "Laser Decontamination Method (Figure-2)" controlled automatically that can remove radioactive materials and anticorrosion coatings before dismantling.

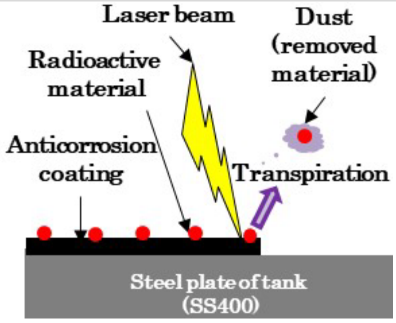


Figure-2 Laser decontamination image

2. Laser decontamination method for dismantling flange tank

2-1. Cold tests (with uncontaminated test piece)

We carried out cold tests to confirm the effective coating removal method with continuous wave fiber laser (2kw) used for laser material processing machines recently. We tried and selected the combination of parameters (laser power, speed and frequency), with the test piece which coated tar epoxy resin on steel plate (SS400) same as a real tank. (Figure-3)

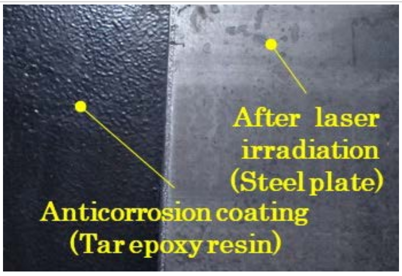


Figure-3 Cold tests result

2-2. Hot tests (with contaminated test piece)

We performed the parameter examination like clean test to confirm the decontamination effect by laser with the test piece cut out from the dismantled tank. Because of difference of the properties by the influence of the encrustations, we reset most suitable parameter. As a result of hot tests, we obtained decontamination factor (DF) 10~1000. Then we developed exclusive laser irradiation and collecting waste system (Figure-4) to reduce radiation exposure. (patent application No. 2018-139336).

Figure-4 Laser decontamination method in flange tank

3. Conclusion

According to both tests, it was confirmed laser decontamination method for inside of flange tank was effective. Utilizing these results, we continuously execute this method to decontaminate flange tank effectively.

E03 Uranium-Containing Particles Found in the Fukushima Daiichi Primary Containment Vessel Interior

Masato Mizokami[1,4], Kenichi Ito[1,4], Akihiro Suzuki[2], Ikken Sato[3,4],
Masaki Kurata[3,4], Mutsumi Hirai[1,4], Shinya Mizokami[1,4]
[1]TEPCO HD, [2]NFD, [3]JAEA, [4]IRID

Abstract

With the recent progression of decommissioning work, radioactive contaminate samples have become obtainable from the Primary Containment Vessel (PCV) interior of Fukushima Daiichi accident reactors. The samples were transported from Fukushima Daiichi (1F) to the hot laboratories of Japan Atomic Energy Agency (JAEA) and Nippon Nuclear Fuel Development Co. Ltd (NFD) for general and detailed examinations. This presentation places the focus on characterization of uranium-containing particles found in the samples.

Introduction

A small portion of the samples were used for Scanning Electron Microscope (SEM) observation. EDS (Energy Dispersive X-ray Spectroscopy) analysis revealed several spots of high uranium EDS signal, possibly arising by the presence of uranium-containing particles. We selected some of the spots for transmission electron microscope (TEM) examinations. About 20 spots were examined so far to characterize the particle details.

Characterization of uranium-containing particles

The microstructures and compositions of particles were characterized by TEM/EDS, and generation mechanisms were discussed.

<u>Type I: Particles generated probably from molten corium</u>

Typical particles consist of dense crystalline Urania-zirconia. An example of TEM images is shown in Fig.1. The compositions were examined by EDS at several spots within the particles. The Zr/U ratios obtained were relatively uniform within the particles, while in one particle Zr-rich region was detected in the matrix, probably showing phase separation of α-Zr(O). The Type I particles have a possibility to carry compositional features representing main fuel debris.

<u>Type II: Vapor condensation</u>

Some particles have round surface and low Zr content. An example is shown in Fig.2. This looks like an agglomerate of 100nm size particles. We also found a dense spherical particle with a diameter similar to that of the agglomerate of Fig.2.

Some other particles look different from the above types (Fig.3), having a separated regions, gaps and micro cracks.

Conclusion

We detected the two types of particles: One is generation from molten corium (Type I), and the other is from vapor condensation (Type II).

ACKNOWLEDGEMENT

This presentation is based on the results of Subsidy Program "Project of Decommissioning and Contaminated Water Management".

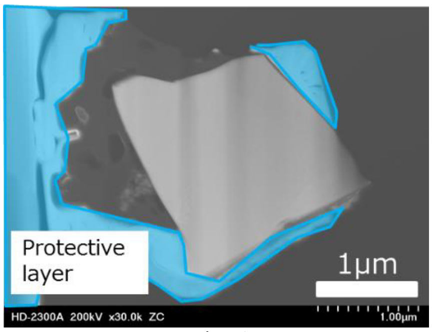


Fig. 1

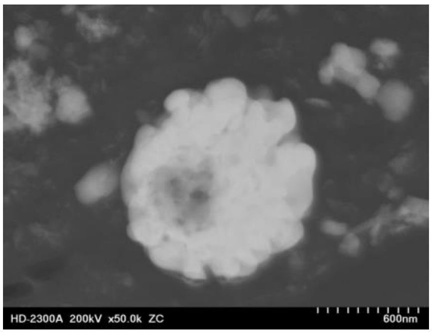


Fig. 2

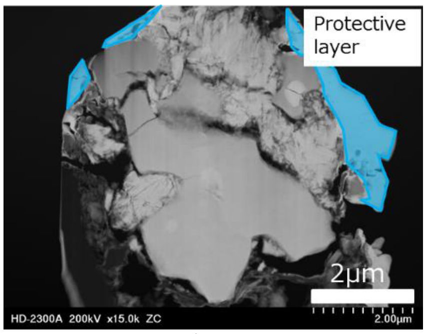


Fig. 3

E04

Units 1-2 Exhaust Stack: Local Company's Big Challenge -- Yes, it's possible!

1・2号機排気筒：地元企業のビッグチャレンジ ～ Yes, it's possible!

Yukihide Sato[1], Isamu Okai[1] and Mariko Chuman[1]

[1]ABLE Co. Ltd.

Abstract

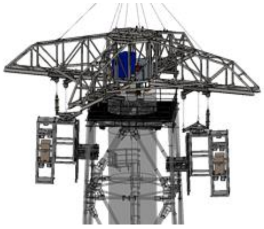
The high dose rate around Fukushima Daiichi NPP Units 1-2 Exhaust Stack has been an impediment to the D&D work. And besides, damage and cracks were detected in its support. ABLE is challenging these difficult tasks as the prime contractor, and demonstrating its high technical capabilities.

1. Removal of Accumulated Radioactive Water from Stack Drain Sump Pit

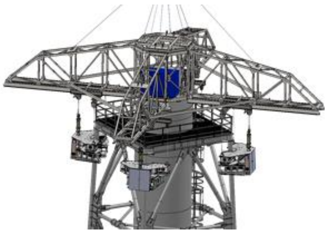
- ✓ No as-build drawing, very high dose rate (max. 3.6 Sv/h), narrow space
- ✓ Great efforts in robot development of only 5 months, including mock-up test and training
- ✓ Field work completed in only 2 months resulting from the perfect preparation in off-site

2. Dismantling of the Upper Half of Stack

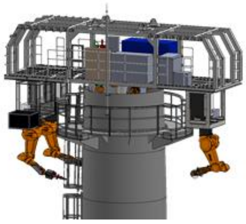
- ✓ 9 to 12 mm^{T} steel stack, highly contaminated inside, dismantling from 120 m to 59 m^{H}
- ✓ Never experienced method: remotely-operated cutting machine suspended from crane
- ✓ FS from 2017, robot development, and elaborate mock-up test & training of 7 months



Cutting of Diagonal Braces



Cutting of Principal Posts



Cutting of Stack Shaft

Mock-up test & Training

Stand-by at the Site

E05

Preparation Work for Unit-2 Fuel Removal at Fukushima Daiichi Nuclear Station

Takashi Inoue, Motohiro Taniyama, Jyunro Nakagoshi,
Yasushi Kageyama, Miho Miyazaki, Ippei Matsuo, Kihei Ogawa
Kajima Corporation

Abstract

Kajima has started preparation work for Unit-2 fuel removal in 2015. Until now, west platform construction (1), new opening work to access the operating floor of the reactor building from the outside (2) and prevention of increase in contaminated water due to rainwater intrusion to contaminated buildings (3) have been carried out in a high dose rate environment. This paper summarizes the challenges in design and construction.

1. West platform construction

- Overhang design: West platform was overhung to secure adequate work space to build new opening to access the operating floor since the west area of the reactor building was very limited. The platform was connected with the reactor building through damping device to resist earthquake.
- Modular construction: Modular construction was adopted to reduce radiation exposure of workers during assembly of west platform. After each module was assembled at the yard in a low dose rate environment, the assembled modules were installed by large cranes.
- Damping device: Buckling-restrained brace and ball joint were developed for the connection between west platform and the reactor building. The earthquake resistance was achieved by plastic deformation of steel tube which was filled with mortar to prevent bucking.
- Remote-controlled device: Remote-controlled devices for rebar position survey and drilling for anchor placement were developed to reduce radiation exposure of workers during installation of the buckling-restrained braces.

2. New opening work in a west outer wall

The new opening work in a west outer wall was carried out inside the temporary house for dust scattering prevention. After the wall was cut and divided to several blocks by concrete core cutter, the blocks were removed by remote-controlled machine.

3. Prevention of increase in contaminated water due to rainwater intrusion to contaminated buildings

- Contaminated protection layer for waterproof roof which contaminates rainwater was removed by remote-controlled machinery.
- Wireless active personal dosimeters (APD) were introduced to reduce further radiation exposure of works. They enabled remote-monitoring and remote-instruction for work and evacuation.
- High flowing, fiber reinforced, and heavy concrete was developed to reduce manual repair work for damaged reinforced concrete roof in high dose rate environment and to enhance radiation-shielding capability.

West platform construction

New opening work in a west outer wall

Removal of contaminated protection layer

Note: The remote-controlled technologies for the above project were developed in collaboration with local companies.

H01

Variations of Hydrogen and Vapor Concentrations in a Simply Simulated Radioactive Waste Long-Term Storage Container with a Newly Developed PAR

Kazuyuki Takase[1], Genya Kawasaki[1], Kohei Ueta[1], Ky Tri Vu[2] and Isamu Kudo[2]
[1]Nagaoka Univ. of Technology, [2]ADVAN ENG. Co., Ltd.

Abstract

In the decommissioning of nuclear reactors, hydrogen which is the flammable gas is generated in storage containers by radiolysis of water. Therefore, in order to improve the safety of waste storage containers for hydrogen, it was experimentally confirmed that a newly developed passive autocatalytic recombiner (PAR) is effective in reducing hydrogen concentration.

1. Introduction

The objective of the present study is to confirm experimentally that a newly developed PAR is effective for reducing the hydrogen concentration in the waste storage containers.

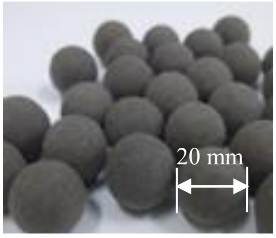


Figure 1. developed catalyst

2. Evaluation of PAR on Hydrogen Concentration Reduction

A catalyst in which platinum is coated on the surface of alumina sphere has been developed as a hydrogen recombination catalyst. By changing the diameter of the alumina sphere, it becomes possible to make PARs with any size. Fig. 1 shows the appearance of the presently developed PAR with a diameter of 20 mm, which was made by ADVAN ENG Co. Ltd.

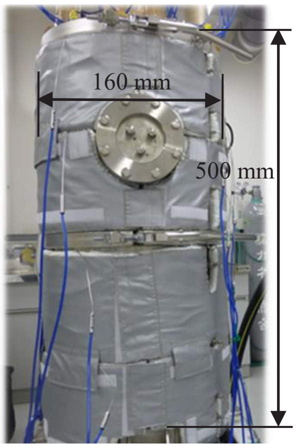


Figure 2. A cylindrical container

Figure 2 shows a container used in the experiment which is 400 mm in diameter and 900 mm in height. A hydrogen concentration sensor and a relative humidity sensor are installed in the container. In the experiment, hydrogen gas of a constant flow rate was injected into the container from the central portion of the bottom plate of the container.

Figure 3 shows experimental results of hydrogen concentration and relative humidity when the hydrogen flow rate of 50 cc/min was injected during 200 s. Here, the number of 3, 6 or 9 indicates the number of installed catalysts. Hydrogen and oxygen are combined with PAR and vapor occurs. As a result, the hydrogen concentration is reduced. Reduction of hydrogen concentration depends on the number of catalysts.

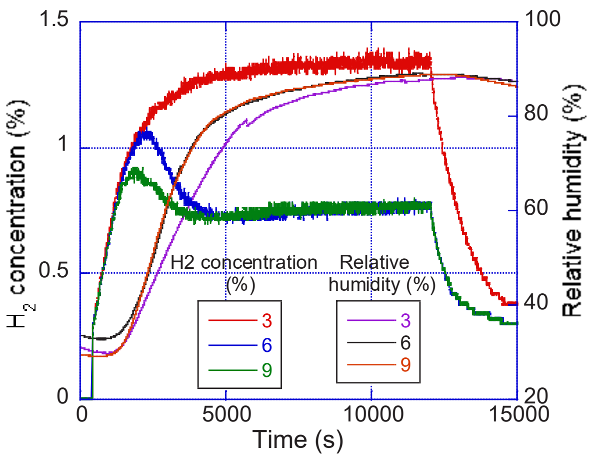


Figure 3. Measured H_2 concentration

3. Conclusion

It was clarified experimentally that the hydrogen concentration in the container can be reduced by the presently developed catalyst.

H02

Research on hydrogen safety technology utilizing the automotive catalyst

T. Aida[1], T. Yamoto[1], T. Kita[1], H. Ono[1], K. Takenaka[1], D. Matsumura[2], M. Taniguchi [3], E-A. Reinecke[4], H. Tanaka[1]

[1]Kwansei Gakuin University, Hyogo, Japan,
[2] Japan Atomic Energy Agency, SPring-8, Hyogo, Japan,
[3] Daihatsu Motor Co., Ltd., Shiga, Japan, [4] Forschungszentrum Jülich, Germany

Abstract

The passive autocatalytic recombiner (PAR) for hydrogen oxidation which does not need electric power supply has been developed. Researches on the effect of geometric parameter of monolithic substrate on the natural convection and the catalytic activity for oxidizing hydrogen were investigated in order to exhibit sufficient performance even with mass production catalysts.

1. Introduction

It has been required to transport and store safely high-dose radioactive waste toward decommissioning of Fukusima Daiich Nuclear Power Station. Generation of hydrogen in the container due to radiolysis of water is serious problem. The objective of this study is to reduce hydrogen by recombining to water with monolithic automotive catalyst without external power source.

Fig.1. REKO-4 reactor.

2. Experimental

The experiments were conducted in REKO-4 reactor in institute of Jülich, German. The appearance of REKO-4, 5330 L in the volume, is shown in Fig. 1. In steady state test, the natural convection velocity was calculated from the amount of injected hydrogen since it was equivalent to that of hydrogen oxidized by the catalyst. In a dynamic test, hydrogen gas was injected until the concentration reached 6%, and then the injection was stopped and all valves were closed, after which time changes in hydrogen concentration were measured. These series of tests used a chimney of 300 mm.

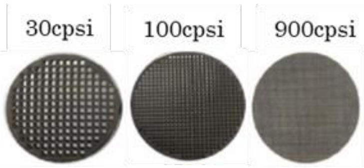


Fig.2. Geometric parameter.

3. Results & Discussion

3-1. Steady state test

The effect of geometric parameter on the natural convection was evaluated (Fig.2). The flow rate of natural convection was 27 times improved by lowering cell density from 900 to 30 cpsi (Fig. 3).

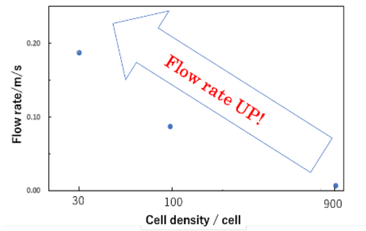


Fig.3. Flow rate / cell density.

3-2. Dynamic test

By lowering the catalyst cell density to 30 cpsi, the hydrogen reduction time could be significantly reduced and achieved to lower hydrogen concentrations (Figs.4 and 5). It has been confirmed that lowering the cell density of the catalyst provides significant benefits from both the steady state and the dynamic tests.

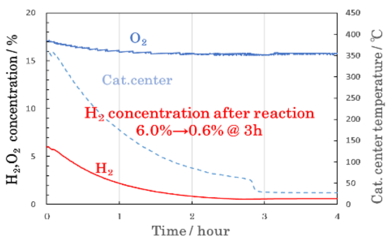


Fig.4. The catalytic activity (30 cpsi).

4. Conclusion

The flow rate was improved by coarse cell density and it can reduce hydrogen concentration efficiently to safe level. Sufficient performance was exhibited even with mass production lower limit size catalysts (30 cpsi, 10 mm).

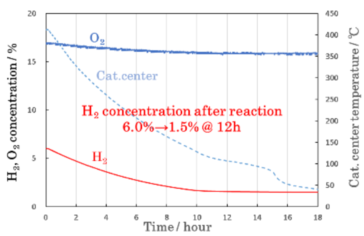


Fig.5. The catalytic activity (900 cpsi).

References

[1] H. Ono, et.al., Research on hydrogen safety technology utilizing the automotive catalyst, *E-Journal of Advanced Maintenance*, Vol.11, No.1 (2019) 40-45.

Acknowledgement

A part of this study is the result of "R&D on technology for reducing concentration of flammable gases generated in long-term waste storage containers" carried out under the "Center of World Intelligence Project for Nuclear Science and Technology and Human Resource Development" by the Ministry of Education, Culture, Sports, Science and Technology of Japan.

H03

Research on hydrogen recombination catalyst under humidity environment

T. Yamamoto[1], T. Aida[1], K. Fujita[1] T. Kita[1], H. Ono[1], D. Matsumura[2], M. Taniguchi[3], H. Tanaka[1]

[1]Kwansei Gakuin University, Hyogo, Japan, [2]Japan Atomic Energy Agency, SPring-8, Hyogo, Japan, [3]Daihatsu Motor Co, Ltd., Shiga, Japan

Abstract

The catalysts have been developed for transport and storage debris from decommissioning of Fukushima Daiichi Nuclear Power Station. In particular, catalysts that perform well in humid environments are focusing in this study. It was confirmed that the catalyst performance is maintained even in a high humidity environment by subjecting the catalyst to water repellent treatment.

1. Introduction

The technology for transport and storage a high-dose radioactive waste is required for decommissioning of Fukushima Daiichi Nuclear Power Station. It is a serious problem that hydrogen generated by radiolysis of water in the container. The hydrogen oxidation activity was influenced under 100% relative humidity. As a countermeasure for this, the catalyst was treated to be water repellent and the effect was evaluated.

2.Experimental

The catalytic activities were evaluated by using TPR (Temperature Programed Reactor).

- Catalyst: (1) Pd/Al_2O_3
  (2) Pd/WR (Water Repellency)
  (3) Pd/SWR (Super Water Repellency)
- PGM loading: 0.5 wt% Pd/Supporting materials
- Pretreatment gas: Ox ;10% O_2 / Balance He / 200 ℃
  Red;10% H_2 / Balance He / 200 ℃
- Analysis condition: -120 ℃ ~ RT / without heater
  RT ~1 00 ℃ / use heater
  DRY / WET (absolute10%)
- Gas composition: 6% H_2, 4% O_2, Balance, He
  (50 ml/min)
- Gas analyzer: Q-mass

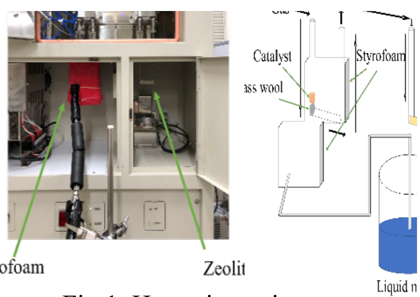


Fig.1. Home-in equipment.

3. Results & Discussion

Fig.2 shows that H_2 conversion at 20 ℃. All catalysts converted 90% or more of hydrogen under DRY conditions. The oxidation treatment did not affect the conversion at room temperature. The performance of Pd/Al_2O_3 was significantly reduced under WET conditions. It is considered that the water generated by the reaction covered catalytic active site. On the other hand, Pd/WR and Pd/SWR were effective. Pd/SWR converted 10% more hydrogen than Pd/WR under WET conditions. Stronger water repellency represented greater catalytic performance improvement. The hypothesis of the water repellant effect is shown in Fig 3.

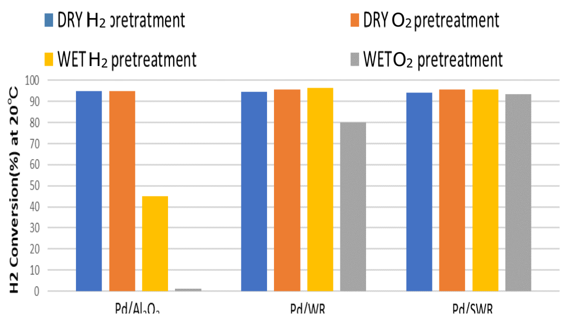


Fig.2. H_2 conversion at 20 ℃.

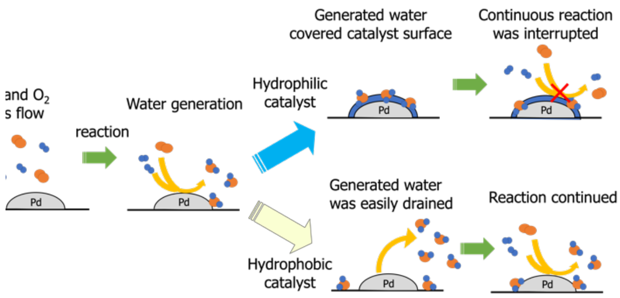


Fig.3. Hypnosis of water repellency.

4. Conclusion

✓ Water repellency improved the catalytic activity.
✓ The research should be continued aiming at the practical use of hydrogen recombination catalyst.

Reference

[1] T. Kita, et. al., "Hydrogen safety technology utilizing the automotive catalyst", poster H03, *The 3rd International Forum on the Decommissioning of the Fukushima Daiichi Nuclear Power Station* (2018).

[2] H. Ono, et.al., Research on hydrogen safety technology utilizing the automotive catalyst, *E-Journal of Advanced Maintenance*, Vol.11, No.1 (2019) 40-45.

K01

Waste Management Symposia: Annual Conference in Phoenix, Arizona, Exchanging Knowledge from Around the World

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

[1]WM Symposia, Inc. Board Member, [2]WM Symposia, Inc. Deputy Managing Director and [3]WM Symposia, Inc. 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 WM2020 Conference, the 46th version, will be held March 8-12, 2020 at the Phoenix Convention Center in Phoenix, Arizona. Conference theme is "Reducing Long Term Environmental Liability Through Efficient, Effective Clean-up". WM2020 will feature over 500 papers and more than 40 panel discussions in over 140 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. WM2020 has special programs aimed at encouraging participation of the world's leading companies in the exhibit hall, including a program for product demonstration.
The Conference focuses also on promoting the next generation of students and radwaste management professionals by providing a platform for exchanges of expertise.

3. Conclusion

The Conference promotes, among Japanese and professionals from around the world, a broad exchange of knowledge on technologies, operations, safety, waste management issues and the decommissioning and dismantling. The deadline for submittal of Abstracts for WM2020 is August 23. Details are shown on the poster.

Reference: www.wmsym.org.

K02

Influence of Na and Ca Concentrations on Sr Adsorption Performance of Cs-Sr Simultaneous Adsorbent

Akira Tsutsumiguchi[1], Yuko Kani[1], Tsuyoshi Ito[1], Takashi Asano[2], Takatoshi Hijikata[3], Tadafumi Koyama[3] and Yasuhiro Suzuki[4]

[1]Hitachi, Ltd., [2]Hitachi-GE Nuclear Energy, Ltd.,

[3]Central Research Institute of Electric Power Industry, [4]JGC Corporation

Abstract

Treating evaporation-concentrated liquid waste (evap. waste) stored in tanks at Fukushima with high radioactive Cs and Sr concentrations has been examined using Cs-Sr simultaneous adsorbent. As a result of the dilution effect evaluation, the Sr breakthrough point of the adsorbent was markedly increased by diluting the evap. waste 4 times or more.

1. Introduction

In this study, we investigated the applicability to treatment of evap. waste stored at Fukushima which is present in a small amount but has high Na (max 50000 ppm) and Ca (max 300 ppm) concentrations using Cs-Sr simultaneous adsorbent [1]. The Na and Ca concentrations are higher than the Sr concentration, and even if the adsorbent has high Sr selectivity, the lifetime is shortened by the influence of Na and Ca. Therefore, we experimentally evaluated the influence of Na and Ca concentrations on Sr adsorption performance of the adsorbent to apply the Cs-Sr simultaneous adsorbent efficiently for the evap. waste.

2. Results and Discussion

A column test was conducted to evaluate the Sr adsorption performance of Cs-Sr simultaneous adsorbent. Na and Ca concentrations of the test liquid were set as parameters based on the evap. waste. From the test results, lowering of Na concentration was effective from the start even at high concentration, and lowering of Ca concentration was effective when decreased to 80 ppm or less (Figure 1). As a result of evaluating dilution effect of concentration reduction, Sr breakthrough point of the adsorbent was markedly increased by diluting the evap. waste 4 times or more (Figure 2).

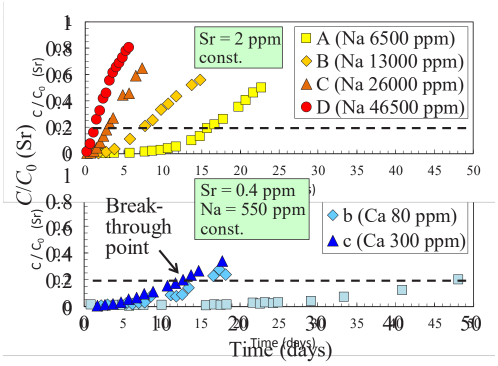


Figure 1. Sr breakthrough curve

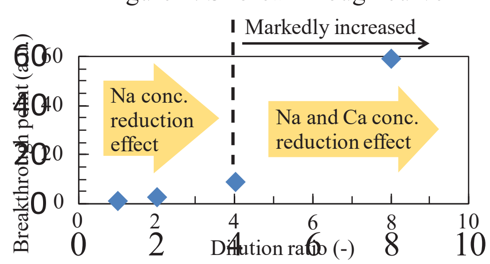


Figure 2. Dilution ratio and breakthrough point correlation

3. Conclusion

The applicability of the Cs-Sr simultaneous adsorbent to treatment of the evaporation-concentrated liquid waste was confirmed from the results of the column test.

Reference

[1] Y. Kani et.al, Proc. WM2014 Conference, Phoenix, Arizona, March 2-6, No. 14110 (2014).

K03 Study on Post-Accident Waste Management Scenarios for Fukushima Daiichi Nuclear Power Station

Subcommittee on Radioactive Waste Management
Review Committee on Decommissioning of the Fukushima Daiichi NPS
Satoshi YANAGIHARA and Hiroshi MIYANO
Atomic Energy Society of Japan

Abstract

Long-term radioactive waste management strategy is integral to decommissioning and site remediation at Fukushima Daiichi Nuclear Power Station (1F). Potential waste management approaches were identified and characterized in order to support future decision making on the strategy. Four radioactive waste management scenarios were developed considering alternative interim and final end states of decommissioning and remediation. They were semi-quantitatively evaluated from waste generation through disposal in terms of activity timelines, amount of waste, waste management options, and timescale to achieve the end state.

1. Introduction

The strategy for managing radioactive waste and materials at Fukushima Daiichi Nuclear Power Station (1F) needs to be developed alongside with the decommissioning and site remediation strategies and plans. The objective of the subcommittee is to identify the requirements and constraints on decommissioning and remediation of 1F from the waste management perspective, and provide information on alternatives of the radioactive waste management. There are multiple approached to manage waste that will be generated from decommissioning and site remediation of 1F, and optimization of radioactive waste management scenarios is essential. This study aims to semi-quantitatively evaluate potential waste management scenarios for multiple potential interim and final end-states of decommissioning and remediation.

2. Methodology

The site at 1F was divided into six zones in terms of decommissioning of nuclear facilities, and three zones in terms of site remediation. In order to construct potential decommissioning/site remediation scenarios, decommissioning/site remediation activities at 1F were divided into four phases: 1) dismantling of reactor and turbine facilities (Phase 1), 2) dismantling of other nuclear facilities (Phase 2), 3) site remediation (Phase 3), and 4) nuclear site management/radioactive waste storage and environmental stewardship (Phase 4). Four scenarios were developed with two possible interim end state of i) complete removal of all facilites/contamination, and ii) partial removal and continued care and maintenance, and two decommissioning strategy of i) immediate dismantling, and ii) deferred dismantling.

3. Results and Discussion

The outline of the scenarios are as follows; S-1: Immediate dismantling/ complete removal of all radioactive contamination, S-2: Immediate dismantling/ partial removal of facilities/ contamination followed by continued maintenance/environmental stewardship, S-3: Deferred dismantling/ complete removal after safe enclosure of nuclear facilities, S-4: Deferred dismantling/ partial removal after safe enclosure followed by maintenance/ environmental stewardship (Fig 1). The S-1 can achieve the end state at the early stage, but disposal of a large quantity of radioactive waste also needs to be realized. Only a limited area can be released from the regulatory control in the S-2 because of the continued C&N and environmental stewardship. The achievement of the end state will be further delayed in S-3 and S-4 when all of the activities occur after safe enclosure period.

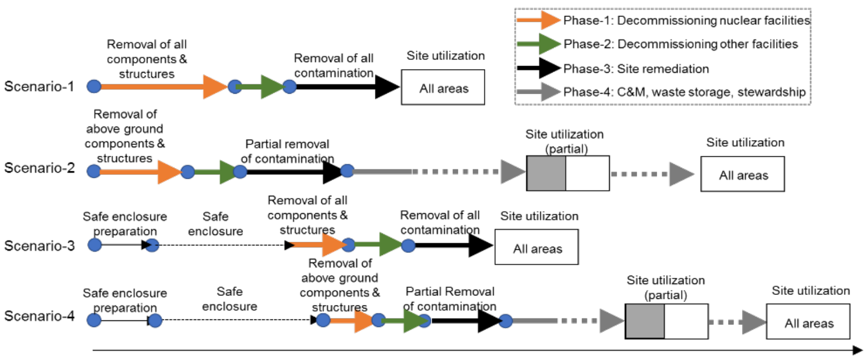


Fig. 1 Timeline of various scenarios divided into four phases

4. Concluding Remarks

Based on the Mid-and-long-term roadmap towards the Decommissioning of 1F, fuel debris removal will start in 2021 to complete decommissioning activities by around 2050. These activities will generate waste, and radioactive waste management strategies need to be developed in time. The development involves not only technical considerations, but discussions with stakeholders on important matters including the end state conditions of the 1F site and allocating waste storage and disposal facilities. This scenario analysis on the waste management alternatives aims to support future discussion and decision making from the long-term perspective.

K04

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

Milena Prazska and Marcela Blazsekova , Wood (Slovakia)
Hisashi Mikami , Nobuyuki Sekine and Mika Mochimaru , 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).

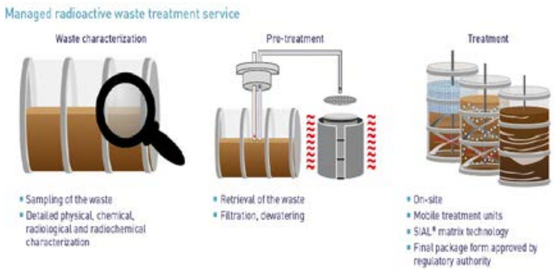


Figure 1 SIAL® solidification treatment process

Figure 2 Sludge and slurry waste streams cross section observation

3. Performance Record Example

About 650 m^3 of radioactive waste streams (resins, sludge and crystalline borates) stored in 14 tanks situated in auxiliary building of NPP Unit V1 in Jaslovske Bohunice. 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.

J01

Preparation of VR data for utilization to decommissioning of Fukushima Daiichi Nuclear Power Station

Rintaro Ito[1], Takahiro Ohno[1], Yoshihiro Tsuchida[1], and Takashi Okada[1]

[1]Naraha Center for Remote Control Technology Development, Japan Atomic Energy Agency

Abstract

The Naraha Center for Remote Control Technology Development (Naraha Center) provides an operation training system for decommissioning work using immersive virtual reality technology (VR system). The current status of the preparation of VR data of Fukushima Daiichi Nuclear Power Station (1F) in the system is described below.

1. Introduction

The VR system installed at the Naraha Center can display the equipment and structure of the 1F facility in full scale with three dimensions and with dose distribution. This system is utilized for planning and training of works related to 1F decommissioning. In order to make more effective use of this system, it is required that the internal situation is accurately reproduced so that users can plan and train in the same environment as the site. Therefore, continuous preparation works on the 1F VR data have been carried out.

2. Current status of the VR data

The data prepared so far are from the first floor to the basement floor in the reactor buildings of Units 1 to 3. These data are created by the point cloud data measured by the robot which entered in the reactor buildings after the accident, and the deficiency data are complemented with the data before the accident. In the Unit 2, some parts of PCV have been prepared for the future debris retrieval works. Furthermore, as for the inside of the pedestal, it was difficult to obtain the data, then the Unit 5 data were obtained by point cloud measuring and converted it into 3D CAD data.

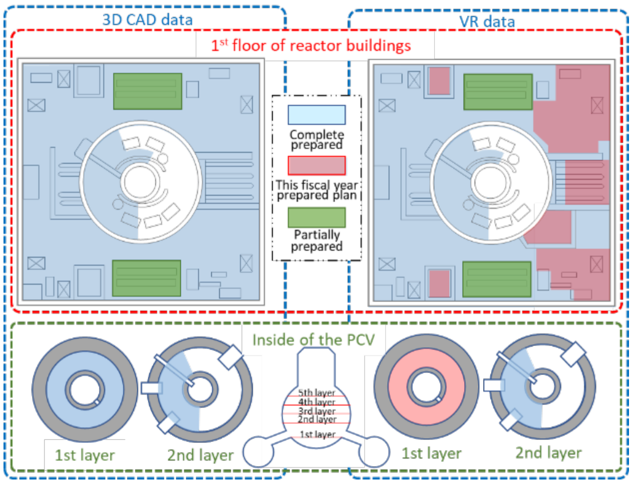


Fig1. Status of the Unit 2 data

In this fiscal year, the preparation of the VR data is planned from 3D-CAD, a part of Unit 2 PCV acquired last year and small rooms on the first floor of the reactor building. In addition, it is also planned to collect survey data, which are inside of the reactor building acquired by International Research Institute for Nuclear Decommissioning, and to expand the data.

The prepared data can be used in the system, and can also be used outside of our center for the 1F decommissioning work as well as the research and develop on the 1F decommissioning.

3. Conclusion

By utilizing the data in this system, it is believed that this system contributes to the safe and reliable decommissioning work, and also to the improvement of the decommissioning work. Our future works are to expand the data based on needs of users, and to develop a mechanism for applying the ever-changing on-site situation.

J02

Evaluation Method of Robot Mapping Performance for Decommissioning

Taichi Yamada[1] and Kuniaki Kawabata[1]

[1]Japan Atomic Energy Agency

Abstract

For decommissioning of Fukushima Daiichi nuclear power plant (1F), workers need to know about work site information for working efficiency and safety of themselves. Mapping technology in robotics called Simultaneous localization and mapping (SLAM) is useful for sharing the site information. Though, there are technical problems for applying SLAM to 1F tough environments. This paper shows the development of the SLAM evaluation method aimed to enhance the research of SLAM for 1F.

1. Introduction

For 1F decommissioning, workers can work only for a very short time due to high radiation exposure. Therefore, workers need to transfer the task to next workers, and the information sharing is important. The visualization of site data makes it easy to share the information. SLAM allows to visualize data of the site by creating maps based on robot sensor data. However, there are technical problems to apply SLAM to 1F environments because of high radiation, darkness, drops of water and so on. Furthermore, because the experiment on the real 1F site is impossible, research to apply SLAM to 1F is difficult. Therefore, in order to enhance SLAM research for 1F decommissioning, we aim to develop the evaluation method and test data for SLAM algorithms. This paper represents the idea to facilitate SLAM algorithm test without actual field and the quantitative evaluation of SLAM algorithm.

2. SLAM Evaluation Method

SLAM algorithms compute the robot position based on the robot sensor data such as camera images. Therefore, by using the log of robot sensor data, it is possible to test SLAM algorithm without any actual equipment, robot and testing field. Moreover, the performance comparison between SLAM algorithms can be conducted based on testing results using the same log, that means, under the same condition. Thus, we prepare the log of sensor data when a robot moves on the mockup field simulating 1F site in Naraha Center for Remote Control Technology Development [1] for SLAM algorithm test. Regarding the SLAM evaluation, we can evaluate quantitatively SLAM algorithm by comparing the estimated robot position by the SLAM algorithm and the accurate robot position. In this research, we obtain the accurate robot position using motion capture at the same time of the sensor data logging. Specifically, our current research is focused on the collecting the log of robot sensor data and the accurate robot position data by motion capture when the robot moves on the mockup field of platform in PCV like shown in Figure 1.

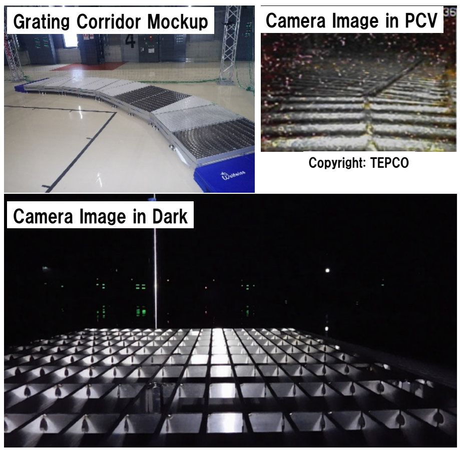


Figure 1. Mockup Filed of PCV

3. Conclusion

This paper describes a SLAM evaluation method using the sensor data log when the robot moves on mockup field. In the future works, we will verify this SLAM evaluation method with the sensor data log.

References

[1] Naraha Center for Remote Control Technology Development, URL: https://naraha.jaea.go.jp/en/index.html

J03

Development of Corrosion Database in Specific Environment

Tomonori Sato[1], Kuniki Hata[1], Yoshiyuki Kaji[1], Hiroyuki Inoue[2], Mitsumasa Taguchi[3], Hajime Seito[3], Eiji Tada[4], Hiroshi Abe[5], Eiji Akiyama[5] and Shunichi Suzuki[6]

[1]Japan Atomic Energy Agency, [2]Osaka Prefecture Univ., [3]National Institutes for Quantum and Radiological Science and Technology, [4]Tokyo Institute of Technology, [5]Tohoku Univ., [6]Univ. of Tokyo

Abstract

Since risk of corrosion degradation for plant materials in Fukushima-Daiichi Power Plant (1F) site have been increasing with time duration and/or environmental changes by decommissioning procedure, we will build a corrosion and radiolysis database in irradiated condition including estimating data area for 1F corrosion.

1. Introduction

Risk of corrosion degradation for plant materials in 1F site have been increasing with time duration and/or environmental changes by decommissioning procedure. Preventing methods for these corrosion risk are developed based on corrosion mechanism. Preventing methods for these corrosion risks are developed based on corrosion mechanism. The implementation system of this research is shown in Fig.1.

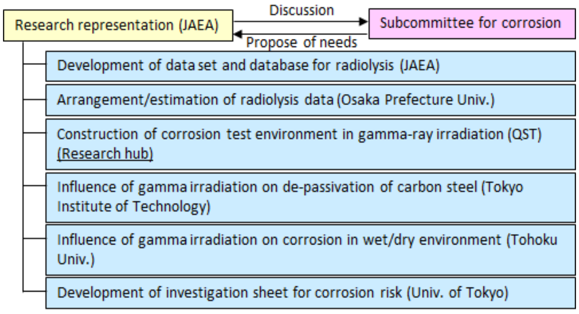


Fig. 1 Implementation system

2. Research plan and results

2-1. Data area for evaluation

In 1F decommissioning process, corrosion data is needed in relatively high dose rate (100 - 10,000 Gy/h) and 1 μS/cm - 10 mS/cm in conductivity conditions (Fig. 2). The gamma-ray irradiation test environment was constructed in the gamma-ray irradiation facility, Takasaki advanced radiation research institute, QST as a research hub for the corrosion under irradiation.

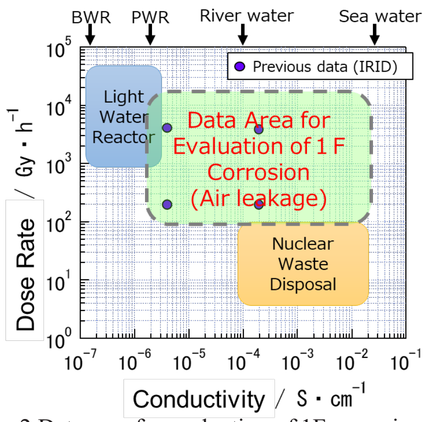


Fig. 2 Data area for evaluation of 1F corrosion

2-2. Major results

The obtained major results are listed in follows;

- The corrosion test environment under irradiation with controlling pH, temperature dissolved O_2 concentration and ion concentration was constructed (Fig.3).
- The corrosion behavior changes with [$B_4O_7^{2-}$]. The de-passivation occurred with the increase of the [Cl^-] in $B_4O_7^{2-}$ and Cl^- co-existent condition.

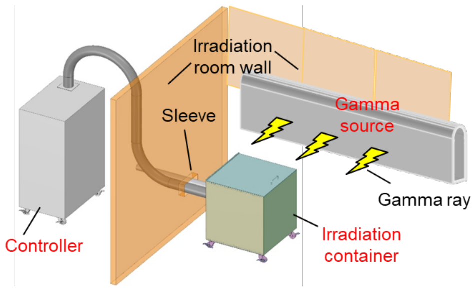


Fig. 3 Gamma-ray irradiation test system

- Corrosion rate at the waterline for half immersion was larger than simple immersion. Corrosion rates under irradiation were larger than non-irradiation.

This work was conducted as a part of "Analysis of Corrosion Mechanism in Specific Environment" funded by the Ministry of Education, Culture, Sports, Science and Technology (MEXT) of Japan.

J04

Development of a Robot Simulator System for Operator Proficiency Training

Kenta Suzuki, Shunji Mukaida, Fumiaki Abe and Kuniaki Kawabata

Japan Atomic Energy Agency

Abstract

This paper describes the current development status of a robot simulator system for nuclear decommissioning operations by using remotely operated robots. This system is utilized for operator proficiency training and implemented function of underwater behavior, communication traffic control, image view presentation, etc. are for simulating the situations and conditions of decommissioning tasks.

1. Introduction

After the accident at Fukushima Daiichi Nuclear Power Station (FDNPS), remotely operated robots were deployed to execute the tasks such an environmental and a radiation survey [1]. However, maneuvering the robots was sometimes difficult and also doing the training within FDNPS by using remotely operated robots is difficult because it may worsen the condition of the work site. Therefore, we are developing a robot simulator system for contributing an operator proficiency training.

2. Implemented functions and Current status of our study

In actual decommissioning tasks, remotely operated robots were deployed to execute environmental survey inside the reactor building, dose rate measurement, underwater investigations in the cooling water and so on. The communication condition also affects to maneuver the robots by the operator. A gamma camera carried by the robots was utilized for radiation measurement at remote site. By referring to the conditions and the lessons learned from past tasks, we have started developing a robot simulator system based on Choreonoid and developed extended functions for simulating above-mentioned situations. Figure 1 shows the implemented functions on the system and each function is a plug-in function of Choreonoid. FluidDynamicsPlugin is to calculate the buoyancy and the fluid resistance applied to the robot model [2]. TCPlugin is to emulate communication conditions like time delay, bandwidth limitation and packet loss. ImageNoisePlugin is to provide a magnification, the distortion, the gaussian noise and the salt-pepper noise on captured image from the camera device. DosePlugin is to provide simulated view of a pinhole-type gamma camera utilizing on radiation dose rate data. In addition, digital models of 1^{st} and basement floor of the rector building of FDNPS, unit 1, 2 and 3. Also the model of mock-up testing facilities of Naraha Center for Remote Control Technology Development, Japan Atomic Energy Agency are installed. By combining these functions, the users can set various situations and conditions and do operator proficiency training.

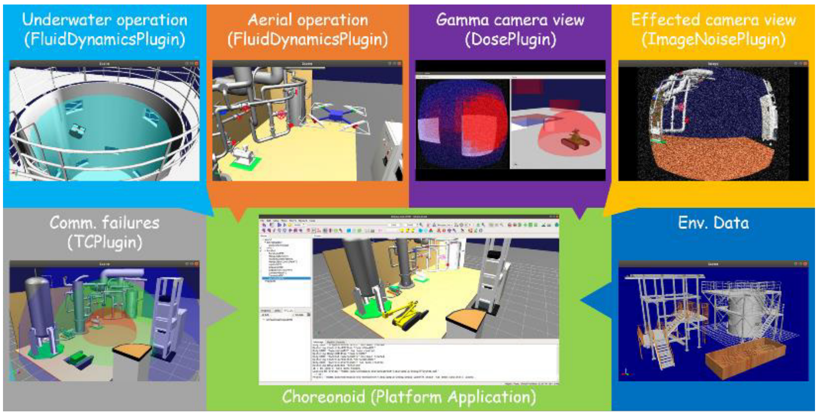


Figure.1 Implemented functions on robot simulator

3. Conclusion

We described the current status of development of the robot simulator and we introduced implemented functions and the digital work site models. The users can train to improve an operator's skill for robot operation.

In future work, we will attempt to develop the functions for simulating a decommissioning task by using multiple robots and also a surface contamination of the robot body.

References

[1] K. Nagatani, S. Kiribayashi, Y. Otake, K. Yoshida, S. Tadokoro, T. Nishimura, T. Yoshida, E. Koyanagi, M. Fukushima and S. Kawatsuma, "Energency response to the nuclear accident at the Fukushima Daiichi nuclear power plants using mobile rescue robots", Journal of Field Robotics 30 (1), 44-63, DOI: 10.1002/rob.21439, (2013)

[2] K. Suzuki, K. Kawabata "Development of a multi-copter simulator and a projection system for virtual operation experience", 2019 IEEE/SICE International Symposium on System Integrations (SII 2019), Paris, France, 2019

J05

Outline and Construction status of Okuma Analysis and Research Center, Laboratory-1

Takahiro Furuse[1], Yuki Ota[1], Van-Khoai Do[1]
[1] Japan Atomic Energy Agency (JAEA)

Abstract

The Japan Atomic Energy Agency has been developing Okuma Analysis and Research Center, a research and development facility, in order to ascertain the properties of radioactive wastes and fuel debris towards the decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Station (1F). In this presentation, we are reporting outline and construction status of Laboratory-1 which will analyse low and medium level samples in the Okuma Analysis and Research Center.

Outline of Laboratory-1

- Radioactive materials with surface dose rate up to 1 Sv/h such as rubble, incinerated ash, and secondary waste from water treatment will be analyzed. Laboratory-1 equips iron cells, glove boxes and fume hoods (Fig. 1).

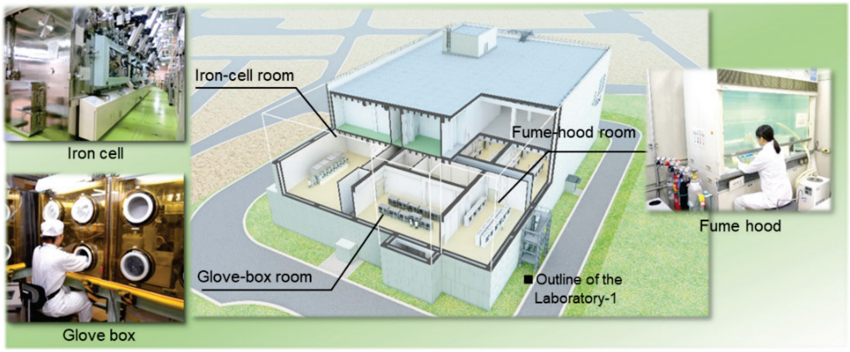


Fig. 1: Conceptional drawing of Laboratory-1

- Radiochemical analysis of 38 nuclides is anticipated necessary to establish technical prospects on safe treatment and disposal of radioactive waste from 1F. Facility design was made to enable variety of analyses, which are listed in Table 1.

Table 1: Main apparatus to be installed

Apparatus to be installed	Analysis target
Alpha ray spectrometer	Alpha nuclides
Gas flow counter	Beta nuclides
Liquid scintillation counter	Low energy beta nuclides
High purity Ge detector	Gamma nuclides
Low energy gamma ray measuring device	Low energy gamma nuclides
ICP-MS	Actinide nuclides
ICP-AES	Trace metal elements

Construction status of Laboratory -1

- Construction work began on April 19, 2017, we have been adopting precast concrete method. This method is that construction products are made in factories in advance to reduce work time in construction site.
- The building is under construction, we are making the third floor now (Fig. 2).

Fig. 2: Construction status of Laboratory-1

J06

Development of analytical methods for 1F-samples

~Analysis of ^{93}Zr by ICP-QQQ-MS and automated sample preparation instruments~

Van-Khoai Do[1,2], Takahiro Furuse[1,2], Yuki Ota[1]

[1] Japan Atomic Energy Agency (JAEA)

[2] International Research Institute of Nuclear Decommissioning (IRID)

Abstract

This presentation outlines our idea of method development for the analysis of ^{93}Zr with application of ICP-QQQ-MS after a simple chemical separation using automated sample preparation instruments

Introduction

The analysis of radioactive samples collected from Fukushima Daiichi Nuclear Power Station (herein referred as 1F) is in a great demand in order to promote the decommissioning of the nuclear facilities. In general, the analysis of those samples often includes a chemical separation step, which eliminate completely influence of sample matrix, and a quantification step using an appropriate technique. ^{93}Zr is one of long-lived fission product nuclides, which emits low beta energy particles. ^{93}Zr is, therefore, a difficult-to-measure isotope. Conventional methods require a time-consuming chemical separation of it from all other beta emitting nuclides. In the previous report [1], it was found that inductively coupled plasma triple quadrupole mass spectrometry (ICP-QQQ-MS) is a promising method for ^{93}Zr quantification because it can simplify the chemical separation step and be capable of enhancing detection limit.

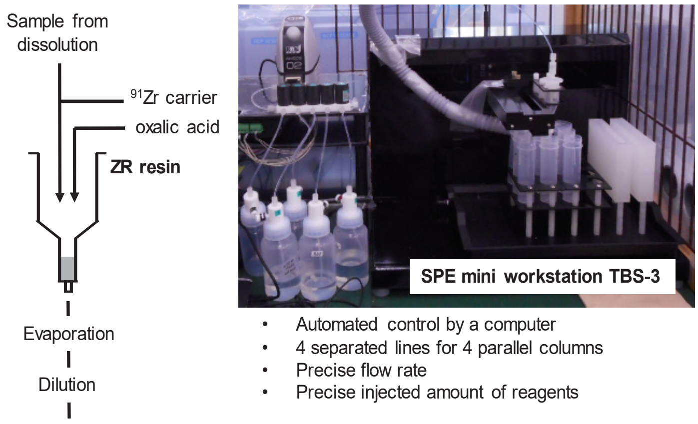


Fig. 1 Separation scheme and automated SPE instrument

Sample preparation using automated instruments

Figure 1 shows a simplified separation scheme for Zr (left) using an automated SPE instrument (right). Samples (rubble, ash, soil) are first dissolved in acids using a microwave. Then, Zr is separated from the sample matrix by a solid phase extraction (SPE) process using ZR-resin (Triskem) before measured by ICP-QQQ-MS.

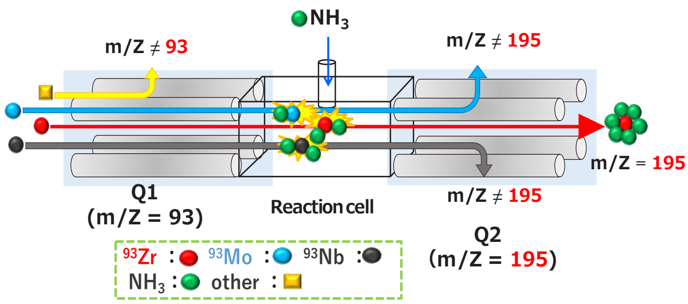


Fig. 2 Interference removal using ICP-QQQ-MS

Quantification of ^{93}Zr using ICP-QQQ-MS.

Interference from natural ^{93}Nb and activation product ^{93}Mo is a major problem in quantification of ^{93}Zr. The interference is expected to be removed partly in the previous chemical separation. Additionally, our preliminary data indicate that ICP-QQQ-MS with two mass filters (Q1 and Q2) and a reaction cell can remove effectively Nb using NH_3 as a reaction gas as illustrated in Fig. 2 [1]. According to our test with non-radioactive isotopes, at mass shift of 102, instrumental sensitivity for ^{90}Zr is 5 orders of magnitude greater than that for ^{93}Nb and ^{98}Mo. After separated from the sample matrix and interferences, collected ^{93}Zr is quantified through ^{91}Zr standard after a mass bias correction.

Reference [1] T. Horita et at, AESJ Spring meeting 2019, 2D10.

This report includes part of the results obtained from the work "Research and Development of Processing and Disposal of Solid Waste", carried out on a supplementary budget allocated for Project of Decommissioning and Contaminated Water Management.

J07

Approach to analysis of hydrogen combustion inside of radioactive waste vessels using CFD softwares based on an open source "OpenFOAM"

Thwe Thwe Aung[1], Atsuhiko Terada[1] and Ryutaro Hino[1]

[1]Japan Atomic Energy Agency

Abstract

As an approach to analysis of hydrogen (H_2) combustion inside of radioactive waste vessels, we performed the simulations for propagation of H_2-air premixed flame by XiFoam solver of OpenFoam package. A new laminar flame speed model deduced from H_2-air explosion experiments by T. Katsumi et al. [1] was implemented in the solver. Flame radius obtained by simulation agreed with the experimental results [1-2].

1. Introduction

Since hydrogen is continuously generated and releases inside of high-level radioactive waste vessels, the awareness must be taken on the risk of hydrogen combustion and explosion. In premixed combustion of H_2-air, flame front becomes unstable and wrinkle due mainly to the hydrodynamic and diffusive-thermal effects, which causes the increase in flame surface area and the faster in flame propagation speed, and the flame accelerates itself. In severe cases, fire and gas explosion occur. In hydrogen safety management, besides the experimental investigations, CFD approaches in predictions of flame propagation phenomena are of important role. Therefore, we introduced the new laminar flame speed model in XiFoam solver [3] and reproduced the propagation of H_2-air flame [1].

2. Simulation Setup and Results

By using the modified XiFoam solver, propagation of H_2-air flame was simulated with setting the parameter obtained by experiment [1]. Mesh for one-eighth of model (0.13m in each side) with uniform mesh size of 0.65mm was generated. Total number of cells was 8,000,000. Equivalence ratio of H_2-air premixture was set to unity, and initial temperature and pressure was set to 298K and 101,325Pa, respectively. Figures 1 and 2 show the temperature distribution of flame at t = 0.007s, and the comparison of flame radius obtained by XiFoam and experiments, respectively. Wrinkle flame formation was observed as in Fig. 1 when the flame propagated outwardly as in experiments [1-2].

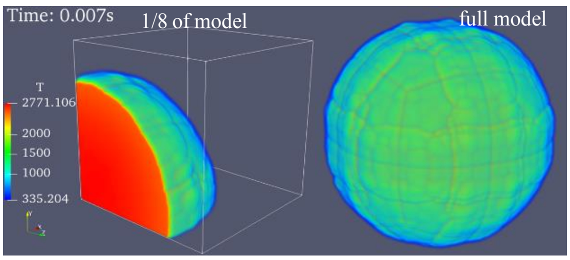


Figure 1. Temperature distributions of H_2-air premixed flame.

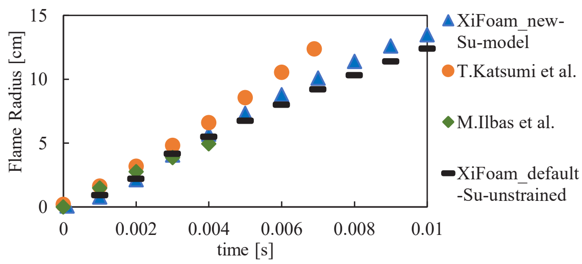


Figure 2. Comparison of flame radius.

3. Conclusion

The results indicated that the modified XiFoam solver of OpenFOAM has capability to reproduce the spherically expanding H_2-air flame and propagation phenomena as in experiments within 0.005s.

References

[1] T. Katsumi et al., Journal of the Combustion Society of Japan, 2017, Vol. 59 No.189, pp. 210-215 (in Japanese)

[2] M. Ilbas et al., International Journal of Hydrogen Energy, 2006, Vol.31, pp. 1768-1779

[3] OpenFOAM User Guide v1712, https://openfoam.com/releases/openfoam-v1712/

J08 Detection of Alpha Particle Emitters Originating from Nuclear Fuels of the Fukushima Daiichi Nuclear Power Station

Yuki Morishita[1]

[1]Japan Atomic Energy Agency/Collaborative Laboratories for Advanced Decommissioning Science (CLADS)

Abstract

We measured alpha emitters obtained from a reactor building in the Fukushima Daiichi Nuclear Power Station (FDNPS) by using an alpha particle imaging detector which combines a very thin (0.05-mm-thick) a cerium-doped $Gd_3(Ga,Al)_5O_{12}$ (Ce:GAGG) scintillator and silicon photomultipliers (SiPMs). The alpha spectrum was in the energy range of 5–6 MeV, which corresponds to the alpha particle energy of ^{238}Pu (5.5 MeV alpha) originated from the nuclear fuel.

1. Introduction

Several papers have reported the presence of alpha particle emitters originating from Nuclear Fuels at the FDNPS [1],[2]. A commercial ZnS(Ag) scintillator-based alpha counter cannot show where the alpha emitters are originating from: nuclear fuels or naturally occurring radionuclide such as Radon (Rn) and progeny. Therefore, we developed an alpha particle imaging detector which is capable of measuring an alpha particle spectrum and 2-dimensional distribution. Some research results were recently published in the journal of Scientific Reports [3].

2. Materials and Methods

Actual measurements for smear samples were performed at the FDNPS site. The detector was composed of a 0.05-mm-thick cerium-doped Ce:GAGG scintillator and SiPM arrays. The floor of the reactor building in FDNPS was wiped off by using smear papers, and the radioactivity of these papers was measured by the alpha particle imaging detector. In addition, we measured a Plutonium (Pu) sample obtained from a nuclear fuel facility by using the same detector for comparison with the smear papers.

3. Results

Figure 1 shows the comparison of alpha spectra between the smear paper and the Pu sample. The alpha spectrum was in the energy range of 5–6 MeV, which corresponds to the alpha particle energy of ^{238}Pu. Moreover, the 60 keV gamma-ray peak from ^{241}Am was identified using a CZT-based gamma spectrometer (Fig.2).

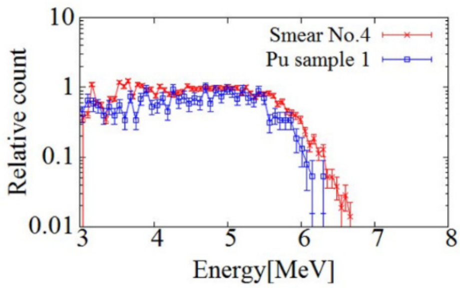


Fig.1 Comparison of alpha spectra between the smear paper and the Pu sample

4. Conclusions

We measured the contamination of the alpha emitters of the FDNPS smear papers by using the alpha particle imaging detector. Based on the measured results, we report actual findings of alpha emitters in the FDNPS reactor buildings originating from nuclear fuels.

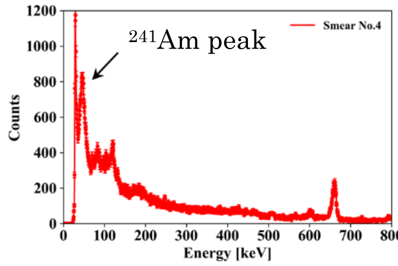


Fig.2 Gamma-ray spectrum of smear paper

References

[1] K. Shozugawa, et al. Environmental pollution 163 (2012) pp. 243-247.

[2]S. Schneider et al. Scientific reports 3 (2013): 2988.

[3] Y. Morishita et al. Scientific reports 9.1 (2019), 581.

J09

On the hydrogen production of geopolymers under gamma irradiation

Vincent Cantarel[1], Makoto Arisaka[1], Isao Yamagishi[1]
[1]Japan Atomic Energy Agency

Abstract
We measured the hydrogen production of geopolymers under gamma irradiation and confront the experimental result to a simple model. Using this model, we were able to highlight the importance of hydrogen recombination [1].

1. Introduction

To ensure the absence of hydrogen explosion risk, it is important to quantify the amount of H_2 gas produced by radiolysis. This is a key factor for intermediate activity wastes in hydraulic binders. To enhance the accuracy of hydrogen production prediction, experiments were carried out to quantify the effects of gas diffusion throughout geopolymers, interesting matrix for in the nuclear waste management field [2].

2. Experimental Results – Irradiation of geopolymers

Under ^{60}Co irradiation, the hydrogen production yield (mol of H_2/Joule of energy absorbed) was measured for water saturated geopolymer of different size to increase the diffusion length for the gas (Figure 1). From powder to 40 cm high cylinders a strong decrease of the hydrogen yield was measured, even after the hydrogen trapped inside the sample was allowed to diffuse out of the matrix.

3. Modelling of the hydrogen gas in geopolymers

For the modeling of the hydrogen production, 3 phenomena were taken in account, production, recombination and diffusion of H_2. With f a function representing the hydrogen concentration, it was found that approximation of the solution of equation 1 allows to predict the hydrogen of all samples with a reaction constant of 6.10^{-4} s^{-1}, all other parameters fixed on literature values.

$$\frac{\partial f}{\partial t} = P - kf + D\frac{\partial^2 f}{\partial x^2} \quad (1)$$

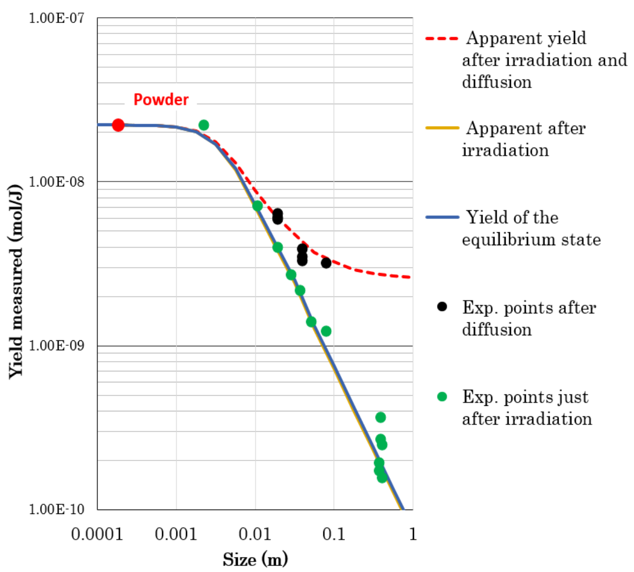


Figure 1 : Hydrogen production of geopolymers samples of different sizes and model predictions

4. Conclusions

When the sample is large, the H_2 accumulates in the sample and is available for recombination, making the global yield a lot lower than powdered samples.

References
[1] Cantarel, V.; Arisaka, M.; Yamagishi, I. *On the hydrogen production of geopolymer wasteforms under irradiation* J. Am. Ceram. Soc. 2019 00:1–11. https ://doi.org/10.1111/jace.16642
[2] Cantarel, V.; Motooka, T.; Yamagishi, I. *Geopolymers and their potential applications in the nuclear waste management field; A Bibliographical study*- JAEA-Review 2017-014

J10 Development of an Adaptive Control System for Laser Cutting of Fuel Debris and Structural Components for the Decommissioning of the Fukushima Daiichi Nuclear Power Plant

Koichi Saruta, Naomitsu Kamei, Yuji Sato, and Toshiharu Muramatsu

Applied Laser and Innovative Technology Institute, Japan Atomic Energy Agency (JAEA)

Abstract

An adaptive control system for laser cutting of fuel debris and structural components is studied for the decommissioning of the Fukushima Daiichi Nuclear Power Plant. An overview of the research and development of the adaptive control system and some preliminary experimental results are presented.

1. Introduction

Laser cutting is one of the promising cutting techniques in the decommissioning of nuclear power plants since it provides many advantages such as remote controllability, high efficiency, and good compatibility with robot technologies. To make the best use of these capabilities and successfully apply laser cutting to fuel debris and structural components in damaged nuclear power plants such as the Fukushima Daiichi Nuclear Power Plant, an adaptive control system will be required because of the inherent difficulty and insufficient knowledge of cutting those targets.

2. Overview of adaptive control system

As shown in Fig. 1, the proposed adaptive control system [1] employs various environmental sensors such as a laser scanner to capture indefinite shapes, a laser-induced breakdown spectroscopy sensor to identify multi-ingredient materials, and a two-color thermometer to monitor melting conditions. Based on the information from the sensors, the robot motion, laser irradiation, and assist gas flow are adaptively controlled to perform laser cutting processes under various conditions. To demonstrate the adaptive control performance, a series of cutting experiments were conducted using a test piece simulating fuel debris, which consists of a SUS304 plate and alumina pellets, as shown in Fig. 2 [2]. The results confirmed that the adaptive control system enables laser cutting of the simulated fuel debris in both air and water.

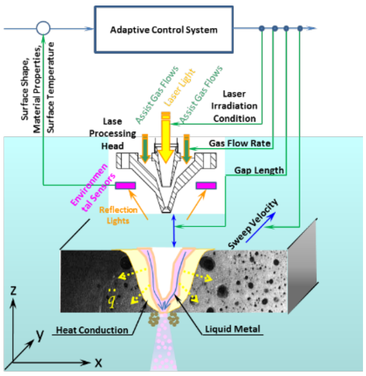


Figure 1. Adaptive control system.

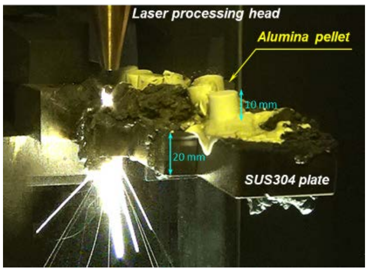


Figure 2. Demonstration of the adaptive control system for laser cutting of the simulated fuel debris.

3. Summary

The preliminary results obtained from the experiments are very encouraging in the sense that the proposed adaptive control system is useful for laser cutting of the fuel debris and structural components at the Fukushima Daiichi Nuclear Power Plant. In the next step the adaptive control system will be integrated into the robot arm system installed in the Fukui Smart Decommissioning Technology Demonstration Base [3] of JAEA to demonstrate laser cutting of simulated fuel debris and structural components in full-scale experiments.

References
[1] Muramatsu et al., JAEA-Research 2013-024 (in Japanese). [2] Muramatsu et al., JAEA-Research 2014-018 (in Japanese). [3] https://fsd.jaea.go.jp/

J11 Prediction of the dose rate of various fuel debris to be sampled from Fukushima Daiichi Nuclear Power Station

Kenichi Terashima[1,2], Keisuke Okumura[1,2]

[1]International Research Institute for Nuclear Decommissioning (IRID),

[2]Japan Atomic Energy Agency (JAEA)

Abstract

The dose rate around the surface of retrieved fuel debris is important information for radiation shielding, handling, transport, analysis and storage of the fuel debris. Therefore, we have developed the dose rate prediction method applicable to various fuel debris sampled from the Fukushima Daiichi Nuclear Power Station in the near future. Using the method, we predicted the dose rate of the fuel debris to be sampled in 2021 and clarified the sensitivities of contributing parameters.

1. Introduction

Samplings of a small amount of fuel debris are planned at the Fukushima Daiichi Nuclear Power Station (1F). However, it is anticipated that various types of fuel debris will be sampled depending on the unit and the sampling position, which makes it difficult to predict the dose rate of fuel debris required for radiation shielding and workers' exposure management. Therefore, by combining the fitting formula based on the results of many photon transport calculations by the Monte Carlo code PHITS with theoretical models, we have developed an efficient prediction method of the surface dose rate defined as ambient dose equivalent around the surface of fuel debris.

2. Method

It is considered that the surface dose rate depends on fuel debris size, radiation source for each nuclide, release ratio of volatile FP at the 1F accident, burnup, elapsed time, etc. The combination of these parameters is enormous, and the information to estimate the value of each parameter is quite limited. Therefore, we have developed an equation for predicting the surface dose rate by the following procedure: (1) Making a radiation source composition by burnup and activation calculations, (2) Making a photon source (line spectrum) for each radioactive nuclide, (3) Dose rate calculations by PHITS for various combinations of parameters for spherical fuel debris assuming homogeneous composition with unit photon source for each nuclide. (4) Making a prediction equation based on the fitting formula obtained from the results of many PHITS calculations, and on the theoretical models on radiation source decay.

3. Conclusion

The obtained equation for the prediction of the surface dose rate D is expressed as follows,

$$
\begin{aligned}
D\left(r,\ \rho, u, w_{Zr}, w_{Fe},\ t, N_Z,\ B, \vec{f}\right) &= \sum_i g_i(1-f_i/100)\cdot P_i\left(r,\ \rho, u, t, B\right)\cdot R_i(r, u, N_Z) \\
&+\left\{g_{Co60}^{Zr}\cdot P_{Co60}^{Zr}\left(r,\ \rho, w_{Zr},\ t\right)+g_{Co60}^{Fe}\cdot P_{Co60}^{Fe}\left(r,\ \rho, w_{Fe},\ t\right)\right\}\cdot R_{Co60}(r,\ u, N_Z) \\
&+g_{Sb125}^{Zr}\cdot P_{Sb125}^{Zr}\left(r,\ \rho, w_{Zr},\ t\right)\cdot R_{Sb125}(r,\ u,\ N_Z).
\end{aligned}
$$

where, r: radius, ρ: bulk density, u, w_{Zr}, w_{Fe}: concentration of uranium, zirconium, iron in fuel debris, respectively, t: time elapse from the accident, N_z: number density of proton, B: burnup, f_i: release ratio from fuel debris, i: contribution radioactive nuclide, g_i: correction coefficient (usually 1.0), P_i: Function of photon intensity, R_i: Response function of surface dose rate for unit photon source by nuclide.

Figure 1 shows an example of the surface dose rate for the fuel debris assuming sampled from Unit-2 in 2021.

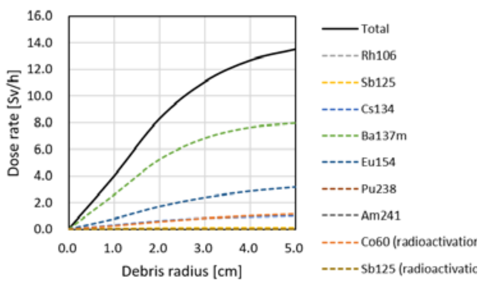


Figure 1. An example of surface dose rate of fuel debris and contributing nuclides

This study includes the result of Subsidy Program "Project of Decommissioning and Contaminated Water Management (Development of Analysis and Estimation Technology for Characterization of Fuel Debris)" in the FY2016 supplementary budget.

J12

Innovative laser remote analysis for surveillance of nuclear debris in decommissioning of Fukushima Daiichi Nuclear Power Station

Ikuo Wakaida[1], Hironori Ohba[1,2], Masabumi Miyabe[1], Katsuaki Akaoka[1], Masaki Ohba[1], Koji Tamura[1,2], Morihisa Saeki[1,2], Yuji Ikeda[3], Takunori Taira[4]

[1]Japan Atomic Energy Agency/CLADS, [2]QST/QuBS, [3]Imagineering Inc., [4]NINS/IMS

Abstract

Radiation resistant optical fiber based laser induced breakdown spectroscopy (Fiber LIBS) and laser ablation resonance absorption spectroscopy are developing for elemental and isotope analysis of sampled nuclear debris in the on-site hot cell facility at the Fukushima Daiich Nuclear Power Station. Microwave assisted LIBS for higher sensitivity and Microchip laser use LIBS (MC-LIBS) for flexible and easy light delivery, are also performed for the surveillance.

1. Introduction

For the decommissioning of “Fukushima Daiich Nuclear Power Station (F1NPS)” development of rapid, easy, onsite and especially in-situ remote analysis/surveillance techniques under the severe environments such as extremely high radioactive condition will be strongly required. In order to accomplish this requirement, the concept of “Probing by light and Analyzing by light” with radiation resistant optical fiber will be one of the simple, powerful and applicable choices as the innovative development based on LIBS and related technologies. In near future, at the first time, analysis/surveillance of sampled debris with strong radiation activity will be required before the transportation of debris samples to the Hot-facility at the outside of F1NPT to make fine and detailed analysis. And the next, we will introduce again the planning of in-situ surveillance of debris in damaged core.

2. Schematics of surveillance system

2-1. Basic concept of laser remote analysis

Basic concept of laser remote analysis will be shown in Fig1. LIBS for elemental analysis and laser ablation resonance absorption spectroscopy in reduced pressure for isotope analysis are integrated as the non-contact, no-preparation, rapid and in-situ analysis system. The laser light and the atomic emission will be able to delivery through an optical fiber, so, this concept will be suitable for remote analysis even under a high radiation activity.

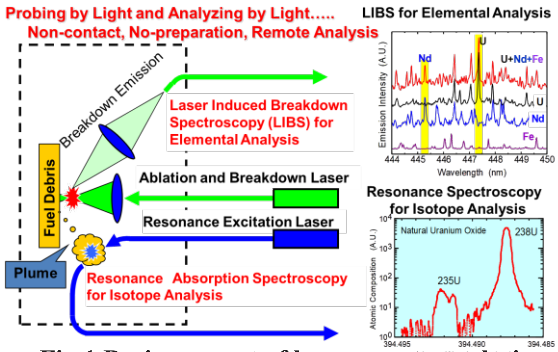


Fig.1 Basic concept of laser remote analysis

2-1. Surveillance system for the on-site hot cell facility

For fine and detailed analysis of debris in the Hot-facility at the outside of F1NPT, confirmation of the existence of the nuclear fuel material in sampled debris by pre-analysis/surveillance will be required before transportation. The basic system for the on-site hot cell facility at F1NPT are now introduced and under developing as shown in shown in Fig.2. The activity of microwave assisted LIBS and MC-LIBS are also under developed.

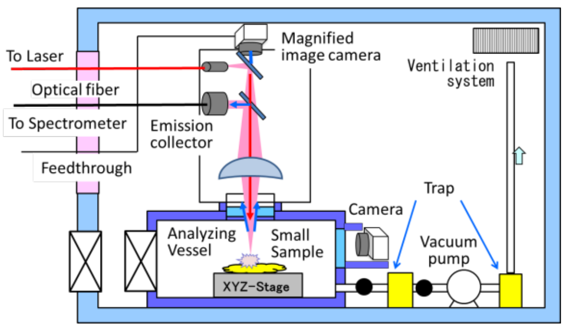


Fig.2 LIBS system for on-site Hot Cell

Present study includes the results of “Advanced study on remote and in-situ elemental analysis of molten fuel debris in damaged core by innovative optical spectroscopy” entrusted to Japan Atomic Energy Agency by the Ministry of Education, Culture, Sports, Science and Technology of Japan (MEXT).

J13

Research and Development on Solidification Technology for Fukushima Daiichi NPS Accident Waste

Jun Kato[1,2], Taniguchi Takumi[1,2], Takeshi Osugi[1,2], Osamu Nakazawa[1,2], Tomoyuki Sone[1,2], Ryoichiro Kuroki[1,2]

[1]Japan Atomic Energy Agency (JAEA), [2]International Research Institute for Nuclear Decommissioning (IRID)

Abstract

We extracted task for selecting solidification technologies available to contaminated water treatment secondary wastes occurred by Fukushima Daiichi NPS accident. The extracted task is lack of information to select solidification technology available to contaminated water treatment secondary wastes. The information lacked was obtained by investigation and experiment for solve this task.

1. Task for selecting solidification technology

Among solid wastes caused by Fukushima Daiichi NPS Accident, rubble, the wastes account for the majority of total (i.e. trees and the waste generated by power plant dismantling (i.e. concrete, metals, and so on)) was removed from target examined because these properties are similar to the existing nuclear facility waste except radioactive inventory. On the other hand, contaminated water treatment secondary wastes (i.e. zeolite, slurry, resin, and so on) have diversity and they have not processed yet. The purpose of this study is to select solidification technology, which will be applicable to the contaminated water treatment secondary wastes. Initial candidate technologies were suggested based on our technology catalog [1] and they were further limited at the past approach. The candidate technologies were limited to high temperature processing (i.e. vitrification and melt-solidification) and low temperature processing (i.e. cementation and Alkali Activated Material (AAM) solidification). The task on these technologies applied to the contaminated water treatment secondary wastes was extracted, and it was found that obtainment of the information lacked is necessary to select candidate technology.

2. Examination of task solution

(1) Preparation of comparative approach between different technologies

Though comparative of different solidification technologies is necessary to select applicable technology, the information which is necessary for preparation of the comparative approach is lack. Therefore, existing information (i.e. processing achievement, the number of steps, processing capacity, operability, economic efficiency, solidified waste size, leaching rate, and so on) was investigated on each candidate technologies. The items which is comparable between deferent technologies were selected from information obtained, comparable items (horizontal axis) was arranged for candidate technologies (vertical axis), and the table was created for comparison these technologies.

(2) Investigation for supplement of unsatisfactory information concerning low temperature processing

Among low temperature processing, the information lacked is concerning the influence factor, liquidity evaluation method, influence of chemical components, and waste temperature in the case containing radioactive, in cementation. These unsatisfactory informations were supplied by investigating available information except for concerning temperature of waste with radioactive. The information concerning the temperature of waste containing radioactive was investigated by simulation of solidified radioactive waste.

(3) Experiment for supplement of unsatisfactory information concerning low temperature processing

Among low temperature processing, the information lacked is concerning usability evaluation (i.e. measurement of fluidity and hardening characteristic), analysis concerning solidified substance performance (i.e. measurement of compressive strength, dissolution, and hydrogen generation by γ-ray irradiation), and analysis concerning standard (i.e. comparison of materials synthesized multiple institutions), in AAM processing. We did experiment for supplement these unsatisfactory informations. The data obtained were comparison with the data of cementation.

4. Next approach

In the next stage, the unsatisfactory information on high temperature processing is going to be supplemented. The information concerning solidified waste which contains the contaminated water treatment secondary wastes is lacking. Therefore, the information is going to be supplemented by investigation and experiment.

This report includes part of the results obtained from the work "Research and Development of Processing and Disposal of Solid Waste", carried out on a supplementary budget allocated for Project of Decommissioning and Contaminated Water Management. This work was studied by Central Research Institute of Electric Power Industry, Taiheiyo Consultant Corporation, and JAEA.

References

[1] Jun Kato, Akinori Nakagawa, Takumi Taniguchi, Tetsuro Sakakibara, Osamu Nakazawa, Yoshihiro Meguro (2017): "The Catalog of solidification and volume reduction technologies for the treatment of radioactive waste generated by the decommissioning of Fukushima Daiichi Nuclear Power Station", JAEA-Review 2017-015. (in Japanese)

J14

Sensitivity analysis of core slumping and alternative water injection in Fukushima Daiichi Nuclear Power Plant Unit 3

Xin Li[1], Ikken Sato[1] and Akifumi Yamaji[2]

[1]Japan Atomic Energy Agency, [2]Waseda University

Abstract

This study aims at identifying the modeling uncertainties and addressing the sensitivity parameters in Fukushima NPP Unit 3. A more detailed Control Volume (CV) division model of the reactor core region has been developed to better simulate the thermal-hydraulic behavior of liquid water and steam, which is considered to be crucial in simulating the core uncovery and degradation process.

1. Introduction

In order to predict the possible status of the corium materials in Fukushima NPP Unit 3, by utilizing severe accident analysis code MELCOR, the current study aims at identifying the modeling uncertainties and addressing the sensitivity parameters in the first 55 hours after SCRAM of Unit-3.

2. Results

2-1. Core degradation behavior before RPV depressurization (09:00, March 13th, i.e. 42:14 h after SCRAM)

Good agreement of the calculated RPV pressure and the water level with the measurement data was achieved by tuning the the flow rates of steam extracted by RCIC/HPCI turbine from the core and the flow rates of water injected to the core by RCIC/HPCI pump (see Fig. 1). Water level history shown in Fig. 2 suggested that water level should have stayed above BAF before RPV depressurization.

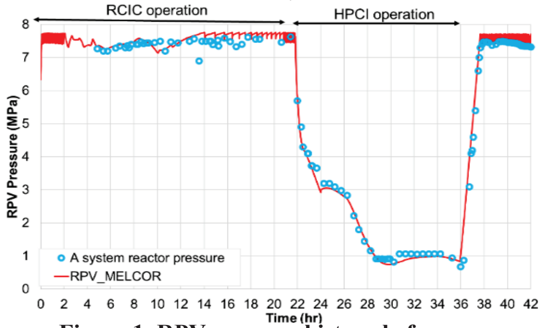


Figure 1. RPV pressure history before RPV depressurization

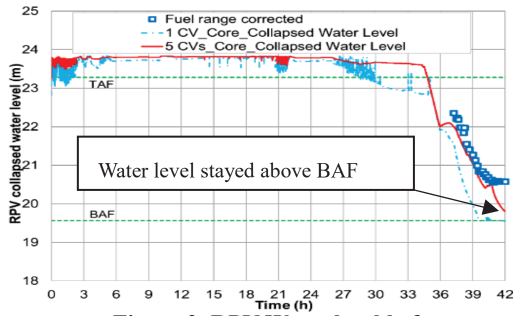


Figure 2. RPV Water level before RPV depressurization

2-2. RPV water level history before depressurization

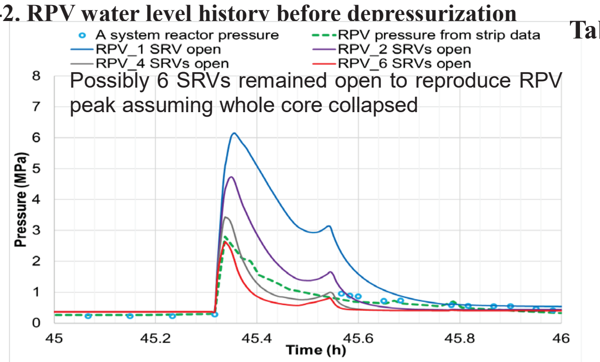


Figure 3. Sensitivity analysis of RPV history with different opening SRVs

Table 1 Summary of lower head failure time and RPV water inventory conditions in sensitivity cases of

Case	Lower head failure time (h)	RPV water level at lower head failure (m)	Lower plenum water mass at lower head failure (kg)	Total discharged debris mass into pedestal (ton)
100%	No failure	N/A	N/A	N/A
75%	53.25	23.47	9360.70	19.19
50%	51.04	19.43	1021.28	19.44
25%	51.83	15.28	168.90	19.46

3. Conclusion

- 6 SRVs opened during RPV depressurization at ca. 42:09 h after SCRAM and possibly remained open when the major core slumping took place with the current MELCOR modeling conditions.
- Freshwater injection reaching RPV is most likely to be no more than 50% of that reported.

F01

Fuel Debris cutting and aerosol collection for Fukushima Daiichi Decommissioning

Damien Roulet[1], Rémi Delalez[1], Christophe Chagnot[2], Ioana Doyen[2], Christophe Journeau[2],Christophe Suteau[2], Emmanuel Porcheron[3], Thomas Gelain[3]

[1]ONET Technologies, [2]CEA, [3]IRSN

Abstract: A French consortium studies laser-cutting technology for Fukushima Daiichi fuel debris. Dedicated laser heads have been optimized and validated on simulants. Preliminary investigations of on-site implementation of such cutting systems have been conducted. Two approaches are being studied to mitigate the risk linked to the generated radioactive aerosols during laser cutting: spray scrubbing and local collection near the cutting point.

1. Introduction

Fuel debris in Fukushima Daiichi must be cut into pieces before their retrieval. A French consortium (ONET, CEA, IRSN) has been studying laser-cutting technology for cutting of fuel debris [1]. This project is in line with the implementation experience in UP1 French reprocessing plant decommissioning. Laser cutting has a great advantage compared to many cutting techniques for its deployment by robots and remote systems. Tools are compact, without contact with the material to be cut, and well fitted for being deployed by remote manipulators in extreme conditions. Figure 1 sketches a possible implementation to cut ex-vessel fuel debris.

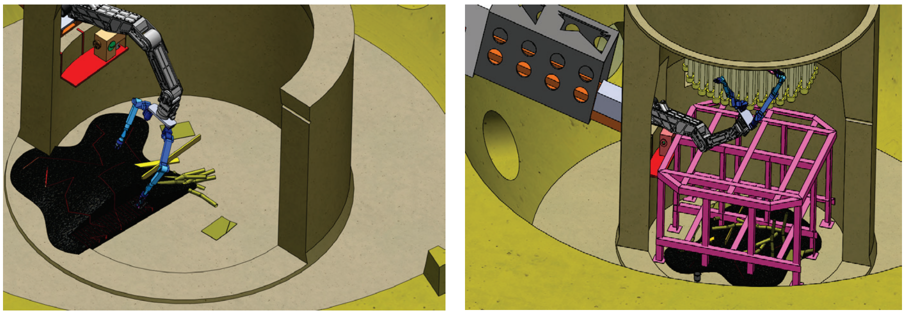


Figure 1: Example of laser head deployment in a 1F pedestal [1]

2. Laser cutting technology

Dedicated laser heads have been optimized for in-gas and underwater applications. Experimental validations on metallic and ceramic fuel debris simulants have been successfully carried out.

3. Aerosol collection

One of the important safety issue is linked to the generated aerosols. Two approaches are being studied to mitigate the risk linked to the generated radioactive aerosols during laser cutting.

One concept is the use of spray scrubbing to significantly reduce the aerosol concentration. This technology has been investigated and the preliminary results show good performance for a possible implementation during fuel debris retrieval [3]. Another approach is related to local collection of aerosols near the cutting point.

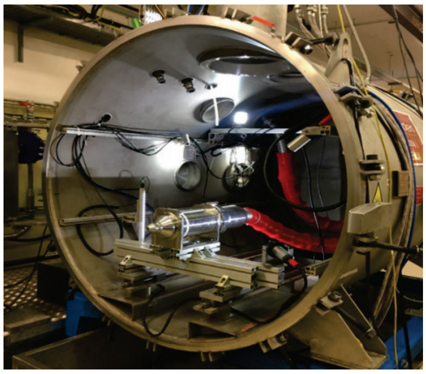


Figure 2: Laser head for underwater cutting in the DELIA test facility [2]

References:

[1] D. Roulet et al., Development of innovative laser cutting technology for fuel debris retrieval by robotic systems, FDR2019, J-Village, Naraha, Japan, 2019.

[2] C. Journeau et al., Fukushima Daiichi fuel debris simulant materials for the development of cutting and collection technologies, J. Nucl. Sci. Technol. 55:9, 985-995, 2018.

[3] E. Porcheron et al., Study of spray scrubbing as mitigation means against aerosol dispersion during the Fukushima Daiichi fuel debris retrieval, ICONE27, Tsukuba, Japan, 2019.

Acknowledgements: This work has been carried out thanks to subsidized project of Decommissioning and Contaminated Water Management (Advancement of Fundamental Technologies for Retrieval of Fuel Debris and Internal Structures) funded by the Japanese Ministry of Economy, Trade and Industry (METI) and managed by the Mitsubishi Research Institute.

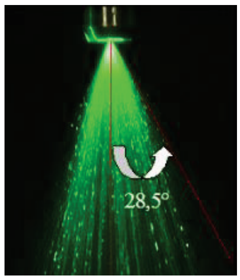


Figure 3: Spray scrubbing experiment in TOSQAN facility [3]

F02

Evaluation of the safety of geopolymer with respect to the hydrogen production JAEA – CEA Collaboration

Yoshikazu Koma[2], Vincent Cantarel[2], David Lambertin[3], Veronique Labed[3], Makoto Arisaka[2], Isao Yamagishi[2]
[2] Japan Atomic Energy Agency
[3] CEA – DEN – DE2D – SEAD - LCBC

Abstract
From 2019, a scientific collaboration is conducted between CEA and JAEA, aiming at enlarging the knowledge on hydrogen production of geopolymer wasteforms. The hydrogen recombination in geopolymers [1] will be further investigated to have a precise estimation of the effective hydrogen accumulation in realistic storage conditions.

1. Introduction
Recently, geopolymer gained interest for the treatment of nuclear wastes, due to their specific chemistry and their ability to retain Cesium. However, this matrix contains water, susceptible to produce hydrogen gas by radiolysis. For the immobilization of waste with non-negligible activity, accurate prediction of the hydrogen accumulation around the waste packages is essential to mitigate the hydrogen explosion risk.

2. CEA experiment
Geopolymer samples will be irradiated at CEA site Marcoule, using the Gammatec facility (Figure 1). It is possible to measure in situ the hydrogen production using an experimental setup designed in CEA. The hydrogen generated by the sample under ^{60}Co irradiation will be measured during the experiment in a laboratory contiguous to the irradiation cell.

3. JAEA samples and hydrogen recombination model
JAEA will prepare the samples for irradiation and diffusion experiment. JAEA will be in charge of the data treatment and the use of a model to extract the effective hydrogen production and recombination rates. At last, diffusion experiment will be carried out to obtain a precise value of the hydrogen gas diffusion constant in the geopolymer wasteform. Simulations in 3D will be used to estimate the hydrogen accumulation around geopolymer based wasteforms.

Figure 1 : Gamma irradiation facility (Gammatec – CEA Marcoule Site) with ^{60}Co source

References

[1] Cantarel, V.; Arisaka, M.; Yamagishi, I. *On the hydrogen production of geopolymer wasteforms under irradiation* J. Am. Ceram. Soc. 2019 00:1–11. https ://doi.org/10.1111/jace.16642

F03

Orano: Working in Partnership with Local Communities Over 50 Years of Regional Integration

Xavier VERDEIL, Maryline BRETON, Daphné OGAWA
Orano Group

Abstract

For many decades, Orano has been actively involved in promoting regional economic development and engaging with stakeholders at all of its sites worldwide. Orano's experience with the la Hague site: going beyond regulatory requirements for successful integration in the region.

1 – Introduction

With more than 5200 employees, the Orano Group is the number one industrial employer in the Normandy region. Approximately €400 million are spent in purchases from Norman companies each year.
The Orano La Hague site also participates in economic development in the region by actively supporting the creation of jobs in all sectors of the economy, including the Social and Solidarity Economy.
Furthermore, Orano La Hague engages in regular dialogue and consultation with all stakeholders.

2 – Involvement of Orano Group: the Example of La Hague site

2-1 – Dialogue with Stakeholders

The Orano La Hague site communicates on a regular basis with its external stakeholders, at the Local Information Commission for instance, or using the Nuclear Safety Transparence report. Thousands of visitors come to the site every year, including many journalists and decision-makers. The site also regularly exchanges with regional actors and elected officials, communicates news from the site to the press or on social media, and participates in public meetings.

2-2 – Attractiveness and Employment

Orano La Hague has established close connections with certain universities and schools in the region and with all employment actors, including the main occupational branch. It has contributed to the launch of several training programs and employees teach specific courses. Orano La Hague participates in many professional or job fairs for a variety of audiences (schools, universities, job seekers). In 2018, the site organised its own job fair, welcoming over 4000 visitors.

2-3 – Supporting the Nuclear Industry

Orano La Hague is a founding member of Nucleopolis by Normandie Energies, a cluster dedicated to economic development of the Norman nuclear industry. It therefore participates in various actions organised in the sectors of business (Norman Nuclear Industry Business Day, Focus'Business Day on the site, etc.), innovation and technical skills, such as the operational excellence label, Exc'op, which Orano sponsors.

Figure 1. BtB meeting as part of the Focus'Business Day with suppliers

2-4 – Participation in the Local Economy

Orano is a partner in organising several sporting and cultural events in the local community. It is also a member of economic development structures: the Alizé Manche Committee (which provides small companies with free access to the skills of larger companies) or Normandie Incubation, a startup incubator.
To encourage employment opportunities, Orano provides financial support to local companies and non-profits, including those participating in the Social and Solidarity Economy. Since the end of 2016, around 220 local jobs were created with the support of €500,000 from Orano.

F04

Investigation and Characterization in hostile nuclear environment

Kenny Le Flanchec, Cédric Escoffier, Simon Haen, Sébastien Bouzat, François Catsivelas, Frédéric Chambon, Daphné Ogawa

Orano Group

Abstract

Investigation and Characterization are key activities to build robust Decommissioning scenario. Orano has designed a significant number of innovative solutions and successfully implemented them in very complex environments. The characterization and the sampling of high activity deposit settled at the bottom of fission product evaporators is one of those Orano solutions which are currently being adapted to perform corium (Fuel debris) sampling for Fukushima reactors.

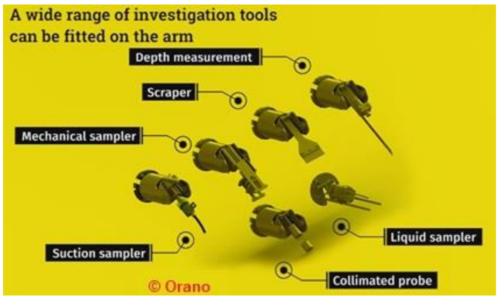
1. Introduction

Collecting reliable input data is essential to build robust decommissioning scenario and define waste routes. Therefore, many innovative solutions have been designed by Orano and successfully implemented in very hostile environment, such as the investigation within fission product evaporators.

2. Investigation within fission product evaporators at the CEA Marcoule site

Between 2016 and 2018, on behalf of the French Atomic Energy Commission (CEA), Orano DS team conducted investigation work at Marcoule on three evaporators that were operated from the 1960s to the 1990s to concentrate fission products due to the fuel reprocessing operations. The first step was to core drill through a one-meter thick heavy concrete block wall and then drill through the evaporator to access inside of the vessel. A dedicated technology of trepanning was developed to cut the 12 mm thick Uranus steel vessel.

Different kind of investigation had to be performed: visual inspection, dose rate measurement, smear, vessel bottom head deposit depth measurement, deposit sampling. Due to the highly radioactive environment (dose rate from 1 to 10 Gy/h at the vessel bottom), Orano designed, procured and tested two multitask mechanical robotic arms able to carry customized tools like depth measurement device, scraper, mechanical sampler, suction sampler, collimated radiological probe, from top and side access.



To limit the ambient dose rate at the operating position and meet the dosimetry objective initially defined for the workers, a new biological protection called IRIS (Investigation Radiological Innovation Safety), designed in the framework of Orano's R&D activities, was applied for the first time. IRIS is a diaphragm-like radiological shield which allowed Orano to divide by ten the integrated dose by our team and thus meet the ALARA objective. The project is now successfully completed. As specified by our client, nineteen deposit samples have been taken from the three evaporators and more than one hundred radiological and physical measurements have been performed. Forecasted operator radiation exposure was respected.

3. Fuel Debris (corium) sampling tool for Fukushima reactors

Based on its large experience in the design of various sampling tools and their implementation in several projects like the Marcoule evaporator one, Orano is now adapting a specific solution to meet the need of characterization of fuel debris fragment within Fukushima reactors. Indeed, in cooperation with MHI within IRID project, Orano is currently performing the design of a grasping-type device to recover several samples of corium fragments. Conceptual design and preliminary trials with 3D-printed sampling tools have already been successfully completed. A new phase concerning basic design and steel body prototyping is ongoing.

4. Conclusion

Orano D&D teams have acquired over 20 years of hands-on experience in investigation and characterization in highly radioactive environment. Innovative robotic equipment was designed, tested and successfully implemented to support the D&D program of the first generation of French reprocessing plants. These proven solutions can now be shared and adapted to the needs of new challenges such as Fukushima-Daiichi.

F05

TEΩ600, the industrial solution for robotics in hostile nuclear environment (up to 1MegaGray cumulated dose)

Alain Coudray[1], Xavier Verdeil[2], Daphné Ogawa[3]
[1] Orano Temis, [2] Orano Cycle, [3]Orano Japan

Abstract

The TEΩ600 is a modular, economic and robust solution, based on an industrial arm to guaranty performance, reliability and spare part availability. TEΩ600 is designed for high precision and dexter distant work using remote control with force feedback, and for complex and/or repetitive tasks in automatic mode by running programmed cycles. A wide range of tool (handling, cutting, cleaning, machining, welding…) can be used with TEΩ600 thanks to its performant payload of 60kg in all its workspace. Embedded on a mobile carrier, it can be easily adapted for Fukushima requirements with a long-distance remote control station (several kilometers).

1. Introduction

Decommissioning works at 1F site are very challenging in terms of dose rate, contamination risk and access difficulty. The TEΩ600 is a telerobotics solution with force feedback technology designed for remote operations requiring sensitivity and contact-like control to be performed in hostile environment as nuclear sites.

D&D projects are often singular and have restraint budget, thus, adapting off-the-shelf industrial robot allows proposing reliable solutions at low price with little development.

2. Innovation

Based on Orano Temis extensive experience, TEΩ600 is specially designed such as follows for nuclear environments:

- Radiation hardening at high cumulated dose
- Improved maintainability (rapid, semi-automatic) in hostile environment (contaminated, high dose rate, limited space and time)
- Management of the contamination (avoid contamination insertion, decontamination)
- Evolved handling of malfunctions
- Accessibility to target area thanks to the use of a robust and powerful industrial crawler
- Improvement of operability (visibility, sensors, 3D model vision), etc.

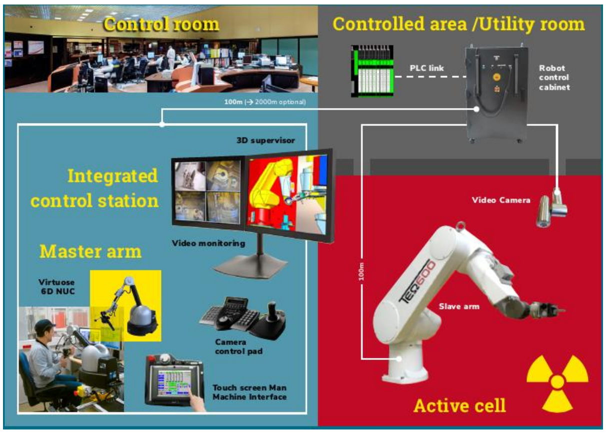


Figure 1: TEΩ600 Architecture

3. Key features

- Cartesian position and force-feedback Master-Slave control
- Prevention of internal contamination by air purging to the casing
- Surface smoothing and dedicated for easy decontamination
- Water proof, dust proof
- Equipped with cameras in the proper position
- Can be attached to various tools by modular design
- Able to be embedded to various carrier as lifting crane, or crawler
- Able to be operated with protect gear and mask

Figure 2: TEΩ600 on a mobile carrier (MC-TEΩ600)

4. Conclusion

Orano has acquired unique experience in remote solutions for performing D&D projects. Orano can propose integrated solutions (soft and hard) for Fukushima-Daiichi and its ever more demanding challenges ahead.

Figure 3: ATOX, ANADEC and Orano collaboration: Operator training and operation tests type for Fukushima (ATOX, Kashiwa)

F06 Tools and technical means for under-water decommissioning in highly contaminated environment

Jean-Michel CHABEUF[1], Malo DIENNET[2], Alexandra SYKORA[3], Daphne OGAWA[4]

[1]Orano Dismantling & Services, [2]Ecole Centrale Supelec, [3]Orano GmbH, [4]Orano Japan

Abstract

Segmentation of Reactor pressure vessels and their internals have been performed by Orano on a variety of reactors in Germany and the United States Mainly. The underwater segmentation of core waste is also routinely performed with industrialized solutions. Water purification is a key success factor for under water segmentation. Finally, innovation allows improving reliability and safety of segmentation operations.

1. Introduction

In the areas of the remote segmentation of reactor pressure vessels (RPV) and internals, as well as associated metallic structures such as control rods or fuel racks, Orano has developed and deployed successful technologies since the end of the nineties for nuclear power plants and research reactors worldwide, delivering projects on time and within budget.

2. Remotely operated under water segmentation

2.1 RPV Internals & RPV

Dismantling, conditioning and packaging activities of RPV and internals are performed with proven remote-operated technology and underwater**.** Operations have been conducted in several BWR and PWR NPP such as Würgassen, Stade, Mülheim-Kärlich, Brunsbüttel, Phillipsburg and Neckarwestheim in Germany.

2.1 Pool internals and core waste D&D

Orano has delivered the segmentation of core waste in several reactors, such as Biblis or Isar 1. In the latest project at Brunsbüttel facility, controls rods, neutron flux measuring lances, fuel element racks, a basket with technological waste and filter cartridges, have been cut and packed as shown on the figure 1.

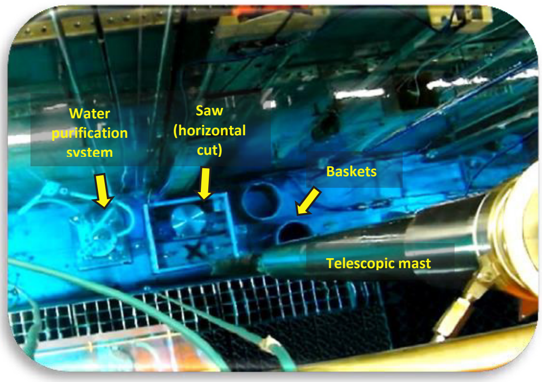


Figure 1 : Control rods

2.2. Water filtration and decontamination

Figure 2 : Rough filtration

Figure 3 : Fine filtration

The setting up of an auxiliary filtration and decontamination system is often necessary to decrease the visibility loss due to particles in suspension. During the D&D activities of Orano, both fine and rough filtration are used (resp. figure 2 and 3). The first one is a flexible machine with several setting possibilities allowing to filtration of fine particles of only 2 µm; and the second one is a rough filter which carries out particles of 20 µm through hydrocyclones, to be separated in a basket retrieved remotely.

3. Conclusion

Industrialized and proven segmentation and filtration solutions are deployed on a variety of projects, and Orano continues innovation to further improve the timely delivery and safety of its projects.

F07

DEM&MELT In-Can Melting Technology for the Vitrification of D&D and Remediation Waste

Regis Didierlaurent[1], Thierry Prevost[1], Daphné Ogawa[1], Hubert Alexandre Turc[2], Kohei Shibata[3]

[1]Orano, [2]CEA, [3]ANADEC

Abstract

The DEM&MELT In-Can vitrification process is designed to process intermediate and high level waste coming for D&D operations. It is designed to treat liquid and solid waste, to produce a small amount of secondary waste and to minimize investment and operating costs. This In-Can vitrification process is also developed with a compact and modular design and can be adapted regarding to nuclear operators' needs and requirements. Furthermore, this robust and simple process is flexible enough to accommodate most uncertainties on waste composition. Considering the DEM&MELT features, this process is deemed as an appropriate tool for the vitrification of the Fukushima Daiichi water treatment secondary waste. The DEM&MELT process is developed through the French consortium which gathers Orano, CEA, ECM technologies and Andra.

1. Introduction

Following the earthquake and the tsunami on March 11th 2011 in Fukushima, large quantities of contaminated water have been generated due to the cooling of the damaged reactors at Fukushima Daiichi Nuclear Power Plant (1F). Several water treatment systems have been used to decontaminate this water and have led to the production of many secondary solid wastes so called FETW (Fukushima Effluent Treatment Waste). These waste, mainly contaminated with cesium and strontium, are temporarily stored, waiting for a process to stabilize them for final disposal. ANADEC, Orano and CEA have proposed to stabilize the FETW using the In-Can vitrification technology. A dedicated tests program on Fukushima Daiichi wastes, supported by Japanese government funding, has started in Japanese FY2018 and will continue until FY2020 in the framework of The Subsidy Program "Project of Decommissioning and Contaminated Water Management". The main goals are to demonstrate the feasibility and to provide a first preliminary design of an industrial unit adapted to 1F waste treatment.

2. Evaluation of the In-Can melter for Fukushima water treatment secondary waste – Main results

To demonstrate the FETW treatment feasibility by the In-Can vitrification technology, a step-by-step work process has been implemented. The first step has consisted in studying various waste assembly scenarios and glass matrix formulation by smartly optimizing treatment time, cost and final volume of waste packages. During this step, glass samples were elaborated at small scale and characterized (~100 g of vitrified material for each sample). It has been shown that it is possible to obtain a dense vitreous matrix with high waste loadings, ranging from 50wt% up to 80wt% depending on the waste scenario assembly. In a second step, bench scale tests have been performed on intermediate scale mock-ups (~1kg of vitreous matrix). They have confirmed the feasibility of the various scenarios and have led to the determination of a first set of operating parameters. In the final step, a pilot scale demonstration test has been successfully performed on the In-Can-Melter vitrification pilot (~100kg of glassy material) for a reference scenario with representative non-radioactive surrogate waste. Analysis of glass and off gas samples performed during pilot test allowed assessing cesium behavior in the process. The waste is well incorporated into the vitrified material. The efficient temperature control and the highly efficient Off Gas Treatment System allowed an efficient volatility management for Cs, which remains lower than 0.5 wt%.

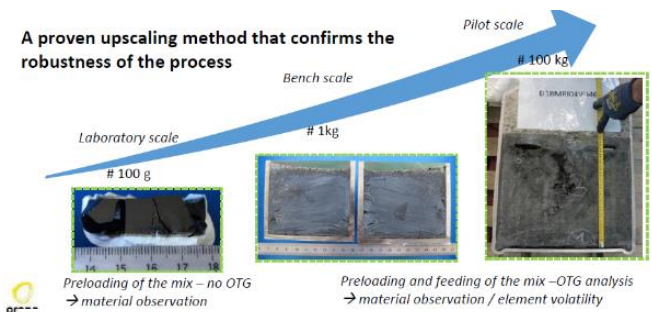


Figure 1. R&D method from laboratory to pilot scale tests

3. Conclusion

An evaluation of the In-Can vitrification process industrial implementation and the associated waste management was performed. The feasibility of the In-Can vitrification process for FETW has been demonstrated for 90 % in volume of the total inventory and 98.6 % of the total activity during FY2018 study. A complementary R&D Program is currently ongoing to complete the demonstration that the In-Can vitrification is fully adapted to the FETW characteristics.

F08 New peelable coating RTV FA 878 when dismantling the nuclear facilities

J.P. Thome, BCSN & M. Sawamoto, eEnergy Corp.

Abstract: newly developed coating now used at nuclear facilities for the purpose of surface coating against contamination spread-out, strengthening leak-tightness as well as decontamination

1 Characteristics of coating:

Translucent elastomeric compound

Low Hardness Shore A (28)

Sprayed with instant cure without linear shrinkage

Good mechanical properties (tensile, strength, elongation at break, tear, resistance)

Outstanding lifetime under irradiation

Injectable underwater

Easy retrieval

Self repairable

2 Use

Protective flexible skin against shocks or contamination

Reinforce depression tightness of airlocks & vent ducts

Fixing of loose contamination & high efficiency decontaminating strippable coating

Waterproof coating for submerged equipments in pools

Tightness coating against corrosion

3 Conclusion

We hope this material will be helpful for the cases at Fukushima site where there are needs against dispersion of contamination, for leak-tightness, and for decontamination.

F09

Introduction of ROBATEL Industries business and their experiences regarding hot cells project

Christopher DANE[2], Natalia ZOLNIKOVA[2], Masahiro Sawamoto[1] and Keiichi Inoue[1],

[2]ROBATEL Industries, [1]e-Energy Corporation,

Abstract

The company ROBATEL Industries is a French SME founded in 1830, whose activities are exclusively turned towards the nuclear industry for over 60 years. ROBATEL has developed a comprehensive range of skills and experience.

1. Introduction

The activities of the company cover both aspects "design" (studies, calculations, safety assessment…) and "production" (production, commissioning tests, maintenance, on-site interventions and loading assistance). Thanks to its long experience, ROBATEL is capable of putting its expertise to the service of major high stake projects for the industry; developing, case-by-case, hot cells (among other fields) and all their associated operating tools (remote handling equipment, control equipment and others...) to meet our customer specific needs and requirements.

2. Our experiences

Quantities of hot cells fabricated in recent history by ROBATEL:

Over the past thirty years, 46 hot cell facilities and cementing units were completed:

- More than 200 hot cells
- For 14 different clients
- In 8 countries

And more precisely since 2000:

- More than 45 hot cells
- 5 for different customers
- In 7 countries.

3. Conclusion

We hope above our experiences will be helpful for decommissioning program in Fukushima 1st nuclear power station.

F10

Mechanical Installations and Equipment for Decommissioning and for Waste Storage

廃止措置および廃棄物保管用機械設備および機器

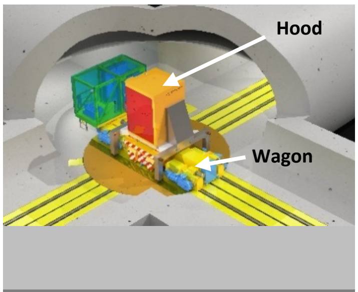
Craire PAOLONI1, Vincent GILLET1, Renaud BOURDY1, Masahiro Sawamoto2, and Keiichi Inoue2
Cegelec CEM1, e-Energy Corporation2

1. Introduction

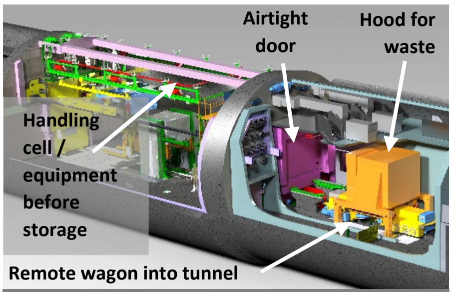
Cegelec CEM is a French company, belonging to VINCI Energies Corporation, specialized in mechanical processes, facilities and equipment for Decommissioning and Nuclear Plants operations.

2. Handling system for the French Future Final Disposal (CIGEO)

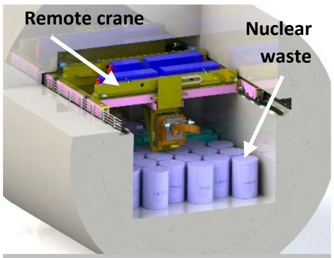
Cegelec CEM realizes concept design and studies of all the handling process equipment of high and medium activity nuclear wastes canisters **from the level 0 to the level -500 m (**under the ground level): cranes, waste hoods and maintenance facilities.



-500 m lower from level 0



Handling equipment under the ground



-500 m lower from level 0

3. Nuclear Waste interim storages and in-cell handling equipment

Cegelec CEM performs **engineering and turnkey projects for nuclear operators** in France and overseas, from studies to manufacturing, testing, and on-site installation and commissioning.

Cegelec CEM has **in house engineering skills** for **design** and **calculations** (mechanical, seismic, thermic and various nuclear calculations), to propose **innovative solutions** to its customers.

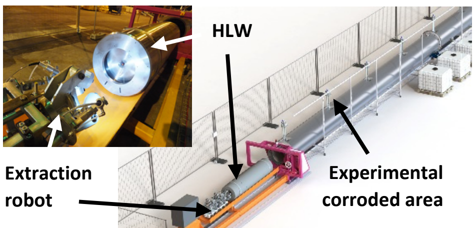

Waste lorry docking on cell


Equipment / handling system in cells

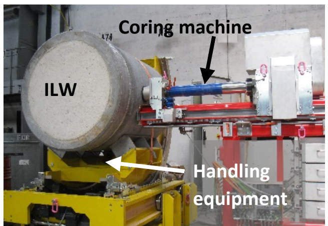

In cell carrying solutions

4. Demonstrators and specific equipment for nuclear activity


Pushing machine for HLW into disposal



Experimental remote extraction of HLW



Coring machine (waste sampling)

5. Conclusion

By its long track record in Nuclear Waste management in Europe, Cegelec CEM can bring to Fukushima decommissioning project its know-how and proven solutions.

6. References

www.cegelec-cem.com

GB01

Pile Fuel Cladding Silo Decommissioning Project – Collaboration

Sarah Mundy, Roger Cowton
Sellafield Ltd.

The Pile Fuel Cladding Silo (PFCS) was built in Sellafield in 1949/1950 for the purpose of receiving and storing the fuel cladding material from the Windscale Pile fuel reprocessing. Its secondary purpose was to store miscellaneous waste from the UK nuclear programme. Due to its construction, PFCS is unsuitable to store nuclear inventory for an indefinite period. Coupled with the unsuitable storage qualities, the uncertainty of the nuclear inventory causes PFCS to be one of the most hazardous nuclear facilities in Western Europe.

To mitigate the ongoing risk associate with operating PFCS, Sellafield Ltd have been designing and building a retrievals system. To manage the uncertainty associated with inventory and structure without the excess of a bounding design, the programme has devised a 2 step approach - early retrievals and full retrievals where early retrievals will use a lead and learn approach. The lead and learn approach allows feedback of knowledge regarding the waste, retrievals process and equipment gathered during early retrievals to be implemented within the full retrievals design. This aims to reduce design periods, failure rates and streamline the retrievals process.

To deliver PFCS retrievals, Sellafield Ltd have partnered with supply chain partners Bechtel Cavendish Nuclear Solutions (BCNS - a joint venture between Bechtel Inc and Cavendish Nuclear Ltd). Running a project with such high levels of uncertainty on a traditional client contractor basis proved ineffective. It was recognised that the strengths of all organisations working in a highly collaborative relationship wold be needed to deliver the work. The project was overhauled in 2015 to focus on collaboration; since then, the projected cost has reduced by £100m with early retrievals being delivered 17 months ahead of schedule.

These improvements would've not been achievable without collaborative working between Sellafield Ltd and BCNS, innovation solutions and a Fit for Purpose mind-set. Stakeholder confidence has been re-affirmed and Sellafield is on the brink of retrieving legacy waste from PFCS!

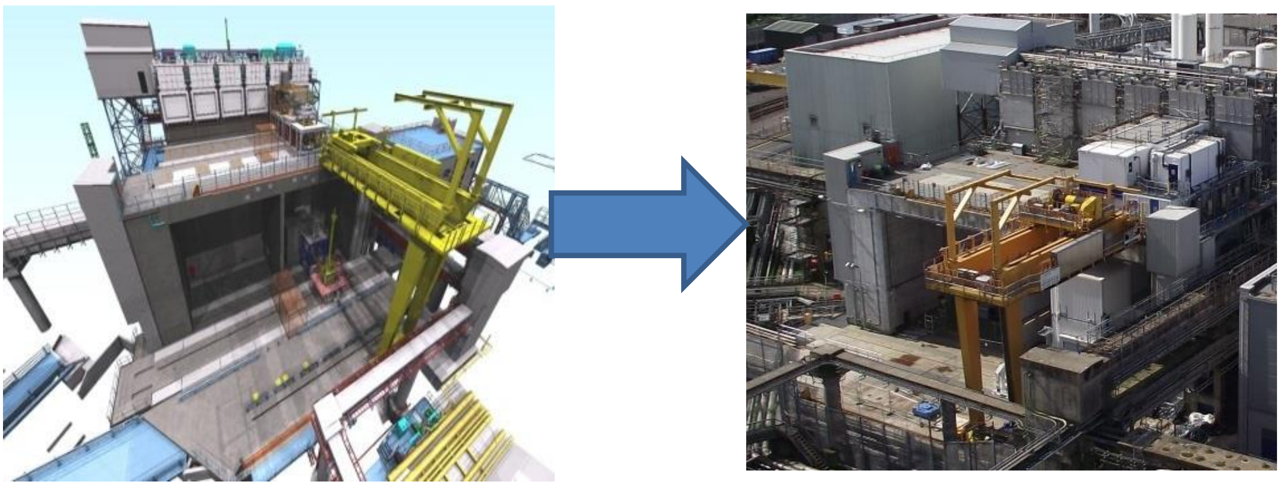


Figure 1 - The retrievals equipment - from a model to fully built.

GB02

Technologies for Detection of Contaminated Particles

Christopher Carter[1], Jonathan Britton[1] and Mark Rouse[2]
[1] Cavendish Nuclear Ltd, [2]Cavendish Nuclear Japan

Abstract

During decommissioning activities the area surrounding the decommissioning site is monitored to confirm that contamination has not been released into the environment.

Cavendish Nuclear has developed technologies and techniques that are used for monitoring the air and the ground for detection of very low levels of radiation that are associated with small particles of contamination.

1. Introduction

Cavendish Nuclear is an experienced operator and developer of radiological environmental monitoring systems. Cavendish Nuclear's expertise ranges from overall strategy development to investigation planning and implementation, laboratory testing, data interpretation, visualization using geographic information system (GIS), risk assessment and data/records management.

Hot particles with alpha emissions are of particular concern if they have entered the environment due to the high potential for harm if they come into contact with operational staff or public.

2. Technologies for Monitoring and Detection of Contaminated Particles

The poster will describe examples of equipment and techniques that can be applied for environmental monitoring at decommissioning sites, including the three technologies shown below.

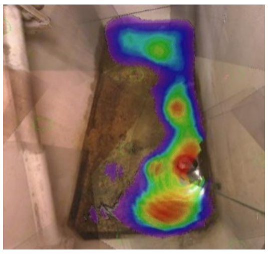
Large Area Radiological Characterization (LARCH)

The LARCH system deploys a high-resolution gamma-ray detector in conjunction with automated surveying technology to produce a series of spectra that are stamped with the date, time and GPS location from which they were acquired.

Environmental Monitoring and Characterization – RadScan®

RadScan is a portable gamma ray imaging system that remotely locates and characterizes gamma radiation from a wide variety of environments including building surfaces, land, cells, gloveboxes, drums and process vessels.

RadScan automatically scans and records gamma spectra from the measurement area and displays this as a colored overlay on a color video image. The image clearly shows where the radioactive contamination is located and its intensity.



Environmental Monitoring – Drop & Go Monitor

The 'Drop & Go Monitor' is a portable instrument which autonomously measures Alpha and Beta contamination and Gamma Dose Rate.

This radiometric data, together with positional and status information, is transmitted via the 3G/4G mobile phone network back to a secure cloud service which can be accessed from any location via a standard internet browser with access to the internet.

GB03 Integrating Waste Management with Project Management: the Key to Successful Decommissioning

Bill Miller, David Luckhurst, Darren Dudley, Steve Caley & Paul Fleming

Wood plc

Abstract

Decommissioning is a waste generation activity. But decommissioning projects are too often underprepared for the challenge of managing and disposing of wastes, leading to project delays and cost increase. At Wood we work with site owners to *integrate* decommissioning plans with waste management plans. This integrated approach delivers substantial safety and cost benefits, and can often lead to a significant reduction in the overall volume of waste to be disposed.

1. The challenge of waste management

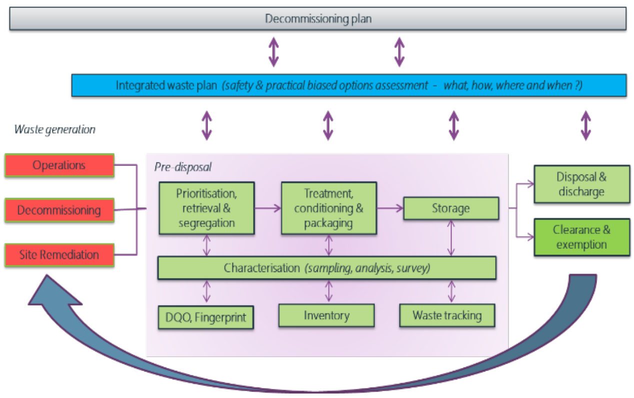
At its most simple, a decommissioning project is a waste generation activity that can produce very large quantities of many different waste materials, both radioactive and non-radioactive.

Unfortunately, there are several examples around the world of decommissioning projects that have underestimated the difficulties associated with waste management, and have failed adequately to plan and prepare. As a result, decommissioning projects are too often delayed because waste treatment and disposal facilities are not available when they are needed or are not suitable for the wastes that are actually generated. This can lead to schedule delays and greatly increased costs, and the loss of regulatory and stakeholder confidence.

Learning from experience, it is important to recognise that waste management (and disposal) is very often the rate limiting step on the critical path for a decommissioning project.

2. The need for waste management plans

It is important to prepare for decommissioning by first developing a robust decommissioning plan and work breakdown structure (WBS), and these must be underpinned by a sensible engineering approach and reliable estimates for cost and schedule.



However, it is recommended that the decommissioning plan is fully integrated with a detailed waste management plan. As a minimum the waste management plan should specify:

- an estimate of the waste inventory, identify all waste materials (both radioactive and non-radioactive) and their volumes
- the best available techniques (BAT) for characterisation, segregation, treatment, condition and packaging of each waste material
- the anticipated interim storage and disposal routes for each waste, and the acceptance criteria
- the infrastructure requirements, including any new facilities needed for all waste treatment, storage and disposal activities
- a reasonable estimate of the schedule for waste management and disposal (which may take decades), and the associated costs for each activity.

3. Integration of decommissioning and waste management plans

At Wood plc we work with nuclear sites to develop and integrate decommissioning plans with waste management plans. This integrated approach *(sometimes called waste-led decommissioning)* has delivered substantial benefits, including costs savings, by optimising work while at the same time reducing the overall volume of waste to be disposed.

For example, we have undertaken several projects to plan and implement waste retrievals and decommissioning of legacy silos on nuclear sites. Our work integrates all planning, licensing, options assessment, safety case, engineering design for retrievals equipment and waste immobilisation equipment, in some cases using Wood's own SIAL® geopolymer.

GB04 Innovative Cost Effective Solutions for In-Situ Liquid Waste Processing

Timothy Milner[1]
[1]ATKINS

Abstract

The poster details the concept of using a Rotating Bed Reactor (RBR) that was pioneered by Swedish company SpinChem and has achieved success in the pharmaceutical and other non-nuclear industries. Atkisn has teamed with SpinChem to develop the SpinIonic® RBR for use in the nuclear industry.

Outline of System

The Atkins and SpinChem team developed the nuclear application for the SpinIonic® RBR and is now actively demonstrating the technology at client nuclear sites. The RBR device retains the solid phase as a packed bed inside a rotating cylinder. As the RBR spins, a continuously circulating flow develops. Reaction solution is rapidly aspirated into the bottom of the RBR vessel, percolated through the solid phase, media, and quickly returned to the storage tank or vessel being treated. By the intelligent design of the SpinIonic® RBR, the axial mixing and convective transport are maximized. The resulting efficient mass transfer minimizes reaction time, boosts material capacity and increases process flow rates.

Conclusion

With the SpinIonic® RBR it is possible to minimize slow reaction kinetics caused by poor mass transfer between the solution and solid phase. The SpinIonic® RBR design is flexible and can be used for numerous applications in situ to treat tanks for radioactive waste or in process to treat continuous waste streams. Utilising the SpinIonic® RBR typically results in faster processes, higher decontamination factors or reduced generation of secondary waste, depending on the type of process. In addition, the SpinIonic® RBR extends the lifetime of the solid phase particles by minimizing grinding and fines, while at the same time simplifying the solid phase collection and recycling or disposal.

The importance of correctly identifying and managing the cost of decommissioning nuclear facilities

Gary Mills, Director, Gleeds Energy Limited, UK

ABSTRACT

It is important that we understand the real cost of decommissioning a nuclear facility, so the correct level of funding can be arranged and then correctly managed to ensure the activities are safely completed. This poster focuses on how the collection and application of detailed data has enabled Gleeds to provide accurate estimates that influence procurement and contracting strategies and enable effective management and control the cost.

SK01 External & Internal Dose Assessment on Spent Resin Treatment Facility for Recycling C-14

Woo Nyun Choi, Ukjae Lee, Bae and Hee Reyoung Kim
School of Mechanical Aerospace and Nuclear Engineering, UNIST, 44919 Ulsan, Republic of Korea

Abstract

External & Internal dose assessment are conducted to ensure radiological safety during the treatment and disposal of spent resin. This study performed a radiological evaluation of the operation of spent resin waste treatment facilities of Wolseong Nuclear Power Plant, a heavy-water reactor nuclear power plant in Korea.

1. Introduction

In the heavy-water reactor, ion exchange resin is used to purify the liquid radioactive waste generated while operating, about 10,000 L or resin is generated annually. The resin stored in the storage tank should be handled in accordance with future decommissioning plans [1].

2. Method and Result

2-1. Modelling the spent resin treatment device

The spent resin treatment device which is a demonstration scale (10 L) (Fig.1.) is modelled based on basic structures which are supported by VISIPLAN (SCK-CEN, Belgium). The device was modelled with stainless steel material.

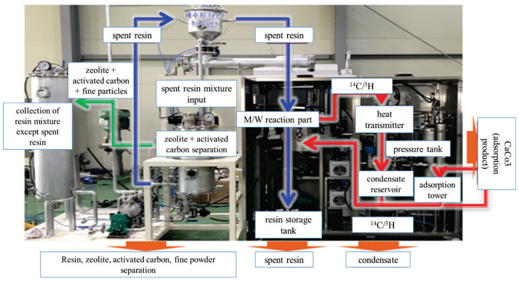


Fig. 1 Actual resin treatment device

2-2. Assessment of external and internal dose

The 100 % leakage of radionuclides is assumed. The worker radiation exposure dose was calculated in accordance with the worker scenario using VISIPLAN. A typical radiation worker's annual working time of 2,000 h was used to derive the annual worker dose. And the internal exposure dose was evaluated based on the activity concentration of the actual spent resin from Wolseong nuclear power plant.

2-3. Result

For 1 h of working time, the received dose was calculated as 5.3E-05 mSv/h. The typical annual working time of radiation workers (2,000 h) was used to derive the annual worker dose based on the received dose per hour. The derived dose value was 0.106 mSv. The ^{137}Cs had the largest influence on the external dose value at 6.30E-02 mSv. And the value of internal dose of for the first year was 10.34 mSv.

3. Conclusion

In evaluating the external & internal exposure of workers to radioactive waste from the spent resin treatment device, the resulting radiation dose was 10.446 mSv, which is below the Korea annual worker dose limit of 50 mSv. Thus, the radiological safety of workers in the spent resin treatment process was confirmed.

References

[1] Fiskum SK, Blanchard DL Jr., Steele MJ, Wagner JJ. Analysis of spent SuperLig® 644 resin used for cesium removal from Hanford tank wastes. *Solvent Extra Ion Exc*. 2006;24(1):65-79.

SK02 A Transportable Ground Radiation Monitoring System for Decision-Making on Release of Nuclear Decommissioning Sites

Chanki Lee, Hee Reyoung Kim
Department of Nuclear Engineering, Ulsan National Institute of Science and Technology

Abstract

We develop a transportable ground radiation monitoring system, which adopts Compton suppression method with NaI and PVT scintillators. In order for the system to be efficient to making decision on the release of nuclear decommissioning sites, we simulate scan minimum detectable concentration (MDC) related to a preliminarily derived concentration guideline level (DCGL) of specific site [1], that is, Kori-1 nuclear power plant (NPP).

1. Introduction to the monitoring system

The system is based on a trailer as well as a gamma spectrometer that analyzes signals coming from a main detector and a guard detector. The main detector is composed of two NaI scintillator, and the guard detector is of three PVT scintillator. Using anti-coincidence circuit, the system reduces background radiation from Compton scattering.

2. Performance evaluation methods

2-1. Detection efficiency

Two cases are set to evaluate detection efficiency with distance from 0.1 m to 0.5 m: Surface contamination and vertical contamination with different depth profiles. Depth profiles are assumed by using Gaussian distribution [2] of MCNP code.

2-2. Estimation of suppressed background

Compton suppression factor (CSF) is calculated to quantify how much background counts are suppressed. Actual data are used to select proper energy bin of MCNP PHL card.

2-3. Scan MDC calculation

By using conservative assumptions from evaluated performances, scan MDCs are calculated. Especially, those of ^{137}Cs and ^{60}Co isotopes are compared with their DCGLs of Kori-1 NPP.

3. Conclusion

Simulation results of scan MDC show that the proposed system should be operated at velocity below 3 m/s, which is the maximum optimal velocity for application to Kori-1 NPP decommissioning. Overall, the results imply that the system can possibly be a rapid but integrated (i.e., gamma radioactivity and dose rate) evaluation tools on site. In addition to the simulation results, more strict validation based on actual test should be conducted.

References

[1] Abelquist, E.W., 2001. Decommissioning health physics: A handbook for MARSSIM users. Taylor & Francis.

[2] Waters, L.S., 2002. MCNPX User's Manual. Los Alamos National Laboratory.

SK03 Dose calculation of dismantling workers for bio-shield of Nuclear power plant

ChoongWie Lee and Hee Reyoung Kim
Ulsan National Institute of Science and Technology

Abstract

The doses of the workers at the time of bio-shield dismantling with various radioactivity distributions were evaluated. The working scenario for dismantling of Bio-shield a was derived, and the total dose during dismantling Bio-shield were evaluated accordingly. In case of external exposure, VISIPLAN, a dose assessment tool, was used. In case of internal exposure, the amount of dust production was calculated, and the dose was calculated.

1. Introduction

The Kori Unit 1 in Korea is planned to be closed and decommissioning accordingly. In the situation of dismantling of nuclear power plant, evaluation of worker's dose is essential for securing worker's safety. A radiological safety assessment is required for the safety of workers and authorization in the decommissioning.

2. Methods and Results

2-1. Dose calculations methods

Since there is no current radioactivity information for Kori Unit 1, it is necessary to calculate the current radioactivity distribution based on the initial conditions of the core. For this purpose, the radioactivity distributions for the five major radionuclides (^{3}H, ^{55}Fe, ^{60}Co, ^{152}Eu, ^{154}Eu) were calculated using MCNP code using information such as neutron flux, nuclide concentration, and operation history. Then, the dose rate was calculated using the dose assessment tool, VISIPLAN, and the external exposure was evaluated according to the dismantling work scenario. The internal exposure was calculated by modeling the amount of dust production by concrete dismantling and taking into consideration the dose coefficients.

2-4. Results

The results of simulating the dose rate inside the radioactive concrete using VISIPLAN are shown in Fig 1. As a result of the dose evaluation, the external exposure of the worker was 80 mSv and the internal exposure was 3 mSv. It was shown that the external exposure was more affected than the internal exposure.

3. Conclusion

As a result of the dose evaluation, the calculated dose was more than annual dose limit in Korea (20 mSv/y). Therefore, it is expected that protective measures such as remote control and protective clothing will be needed for the safety of workers dismantling the bio-shield.

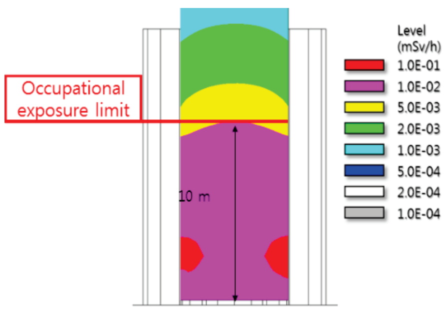


Figure 1. Dose rate at Bio-shield

References

[1] D. Lee, H.R. Kim, Preliminary Source Term Assessment on Kori unit 1 Radioactive Bioshield for Decommissioning, Transactions of the Korean Nuclear Society Spring Meeting, (2018).

SK04

Comparison of Detection Efficiency according to the number of Scintillator in Continuous Tritium Monitoring System

Jae Hoon Byun, Ki Joon Kang and Hee Reyoung Kim

Department of Nuclear Engineering, Ulsan National Institute of Science and Technology, 50, UNIST-gil, Ulju-gun, Eonyang-eup, Ulsan, 44919, Republic of Korea

Abstract

The detection efficiency was compared according to the number of scintillators inside the chamber used in the continuous tritium monitoring system. The 13-slot chamber's efficiency is 13.12% and that of 7-slot chamber is 19.41%. The efficiency of the 7-slot chamber was 6.29% higher than that of 13-slot chamber.

1. Introduction

In decommissioning sites, underwater tritium should be monitored. The energy generated by tritium is so low that the detection efficiency is low due to the self-absorption [1]. So, the tritium monitoring system using electrolysis was used. In this study, the detection efficiency was analyzed for both conditions of the detection chamber.

2. Materials and Methods

2-1 Experimental Setup

To compare the detection efficiencies of 13-slot chamber and 7-slot chamber, the study was carried out. The 170 kBq/m^3 of tritiated gas was produced and measured for 3 mins.

Figure 1. 7-slot detector chamber

2-2 Comparison of two chambers

Comparing the characteristics of the two chambers, the 13-slot chamber has a relatively large number of scintillators, but the amount of gaseous tritium in the chamber is small. In contrast, the 7-slot chamber has a relatively small number of scintillators, but the amount of gaseous tritium in the chamber is large.

2-3 Results

13-slot chamber's average efficiency is 13.12%, 7-slot chamber's average efficiency is 19.41%. The efficiency of the 7-slot chamber was 6.29% higher than that of 13-slot chamber. Of course, as the number of scintillators decreases, the volume of the scintillators decreases, but, it was found that the amount of tritiated gas inside the chamber influenced the detection efficiency more. Also, 7-slot chamber is cheaper than 13-slot chamber.

3. Conclusion

The detection efficiency was compared according to the number of scintillators inside the chamber. The efficiency of the 7-slot chamber was higher than 13-slot chamber. So, the device with simple composition, light mass and low-cost for practical applications can be used in real time detection in decommissioning sites. In the future, MDA calculations should be carried out, and further studies should be conducted to increase the efficiency.

References

[1] Soreefan, Aurelie M., and Timothy A. DeVol. "Proportional counting of tritium gas generated by polymer electrolyte membrane (PEM) electrolysis." Journal of radioanalytical and nuclear chemistry 282.2 (2009): 517.

SK05 Real-time Monitoring of Tritium in Water with Enhanced Detection Efficiency

Jun Woo Bae, Ki Joon Kang and Hee Reyoung Kim
Department of Nuclear Engineering, Ulsan National Institute of Science and Technology, 50, UNIST-gil, Ulju-gun, Eonyang-eup, Ulsan, 44919, Republic of Korea

Abstract

A plastic scintillation-based tritium detection chamber was improved. The volume of the chamber was enlarged and designed with consideration of higher detection efficiency. The maximum detection efficiency and minimum detectable activity (MDA) for tritiated water was evaluated as 43±4.3% and 3.4 kBq/L, respectively.

1. Introduction

Tritium contamination is one of major concern of Fukushima Daiichi Nuclear Power Plant site [1]. In previous study, an electrolysis-based real-time tritium detection system was developed to monitor the abundance of samples and constant monitoring of tritium in water. The MDA of the system was 17.8 kBq/L for 15,000 s of measurement time, but it was quite high to be used as a low-level contamination monitor for aquatic environment. In this context, the detection chamber was improved to lower the MDA.

2. Methods and Results

Figure 1 shows the improved tritium detection chamber. The volume of the chamber was 32 times increased comparing with the previous study. To increase the detection efficiency, 13 pieces of plastic scintillators were placed parallelly. The 170 kBq/m^3 of tritiated gas was produced by electrolyzing 82 kBq/L tritiated water sample. The produced gas was measured for 3 mins1). The maximum detection efficiency and MDA was evaluated as 43±4.3% and 7.0 kBq/m^3 for tritiated hydrogen gas, or 3.4 kBq/L as a concentration of the tritiated water sample.2) respectively. The MDA was 5.28 times lowered comparing with previous study where the measurement time was 50 times shorter.

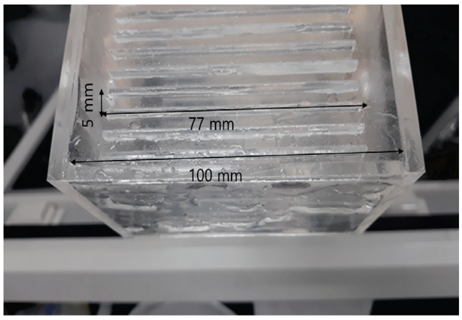


Figure 1. Improved tritium detection chamber

3. Conclusion

Tritium detection chamber was improved which was used in the electrolysis-based real-time tritium monitoring system developed in previous study. The tritium detection chamber was newly designed, and it showed better efficiency and lower MDA. Though the improved detection system does not satisfy an average tritium concentration of north west Pacific Ocean (~0.1 Bq/L), it is expected to be applicable for regular monitoring of release of contaminated water during decommissioning of the Fukushima Daiichi Nuclear Power Plant.3)

References

[1] Kumamoto, Yuichiro, et al. "Impact of Fukushima-derived radiocesium in the western North Pacific Ocean about ten months after the Fukushima Dai-ichi nuclear power plant accident J. Environ. Radioact. 140, 114-122 (2015).

SK06

The Measurement and Comparation Analysis of Change of Radioactivity Concentration of the Water Containing HTO Molecule by Proton Exchange Membrane Electrolysis

Ki Joon Kang, Jun Woo Bae, Hee Reyoung Kim

School of Mechanical, Aerospace and Nuclear Engineering, Ulsan National Institute of Science and Technology, Ulsan 44919, Republic of Korea

Abstract

The liquid tritium solution should be electrolyzed by proton exchange membrane (PEM) cell. And the change of radioactivity concentration should be checked. The change before and after electrolysis can be detected by liquid scintillator counter (LSC), quantulus 1220 from Parkin Elmer [1]. The change of liquid tritium mass should be also measured by electronic scale. And then finally consumed radioactivity of liquid tritium can be calculated.

1. Introduction

In decommissioning sites of nuclear power plant, underwater tritium should be monitored and protected by safety monitoring system. Existed detection method that need a long time for sampling, and treatment cannot deal with realtime monitoring. However, if PEM cell is used in decommissioning sites of nuclear power plant and gets steady power supply, realtime onsite monitoring of underwater tritium can be detected.

2. Method and Result

2-1. The Change of Radioactivity Concentration

LSC cocktails using liquid scintillator, liquid scintillator and the masses of the water containing HTO before and after electrolysis are measured. The change of radioactivity before and after electrolysis is same with the number of HTO molecules electrolyzed. The current condition of electrolysis is changed by 2 A, from 9 A to 5 A. For 45 minutes, radioactivity (Bq) of the water containing HTO from each current were changed from 535970, 476197, 455842 to 480810, 460586, 440269. The ratios of HTO molecules electrolyzed to total HTO molecules were 10.29 %, 3.27 %, 3.42 %. These results can be used as reference value for the calibration of realtime detection by betaionix monitor. And because the ratio of difference is not large, the results can be used to detect tritium only.

2-2. The comparation with PMT measurement

HT gases were moved as follow **Figure 1**. And it was also detected by using plastic scintillator and photomultiplier. After detection chamber, HT gases are measured by beta ionix monitor. The radioactivity by betaionix monitor was 25.4 Bq, 12.7 Bq, 11.8 Bq. The detection efficiencies were 35.01 %, 13.04 %, 6.43 %. It was a higher than the number of HTO molecules electrolyzed per 1 second, 20.43 Bq, 7.49 Bq, 7.02 Bq.
However, the calculation values of detection efficiency by LSC, 43.56 %, 22.70 %, 10.82 % were higher than the calculation by betaionix.

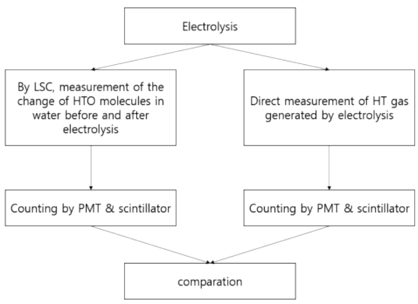


Figure 1. The scheme of experimental process.

3. Conclusion

Direct measurement by betaionix is more incorrect than measurement by LSC. Because HT gas diffusion can be affected by temperature or pressure. Measurement by LSC is more correct, however it also takes a long time for measurement. If LSC is used, realtime detection will not be possible. Therefore, it is aimed to ensure the validity of the value of betaionix through repeated experiments and correction factors, so that it can be used in realtime onsite detection.

References

[1] 1220 QUANTULUS Ultra Low Level Liquid Scintillation Spectrometer, Perkin Elmer.

SK07 Fundamental Experiment of In-situ Beta Monitoring System for Decommissioning Site

Min Ji Kim, Uk Jae Lee and Hee Reyoung Kim

Department of Nuclear Engineering, Ulsan National Institute of Science and Technology

Abstract

Radiation monitoring is required to ensure the safety of decommissioning sites. In this study, beta radionuclide ^{90}Sr in water samples is monitored by using plastic scintillators and multiple photomultiplier tubes.

1. Introduction

The measurement system in which the source directly attaches to the plastic scintillator is proposed because it is difficult to measure beta radionuclides in the field. The plastic scintillators do not react well with water and gamma rays, but react well with the beta rays. The detection efficiency and MDA (Minimum Detectable Activity) of ^{90}Sr are calculated.

2. Method and Result

2-1. Experimental System

Figure 1 shows the connection of experimental system. This system uses coincidence counting technique with simultaneous circuit in order to increase detection efficiency.

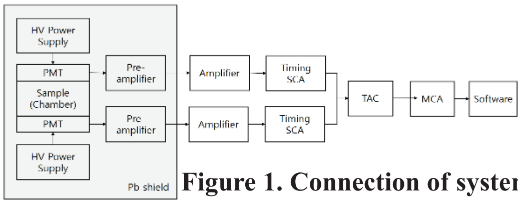


Figure 1. Connection of system

The MDA is calculated as shown in the Eq (1).

$$MDA = \frac{2.71 + 4.65 \times \sqrt{N_b \times T}}{T \times \epsilon \times 1/100 \times V_c} \quad (1)$$

N_b is background count rate (cps), ϵ is efficiency (%), T is count time (sec), and V_c is volume of sample (cm^3).

2-2. Results

Table 1. Detection Efficiency and MDA

Case	Activity(Bq)	Efficiency(%)	MDA(Bq/g)
3.15	11.918	16.305±1.234	0.014±0.0010
6.3	23.837	19.327±1.511	0.011±0.0009
9.45	35.755	17.854±1.328	0.013±0.0012

The experiments are conducted by adding 3.15 ml of ^{90}Sr source whose activity is 3.7765 Bq/g to confirm activity increases according to the amount of source. The Table 1 shows result of 3 case of experiments. The activities, detection efficiencies, and MDAs are calculated, respectively.

3. Conclusion

A system for the monitoring of beta radionuclide in water samples was proposed and MDA and detection efficiency about ^{90}Sr were estimated. Through this study, it is possible to develop a monitoring system for water beta radionuclide such as tritium using a coincidence counting technique.

References

[1] Lee, UkJae, et al. "In situ beta radiation monitoring system with enhanced efficiency for water samples from decommissioned nuclear environment." Review of Scientific Instruments 90.2 (2019): 025103.

SK08 Pre-treatment System for In-situ Tritium Monitoring in Seawater near Decommissioning Site

Se Won Park and Hee Reyoung Kim

Ulsan National Institute of Science and Technology, Ulsan 689-798, Republic of Korea

Abstract

The Seawater pre-treatment system for in-situ tritium monitoring was developed, which can be applied to decommissioning sites. This pretreatment system consists of Micro Filtration filter (MF filter), Active Carbon filter (AC filter), Reverse Osmosis membrane (RO membrane), ion-exchange resin, 0.2μm final filter etc. In this study, the performance of seawater pretreatment system is verified using water quality standards.

1. Introduction

Tritium should be monitored strictly to ensure radiological safety, while its short range makes in-situ monitoring difficult. Here, proton-exchange membrane (PEM) electrolysis method is considered, where hydrogen ions act as an electrolysis medium. Therefore, pure water should be provided by an efficient seawater pretreatment system.

2. Methods

2-1. Description of Seawater pretreatment System

The MF filter removes most impurities such as silica and bacteria, the AC filter removes organic matter, and the RO membrane removes salinity of the seawater. The ion-exchange resin is used to enhance the purity of water, and 0.2μm final filter to remove micro-organisms and particles once more [1].

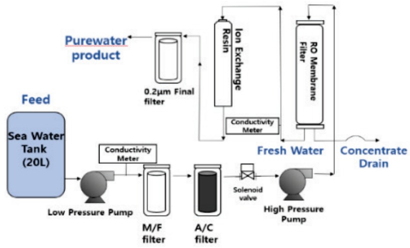


Figure1 Design of Seawater Pretreatment System

2-2. Water Quality analysis

To identify the purity of water generated from the system, pre-treated water was analyzed based on water quality standards.

Table 1 Water Quality Standards

3. Results

For the results, conductivity is 0.080μS/cm^{-1} (ASTM TypeII conductivity value is 1.0 μS/cm^{-1}). Other analysis items were analyzed as a detection limit, which is lower than the Type II standard value.

4. Conclusion

The purity of seawater through the pre-treatment system can be predicted between Type I (Ultra-Pure Water) and Type II. Therefore, seawater pre-treatment system provides proper water quality to be applied to the PEM electrolysis method.

References

[1] Nikolay Voutchkov, "Considerations for selection of seawater filtration pretreatment system", Desalination 261(3), October 2010, Pages 354-364.

SK09 Fit-Goodness Evaluation of Peak Shape Fitting Function for Airborne Alpha Beta Detection System

Si Hyung Seong, Hee Reyoung Kim

Department of Nuclear Engineering, Ulsan National Institute of Science and Technology

Abstract

An airborne alpha beta detection system using passivated implanted planar silicon (PIPS) detector was proposed. Comparison of three fitting functions was performed to improve the accuracy of alpha beta spectrum analysis of the detection system. The spectrum was simulated using Monte-Carlo simulation.

1. Introduction

The difficulty of alpha beta spectrum measurement in air should be considered because of alpha beta particle energy loss easily due to high reactivity[1]. The application of various fitting functions should be evaluated to good-fit the peaks of each nuclide through background subtraction in the spectrum[2].

2. Materials and methods

2-1. Monte-Carlo simulation

The structure of the airborne detection system shown in figure 1 was simulated using the MCNP6 code.

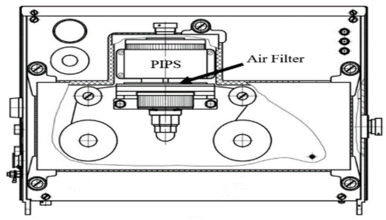


Figure 1. Diagram of detection system

2-2. Fitting functions

The three fitting functions are applied to spectrum. Gaussian and Lorentz, peak shape fitting as shown Eq (1).

$$y(x) = G(x) + T(x), where\ G(x) = H_g \times e^{-\frac{(x-x_g)^2}{2\sigma^2}}, T(x) = H_s \times e^{\frac{x-x_g}{\sigma \times t_s}} \times erfc(\frac{x-x_g}{\sqrt{2}\sigma} + \frac{1}{\sqrt{2}t_s}) \quad (1)$$

$G(x)$ is gauss equation which describe peak of pulse, $T(x)$ describe tailing of energy, H_g is peak amplitude, x is channel number in interest region, x_g is peak channel, H_s is peak tailing amplitude, t_s is extent of peak tailing.

3. Conclusion

Table 1 shows the comparison of the root mean square and the Chi-Squared test. As a result, it was confirmed that the peak shape fitting function showed 3 ~ 6% higher fit-goodness compared to other fitting functions. Since background removal and nuclide analysis are performed based on peak discrimination ability, it is expected to improve the accuracy of nuclide monitoring when using peak shape fitting method.

Table 1. RMS and Reduced Chi-Squared Test

Fitting Functions	RMS	Chi-Squared Test
Gaussian	0.9652	6,79E-9
Lorentz	0.9382	4.53E-8
Peak Shape	0.9965	3.42E-10

References

[1] 1. E. Garcia-Torano (2006) “Current status of alpha-particle spectrometry”, Applied Radiation Isotope, 64, 1273-1280

[2] Bortels. G (1987) “Analytical function for fitting peaks in alpha-particle spectra from Si detectors”, IJRAI P38, 831-837

US01

EnergySolutions Decommissioning Experience

Colin Austin1 and Makoto Kikuchi2

1 EnergySolutions, 2 MKC Consulting

Abstract

EnergySolutions (ES) is a leading international Decommissioning (D&D) company with a rich history in decommissioning and waste management, both in the US and internationally, stretching back over 30 years.

1. Successful Project Performance

Worldwide, only 15 commercial nuclear power plants have completed D&D, 12 of these were in the US. ES managed, or played a major role in all US projects. Since the international decommissioning industry is relatively new, successful performance requires critical evaluation of past performance and adoption of lessons learned. By applying this practice, ES is currently completing decommissioning of two 1,090 MWe PWR reactors (Zion) for less than $1B and in just over 8 years.

2. Project Success through Integration

Successful D&D projects need clear understanding and definition of the scope, strong disciplined program management, and extremely active stakeholder engagement and participation.

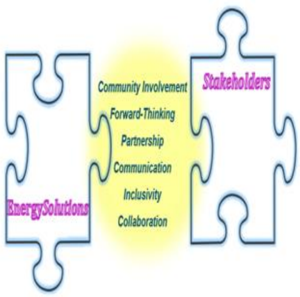
a. Fully Integrated Project Scope, Cost, Schedule and Risk

There is a major difference between D&D projects and traditional predictable projects. D&D projects carry significantly more risk. Traditional projects focus on scope, cost and schedule, the so-called “three-headed dragon”, but in less predictable projects, risk is a major component of the project management model. By their nature D&D projects are dependent on the limited industry experience to develop the baseline and are subject to significantly more uncertainty and unpredictability. ES has developed its own approach and tools necessary to manage decommissioning activities in this higher risk environment, and the success of this approach has been demonstrated on the Zion project.

b. Integration of Experience, Project Management and Local Capabilities

A major component of reducing D&D project risk is the implementation of lessons learned from prior D&D projects. International experience has demonstrated inexperienced companies significantly exceed budget, sometimes as much as 400%.

c. Integration of all stakeholder



Stakeholder participation and support is critical to success. Poor relations and badly formulated stakeholder management plans will cause extreme problems in the decommissioning project and potentially result in major cost escalation and schedule slippage. Early, open, honest and continuous communication is essential.

US02

EnergySolutions Waste Management Experience

Colin Austin1 and Makoto Kikuchi2

1 EnergySolutions, 2 MKC Consulting

Abstract

Key to success in decommissioning is an effective radioactive waste management strategy. EnergySolutions (ES) has more than 35 years' experience developing and executing nuclear waste management, treatment, and waste volume minimization processes to address the challenges of optimizing waste strategies to meet different national and international disposition criteria.

1. Full Range of Waste Treatment Capabilities

In the past, the US had the same issue as the rest of the world high cost waste disposal and limited available volume. To meet this challenge, ES developed comprehensive waste management, treatment and volume minimization techniques. These included the full range of mechanical, chemical and thermal techniques including; mechanical decontamination (grit /abrasive blasting/ high pressure spray), liquid / water treatment, chemical decontamination, electrorefining / electropolishing, incineration, metal melting / recycling, steam reforming, vitrification, shredding, compaction, equipment salvage and reuse, repackaging for volume reduction, reclassification etc.

- Processes can be Local or Centralized, Permanent or Temporary (Modular or Mobile)
- Effective Waste Strategy requires Successful Integration of Multiple Processes

2. Recycle of Major Component of Radioactive Waste for Volume Minimization

ES has been recycling waste materials from the nuclear industry for over 30 years, including over 25 years' experience in international metal recycling. ES takes ownership of the metal, produces products used in nuclear licensed facilities, and no waste is returned to customer. JPARK (Tokai) uses 2500 tons of shield blocks made of recycled metal.

Contaminated metal	
Belgium (since1996)	360 tons
Canada (2007)	6843
Germany (2000)	1253
Spain (2000)	92
UK (2006)	280
Total	8828 tons

4. Conclusion

As D&D progresses in Japan, the need for new and effective waste management practices will be essential.

- Radioactive Waste Management, a New Opportunity for Communities
- EnergySolutions Approach – Partner with Local Communities and Industry

US03 Horizontal Storage System for Used Nuclear Fuels, Fuel Debris and Nuclear wastes - NUHOMS® MATRIX system -

Ryan BUCK(i) , Jane HE(i) , Prakash NARAYANAN(i)
Vincent TRAN(ii), Hiroshi AKAMATSU(ii) , Yasuo YAMADA(ii)
(i) TN Americas, LLC (ii) Transnuclear, Ltd

Abstract

The NUHOMS® MATRIX Horizontal Storage Module (HSM) is a two-tiered concrete storage overpack system for dry shielded canisters (DSC) to store used nuclear fuels, fuel debris, and nuclear wastes. It is an evolutionary design of the existing NUHOMS® HSM, MATRIX is characterized by its small footprint, storage efficiency, seismic resistance, and self-shielding. Combined with the TMI-2 fuel debris canister design knowledge, the NUHOMS® Matrix is an optimal and cost-effective solution for the Fukushima fuel debris and used nuclear fuel.

The NUHOMS® experience with fuel debris

The only example of dry storage of fuel debris took place in the USA for the TMI-2 reactor. Debris is stored at the Idaho National Laboratory (INL) within the NUHOMS® system which consists of a metal canister and a horizontal concrete storage overpack. Apart from this technology, all other storage method would be a First Of A Kind (FOAK) achievement. The NUHOMS® horizontal concrete storage system went through several developments and its performance was significantly enhanced since its adoption for TMI-2 melted fuel debris. For instance, the use of a dual level structure reduced considerably the footprint compared to the system used in TMI (45% reduction).

The NUHOMS® experience with used fuel

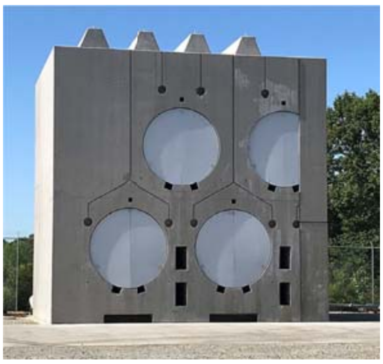
Orano TN's NUHOMS® systems have securely stored used nuclear fuel in the United States for more than two decades, with installations at 33 sites around the country representing more than 45,000 stored used fuel assemblies. The NUHOMS® MATRIX HSM system is scheduled for approval by US NRC in 2019 and was already selected by a US Utility, Wolf Creek Nuclear Operating Company (WCNOC), for loading in 2020.

The NUHOMS® experience with wastes

The NUHOMS® Matrix System is also capable of storing the NUHOMS® Radwaste Canister (RWC), a canister that is used to store dry irradiated and/or contaminated nonfuel hardware. The NUHOMS® system dry shielded canisters can be adapted internally to safely store many forms of nuclear waste (intact used fuel to encapsulated fuel debris to radioactive wastes), all of which could be stored in the NUHOMS® MATRIX HSM array. This technology can be applied for the whole decommissioning process; its flexible design makes it a universal overpack that can meet nuclear operators' needs and requirements.

Conclusion

More than 1,200 systems were loaded in the USA, all associated hardware, auxiliary systems, procedures for loading/unloading, transfer, and monitoring are well developed. The small footprint and multiple uses makes the NUHOMS® MATRIX perfectly suited for decommissioning activities.



Dual level horizontal concrete module

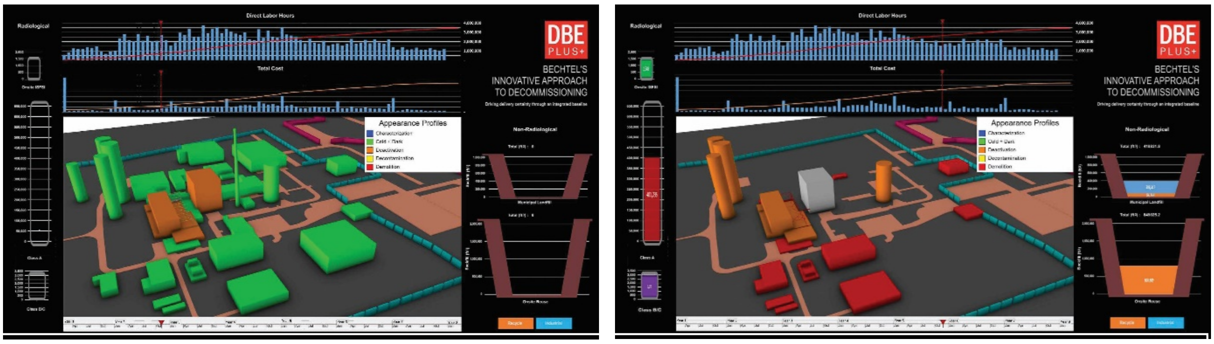
Loading operation in horizontal

US04

DBE *PLUS+*
Innovative Approach to Decommissioning

Takehiko Miyazaki, Bechtel[1]

Abstract: This poster illustrates Bechtel's new approach to developing site-specific Decommissioning Baseline Estimates using over 20 years of historical decommissioning results and site-specific data. A feature of this approach takes the 3D site-specific model developed from historical waste and construction quantities and integrates schedule/cost to create 4D/5D modeling solutions that demonstrates D&D sequencing, cost, schedules, waste generation, and manhours.



<u>**Screenshots from Bechtel's DBE *PLUS+* 4D/5D modeling solution.**</u>

Introduction: Bechtel's Decommissioning Baseline Estimate methodology, DBE *PLUS+*, is an advanced schedule and estimating baseline tool developed to bring site-specific D&D solutions to our customers. DBE *PLUS+* leverages more than 100 years of project execution history and decommissioning baseline estimating expertise. It draws upon the experiences and historical data across the nuclear lifecycle of commercial and government customers—as well as the latest data from current nuclear new build and decommissioning projects such as Vogtle Units 3 & 4, Sellafield Pile Fuel Cladding Silo, and Savannah River Site. Bechtel Subject Matter Experts (SMEs) analyze each aspect of a nuclear power plant including the following:

- End State Requirements
- Large Component Removal
- Reactor/Internals Segmentation
- Cold & Dark Requirements
- Spent Fuel Requirements
- Waste Management Requirements
- Risk Management
- Site Security Requirements
- Regulatory Requirements
- Work Force Requirements

Then the Bechtel SME's integrate cost and schedule into a site-specific, executable plan that creates a timeline-based output of:

- Generated Waste Quantities
- Equipment Requirements
- Work Breakdown Structure
- Waste Management Costs
- Staffing Plans
- Stakeholder Management Plans
- Risk Management Plans
- Labor Hours Projected and Expended

Conclusion: Bechtel's DBE *PLUS+* utilizes actual D&D results from completed projects and lessons learned to create an accurate execution plan for managing and optimizing project resources, schedules, and cost. The 3D/4D/5D outputs provide an effective method to demonstrate progress and D&D sequencing to all stakeholders.

References: [1] Bechtel is a trusted engineering, construction and project management partner to industry and government. Since 1898, we have helped customers complete more than 25,000 projects in 160 countries on all seven continents.

US05

Application of UK Waste Management Performance at the Fukushima Daiichi Decommissioning

Andy Beckwith, Max Ehrhardt
AECOM

Abstract

The United Kingdom's radioactive waste management approach minimizes the volume of waste that requires final disposal in their limited available space. As Fukushima Daiichi and other domestic decommissioning projects face similar waste management challenges, application of this approach to radioactive waste in Japan can replicate the successful transformation in decommissioning seen in the UK.

1. Introduction

Waste management is a significant challenge at Fukushima Daiichi both in stakeholder concerns, logistical challenges, and long term costs. From a safety point of view, the higher activity waste presents the most acute risk, but the volume of Level 2, Level 3, and Clearance Waste presents a long term drain on resources. This situation is similar to the UK, in that Japan has a large source of radioactive waste without authorized disposal locations. In response to this limitation, the radioactive waste management approach used in the UK is to minimize the volume of waste that requires disposal at their only radioactive disposal sites: Low Level Waste Repository (LLWR) and Dounreay's Low Level Waste Vaults.

2. United Kingdom Waste Management Approach

2-1. A New Waste Management Policy

To account for large-scale decommissioning and environmental remediation projects, the UK Government announced the LLW Policy in 2007. Since 2011, the UK National Waste Program has been developed and executed to accomplish the goals of the Policy. Before the UK National Waste Program was developed, it was projected that the LLWR site's disposal capacity would be fully exhausted by 2023. As a result of the successful implementation and execution of the UK's National Waste Management Program, the LLWR site's useful capacity and life has been extended by one hundred years.

2-2. Waste-Informed Decommissioning

The success of the UK's LLW Policy has resulted in an extensive focus on detailed characterization, application of the appropriate waste hierarchy, and application of processing technologies to minimize or avoid waste disposal. To preserve the limited disposal capacity in the UK, a waste-disposal-avoidance hierarchy is diligently applied to decommissioning projects. The "Waste-Informed Decommissioning Process" ensures that limiting disposal waste volume is a key objective of the decommissioning strategy to provide the best overall solution for the country.

2-3. Diverse Technologies for Waste Processing

The National Waste Program (NWP) operates through individual services contracts between the LLWR and all UK waste generators. The waste management approach is integrated into the entire Supply Chain of the waste generators which allows cross-utilization of services and significantly minimizes the cost for the following Supply Chain elements.

- Safety Case Analysis
- Waste Characterization
- Transportation
- Packaging
- Metals Processing
- Combustible Processing
- Very Low Level and Low Level Waste Disposal
- Waste Tracking Systems

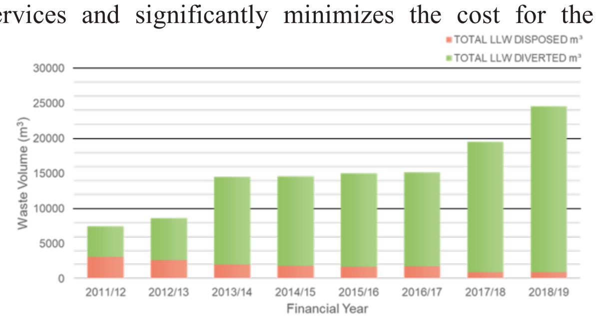


Figure 1. Reduced LLW Disposal under the NWP

2-4. Applying the Waste Hierarchy

Avoidance of Radioactive Waste generation is the best option as it has extended the life of the LLWR site and greatly reduced the overall nuclear liability from generation through disposal. Avoidance has best been achieved by dispatching waste management experts to work directly with the decommissioning sites to ensure that correct Decommissioning processes are applied to reduce generation of as much radioactive waste as possible. Other methods used to minimize disposal at LLWR include Metal Size Reduction, Decontamination via Shot Blasting, Metal Melting, Incineration of Combustible Waste, and Shredding and Compaction.

3. Application to Fukushima Daiichi

The method used at the UK was a combination of government action (waste hierarchy), a good contracting model for the generators and repository, and creation of alternative methods for disposal of the waste. At Fukushima, incineration already exists and is being expanded. However, inclusion of metal recycling and supercompaction, application of a "Very Low Level Waste" arrangement, central transport capability, standard radioactive characterization, and low level liquid processing, all managed through a single group incentivized to reduce the volume of low level waste could replicate the performance of the UK over the next 10 years.

List of Organizations

Organization	ID No.
ABLE Co. Ltd.	E04
ADVAN ENG. Co., Ltd.	H01
AECOM	US05
AESJ (Atomic Energy Society of Japan)	A01, K03
Aizuk, Inc.	B04
ANADEC (Orano Atox D&D Solutions Co., Ltd.)	F07
Ascend Inc.	B03
ATKINS	GB04
BCSN	F08
Bechtel	US04
Cavendish Nuclear Ltd.	GB02
Cavendish Nuclear Japan	GB02
CEA (French Alternative Energies and Atomic Energy Commission)	C02, F01, F02, F07
Cegelec CEM	F10
CORNES Technologies Limited	B10
CRIEPI (Central Research Institute of Electric Power Industry)	C02, C04, K02
Daihatsu Motor Co., Ltd.	H02, H03
East Japan Accounting Center Co. Ltd.	B03
Ecole Centrale Supelec	F06
e-Energy Corporation	F08, F09, F10
EnergySolutions	US01, US02
Forschungszentrum Jülich	H02
Fuji Electric Co., Ltd.	K04
Fukushima Consortium of Robotics Research for Decommissioning and Disaster Response	B01
Fukushima Midori Anzen Inc.	B08
Gleeds Energy Limited	GB05
Gunma University	C02
Hamamatsu Photonics K.K.	B09
Hitachi, Ltd.	K02
Hitachi-GE Nuclear Energy, Ltd.	K02
Imagineeing Inc.	J12

IRID (International Research Institute for Nuclear Decommissioning)	B05, B06, C05, E03, J06, J11, J13
IRSN (Institut de Radioprotection et de Sûreté Nucléaire)	F01
JAEA (Japan Atomic Energy Agency)	A01, C01, C03, C05, E03, H02, H03, J01, J02, J03, J04, J05, J06, J07, J08, J09, J10, J11, J12, J13, J14, F02
JAMSTEC (Japan Agency for Marine-Earth Science and Technology)	C04
JAXA (Japan Aerospace Exploration Agency)	C01
JGC Corporation	K02
Kajima Corporation	E05
Kobe University	C02
Kurion Japan K.K	A03, A04, A05
Kwansei Gakuin University	H02, H03
Kyoto University	C01, C04
Mitsubishi Research Institute, Inc.	A02
MITSUFUJI Corporation	A06
MKC Consulting	US01, US02
Nagaoka University of Technology	H01
Nagoya University	A01, C03
NDF (Nuclear Damage Compensation and Decommissioning Facilitation Corporation)	A01
NFD (Nippon Nuclear Fuel Development Co., Ltd.)	C05, E03
NIT (National Institute of Technology), Fukushima College	B03
NIT (National Institute of Technology), Kisarazu College	C01
NINS (National Institutes of Natural Sciences)	J12
Nuclear water soil solutions	D01
ONET Technologies	F01
Orano	F07
Orano Cycle	F05
Orano Dismantling & Services	F06
Orano GmbH	F06
Orano Group	F03, F04
Orano Japan	F05, F06
Orano Temis	F05

Osaka Prefecture University	C01, J03
Osaka University	A01
QST (National Institutes for Quantum and Radiological Science and Technology)	J03, J12
RIKEN	C01
ROBATEL Industries	F09
Sellafield Ltd.	GB01
Shanghai Jiao tong University	D03
SSC RIAR (Research Institute of Atomic Reactors) JSC	C06
Taisei Corporation	E02
Takawa Seimitsu Co. Ltd.	B03
TENEX JSC	C06
TEPCO (Tokyo Electric Power Company Holdings, Inc.)	E01, E02, E03
The University of Tokyo	C03, C04, J03
Tigers Polymer Corporation	B07
TN Americas, LLC	US03
Tohoku Institute of Technology	C03
Tohoku University	C02, C03, D02, D03, J03
Tokyo Bosai Setsubi Co., Ltd.	B02
Tokyo City University	D01
Tokyo Institute of Technology	J03
Toyo Union Co., Ltd.	E02
Transnuclear, Ltd.	US03
UNION SHOWA K.K.	D02, D03
UNIST (Ulsan National Institute of Science and Technology)	SK01, SK02, SK03, SK04, SK05, SK06, SK07, SK08, SK09
Université de Lyon	C02
Veolia Nuclear Solutions, Inc.	A04, A05
Veolia Nuclear Solutions (UK) Ltd.	A04
VNS Federal Services Inc.	A05
Waseda University	J14
WM Symposia, Inc.	K01
Wood	K04
Wood plc	GB03