ABSTRACTS of the Technical Poster Session

The 3rd International Forum on the Decommissioning of the Fukushima Daiichi Nuclear Power Station

-Moving forward together-

Mon, August 6, 2018

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.

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Cover photos: Unit 3 of the Fukushima Daiichi nuclear power station in Japan as at 24 August 2011 and 21 February 2018 (TEPCO, Japan).

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A01 Overview of Research & Development for Decommissioning of the Fukushima Daiichi Nuclear Power Station

Setsuo Nakashima, Takuto Miyamoto , Hidenori Nomura and Takuro Ueda
Nuclear Damage Compensation and Decommissioning Facilitation Corporation (NDF)

Abstract

Overview of R&D for Decommissioning of the Fukushima Daiichi Nuclear Power Station (1F) is described. From basic and fundamental research to applied and practical research are carried out by various institutions, for example the International Research Institute for Nuclear Decommissioning (IRID), the Japan Atomic Energy Agency (JAEA), Universities, etc. The NDF has been undertaking optimization of R&D activities carried out by each institution for a better effectiveness and efficiency as a whole.

1. Overview of R&D for 1F Decommissioning

Several R&D Projects toward to practical outcome are carried out by R&D institutions as the IRID. Basic and fundamental researches are carried out by the JAEA and basic research institutions (e.g. Universities). The Tokyo Electric Company Holdings, Inc. (TEPCO) carry out R&D that is highly practical research. The main R&D themes are "retrieval of fuel debris" and "processing and disposal of radioactive waste". Fig.1 is a conceptual picture of the division of roles among main R&D institutions for the 1F Decommissioning.

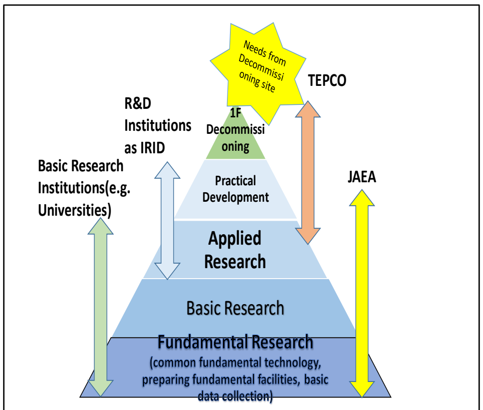


Fig.1 Overview of R&D for 1F Decommissioning

2. Structure of R&D for 1F Decommissioning

Whole Picture of structure of R&D for the 1F Decommissioning shows Fig.2. There are variety of organizations engaged with the 1F decommissioning R&D projects. The NDF has been undertaking optimization of R&D activities carried out by each institution for a better effectiveness and efficiency as a whole. Decommissioning R&D Partnership Council conducts coordination and cooperation among each organization.

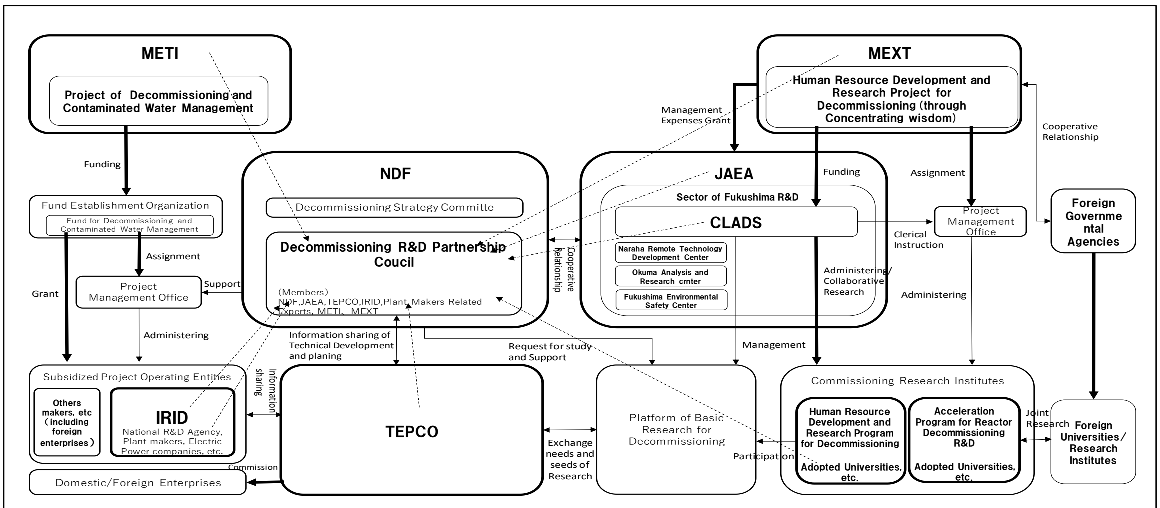


Fig.2 Structure of R&D for 1F Decommissioning

A02

Activities of Review Committee on Decommissioning of Fukushima Daiichi Nuclear Power Station in Atomic Energy Society of Japan

Noriko Asanuma[1], Michitaka Sasoh[2] and Hiroshi Miyano[3]
[1]Tokai Univ., [2]Toshiba Corp., [3]Hosei Univ.

Abstract

In view of long term activities under the well-organized structure, Review Committee on Decommissioning of Fukushima Daiichi NPS is established. Five subcommittees are installed and each activity are introduced.

1. Introduction

Review Committee on Decommissioning of the Fukushima Daiichi Nuclear Power Station, which is directly connected under the board of directors of Atomic Energy Society of Japan (AESJ), is established in 2014. In our committee, the knowledge of academia is widely gathered, and activities applied each expertise is ongoing, for example extraction of the problems and discussion of the countermeasures. Based on these activities, helpful suggestion and proposition can be submitted, that is the aim of the action in our committee.

As shown in Figure 1, 5 subcommittees are installed and each activity are currently accelerated. Based on the activities of subcommittees, the actions of our committee are introduced.

2. Activities of Subcommittees

In Subcommittee on **Risk Assessment and Management**, risk assessment methodology of fuel assembly retrieval from a spent fuel pool as an example was investigated. With this result, the risk assessment method of the decommissioning of the actual nuclear power system is developed. In the AESJ investigation report, unresolved issues of the accident are summarized. In Subcommittee on **Follow-up the Proposal for Lessons Learned and the Unresolved Events**, the unresolved issues of the accident and the up-to-date knowledges on these issues are summarized, and the final report was released [1]. Subcommittee on **Robotics** is a joint committee which is constructed by Robotics Society of Japan and AESJ with cooperation in 2015. In 2016, robot technology competition for decommissioning, where typical environment and task for fuel debris retrieval are defined, was conducted and creative ideas were collected from public and both society members. Continuing on, a new competition on robot idea of reactor inside survey will be held. In Subcommittee on **Structural Performance of Building**, aseismic soundness of the reactor building No.3 damaged by the hydrogen explosion was verified and the interim report will be released. In Subcommittee on **Radioactive Waste Management**, the end state as well as the scenarios are discussed and evaluated thoroughly and carefully in terms of factors such as feasibility, radiation safety, technology availability, and consistency with the national framework.

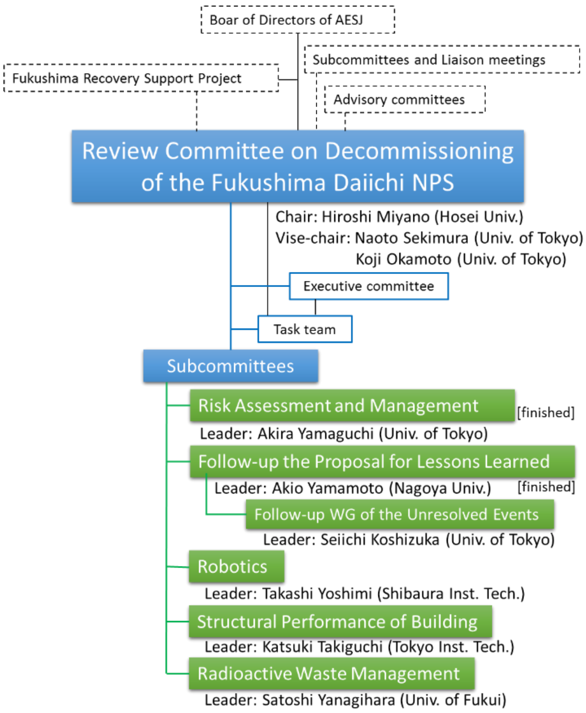


Figure 1. Organization chart of the committee.

Reference

[1] The final report (in Japanese) can be downloaded from "http://www.aesj.net/activity/activity_for_fukushima/public".

A03 Latest Activity of AESJ Research Committee on Fission Product Behavior under Severe Accident

Yosuke Katsumura [1], Shunsuke Uchida [2], Masahiko Osaka [2],
Hidetoshi Karasawa [3], Koichi Nakamura [4] and Junichi Takagi [5]
[1] Japan Radioisotope Association, [2] Japan Atomic Energy Agency,
[3] Institute of Applied Energy, [4] Central Research Institute of Electric Power Industry,
[5] Toshiba Energy Systems & Solutions Corporation

Abstract

In order to revive research activities related to fission product (FP) behavior under severe accidents (SA), a research committee on FP has been organized in the Atomic Energy Society of Japan (AESJ). Major targets of the committee and its procedures for contributing to reliabilities of operating plants and decommissioning of Fukushima Daiichi NPP are introduced.

1. Introduction

Understanding FP behavior under SAs is one of the key issues for plant reliability based on defense in depth as well as decommissioning Fukushima Daiichi NPP. Unfortunately, as a consequence of long quiet period in fuel defects and anomaly plant operations, number of researchers and engineers related to FP behavior decreased and decreased to have apprehension of inadequate technical transfer of knowledge base on FP behavior for long-term decommissioning. Under the background, the AESJ reorganized the research committee on FP.

2. Major Targets of the Committee and its latest activities

The committee put its major targets on (1) contribution to Fukushima decommissioning and (2) increase abilities and reliability of advanced SA analysis codes.

Three working groups (WGs) are collaboratively promoted to establish the targets under supports of the AESJ divisions, e.g., water Chemistry, Nuclear fuel, Thermal Hydraulic, Environment and Safety divisions (**Figure 1**). Based on the survey results of the Phébus FP projects [1], latest knowledge on FP in Fukushima Daiichi NPP should be compiled to a new technical document for technical transfer to next generations.

3. Conclusion

Over a year, the committee has carried on several numbers of technical discussion meetings to polish up the approach toward the final targets. Necessary modifications on its approaches toward the targets can be flexibly accepted based on the discussions at the poster presentation.

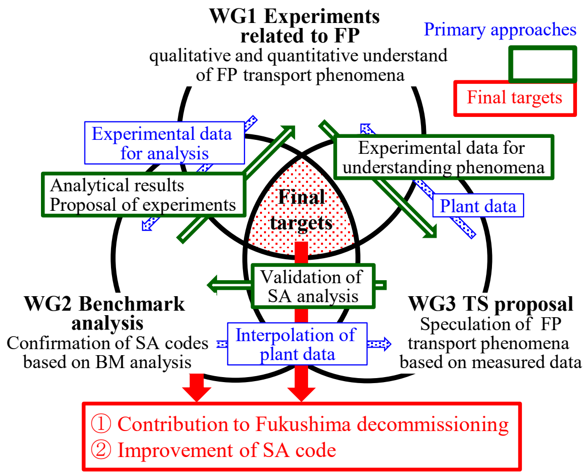


Figure 1 Major roles of 3 WGs and their final targets

References

[1] AESJ Division of Water Chemistry, "Fission Product Behavior Obtained from Phébus FP Projects – Application for Decommissioning Plans of Fukushima Plants and Severe Accident Analysis ", AESJDWC Report#2017-0001, Atomic Energy Society of Japan (May, 2017).

A04 Relationship between the Research and Development of the Project of Decommissioning and Contaminated Water Management and Connection to Preliminary Engineering

Masaaki Matsumoto[1], Naoki Kondo[1], Keishun Nakamura[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 development with high technical difficulty. In this presentation, we will show a briefing on the series of processes including the link between the various subsidized projects of this program and the results obtained in the projects until applied 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 research and development. For this reason, 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 research and development with high technical difficulty. In fiscal 2017, 16 subsidized projects were implemented by subsidized operating entities. The contents of research and development in these projects were diverse. In order to apply the results obtained in the subsidy projects to the decommissioning of Fukushima Daiichi NPP, mutual coordination among subsidized projects are essential.

2. Relationship between the subsidized projects of Decommissioning and Contaminated Water Management program and connection to preliminary engineering

In accordance with the next R & D plan [1], the subsidized projects are divided into “Internal Investigation”, "Development of debris retrieval method", "Development of Fundamental Technologies for Retrieval of Fuel Debris", "Improvement of Work Environment for Retrieval of Fuel Debris”, and "Processing of Solid Waste". As a rough flow, Development of Debris Retrieval Method, Development of Fundamental Technologies and Improvement of Work Environment are carried out based on information inside the reactor vessel obtained by Internal Investigation. In addition, the results of investigation in Development of Debris Retrieval Method are reflected in the preliminary engineering by the operating entities including concretization of fuel debris removal process. Processing of Solid Waste also study in parallel with studying on debris retrieval. In this way, current projects are organically related, and preparations are being made for engineering by subsidized entities.

References

[1] Ministry of Economy, Trade and Industry, Agency for Natural Resources and Energy, Progress of R & D Project and Direction of Next Plan, Decommissioning and Contaminated Water Management Team Meeting / Project Management Office Meeting (51st) (Japanese only)

A05 Advanced Research and Education Program for Nuclear Decommissioning

Toru Obara[1], Kenji Takeshita[1], Yukitaka Kato[1], Hiroshi Akatsuka[1], Hiroshige Kikura[1], Takehiko Tsukahara[1], Katsumi Yoshita[1], Koichiro Takao[1], Jun Nishiyama[1], Masayuki Harada[1], Nobuyuki Iwatsuki[1], Koich Suzumori[1], Gen Endo[1], Kenji Kawashima[2], Noriko Asanuma[3], Tsuyoshi Arai[4], Naoyuki Takaki[5]

[1]Tokyo Institute of Technology, [2]Tokyo Medical and Dental University, [3]Tokai University, [4]Shibaura Institute of Technology, [5]Tokyo City University

Abstract

Advanced Research and Education Program for Nuclear Decommissioning has been started in 2014 for the sophisticated education and advanced research for the decommissioning of Fukushima Daiichi NPS.

1. Introduction

Advanced Research and Education Program for Nuclear Decommissioning (ARED) has been started in 2014 by Tokyo Institute of Technology with the collaboration of Tokyo Medical and Dental University, Tokai University, Shibaura Institute of Technology, and Tokyo City University with financial support of the Mistry of Education, Culture, Sports, Science and Technology (MEXT). The purpose of the program is to perform sophisticated education of graduate school level for the decommissioning of Fukushima Daiichi NPS and to proceed advanced researches to solve the technical issues in the decommissioning.

2. Educational Activities

Some classes for the graduated school students have been offered by the program, which include laboratories using nuclear fuel and radioactive materials. The students perform experiments in radioactive hazard laboratories to obtain the fundamental and scientific knowledge of the materials and the method to tread them. In addition to that here is a class of laboratory for the robotics and remote sensing. The program includes lectures for the decommissioning of damaged cores and for the decommissioning of the reactors which are shut down by the life. The internships and seminars for the decommissioning are also offered.

3. Research Activities

Several advanced researches for the decommissioning are in progress including analysis method of contaminated material with microanalysis technique, decontamination and stabilization technology of contaminated materials, robotics and remote sensing technology, and criticality safety study in the reloading of the fuel debris. They are performed by the collaboration of the universities. Seminars about the research progress are held regularly to exchange information with nuclear industries.

4. Concluding Remarks

The educational activities and researches have been proceeded successfully. Some research topics have been moved to the phase of joint study with nuclear industry for the practical use. To continue the sophisticated education will improve the skill of researchers and engineers who involve in the decommissioning.

A06

Contribution to Fukushima Daiichi unit 5 and 6 strategic decommissioning utilizing experience of decommissioning in Japan and oversea

Toyoaki Yamauchi[1], Colin.R.Austin[2]

[1]The Japan Atomic Power Company, [2]Energy solutions

Abstract

The Japan Atomic Power Company(JAPC) and Energy Solutions(ES) can contribute to the safe and efficient decommissioning of Fukushima Daiichi unit 5 and 6 by utilizing decommission knowledge of both ES, specialized in decommissioning in US, and JAPC in Japan.

Domestic first decommissioning of commercial nuclear power plant

JAPC has executed domestic first decommissioning of commercial Tokai nuclear plant since 2001. In addition, since 2017 JAPC has also started decommissioning of Tsuruga No.1 nuclear plant(BWR) efficiency introducing decommissioning project management by setting Tsuruga Decommissioning Project Center this April.

Removing work of the Turbine and Generator at Tokai nuclear plant

Fukushima reconstruction support with 180 personnels from JAPC group

JAPC group member at Fukushima Daiichi

JAPC concluded the cooperation agreement with TEPCO in March 2015, and currently about 130 personnels of JAPC group support decommissioning of Fukushima Daiichi with TEPCO.

Also, about 50 personnels of JAPC group work for radioactive waste treatment in Fukushima, total about 180 JAPC personnels support Fukushima reconstruction.

Utilizing the decommissioning know-how from overseas

JAPC agreed with ES to obtain and utilize the ES decommissioning know-how in April 2016, and agreed to utilize that know-how for the domestic decommissioning.

So far, we have conducted exchanging each decommissioning members and making efforts for the safe and efficient decommissioning of Tsuruga Unit1 and so on, considering the joint business in the future.

Workshop at Zion in US

Summary

JAPC and ES believe that early decommissioning of Fukushima daiichi unit 5 and 6 could contribute to decreasing the maintenance cost and project risk and utilizing these units for decommissioning of the whole Fukushima Daiichi.

The key factors for successful decommissioning are the proper decommissioning mind set and proper project management, especially waste management of the removed equipment. First, the project of decommissioning of unit 5 and 6, which are similar to normal plants, has to be treated safely and efficiently, after that utilizing the knowledge obtained from that experience for the decommissioning project of other units will have to be conducted, which could lead to decommissioning of whole Fukushima daiichi safely, steady and efficiently.

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 exhibitions.

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, demonstrations and exhibitions of robots 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 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.

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 1. Robots demonstration events

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

Robots Developed by the Members of 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

Aiming for the entry into the market of decommissioning, decontamination and disaster response robotics, member companies of the Fukushima Consortium of Robotics Research for Decommissioning and Disaster Response have developed robots with unique features. Below are some examples.

1. Robots developed by the Consortium members

1-1. Drone for high sensitivity radiation measurement 「NJH950-Ⅱ」（NESI, Inc.）

Drone for high sensitivity radiation measurement 「NJH950-Ⅱ」 can measure surface radiation while flying over measuring area without stopping at the measuring point and instantly visualize the results by installing nine GAGG high sensitivity scintillators. Also, by increasing the battery and reducing the weight of the load, we could dramatically extend the flight time.

Figure 1. NJH950-Ⅱ

1-2. Autonomous Cleaner ReFRO – β Version – （Haloworld Inc.）

Autonomous Cleaner ReFRO – β Version – can create a 3D map with SLAM using laser sensors etc.. Also, it runs autonomously based on the 3D map, and if there is an obstacle, it can be avoided. It can clean indoor and outdoor surfaces.

Figure 2. Autonomous Cleaner ReFRO – β Version –

1-3. Fire Fighting Robot（Kouyou Co., Ltd.）

This fire-fighting robot uses water pressure as the remote control medium. Therefore, there is no electrical component in the main body, and secondary damage doesn't occur. Also, by stretching the hose, it is possible to extend water discharge and remote operation distance.

Figure 3. Fire Fighting

2. Others

In addition to the robots introduced above, many other products developed by the Consortium members are available.

We are pleased to introduce our member companies capable of helping your new product development. They can provide proper technical solutions in your business related to the decommissioning and disaster response. So, if you are interested in such services, please contact the secretariat of the Consortium.

B03

Organization Profile of IRID

Naoaki Okuzumi[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

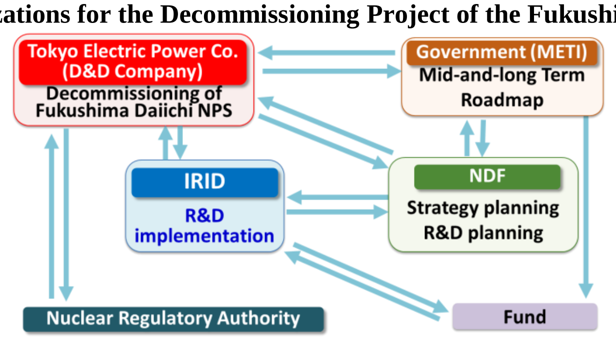
1-1.Scope of Work

●R&D for nuclear decommissioning

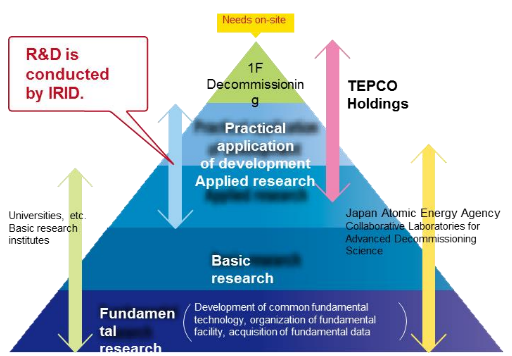
●Promotion of cooperation on nuclear decommissioning with relevant international and domestic organizations

●Human resource development for R&D

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

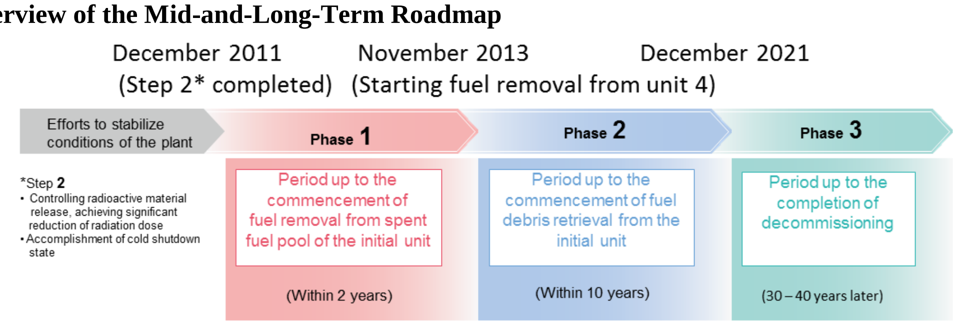


1-3.IRID R&D Scope



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

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



( ) represents the period from completing phase 2

B04

Overview of IRID R&D Projects

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

Abstract

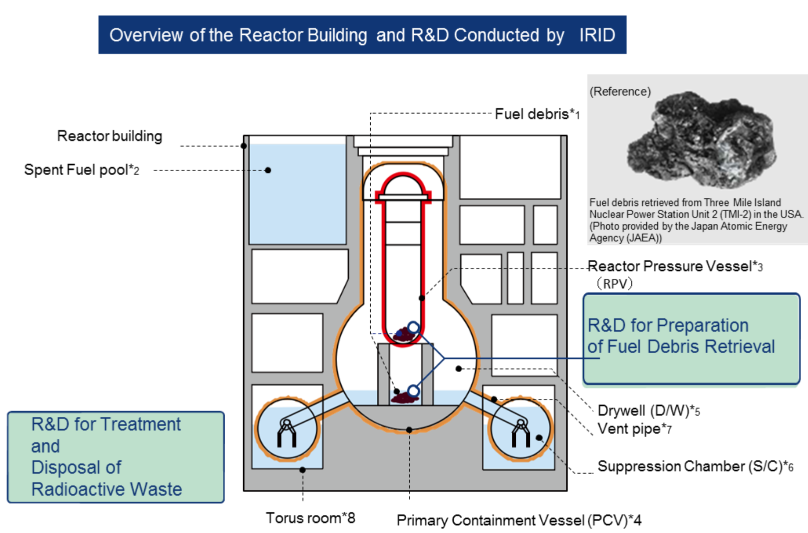
Towards the decommissioning of the Fukushima Daiichi NPS, four entities cooperate closely to work together as one team. IRID is a complex entity consisted of eighteen organizations that play the leading role in R&D for the decommissioning of the Fukushima Daiichi NPS. IRID is conducting R&D projects; “Preparation of Fuel Debris Retrieval” and “Treatment and Disposal of Radioactive Waste”. In preparation for determination of fuel debris retrieval methods in 2019, IRID continues to develop technologies for; 1.Intensive investigation on the existence of fuel debris and damaged conditions inside the reactor, 2.Potential risk management and verification for nuclear safety, 3.Reliable remote operation under high radiation environments.

1. Progress of Research and Development

As R&D projects for the preparation of fuel debris retrieval, IRID is undertaking research based on three elements. Firstly, detection technology that enables to directly retrieve fuel debris in the PCV has been developed. In April 2015, a robot successfully entered the PCV at Unit 1. In FY 2016, the preparation of fuel debris investigation outside the Pedestal was conducted. At the same time, fuel debris investigation robots inside the Pedestal, and remotely operated drilling device for the PCV penetration have been developed to reduce the exposure of workers at Unit 2. Additionally, investigation of fuel debris inside the Pedestal, an underwater swimming robot for Unit 3, and a new survey device equipped with a telescopic pipe and cameras for Unit 2 have been developed. When these robots remotely access fuel debris, it will enable to investigate a distribution and compositions of fuel debris. Severe Accident Analysis Code is upgraded to identify fuel debris inside the reactor, and investigations through the cosmic-ray Muon are conducted. The distribution of fuel debris in the reactor is investigated from outside the reactor building through the Muon at Unit 1. It resulted that the possibility is small that a large amount of fuel remains in the reactor core. The Muon transmission measurement was conducted at Unit 2 from March to July 2016, and at Unit3 from May to September 2017. Essential technologies such as access to fuel debris in the RPV or the PCV are ongoing. It is necessary to secure nuclear safety risks.

2. Future Development

With an aim to advance a very tough task of the decommissioning, IRID intends to proceed with R&D, collecting knowledge and expertise from around the world. Specifically, not only hardware support from oversea organizations is required for removal and storage of damaged fuel, but also experiences of safety management and process.



B05 The Future of Remote Access for 1F Decommissioning (VNS Integrated Nuclear Remote Access)

Akira Ikeda[1], Mamoru Numata[1], Matt Cole[2], Scott Martin[2] and Marc Rood[2]
[1]Kurion Japan k.k, [2]Veolia Nuclear Solutions, Inc.

Abstract

Veolia Nuclear Solutions (VNS) has an extensive record tackling the most difficult decommissioning challenges around the world. Taking advantage of our deep project management experience and best-in-class technology, VNS has been – and will continue to be – a trusted solutions provider for 1F. Utilizing the latest access technology, VNS has already had a positive impact on several challenging areas at 1F.

1. Introduction

Veolia Group is the global leader in optimized resource management. VNS provides the most comprehensive range of technologies, expertise and services for decommissioning, facility restoration and treatment of radioactive waste. VNS is already engaged and cooperating on the 1F decommissioning, and is optimistic that the company’s successful Access technology can be further deployed to help encourage the effective decommissioning of 1F. This paper provides a brief introduction to VNS achievements in Japan and throughout the world.

2. VNS Access technology

2-1. VNS Experience Throughout the World

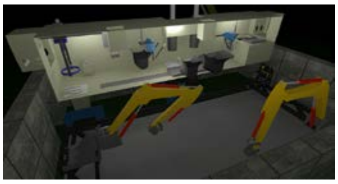
Project: Rotary Deployment Arm

- Location: Trawsfynydd NPP, Wales, UK Date: 2009 - 2011
- Two (2) remote tank-cleaning arms for four (4) different storage tanks. Waste forms included liquid, sludge and solid (Fuel Element Debris).

Trawsfynydd NPP

Project: Whiteshell Standpipes and Bunker Retrieval

- Location: Whiteshell Laboratories, Canada Date: 2016 – Present
- Access and retrieve waste from 177 buried standpipes and several deep bunkers. Waste must be removed and processed for long term disposal.



Whiteshell

2-2. VNS Experience in Japan

Project: Fukushima Inspection and Repair Manipulator

- Location: Fukushima, Japan Date: 2012 – Present
- Two (2) long-reach manipulator systems to inspect and repair leaks that have occurred in the primary containment vessel of the Unit 2 Reactor at 1F.

2-3. VNS Capability in the Future on 1F

Project: D-pit sludge retrieval and Fuel Debris Retrieval in the future

Fukushima

3. Conclusion

VNS couples cutting-edge technologies with deep management expertise to provide bespoke solutions for today and the future. VNS will continue to benefit the 1F decommissioning by providing both trusted experience and Access technology.

B06

Development of a remote concrete sampling device

Kiyoaki Kato, Akira Horii and Toshihiko Mizushima

ATOX CO., LTD. / IRID (International Research Institute for Nuclear Decommissioning)

Abstract

In order to safely process and dispose of solid waste generated by the accident at Fukushima Daiichi Nuclear Power Station, it is necessary to analyze its properties. Therefore, we have been developing a remote sampling device of the floor concrete for characterization of rubbles inside the reactor building under a high-dose radiation environment.

1. Introduction

As huge amount of radioactive solid waste is expected to generate at Fukushima Daiichi Nuclear Power Station, it is necessary to develop technology that can safely process and dispose the radioactive wastes based on inventory evaluation by analysis of radioactive waste. Therefore, the method of sampling the floor concrete was established and the remote sampling device was developed for characterization of rubbles inside the reactor building under a high-dose radiation environment.

2. Development overview

2-1. Core Bit with wedge of core drilling

The shape of the sample to take from the concrete on the floor was decided as the core, and the sampling method adopted the core drilling method. In the usual core drilling method, actions of drilling, core cut off and core removal are needed. However, it might be a complicated method if all actions are performed remotely. In this study, by providing wedge inside the bit, core cutting off and holding were performed at the same time so that series of operations can be completed only by "drilling" operation.

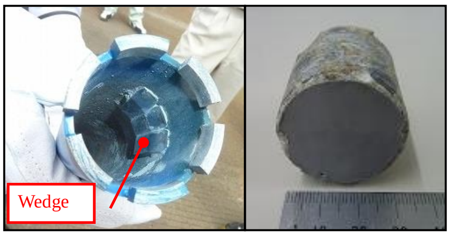


Figure 1. Core Bit and Sampling Core

2-2. Remote sampling device for Kobra

In order to confirm the feasibility of the remote core drilling operation by using core bit with wedge, the remote sampling device for Kobra was developed, which had abundant performances in the field work at Fukushima Daiichi Nuclear Power Station.

Figure 2. Sampling device (For Kobra)

3. Conclusion

It was confirmed that the core sample can be collected remotely from the floor concrete by this study. In the future, we will promote the development of remote sampling device that can be mounted on Packbot, which is widely used in the field work at Fukushima Daiichi Nuclear Power Station, to expand the collectable area for concrete samples.

Acknowledgments

This presentation includes some of the results from the works under the subsidy program “Project of Decommissioning and Contaminated Water Management” by METI.

B07

Usage of remote technologies for D&D operations in hostile environment / Orano's experience serving Fukushima-Daiichi

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[1]Orano Atox D&D Solutions Co., Ltd. (ANADEC)

Abstract

The use of remote technologies is required when performing D&D[i] operations in high dose rate environments. A combination of scenario making, engineering and preparation for on-site operation is needed to propose adequate, reliable and cost effective remote solutions for D&D works. Orano experience in nuclearizing off-the-shelf industrial equipment to develop remote solution perfectly fitted to each problem is unique. ANADEC can benefit from this Orano experience to tackle the various challenges in Fukushima-Daiichi (1F).

1. Introduction

Since 2011 the situation of 1F site has greatly improved, but the remaining works are very challenging in terms of dose rate, contamination risk and access difficulty. The ALARA[ii] principle, and other processes used by Orano when implementing remote technologies on challenging D&D sites such as 1F, are presented below.

2. Principles for implementation of remote technologies and equipment for D&D works

2-1. Definition of a scenario based on ALARA principle

The first step of scenario selection is to put forward several scenarios for a given upper limit of radiation exposure. Only then, through comparison as per key criteria (schedule, cost, dose rate, risk, waste volume, public acceptance, etc.), one scenario among all is adopted to serve as guide for engineering work. While remote technology is not systematically used, it is often key to limit radiation exposure in compliance with the ALARA principle.

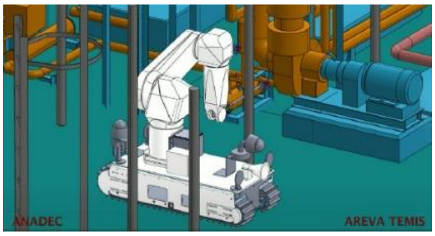


Figure 1. Example of integrated remote solution: 3D simulation for scenario selection

2-2. Nuclearization of reliable off-the shelf equipment

D&D projects are often singular and with restrained budget, thus, acquiring industrial products allows to propose reliable solutions at low price with little development. But adaptations for nuclear environments or on-site access are sometimes required: radiation hardening of sensitive parts, maintenance of contaminated device, boundary making, handling of malfunctions, accessibility to target, improvement of visibility, etc.

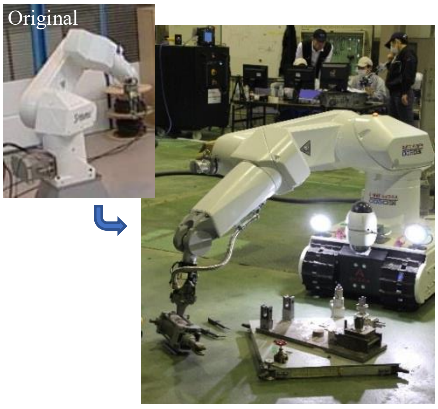


Figure 2. Example of integrated remote solution: Nuclearized industrial robot MC-TEΩ600 Mock-up and training

2-3. Development of dedicated solution

Tailor-made solutions can also be developed to answer specific needs. Ease of operation and maintenance, repeatability for other projects are basic guidelines to design versatile remote technologies for nuclear application.

2-4. Testing and training prior to on-site operation

A series of tests and training are performed in inactive conditions to verify that the operators will be able to solve any issue on-site. Tests and training using a simulator and/or full-scale mockup are carried out by stages as necessary. This step is key to the successful operation in a D&D site.

3. Conclusion

Orano has acquired unique experience in remote solutions for exacting D&D projects. ANADEC benefits fully from this experience to propose integrated solution for 1F, and its ever more demanding challenges ahead.

[i] D&D: Dismantling & Decommissioning

[ii] ALARA: As Low As Reasonably Achievable

B08 Remote Control of a Sensor-less Manipulator for Decommissioning Work

Takahiro Nagai, Masaki Odai, Kishiko Maruyama and Ryosuke Kobayashi
[1]Hitachi, Ltd., Research & Development Group

Abstract

For a maintenance and decommissioning work of nuclear power plants, a remote control system using stereo camera as external sensors is proposed for a sensor-less manipulator. The sensor-less manipulator is controlled in real time by estimating manipulator poses using a state estimation filter. This paper outlines our proposed method and describes results of an evaluation experiment using a sensor-less hydraulic manipulator.

1. Introduction

A remote-controled robot used in nuclear power plants needs to have radiation tolerance. A sensor-less manipulator is preferable in order to suppress radiation effects. In this work, we have developed a remote control method for a multi-joint hydraulic manipulator.

2. Remote Control of Sensor-less Manipulator

2-1. Proposed method

We proposed the concept of a teleoperation system for the sensor-less manipulator as shown in Fig. 1. The current joint states of the sensor-less manipulator are estimated by only stereo camera images. In order to reduce processing load for real-time control, the stereo camera detects only 3-D positions of markers which are attached on every link of the manipulator. The manipulator poses are calculated by using a state estimation filter based on the kinematic model of the manipulator.

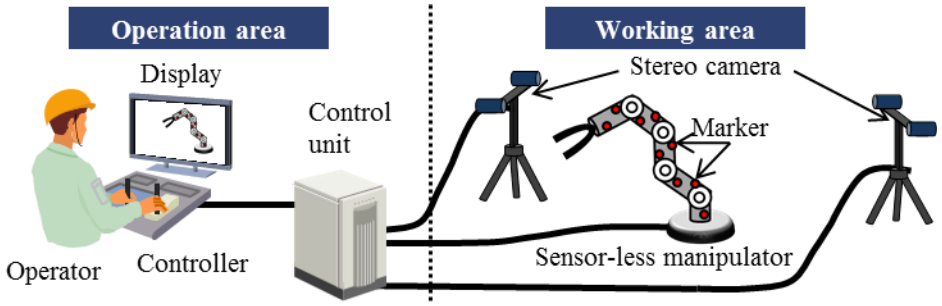


Figure 1. Concept of teleoperation system for a sensor-less manipulator

2-2. Experiment and Results

In the experiment, we evaluated calculation time of the pose estimation processing during operation of the hydraulic manipulator. The hydraulic manipulator with attached markers was set between two stereo cameras. We obtained the average calculation time, shown in Fig. 2, as 60 ms which is less than the refresh rate of the stereo camera (100 ms). We confirmed that the manipulator pose was estimated in real time.

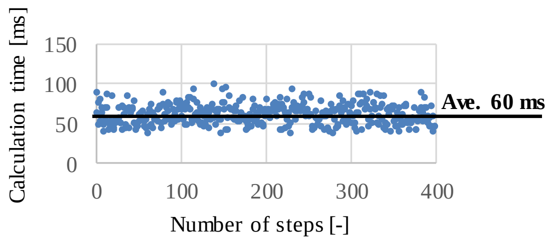


Figure 2. Calculation time of the pose estimation

3. Conclusion

We proposed a remote control system for a sensor-less manipulator using a stereo camera. We confirmed that the manipulator pose was estimated in real time and the manipulator was controlled based on the estimated pose.

B09 Introduction of Mirion Technologies's Radiation Tolerant Camera
The World's Highest Radiation Tolerant Performance Camera

Tadashi Sato, Mikio Katsura, Yuto Tajima
CORNES Technology 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. Radiation Tolerant Performance Camera

1-1. 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)

1-2. 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

1-3. SC985 Color Camera Outstation

The SC985 Color Camera outstation has been developed to provide a color image for Mid-Radiation applications in the nuclear industry. Combining high quality images with a robust pan/tilt/zoom platform for inspection and surveillance applications either underwater or in air. The option of on-board LED lights provides additional scene illumination when required.

- SC985HR Radiation Tolerance: 50kGy (5Mrad), 1kGy/hr dose rate
  SC985 Radiation Tolerance: 10kGy (1Mrad), 1kGy/hr dose rate
- High Quality Color Images with Resolution of 450 TV lines
- 10:1 NB clear glass optical zoom lens

1-4. 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)

B10

High Radiation Resistant Fiberscope using Silica based Imagefiber

Tomoaki Toriya[1], Shinya Taniwaki[1] and Manabu Kudo[1]

[1]Fujikura Ltd.

Abstract

Fujikura developed high radiation resistant fiberscope functional for particularly the high dose rate environment such as 10kGy/hr. This paper introduces some evaluation results of prototyped imagefiber.

1. Introduction

Silica glass with high-OH content core fiber is well known to perform good optical transmission against radiation. However, durability of image observation is limited due to radiation-induced absorption. We accomplished higher radiation durability by introducing fluorine doping and special treatment technologies.

The sample fiber has been evaluated in image quality and transmission efficiency under the dose rate of 10kGy/hr condition.

2. Radiation Durability Enhanced Fiberscope

2-1. Material improvement

The sample imagefiber was made of silica glass core with a tiny amount of fluorine doping, pixel number was 6,000. Sample length was 100m and middle 20m of 100m was irradiated to Co-60 gamma-ray at 10kGy/hr.

Induced loss peak of fluorine doped silica core fiber was about a quarter of conventional high-OH content silica fiber at 630nm after 10kGy irradiation. (Fig.1) However, video image after 10kGy radiation indicates this material improvement is not enough for visualization tool under high dose rate. (Fig.2)

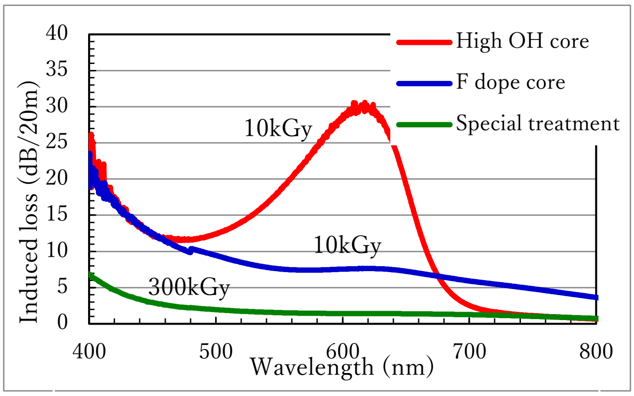
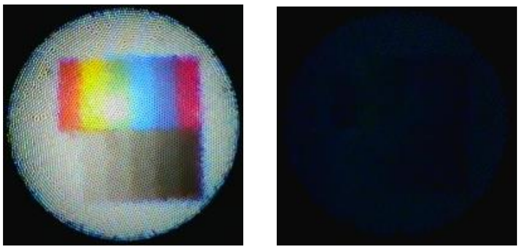


Fig 1. Induced loss after irradiation

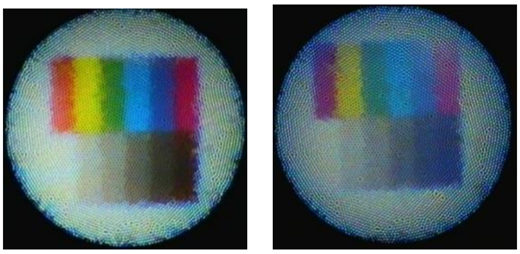


(a) Original (b) 10 kGy irradiation

Fig 2. Video image of original fiber

2-2. Special treatment for enhancing durability

We installed special treatment to the sample imagefiber to enhance radiation durability. This treatment inhibited the generation of absorption defects in the fiber and was stabilized transmission performance. It shows slight degradation on image after 300kGy irradiation at 10kGy/hr. (Fig.3)



(a) Initial (b) 300 kGy irradiation

Fig 3. Video image of enhanced fiber

3. Conclusion

This combination technology of fluorine doped silica glass and special fiber treatment accomplished the highest radiation resistant imagefiber. Fiberscope with using this imagefiber will be the effective visualization tool which performs under high dose rate such as 10kGy/hr. This technology has possibilities of applying to not only fiberscope but also single core fiber products for laser beam delivery, spectroscopy and other application.

B11 High Radiation Shielding Rubber Chips

Kenji Kitazume, Shinji Shimada, Hiroshi Kouda

Toyo Rubber Chip Co, Ltd.

Abstract

The RS-6 shielding rubber chips are new rubber chips which show excellent properties for shielding radiation inside a nuclear reactor building and in an environment with a high radiation. (Patent No.6074491)

Usage and method of application of RS-6 rubber chips

1.These were developed for multiple protection (6 layers of walls) for an emergency situation in an environment handling high radiation such as the inside of a nuclear reactor building.

2.These are very effective as shielding materials in emergency situation and shielding construction and removal are easily done.

3,Also,they are ideal for ensuring the safety of workers in an environment handling a high radiation.

Characteristics of RS-6 rubber chips

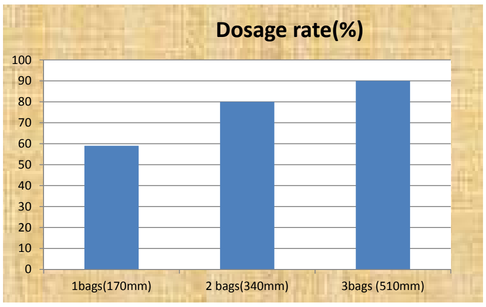
The RS-6 ,the name of high shielding rubber chips developed by us ,are in the form Of rubber chips(particle sizes of 1 ~25mm) and are processable in any desired forms (sandbags containing 10kg chips ~weather resistant flexible container with 500kg chips) . Also the RS-6 chips can be molded to various forms by solidifying.

Shielding properties of RS-6 rubber chips

Shielding dosage rate by thickness of sandbag

Thickness(mm)	Dosage rate(%)
1bag(170)	59
2bag(340)	80
3bag(510)	90

Testing condition :Blank Gamma-ray Co60 160Gy/h, the thickness of one sandbag as 170mm. With 510mm thickness, 1/10 value thickness obtained.



Composition of RS-6 rubber chips

A radiation resistant synthetic rubber with an excellent weather resistance is the main component , and the RS-6 rubber chips are composite materials in which a naturally occurring barium sulfate having a high radiation shielding efficiency is specially combined to form chips.

References

Test place; measured at Takasaki Advanced Radiation Research Institute ,JAEA .

B12 Radiation-Resistant Flexible PVC Material

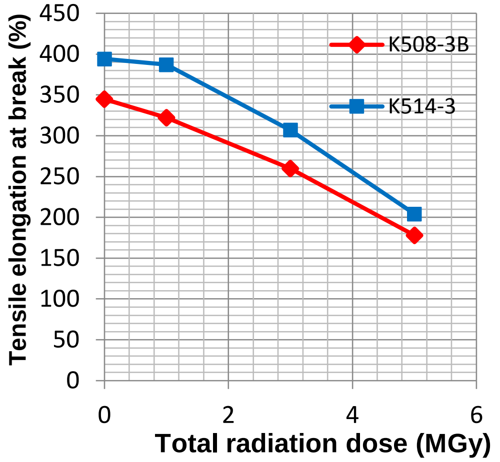
Makoto Oi, Hidenori Shima and Ryosuke Ogata

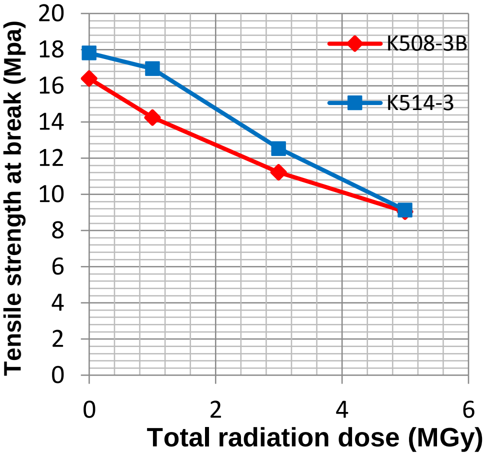
Tigers Polymer Corporation

Abstract

Tigers Polymer has developed radiation-resistant flexible PVC material with the durability of 5 MGy of gamma radiation dose.
This material maintains the elasticity and flexibility even after radiation and it can be molded into hose, tube, sheet, etc. with various shapes and diameters.
Moreover, material property is good and wear resistance is excellent.
The cost performance is satisfactory because of PVC material.

1. Gamma Radiation Test





2. Prototype

Tube

Hose

Sheet

B13 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

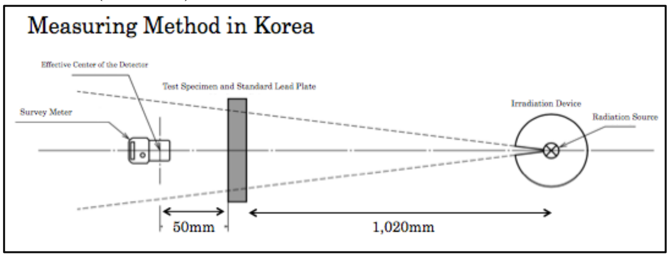
2-1. Thickness of the radiation shielding sheet

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.

C01 Development of a Fiber Optic Radiation Monitor Used in Severe Environments

Takahiro Tadokoro[1], Shuichi Hatakeyama[1], Katsunori Ueno[1], Yuichiro Ueno[1], Keisuke Sasaki[1], Yoshinobu Sakakibara[1], Toru Shibutani[1], Takahiro Ito[1], Mikio Koyoma[1] and Koji Nebashi[2]
[1]Hitachi Ltd., [2]Hitachi-GE Nuclear Energy, Ltd.

Abstract

We have been developing a fiber optic radiation monitor that uses a near infrared photon counting method. The monitor is capable of measuring dose rates from 1x10^{-2} Gy/h to 6.1x10^{4} Gy/h and dose rate linearity is within ±4% of full scale after the heat and pressure resistance test. In addition, measurement conditions of the monitor can be checked while it is operating using a semiconductor laser.

1. Introduction

It is necessary to investigate the current plant status of the Fukushima Daiichi Nuclear Power Station (1F) to provide the planning and the actual decommissioning of 1F. It is also necessary to measure the dose rate in the primary containment vessel (PCV) in any future severe nuclear accident for monitoring plant conditions.

2. Fiber optic radiation monitor

Fig. 1 schematically outlines a fiber optic radiation monitor. The radiation detector part consists of a light emission element (Nd:YAG) which generates 1064 nm photons when a gamma-ray or a laser beam is irradiated to it. In the photon measurement part, each photon is converted to an electron pulse and it is counted. In the control, analysis and database part, counting rates of photons are converted to the dose rate. The semiconductor laser part emits the 808 nm laser beam for checking the measurement condition of the monitor.

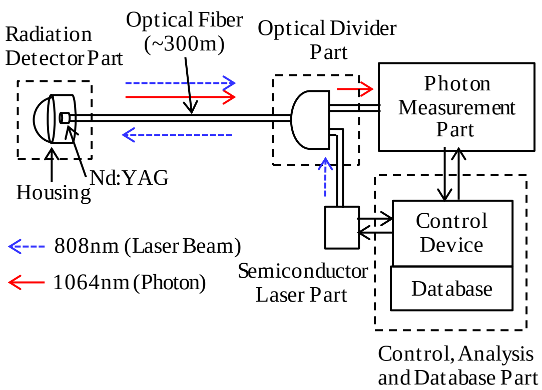


Figure 1. Outline of a fiber optic radiation monitor with a measurement condition check system

3. Test results

The dose rate linearity test results are shown in Fig. 2. We confirmed that the monitor had a capability for measuring dose rates from 1x10^{-2} Gy/h to 6.1x10^{4} Gy/h. We also confirmed the dose rate linearity was within ±4% of full scale after the 84-h heat (300°C) and pressure (1x10^{6} Pa) resistance test.

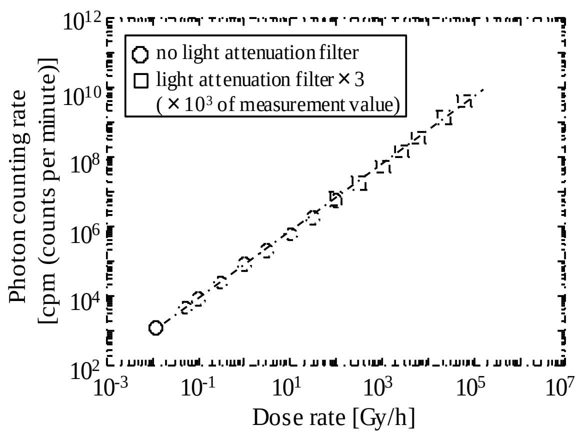


Figure 2. Dose rate linearity test results

4. Conclusion

We have been developing a fiber optic radiation monitor that uses a near infrared photon counting method. As a result of the dose rate linearity test, we confirmed the monitor was able to be used in severe environments.

C02

Progress of 1F PCV contaminant analysis by TEM observation

Akihiro Suzuki[1], Yuji Kitsunai [1], Ikken Sato[2,3]
[1]Nippon Nuclear Fuel Development Co.Ltd,
[2]Japan Atomic Energy Agency, [3]International Research Institute for Nuclear Decommissioning

Abstract

Several contaminant samples from 1F1 primary containment vessel are carefully observed by transmission electron microscope to find micro-scale uranium-containing particles in their chemical forms of U-rich c-(U,Zr)O_2 and Zr-rich t-(Zr,U)O_2, which are also observed in fuel debris of TMI-2 and Chernobyl NPP-4.

1. Introduction

As the progress of the containment vessel (PCV) survey, contaminants from the PCV have been obtained. The observation of the contaminants, which may contain trace amount of materials related to fuel debris and chemical condition during the process of debris formation, is important to know chemical characteristics of debris, FP and their distribution. In April 2017, contaminant samples were obtained from soil-like sediments at the PCV bottom of Fukushima Daiichi NPP (1F) unit 1. After the simplified fluorescent X-ray analysis in the 1F site, the contaminants are transported to Nippon Nuclear Fuel Development Co. Ltd. for further observation by SEM (Scanning Electron Microscope) and TEM (Transmission Electron Microscope) analysis.

2. SEM/TEM observation

A portion of the contaminants picked by the tip of a skewer was used for the SEM observation. EDS (Energy Dispersive X-ray Spectroscopy) analysis of the contaminants showed metallic elements of Fe, Na, Si, Zn, Al, etc. The analysis also detected small spots of U, whose distribution was not uniform as shown in Fig.1. Four spots with high U signals were selected for TEM observation. The TEM samples, which were picked up from the spots and sliced into ~100nm in thickness, were set for the TEM observation stand in a focused ion beam (FIB) device. The origins of the U signals were either a particle or a region where several parts possibly separated from one particle existed. Typical size of each particle or region was from ~300 nm to ~10 μm. Composition and crystallinity of each part or particle were either ($U_{0.8\sim0.9}$, $Zr_{0.1\sim0.2}$)O_2 with a face-centered cubic (FCC) structure, or ($U_{0.1}$, $Zr_{0.9}$)O_2 with body-centered cubic (BCC) structure. Due to the fact that the U-rich (U, Zr)O_2 (UO_2:~90mol%) and the Zr-rich (U, Zr)O_2 (UO_2:~15mol%) were the typical phases of the fuel debris in TMI-2 and Chelnobyl NPP-4, these particles are judged to have originated from the melted fuel and suggests that some of the fuel debris in unit 1 has similar chemical characteristics as those of the former severe accidents. In the presentation, information from the elements other than U will be also introduced.

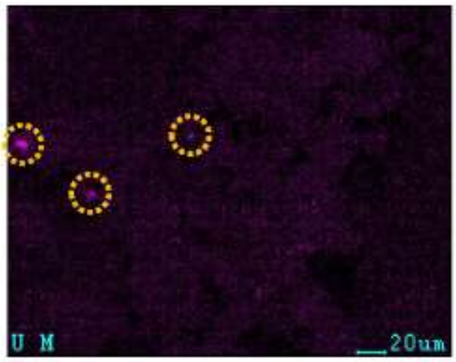


Fig.1 Uranium spots on EDS mapping

ACKNOWLEDGEMENT

This work was conducted as part of JFY2015 supplementary budget "subsidiary fund for Project of Decommissioning and Contaminated Water Management (Upgrading level of grasping state inside reactor)"

C03

Experimental Study on Neutron Induced Gamma Ray Spectroscopy for Characterization of Fuel Debris

Yasushi Nauchi[1], Tadafumi Sano[2], Hirokazu Ohta[1], Shunsuke Sato[1], Motomu Suzuki[1], Hironobu Unesaki[2]

[1]Central Research Institute of Electric Power Industry, [2]Kyoto University

Abstract

The neutron induced gamma ray spectroscopy (NIGS) was tested in the Kyoto university critical assembly facility (KUCA) to develop certification methods of reduction of the infinite multiplication factor (k_{inf}) due to stainless steel (SS) incorporated in the fuel debris and the burn-up depletion of the fuel.

1. Introduction

The fuel debris consists of irradiated fuels, SS, etc. Nevertheless, storage facilities of the fuel debris are designed based on an assumption where fresh fuels are arrayed in water in the optimum condition. To enable efficient storage of the fuel debris, the reduction of k_{inf} of the fuel debris due to the incorporated SS and to the burn-up depletion should be measured. In the present work, certification methods were experimentally studied.

2. Measurement for Capture Credit

The reaction rate ratio of the neutron capture of SS to the fission corresponds to the sub-criticality margin by the capture [1]. In the C-core of KUCA, sub-critical assemblies were mocked up with plates of SS and 93wt% enriched Uranium (93%EU) -Aluminum (Al) alloy. The assemblies were moderated by light water and driven by a ^{252}Cf neutron source. Fission and capture γ rays were counted with a BGO scintillator. The count rate ratio of (6-10 MeV) to (3-5 MeV) is monotonously related to the calculated reaction rate ratio (Fig. 1).

3. Measurement for Residual Enrichment

A reaction rate ratio of fission to ^{238}U(n,γ) is a promising indicator of the fissile enrichment. However, detection of the low energy γ rays from the activation of ^{238}U is difficult for irradiated fuels due to background γ rays (Eγ < 3.5 MeV). Then, prompt emission of a 4060 keV γ ray from a ^{238}U(n,γ) reaction was focused on. In the A-core of KUCA, sub-critical assemblies of 5.4%EU were mocked up. The assemblies were moderated by polyethylene and driven by the ^{252}Cf source. The capture γ rays were measured with a high purity Germanium detector, simultaneously with prompt fission γ rays and decay γ rays from short-lived fission products (Fig. 2)[2].

4. Conclusion

For the efficient storage of the fuel debris, experimental studies of NIGS have been conducted. NIGS is promising to certify reactivity reduction of fuel debris due to SS mixing. Also, the detection of ^{238}U(n,γ) reaction by NIGS is successful. The count rate would be useful for certifications of the residual fissile enrichment.

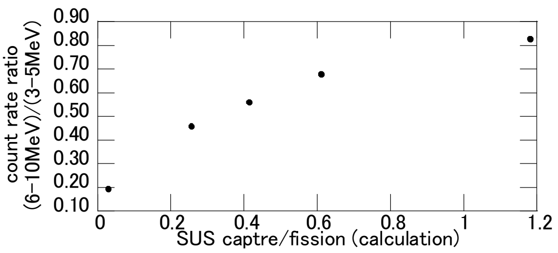


Fig. 1 Capture of SS to fission and count rates.

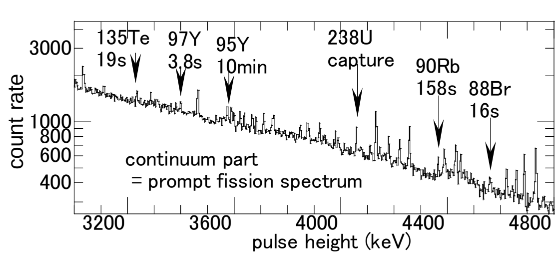


Fig.2 Detection of ^{238}U(n,γ) reactions by NIGS.

References

[1] Y. Nauchi et al., JNST, 52, 1074–1083, 2015. [2] Y. Nauchi et al., Preprints of annual meeting of the AESJ-2018, 1F03.

C04

NDA system for Fuel Debris Characterization at Fukushima Dai-ichi

Ryo ICHIMIYA, Atsuo SUZUKI, Mitsuo TAKEUCHI and Helene LEFEBVRE
Mirion Technologies (Canberra) KK

Abstract

After the Fukushima accident, many projects have been initiated by the International Research Institute for Nuclear Decommissioning (IRID) for the retrieval of fuel debris from the Fukushima Dai-ichi (1F) reactors. The IRID and Mitsubishi Heavy Industries, Ltd. (MHI) engaged in collaborative project to develop of techniques for investigating inside the reactor pressure vessel, by the 2014 supplementary budget of the Government of Japan's Ministry of Economy, Trade and Industry (METI). Under this project, Mirion Technologies undertook a feasibility study for the design of a Non Destructive Assay (NDA) system for fissile mass assessment in corium samples.

1. Introduction

At the accident 1F reactors, fuel debris may consist of several unknown mixture of concrete, stainless steel and melted spend fuel. The damaged fuel may also consist of a mixture of various types of fuels. These highly inhomogeneous materials mean that standard laboratory analysis will lead to difficulties of non-representativeness of samples. In order to solve this problem, the NDA system for fissile mass assessment in corium samples using collimated gamma spectroscopy with HPGe detectors, neutron measurement system with Cadmium wrapped ^{3}He tubes in a moderator (HDPE) and D-T pulsed neutron generator was proposed. This measurement will be done in-situ, therefore the dose rate around the NDA system must not exceed the maximum permissible value.

2. Solution

A bibliography study based on existing designs resulted in the definition of a basic design of a system:

- Collimated Gamma spectroscopy with HPGe detectors
- Neutron measurement system with Cadmium wrapped ^{3}He tubes in a moderator (HDPE), allowing for Passive (PNCC) and Active (DDA) measurement.
- D-T pulsed neutron generator
- The sample capsule is moved into a tube that goes through the whole system.
- Lead and HDPE shielding will prevent damage to the ^{3}He tubes and will maintain a low dose rate than outside the system.

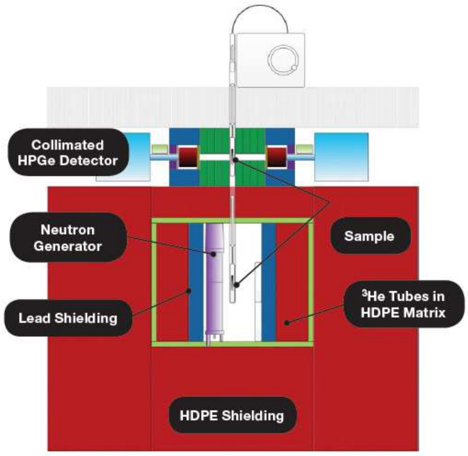


Figure 1. Conceptual System drawing

3. Conclusion

Proven NDA system technologies with standard & new proposed data algorithm were combined to show how the total measurement uncertainty can be minimized. Design was optimized in terms of performance, cost, weight and ALARA principals.

References

[1] **http://www.canberra.com/literature/case-studies/pdf/C0002_Fukushima-Case-Study.pdf**

C05 Application of MTKK technology for D&D and Fukushima remediation

Atsuo Suzuki, Ryo Ichimiya, Mitsuo Takeuchi and Helene Lefebvre
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1. Introduction

Mirion Technologies Canberra KK (MTKK) has more than 30 year experiences of various radiation measurements and safety for Decontamination and Decommissioning. These experiences are actually applied for D&D and Fukushima remediation. Especially, In-Situ Object Counting System (ISOCS) technology is clearly demonstrated as a very useful tool for a large or complex shaped object. In this presentation, some examples of the ISOCS application are reported by MTKK.

2. MTKK activities and achievement

2-1. Application for flexible container (Super-Sack: SS) measurement

The ISOCS system was applied at three sites of Fukushima Prefecture. Six SSs were very inhomogeneous, 10 and 1001 SSs were inhomogeneous. The ISOCS measurements (shown in Fig. 1) indicate very effective and accurate results in all cases. The main uncertainty for measuring the SSs is based on internal inhomogeneity and heterogeneous source distribution. So MTKK and JAEA have developed collaboratively of the Scattering Gamma Equivalent Model for decreasing uncertainty which is derived from these factors. This method decreased Total Measurement Uncertainty (TMU).

Fig.1. SS measurement with ISOCS.

2-2. Application for rubble and block measurement

The ISOCS was applied for rubble and block measurement for getting radioactivity balance before and after screening contaminated soils to reuse. The ISOCS was verified as a strong tool for effective radioactive measurements.

Fig.2. ISOCS measurement in car.

2-3. Application for in-situ and mobile monitoring

The ISOCS and Response Matrix method (RM) were applied for emergency mobile and in-situ monitoring. Both methodologies for mobile monitoring are very unique, so MTKK is collaborating with Fukushima University, NIFS and Shizuoka Prefecture about it. (Fig.2)

2-4. Application for regular D&D

The ISOCS is expected use for regular D&D, so MTKK is collaborating with some Electric Power Company. In 2017, MTKK collaborated with Chubu EPC using the ISOCS application for Reactor Building wall and floor screening at Hamaoka NPP. The results show that the ISOCS possibly contribute to cost reduction and accurate evaluation more than conventional method (Sampling and so on). (Fig.3)

Fig.3. ISOCS measurement for Reactor building wall and floor.

3. Conclusion and Future work

According to a lot of validation tests between the ISOCS measurements and the conventional methods, the ISOCS results were found to be more effective and pragmatic than those of conventional methods in all cases. Therefore, the ISOCS is considered to be utilized D&D and Fukushima remediation. MTKK and our user will try to show the feasibilities of the ISOCS for getting an approval from the regulator.

C06

Gamma-ray measurements in high-dose radiation fields using TlBr semiconductor detectors

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Abstract

Our group fabricated small-volume thallium bromide (TlBr) semiconductor detectors for gamma-ray measurements in high-dose radiation fields. The performance of the detectors was evaluated using a ^{137}Cs checking source at room temperature. Simulation study of the TlBr detectors were performed with PHITS code. Tlbr semiconductor detector exhibited detector performance suitable for Fukushima Daiichi NPP decommissioning.

1. Introduction

TlBr is a compound semiconductor with high atomic numbers (81 and 35) and high density (7.56 g/cm^3). The photon stopping power of TlBr is higher than that of other semiconductor materials such as cadmium telluride (CdTe), silicon, and germanium. Small-volume TlBr semiconductor detectors were fabricated for gamma-ray measurements in high-dose radiation fields such as Fukushima Daiichi NPP.

2. Fabrication of TlBr semiconductor detectors

TlBr semiconductor detectors were fabricated from TlBr crystals. The TlBr crystals were cut into wafers with the dimension of approximately 5 mm × 5 mm × 2 mm. The surfaces of the crystals were polished mechanically. Thallium electrodes were deposited on the wafers by the vacuum evaporation method. 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. Fig.1 shows a TlBr semiconductor detector fabricated in this study.

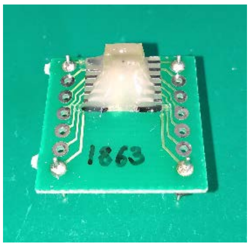

Fig. 1 Small volume TlBr semiconductor detector.

3. Simulation and Experiment

Deposit energy in the TlBr detector irradiated with 662 keV gamma-ray was calculated by PHITS code [1]. Fig. 2 shows the result of the simulation. The 662 keV photo peak was observed from simulation results

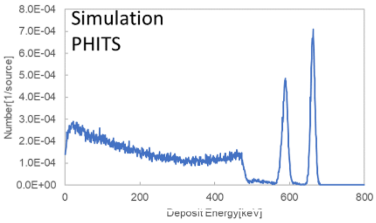

Fig. 2 Simulation results.

The small-volume TlBr semiconductor detectors were tested with a ^{137}Cs checking source at room temperature. The cathode and the anode pixel were connected to charge-sensitive preamplifiers. Negative bias voltage of 400 V was applied to the cathode. The output waveforms from the preamplifier connected to the anode pixel were recoded with a multi-channel digitizer. Fig. 3 shows a ^{137}Cs gamma-ray spectrum obtained from the small-volume TlBr detector using the near cathode events. As can be seen from the Fig. 3, the detector exhibited a clear full-energy peak for 662-keV gamma-rays despite the small active volume. The energy resolution of 4.7% FWHM was obtained from detectors.

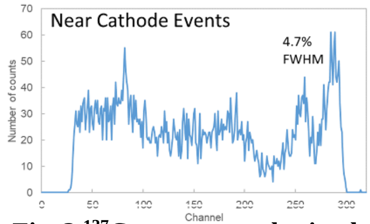

Fig. 3 ^{137}Cs spectrum obtained from a small-volume Tlbr semiconductor detector.

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C07 Piping system, risk management based on wall thinning monitoring and prediction - PYRAMID -

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1. Introduction

In the decommissioning of Fukushima Daiichi Nuclear Power Station, a flow with a high concentration debris of various kinds (concrete, corrosion, metallic…) occurs in piping when removing fuel debris. Pipe wall thinning by Slurry Flow induced Corrosion (SFC) (Fig.1) under solid-liquid two-phase has been anticipated. 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. Description of the research

2-1. Clarification of wall-thinning for SFC by experiment and numerical simulations

Wall-thinning evaluation model based on the wall-thinning rate evaluation in consideration of a mass transfer coefficient evaluated under solid-liquid two-phase flow is developed. SFC is elucidated with experiment due to the disturbance of concentration boundary layer by repeated contact of particles, the reaction rate evaluation, and the tribo-corrosion effects. The effects of solid particles on fluid factor of flow accelerated corrosion in water piping with numerical flow simulation is elucidated. As a result, the location where wall-thinning is the most severe and maximum wall-thinning rate are revealed by experiments and numerical simulations.

2-2. Development of EMAT Monitoring System

A system that can monitor pipe wall thinning by SFC with high accuracy is developed. Also, focusing type EMAT is applied to the monitoring system. It improves accuracy of measurement.

2-3. Evaluation of Engineering Risk

A quantitative evaluation method for engineering risks associated with SFC in power plant is proposed. Specifically, PoF (Probability of Failure) evaluation in consideration of the various errors (inspection / damage progress and applied force) by Bayesian estimation is considered. Moreover, the reasonable plan by integrative evaluation of factors using a risk matrix is considered.

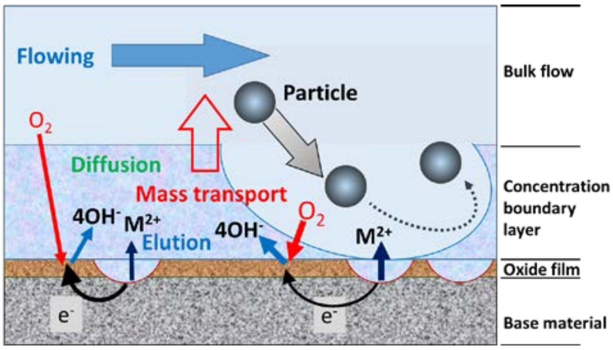

Fig.1 Pipe wall thinning phenomena induced SFC

3. Summary

This project is carried out by an international collaborative research project between Japan and France planned for 3 years since November 2017. Currently, the research is progressing towards the goal.

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.

C08 3D Reconstruction of Unit 3 Primary Containment Vessel Interiors at Fukushima Daiichi Nuclear Power Station Using Structure from Motion

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[1]The University of Tokyo

Abstract

The 3D interior structure can be reconstructed from image sequences captured by underwater robot for internal inspection of Unit 3 Primary Containment Vessel (PCV) based on Structure from Motion (SfM) technique. This recovered structure provides a better understanding of PCV interior environment and information on how best to retrieve fuel debris.

1. Introduction

To advance the plant's cleanup, fuel and other debris submerged in the coolant must be located and mapped. In July 2017, an underwater robot was sent to Unit 3 reactor for investigation [1]. The video of interior environment was recorded. We use this video to reconstruct the 3D structure of Unit 3 PCV interiors based on SfM technique.

2. Method

It is impossible to generate a 3D model of the whole scene in one go due to the huge size of the video. Moreover, the scene continuity is interrupted by floating particle noise. Thus, we propose following steps to reconstruct the 3D model.

2-1. Whole video segmentation

The whole video is divided into several clips based on the scene. The scene filled with floating particles are removed from the video.

2-2. Image extraction

Better quality images are extracted from each clip at three frames per second. The extracted images can keep the view is continuous also certain view disparity for 3D reconstruction.

2-3. Chunk 3D model generation

Extracted images of each clip are utilized to generate chunk 3D model by PhotoScan [2]. Furthermore, the relative trajectory of robot can be estimated.

2-4. Merging into whole 3D model

Separated chunks are merged together as a whole 3D model. Each location of chunk model is determined by arranging the whole robot trajectory as a smooth one. Figure 1 shows a Unit 3 interior model by merging two chunks.

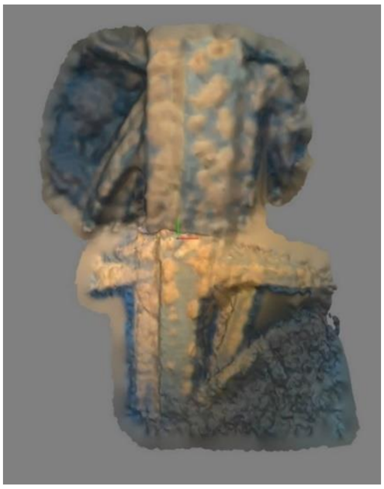

Figure 1. An example of part of Unit 3 PCV interior 3D model

3. Conclusion

The 3D model of PCV interior can be recovered by SfM technique using only image sequences taken by the robot. This reconstructed 3D model may give a better understanding on interior environment of PCV.

Acknowledgements

The authors would like to thank International Research Institute for Nuclear Decommissioning (IRID) and TOSHIBA for providing the image data.

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C09

Research on Numerical Simulation of MCCI using MPS method

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Abstract

Because the status of fuel debris depends on progress of MCCI, numerical simulation of MCCI is needed for decommissioning of Fukushima NPP. In this research, the gas model was focused as a factor affecting the shape of concrete ablation and it was shown that gas model influenced the temperature distribution in corium pool and this affected ablation geometry.

1. Introduction

Results from CCI experiment and simulation of CCI suggested that gas content of concrete and whether gases were generated from bottom or sidewall of concrete cavity influence concrete ablation in MCCI [1,2]. Then, in this research, the numerical simulation of CCI experiment using MPS method was conducted treating location of gas generation as a parameter. In addition, its effect for ablation shape was discussed.

2. Simulation method

MPS method is one of particle methods and employs Navier-Stokes equation and equation of continuity as governing equation. In this simulation, particle distance was 0.01m and the number of particles was 6992. To simulate forced circulation caused by gas generation, corium particles neighboring the concrete whose temperature reached the point of dehydration or decarbonation had smaller density, which was shown in Fig. 1.

3. Results

Fig. 2 shows ablation depth in axial and lateral directions. The results show that gas model in bottom enhanced axial ablation and the model in sidewall enhanced lateral one.

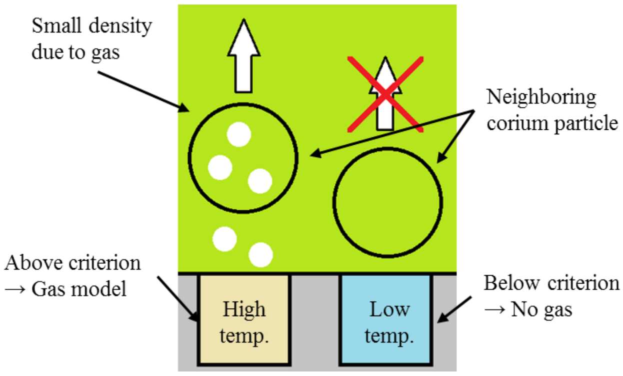

Fig. 1: Image of gas model

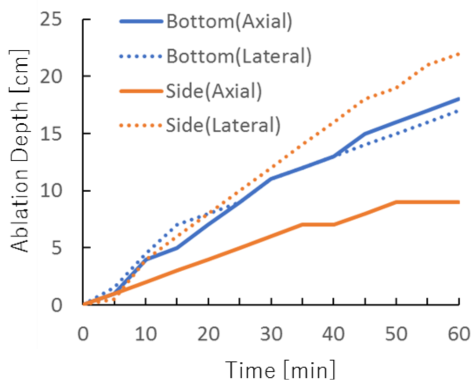

Fig. 2: Simulation results

4. Conclusion

Location of gas generation affected ablation depth. It was supposed that gas generation enhanced ablation of concrete which was close to the generating spot by changing temperature distribution.

References

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C10 Structural deformation monitoring for safe dismantling of nuclear facilities

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Abstract

The dismantling work of Fukushima Daiichi Nuclear Power Plant requires a strict safety control due to its huge weight structure and radioactive waste comprised in it. Recently a monitoring system has been developed to monitor social infrastructures such as roads, allowing more efficient maintenance management and a longer service life of the structure. We propose this technique to ensure safe dismantling work of nuclear reactor facilities.

1. Introduction

Because of its huge weight structure and radioactive waste in it, the dismantling work of nuclear reactor facilities requires a stricter safety control than any other field and necessitates before its completion a long period of time, during which some structural instability could occur and persist. We propose a monitoring system which makes it possible to monitor remotely static and dynamic behavior of aimed structure as structural safety management tool for dismantling work of nuclear reactor facilities.

2. Structural monitoring system

The monitoring system that we introduce, called OSMOS system, is already widely used worldwide in particular in Europe. Figure 1 and Table 1 show the outline of its system: the data measured by the sensors is unified, treated and shared in the Cloud, from which the user can obtain any piece of information in the form of processed data such as graphs or frequency analysis. There are two types of sensors: one is autonomous wireless sensor in which all the functions, such as data measurement, collection and communication are integrated. The other is cable type in which several sensors are independently connected to the data acquisition unit. Both types have an alert function: once the preset threshold value exceeded, an alert is sent to the preselected recipients by mail or SMS with the measured value. The sensors of this system can be installed in a short time. The wireless type sensor is particularly suitable for monitoring of structure in an irradiated environment where the working time is limited.

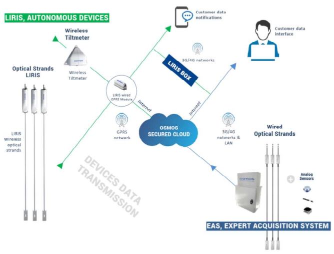

Fig-1 Monitoring system

Table-1 Main characteristics for sensor and system

Wireless Optical Strand	Length	2m or 1m	Wired Optical Strand	Length	2m or 1m
	Measurement rage	4mm(2m),3mm(1m)		Measurement rage	4mm(2m),3mm(1m)
	Sampling frequency	50Hz		Sampling frequency	100Hz
	Temperature renge	-10℃to40℃		Temperature renge	-20℃to60℃
	Resolution	0.02mm		Resolution	0.02mm
	Battery life	1.5to2years		Measurement	Static and dynamic
	Measurement	Static and dynamic			
Wireless Tiltmeter	Measurement rage	+/- 15 degree	Data acquistion unit	Ambient temperature	-10℃to52℃
	Resolution	0.009 degree		Caminet	unsert 19",in height
	Sampling frequency	10Hz		Sampling frequency	100Hz
	Temperature renge	-10℃to40℃		Temperature renge	-10℃to52℃
	Acquistion unit	Battery drive		Measurement	Static and dynamic
	Battery life	minimum 1year		power supply	100V or 260v ca
	Measurement	Static		Connectable sensors	Op.sentor:12 Non Op.senor:18

References

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[2] http://committees.jsce.or.jp/opcet/system/files/monitoring_interim_report_final_1.pdf

D01

Development of New Radioisotopes Selective Adsorption Media in EBARA

Takashi Sakuma[1], Makoto Komatsu[1], Tatsuya Deguchi[1],
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[1]EBARA Corporation

Abstract

EBARA has been developing radioisotopes selective adsorption media for treatment of contaminated water since 2011. During this period, 900 kinds of adsorption media were supplied from some manufacturers and the radioisotope removal efficiency was tested. These media have been successfully used at NPS in Japan to process radioactive contaminated water and contribute the settlement of severe accident occurred in March 2011. This paper introduces the media which has high removal efficiency of radioisotopes.

1. Radioisotopes Selective Adsorption Media

1-1. Advantages

These media have big advantages as follows, 1) large decontamination factor (DFs), 2) increases media troughputs, 3) reduction of waste generation, 4) reduction of radiation exposure to operators by reducing the frequency of media exchange.

1-2. Adsorption Media

EBARA media consist of organic or inorganic material. These media can remove Cs,Sr,I,Sb and Ru. EBARA also developed the media which can remove Cs and Sr at the same time.

1-3. Test Results

The batch test results are shown in Figure 1 and 2.

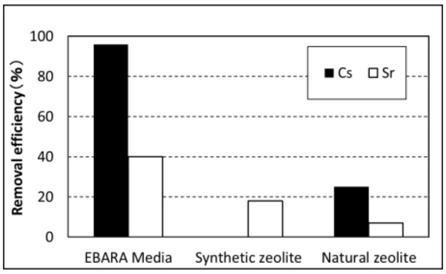

Figure 1 Cs/Sr Removal efficiency under seawater condition

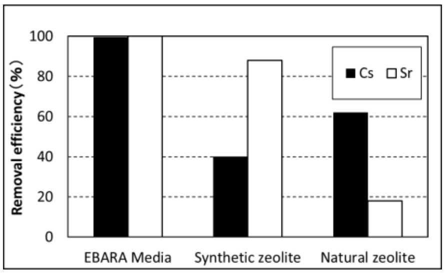

Figure 2 Cs/Sr Removal efficiency under diluted seawater condition

2. Conclusion

EBARA media removes specific ions from contaminated water efficiently and economically, are available in several process to accommodate the needs of any project. We are going to develop the radioisotopes selective adsorption media for farther waste water treatment such as decommissioning.

D02 Development of a Treatment System for Radioactive Spent Ion Exchange Resins

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EBARA Corportion

Abstract

A new ion exchange resin solidification system, featuring the combining of resins with paraffin wax, has been developed. The resulting solid matters, consisting of resins and paraffin wax, were able to be incinerated easily by our own test incinerator and the actual incinerator in nuclear power plants (NPP). Therefore, there is no longer a need to store spent ion exchange resins, as low-level radioactive waste that had been generated from water purification systems at NPP.

1. Introduction

In NPP, a great amount of bead and powdered ion exchange resins are widely used for condensate demineralizers, condensate filters, reactor water clean-up systems, fuel pool clean-up systems, and radioactive waste liquid treatment systems. After usage, spent ion exchange resins are stored in NPP and the amount of radioactive spent resins is gradually increasing. Due to the limited storage capacity, spent resin storage issue is becoming one of the major concerns for stable plant operation. At some plants, spent ion exchange resins are incinerated with other burnable wastes such as paper wastes in existing incinerators. However, due to the effort in reducing burnable wastes in the plants for rational purpose, the lack of the burnable wastes is causing some difficulties to incinerate spent ion exchange resins. Consequently, in order to dispose spent ion exchange resins continuously and stably, we have been developing a new solidification system by combining ion exchange resins with paraffin wax. These solid matters were able to be incinerated easily in our small-scale test incinerators and the actual incinerator in NPP.

2. The solidification system

Spent resins are mixed with paraffin wax, calcium hydroxide and stabilizing additive for the continuous solidification with mixing equipment. After mixing, the mixture is heated and extruded by extruder and cut into the suitable length, and produced solid matters are collected in products tank. Finally, these solid matters are incinerated in existing incinerator.

Figure 1. Products

3. Conclusion

1) Ion exchange resins were able to be solidified stably and continuously combining with the paraffin wax by the addition of the stabilizing additive using extruder.
2) The solid matters were able to be incinerated without burnable materials in small-scale test incinerator and the actual one in NPP. The concentration of sulfur oxide in off gas was kept very low by adding calcium hydroxide.
3) The amount of residual ash after the incineration was very small and the weight and volume reduction ratio were both more than 95%.

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D03

Inhibition of Oxidative Decomposition of Ion-exchange Resins

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Abstract

Hydrogen peroxide (H_2O_2) generated by the radiolysis of water exists in nuclear power plants and it accelerates the oxidation decomposition of the ion exchange resins[1], and finally, it shortens the resin life. The application of Palladium (Pd) doped resins which can decompose H_2O_2 has been considered.

1. Introduction – Pd doped resins

Hydrogen peroxides can be easily decomposed by the catalytic reaction of platinum group metal such as Pd. Pd doped resins are developed and manufactured by Lanxess Company and are strongly basic, gel-type, palladium-doped anion exchange resins.

2. Experimental

2-1. Decomposition behavior of hydrogen peroxide – Long-term column operating test

To evaluate the stability of the H_2O_2 (2 mg/l) decomposition performance by Pd doped resins, long term column (inner diameter: 16 mmϕ, bed height: 10 mm) operating test was carried out (flow rate: 3.5 ml/min, LV=1 m/h). For 4 years in cold tests, Pd doped resins showed the stable H_2O_2 decomposition performance.

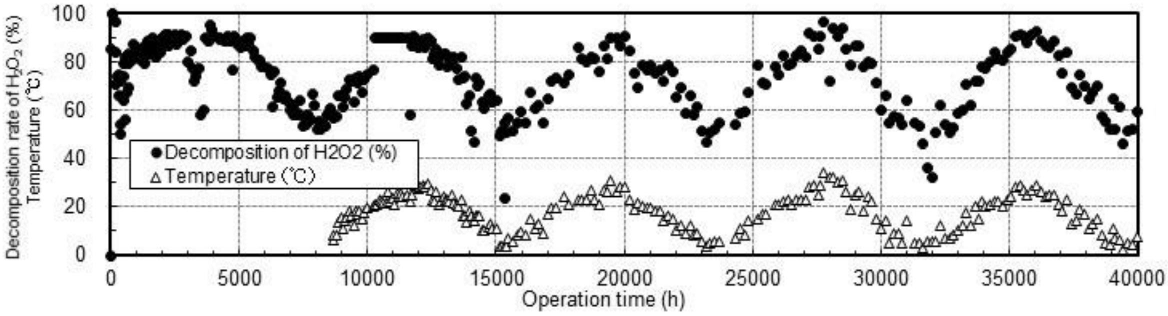

Figure 1. Lifetime of Pd doped resins

2-2. TOC leaching behavior – Inhibition of oxidative decomposition of ion-exchange resins

In order to confirm the improvement effect of the ion-exchange resin degradation tendency by the Pd doped resins, TOC leaching behavior evaluation tests were achieved by using a closed-loop test apparatus (inner diameter: 25 mmϕ, conventional resin bed height: 150 mm, LV=40 m/h, tank volume: 5 liters, H_2O_2: 5 mg/l). Figure 2 shows the results of TOC leaching behavior evaluation test. In case of the loop without Pd doped resins, TOC concentration showed an upward tendency from 500 hours later. On the other hand, in case of the loop with Pd doped resins (26% overlay), the TOC concentration little rose and it was less than 1mg/l at 2500 hours later.

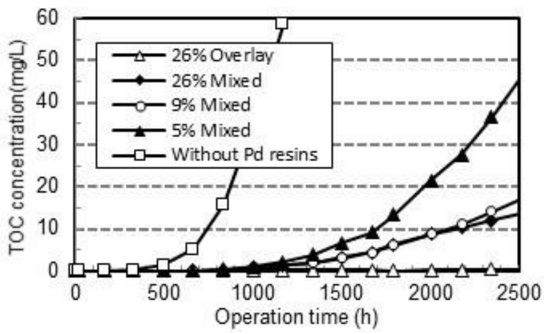

Figure 2. TOC leaching behavior

3. Conclusion

It is expected that Pd doped resins contribute to the extension of the resin life and the reduction in radioactive waste generation.

References

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D04

Alpha nuclide absorbent, TANNIX

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Abstract

MNF developed insoluble tannin absorbent, TANNIX, and it has been successfully applied for alpha nuclide absorption. TANNIX at a nuclear power plant and a variety of nuclear facilities. TANNIX has shown its excellent adsorption capacity and distribution coefficient (Kd) of well above 10^3 against alpha nuclides. It can contribute to Fukushima-Daiichi decommission when alpha nuclide needs to be removed from the cooling water.

1. Introduction

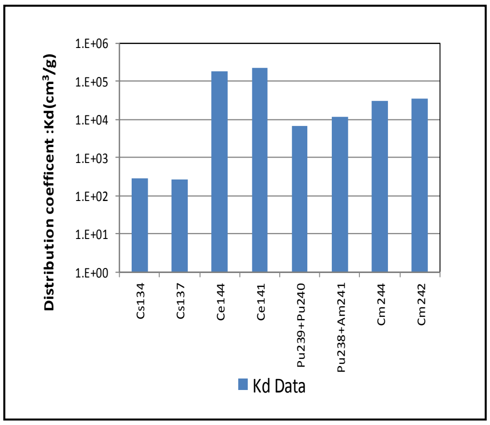
MNF developed insoluble tannin absorbent, TANNIX, originally to remove Uranium from waste water of MNF Re-conversion facility processing from UF_6 to UO_2 powder. TANNIX has processed more than 100,000 m^3 at MNF by using industry size unit. The major features of TANNIX are 1) its high absorption rate for alpha nuclides and 2) almost “0” secondary waste volume after burning.

TANNIX has been used at Paks-2 NPP and serval facilities in Japan such as JAEA plutonium fuel fabrication facility [1]. In this paper, TANNIX performance at Paks-2 NPP is summarized because it would be useful to evaluate its ability to Fukushima-Daiichi cooling water.

2. Joint research and result at Paks-2 in Hungary

During the chemical cleaning of corrosion products on the surface of fuel cladding in Paks-2 VVER-440 reactor overheated to severely damaged [2]. The waste water including some radionuclides and about 20g/dm^3 boric acid (pH=4.1) was generated.

Budapest University of Technology (BUTE) and MNF jointly researched to treat the waste water. BUTE examined equilibrium test with real radioactive waste. The distribution coefficient (Kd) of TRU element was well above 10^3 . After testing, TANNIX was used and the waste water was successfully disposed.



3. Conclusion

TANNIX, developed by MNF, has been used at several facilities including at Paks-2 NPP processing the water with nuclear fuel and boric acid.

MNF thinks that TANNIX can be one of the strong candidates of alpha nuclide absorbent for Fukushima-Daiichi cooling water. It is desirable to confirm the applicability of TANNIX to the cooling water.

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D05

Study on Application of Laser-Decontamination in Decommissioning

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Abstract

Laser decontamination is applicable to the decommissioning of nuclear power plants as a decontamination technology, because secondary waste is expected to be considerably reduced.

1. Introduction

In the field of civil engineering and construction, portable laser paint removal devices (Fig. 1) for steel bridges may be applicable as radioactive decontamination technology for decommissioning measures.

Fig. 1 Portable laser paint removal device

We undertook a study on the feasibility of application of laser decontamination in decommissioning, with confirmation tests on the removal rate of the stainless steel surface and on the dust debris (primary waste) recovery apparatus.

2. Confirmation test of decontamination speed

To estimate the laser decontamination processing speed, we irradiated the surface of steel materials using a 500 W fiber laser, and measured the amount of stainless steel surface material removed. (Fig. 2)

Under optimal irradiation conditions, when estimating the processing speed for a thickness of 10 μm, the removal is 5 hours per m² with the 500 W laser.

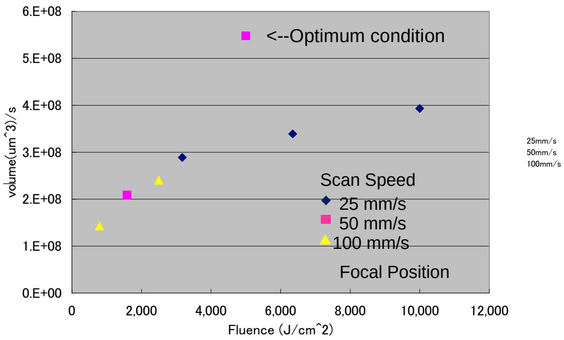


Fig. 2 Laser fluence dependency of surface volume removed

3. Confirmation test of the debris collection apparatus

Even with laser decontamination, it is necessary to collect the decontaminated radioactive debris. In order to prevent dust scattering, we realized a decontamination method in water (Fig. 3).

The mechanism for exhaust and collection of the debris in water is provided at the tip of the laser irradiation part, and the removed debris is recovered together with water and removed with a filter.

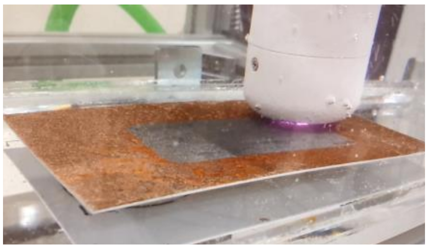


Fig. 3 Prototype underwater laser decontamination device

4. Results and future work

This study revealed that laser decontamination is applicable to decommissioning as a decontamination technology.

Presently, we are working on the selection of decontamination targets (systems, equipment, and parts) and the specification of devices appropriate for this, which will become the key to practical application.

D06 Sorting Technology for Efficient and Safe Soil Processing

Steve Rima[1], Jeffrey Lively[1], Masaru Noda[2], and Haru Hashizume[2]
Wood[1] and Obayashi Corporation[2]

Abstract

Obayashi and Wood have been working on the application of sorting technology for radioactively contaminated soil after Fukushima incident. That includes the adaptation and upgrade of well-proven system in the US to the scopes and specifications for Fukushima offsite cleanup. Its adaptability is further expected to extend into onsite work demands in any challenges. This paper with oral presentation was made at American Nuclear Society Joint Topical Meeting on Remote Control and Robotics in Pittsburgh in August 2016.

1. Conceptual ISF Process Train Design

The sheer volume of solid radiologically impacted materials that need to be processed through the ISFs challenges capacity and throughput of the existing, ready-to-deploy *ScanSort*^{SM} system. The conceptual design for the ISF process train includes processes other than simply assaying and sorting for radioactive content. An appropriate scale for the process train needed to be determined and the electro-mechanical function of the process train needed to be engineered as the *ScanSort*^{SM} system could not function as a “stand-alone” process.

The Obayashi teams design for the ISF process train considered a number of factors to arrive at the design specification. It was determined that a network of nine independent and parallel process lines would be needed to meet the required productivity of 180 cubic meters per hour. Some of the constraining parameters included the limitations for material handling and feeding of 1 cubic meter bags, the desire for built in redundancy and flexibility of scale, as well as the expected volume of material and the proposed time line for ISF operations.

2. Generation II *ScanSort*^{SM} Design

In order to achieve the desired endpoint of a single point of command and control from which an entire process train’s operation would be controlled, Wood has need to reengineer the *ScanSort*^{SM} system. The reengineering efforts fall into two major categories:

- Decoupling of the sensor systems from the electrical power distribution system, the I&C system, and the electro-mechanical control components, and
- Redesign of the *ScanSort*^{SM} operating system software to support the decoupling of the I&C system and the redistribution of command and control functions to non-dedicated electro-mechanical modules

Wood has completed the redesign of the *ScanSort*^{SM} operating system software. In addition to the changes necessitated by the decoupling of sensors system, the redesigned software will provide for the distribution and remote display of real-time system information to a centralized command and control station.

Obayashi and Wood are collaborating on the specifications for the engineering effort required to produce new standardized interfaces for electrical power, I&C signals, and modular mechanical attachment of the *ScanSort*^{SM} system’s components to the mechanical systems sourced completely in Japan.

3. Modification to Process Heavy Clay-Like Soils

Obayashi has previously designed a conveyor, called an “extruder,” to enable efficient processing and sorting of heavy, clay-like materials. The extruder doesn’t simply place material onto the Survey Conveyor belt like the original *ScanSort*^{SM} system, but “squeezes” it out onto a consistent layer on the belt using a roller. For a project in the U.S. with this type of material, Wood has built and deployed an extruder based on Obayashi’s design. The extruder is in current use and has been proven to handle and sort very wet, heavy, clay-like soils with no clogging or other issues that have been experienced by other sorting systems. The addition of an extruder to a *ScanSort*^{SM} system for use in Japan will enable sorting of soils in Japan to be done efficiently while preventing the problems encountered by other sorting systems.

D07

Assessment of Future Countermeasures against Contaminated Groundwater in Fukushima Daiichi Nuclear Power Station

Tadafumi Koyama[1], Yasuharu Tanaka[1], Daisuke Tsumune[1], Takatoshi Hijikata[1]
Mamoru Kamoshida[2], Takeki Yamauchi[2], Masaki Imazu[2] and Toshihiko Fukuda[2]
[1]CRIEPI, [2]NDF

Abstract

Analytical study to evaluate radiological risk, residual contamination and amount of major consumptions was carried out for comparing future countermeasures against contaminated groundwater in Fukushima Daiichi.

1. Introduction

The highly contaminated water in turbine buildings is stably confined by countermeasures, and will be dried-up completely before fuel-debris retrieval. The groundwater around reactor buildings is far less contaminated, and protected by the same countermeasures, however, it is important to evaluate the future radiological risk and assess appropriate countermeasures because groundwater will remain even after debris retrieval.

2. Analytical method

Future distributions of H-3, Sr-90 and Cs-137 in groundwater and soil around reactor buildings were calculated with new 2-D countermeasure model applied to FEGM code.[1] Public radiation dose by ingestion of sea foods was calculated from radioactivity permeated through sea-side impermeable barrier. Consumption of adsorbers (assumed properties of IE-96/A-51) to clean pumped groundwater was estimated from radioactivity transported to subdrain pumps.

Table 1. Effectiveness comparison

	①	②	③	④	⑤	⑥
Soil contamination	ref	→	↘	↓	↓	→
adsorber consumption	ref	↓	↓	↗	↗	↘

(→ same ↘ down ↓ much down)

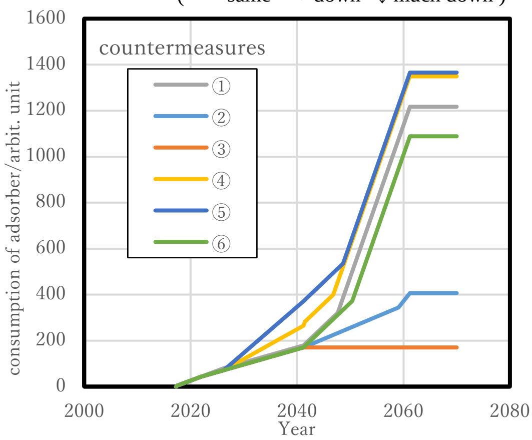


Figure 1. Cumulative adsorber consumption

3. Results and Discussions

Analysis was carried out for the future with reference countermeasure①that consists of subdrain and sea-side barrier after debris-retrieval. The obtained radiation dose, as low as 1E-5 mSv/y, showed the effectiveness of this measure, while adsorber consumption is huge as shown in ① of Figure 1. Hence, comparative study was carried out for estimating possible performance of alternative countermeasures: ② subdrain, frozen soil barrier and sea-side barrier, ③sea-side barrier with permeable reactive barrier to trap Sr, ④subdrain, adaptive pump&treat for known plumes and impermeable barriers, ⑤subdrain, soil flushing and impermeable barriers, ⑥subdrain, injection of chemicals to immobilize Sr-90 and impermeable barriers. All countermeasures were found to keep public radiation dose as low as 1E-5 mSv/y, while reduction of soil contamination or consumption of adsorbers depend on each measure. As shown in Table 1, ④ and ⑤ are the most effective measures to reduce soil contamination. On the other hand, total consumption of adsorbers can be drastically decreased to 1/6 of ① by ③ or 1/3 by ② as seen in Figure 1.

References: [1] T. Koyama et al.," Environmental risk analysis for radionuclides transported in groundwater of Fukushima daiichi", poster K02, 2nd Int'l Forum on the Decommissioning of Fukushima Daiichi NPS, July 3, 2017.

E01

Construction of Unit-3 Cover Structure for Fuel Removal

Shinya Okada, Kenichiro Go, Norio Ryoki, Taimei Hida,
Miho Miyazaki, Ippei Matsuo, Kihei Ogawa
Kajima Corporation

Abstract

We have successfully decreased the radiation dose rates by way of rubble removal, decontamination and radiation shielding that we incorporated ten remote control technology and we have completed construction work of Unit-3 cover structure for fuel removal in February, 2018. This paper summarizes the challenges and accomplishments of the project over seven years.

1. Reducing the radiation exposure of workers

- We have developed remote operation system for heavy equipment. Ten pieces of heavy equipment were operated at the same time controlled from an operation room located approximately 500m away.
- We have also developed a labor-saving joint of steel column members for assembling the base of the cover structure.

2. Removal steel frame rubble collapsed due to hydrogen explosion

- The modeling of the steel flame rubble was performed using 3D leaser scan, and the behavior of steel frame rubble during dismantling was investigated using finite element analysis.
- Various crane attachment to cut and grip steel frames by remote operation were developed.

3. Reduction of dose rates for manned work indispensable for cover structure construction

- Various remote control devices for rubble removal and decontamination were developed for the various floor damage and contamination condition.
- The shielding steel plates were installed with the maximum weight within supporting capability of each floor by remote control.

4. Design of cover structure

- The dorm-shape steel frame were employed for the cover structure in order to secure a large space with a relatively lighter weight.
- The vertical and horizontal loads to the reactor building were reduced as much as possible.
- A labor-saving joint was developed for steel column members in order to reduce manned activities.

5. Safe and efficient manned works on the operating floor

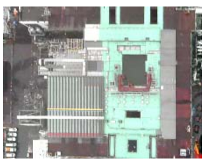
- Amount of manned activities at 1F site were reduced by practicing fabrication outside 1F site.
- Practicing fabrication of the cover structure outside 1F site could eliminate problems in advance and increase the worker’s learning to a higher level.

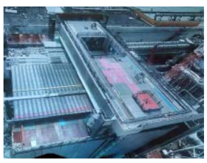
6. Reducing risk of damaging fuel

- The spent fuel pool was covered with protection covers not to damage fuels during rubble removal.
- Cover structure units were set in place by sliding them to avoid construction work over the spent fuel pool.


Before
Rubble Removal


After
Rubble Removal


Radiation shielding
plate installation


Fuel handling machine
girder installation


Completion of cover
structure

E02 Land-side Impermeable "Frozen-Soil" Wall

Atsuhiro Fukada, Isao Abe, Taichi Esaki, Shoji Seno,
Teru Yoshida and Torahito Suzuki
Kajima Corporation

Abstract

The land-side Impermeable Wall with the method of artificial frozen soil, the one of the longest frozen soil wall ever made, was constructed around the damaged reactor buildings, as a measure against the increase of contaminated water. It had been serious issue that the major volume of groundwater had become the contaminated water due to mixing up with the radioactive contamination in the damaged reactor buildings. The Frozen-Wall cuts off this groundwater flow toward those reactor buildings, and prevents from the further increase of contaminated water.

1. Background

Since the accident, the inflow of the groundwater into the reactor buildings was daily estimated up to 400 m^3/day, and it had been required to shut off and reduce this major volume of the flow. To cope with this issue, the Committee on Countermeasures for Contaminated Water Treatment, organized by METI, decided a basic policy in May, 2013, i.e. the installation of the Frozen-soil Walls.

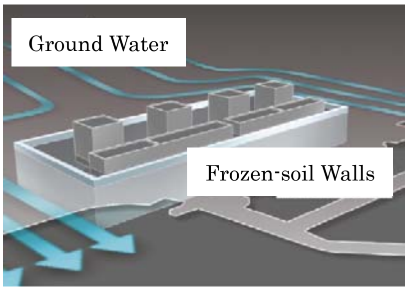


Figure 1. Concept of the Frozen-soil Wall

2. Construction

The construction process to create seamless Frozen-soil wall consists of two phases; 1) the installation of the required facilities and 2) the freezing operation. The facilities-installation was commenced in December 2013. One of the most important facilities in-situ are 1,568 vertical freezing pipes installed into the ground, approximate 30m deep. The following phase for freezing soil has been operated since March 2015, to circulate brine solution through the freezing pipes at -30 degree Celsius. In addition, Grouting method was also complementally and partially applied to advance the development of the Frozen Wall, reducing the velocity of groundwater flow. The Committee and TEPCO announced the completion of the Frozen-Wall in March 2018.

3. Result and Effect

It is recognized that the ground water level inside of the Wall is 5m lower than the outside. Furthermore, the growing-amount of the contaminated water is reduced to 140m^3/day, compared with the amount before close of the Frozen-soil wall, 520m^3/day. According to these results, The Committee evaluated that the Frozen Wall effectively shut off the groundwater flow and prevent from the increase of contaminated water.

References

[1] http://www.meti.go.jp/english/earthquake/nuclear/decommissioning/index.html

[2] http://www.tepco.co.jp/en/decommision/planaction/landwardwall/index-e.html

[3] http://www.kajima.co.jp/tech/c_frozen_soil_wall/index.html (in Japanese)

E03 Technologies for dismantling radioactive contaminated water storage tanks at Fukushima Daiichi Nuclear Power Station

Kazushige Kato[1], Masahiro Nakamura[1], Haruo Nagamine[1],
Naoki Maeshiro[2], Hiroshi Nakamura[3], and Yoshihito Nakahara[4]
[1]TAISEI CORPORATION, [2]Tokyo Electric Power Company Holdings, Inc.,
[3]TOYO UNION Company Limited, [4]TAIYO KOGYO CORPORATION

Abstract

The developing technologies for controlling radioactive dust, when the radioactive contaminated water storage tanks are dismantled at Fukushima Daiichi Nuclear Power Station (1F) are described in this paper.

1. Introduction

The bolted-type tanks for storing radioactive contaminated water constructed at the early stage of decommissioning of 1F have been replaced by the welded-type tanks to avoid water leakage due to aging degradation of sealing material. Since radioactive dust is adhered to the inside of tanks, reducing scattering to the environment and worker's radiation exposure are highly demanded during the dismantling the tanks.

2. Technologies for dismantling bolted-type tank

2-1. Remote spraying device for controlling dust

Before dismantling a tank, the device is inserted from a top of tank to a bottom and the two spray nozzle arms are expanded to right and left. The viscous liquid for retarding dust discharge is sprayed from the nozzles on the inside walls of tank by rotating about the center rod of device from bottom to top of tank by remote control (see Figures-1 and 2).

2-2. Balloon cover for reduce scattering of dust

During dismantling a tank, the balloon is covered at a top of tank after daily work in order to reduce scattering of dust as well as to prevent admitting rain into a tank (see Figure-1 and 3). Since installation and de-installation of cover is daily work, it is crucial to make working hours shorten. The developed balloon cover is frameless and is light enough to install by small crane in a short time. The cover is removable by tightening belts of cover with mounting brackets on a top of side wall at about one meter interval. Thus, the cycle time of the work is to be about 40 minutes.

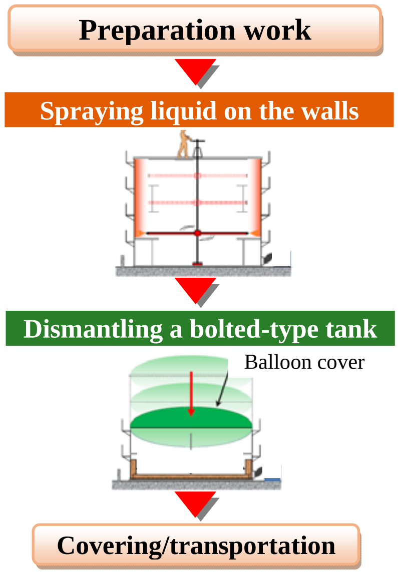


Figure-1 Tank dismantling flow

3. Conclusion

The availability of these technologies was confirmed at the site, and the worker's radiation exposure was kept minimum in consequence.

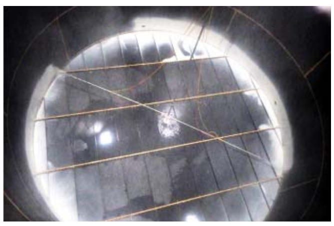


Figure-2 Spraying white liquid


Figure-3 Balloon cover

E04

Dismantling Plan of Exhaust Stack for Units 1 and 2

Hiroshi Noda[1], Yoshihisa Michizuka[1], Takashi Okutsu[1]
[1]Tokyo Electric Power Company Holdings, Inc.

Abstract

Damage has been confirmed in some parts of the exhaust stack by inspection after the earthquake. Currently the exhaust stack has no function. For this reason, we decided to dismantle the upper half of exhaust stack for the purpose of improving the earthquake resistance. Emphasizing on reducing worker exposure dose, we plan dismantling work with unmanned operation using a remote dismantling device.

1. Introduction

The exhaust stack consists of a rectangular steel tower and a cylinder stack which is supported by the steel tower. Height of the cylinder stack is 120 m and inside diameter of it is 3.2 m.

The steel tower part mainly consists of main pillars, diagonal braces and horizontal parts. We plan to partially dismantle the exhaust stack. The dismantling range is from around 59 m to 120 m above the ground as the diagonal nine braces around 66 m and 45 m above the ground are damaged.

2. Exhaust Stack Dismantling Plan

We plan to dismantle the stack and steel tower on a block-by-block basis using a large crane which is used in fuel removal works. Saving labor is taken in by using dismantling device which has cutting and holding functions.

First the stack which is protruding is dismantled and then sequentially the dismantling of steel tower and the stack is repeated.

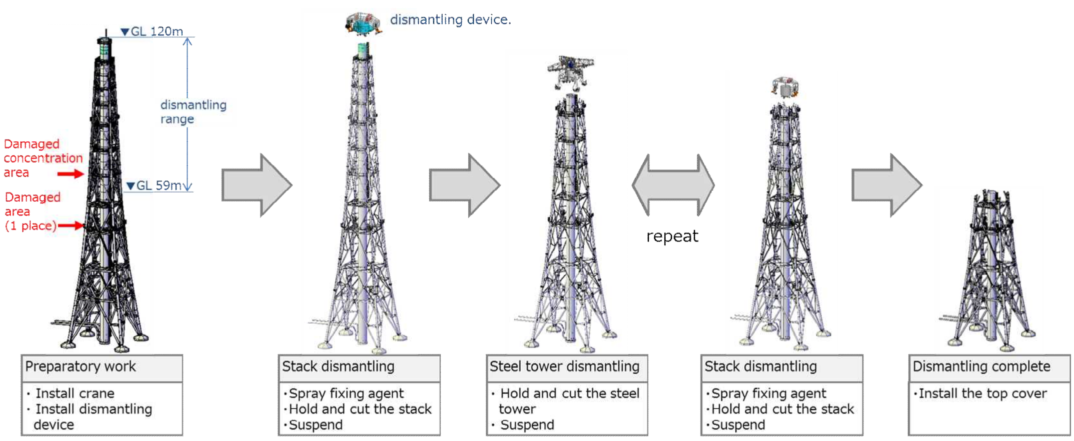


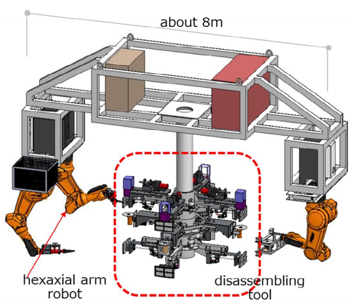
Figure 1. Outline of dismantling step

3. Remote dismantling device

Remote dismantling device is held and fixed with two kinds of holding devices after inserting the disassembling tool into the cylinder stack. The cylinder stack is cut with a chip saw from the inside of the cylinder.

Accessories (such as ladders) outside the cylinder stack that interfere at the time of cutting the cylinder stack are removed with a hexaxial arm robot.

In order to prevent the radioactive dust to scatter when cutting the cylinder stack, fixing agent sprayed inside the stack with another device.



Fugure 2 dismantling device image

E05 Improvement in work environment at Fukushima Daiichi NPS
Yuta Suzuki, Toshiharu Matsuzawa, Shiro Takahira, Koji Hayashi
Tokyo Electric Power Company Holdings, Inc.

Abstract

For workers at Fukushima Daiichi NPS (1F), we have been engaging in the improvement in work environment, light-duty protective clothing, and the reduction of radiation exposure.

1. Introduction

Because the radiation environment at Fukushima Daiichi NPS was extremely changed by its nuclear accident, it was necessary to improve the working environment in order to steadily advance the long-term decommissioning work. Here we would like to introduce how we work on such improvements.

2. Improvement of work environment (dose reduction)

In order to reduce the exposure of workers, we have decontaminated by asphalt pavement, stripping surface soil, shielding, etc. By the end of FY 2015, we could achieve the targeted dose rate of 5 μSv/h* by such decontamination. (Fig. 1).

* The targeted dose rate is set to average of 5 μSv/h at chest position. The Evaluation of dose rates at ground surface level was done by collimator in case the place was affected by direct rays from the plants.

3. Light-duty Protective clothing and finer zoning

In order to keep the decontaminated area at 1F site clean and to reduce the protective gear burden on workers (ex. changed from coveralls to normal work clothes), the controlled area must be divided into some zones according to contamination levels. And it must also carefully controlled to prevent from spreading the contamination from highly contaminated areas, such as inside the Unit 1~4 buildings and the tank dismantling area.

Currently, we can work with normal working clothes and disposable dust mask at about 96% of 1F (Fig. 2).

4. Conclusion

While grasping the radiation environment after the accident, we have taken appropriate measures for radiation protection, and have been continuously improving radiation environments and optimizing radiation management. In order to steadily proceed the decommissioning, we will keep on making our best efforts to improve work environment, light-duty protective clothing and reduce radiation exposure.

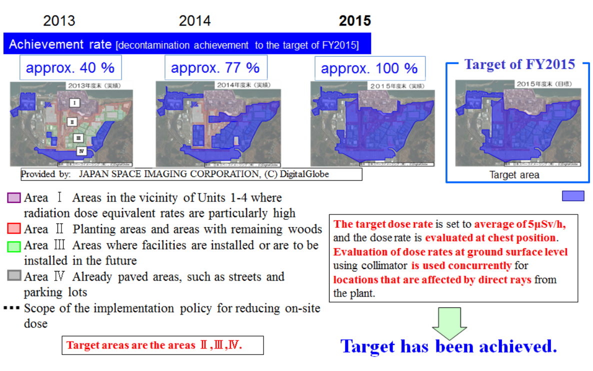


Figure 1 History of dose reduction

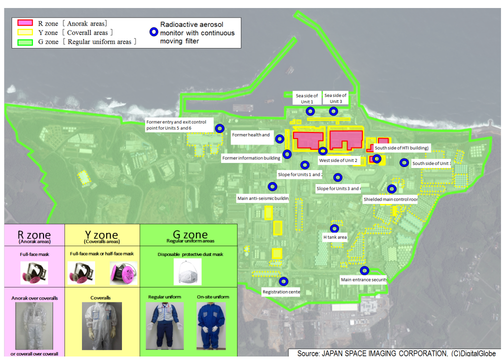


Figure 2 Map after finer zoning

H01

Present Status on R&D of Hydrogen Concentration Reduction Technology in Waste Storage Containers with PAR

Kazuyuki Takase[1], Hisayuki Suematsu[1], Hitoshi Sugiyama[2], Hirohisa Tanaka[3], Masashi Taniguchi[4], Isamu Kudo[5], Ryutaro Hino[6], Atsushi Terada[6], Daijyu Muramatsua[6] and Yasuo Nishihata[6]
[1]Nagaoka Univ. of Technology, [2]Utsunomiya Univ., [3]Kwansei Gakuin Univ.,
[4]DAIHATSU Co. Ltd., [5]ADVAN ENG. Co., Ltd., [6]Japan Atomic Energy Agency

Abstract

In the decommissioning of Fukushima Daiichi nuclear power plant (1F), hydrogen which is the flammable gas is generated in a long-term waste storage containers by decomposition of water due to radiation. Then, R&D on reduction of hydrogen concentration using a storage container with catalysts has been begun.

1. Introduction

In the decommissioning of 1F, the long-term management of fuel debris is necessary. Hydrogen is gradually accumulated in the storage container with time. To reduce the hydrogen concentration, R&D on the long-term radioactive waste storage container with a passive autocatalytic recombiner (PAR) has been begun. The objective is to confirm experimentally and analytically effectiveness of the storage container with PAR.

2. Present Status of R&D

The R&D is performed by six organizations since 2017 and Nagaoka University of Technology is a main organization. Research topics are shown in Fig. 1. Until now, two kinds of prototyped catalysts were developed: the monolith type catalyst made of ceramics; and, the geopolymer catalyst based on geopolymer, alumina and precious metals. The catalytic reaction experiment was conducted for each catalyst, and it was confirmed that it has a good hydrogen recombination reaction.

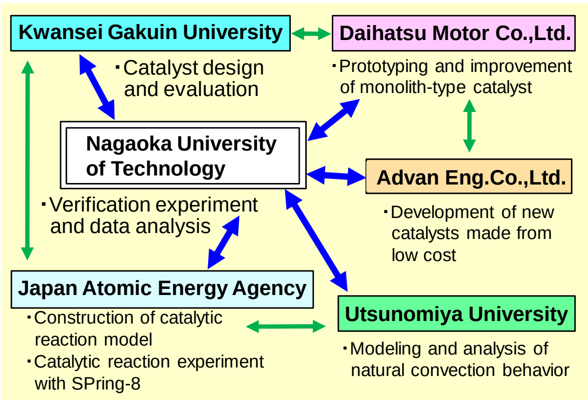


Fig. 1 Research topics of each organization

Figure 2 shows a preliminary result on hydrogen reduction in a small-scale vessel with PAR. This vessel simply simulates a storage container. The number of particulate geopolymer is a parameter. The reduction of hydrogen concentration depended on the contact area of geopolymer catalysts with which hydrogen contacts. Moreover, a simulation method and numerical models are also being developed.

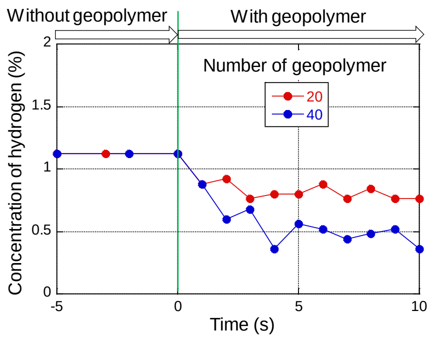


Fig. 2 Change of hydrogen concentration in a small-scale vessel with time

3. Summary

It was found that storage containers with PAR are effective for reducing hydrogen concentration. In the future the present result will be confirmed under a wide range of experimental and analytical conditions.

H02 Evaluation of Performance on Hydrogen Concentration Reduction in Storage Containers with PAR Wetted by Water

Kazuyuki Takase[1], Matsmoto Yoshinobu[1], Isamu Kudo[2], Shogo Taira[2],
Keisuke Tanizawa[2], Gaku Takase[1] and Yoshihisa Hiraki[1]
[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 decomposition of water due to radiation. Then, in order to reduce the hydrogen concentration in a long-term waste storage container, use of passive autocatalytic recombiners (PAR) [1] was considered.

1. Introduction

The objective of this study is to confirm experimentally that PAR functions effectively, even if a part of the PAR is the wet condition.

2. Evaluation of PAR on Hydrogen Concentration Reduction

In this experiment, the condition that much water is mixed with radioactive waste into a container was assumed. Figure 1 shows an experimental apparatus which is a cylindrical container with a diameter of 160 mm and length of 500 mm. Particulate formed geopolymer catalysts are installed in the cylindrical container. The geopolymer catalyst made by ADVAN ENG Co. Ltd. consists of a geopolymer as the base material, alumina as the support material and platinum as the catalyst metal. Hydrogen touches platinum and combines with oxygen, and then water is generated.

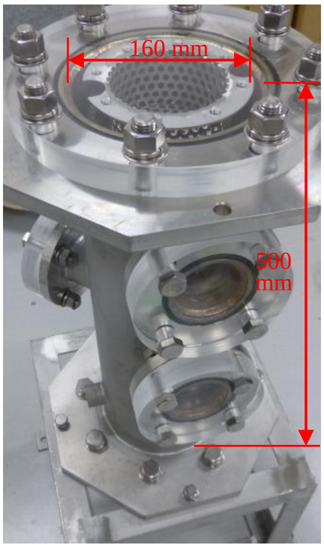


Fig. 1 A cylindrical container

As one of preliminary experimental results, time change of hydrogen concentration in the cylindrical container is shown in Fig. 3. Initially, the inside of the cylindrical container is filled with air with the hydrogen concentration of 1%. Geopolymer catalysts were set in the container on the condition which got wet in water. The hydrogen concentration decreases with time, and it reaches around 0.75 at 200 minutes.

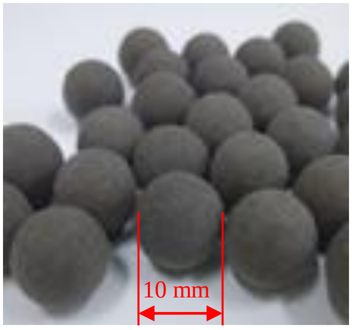


Fig. 2 Particulate formed geopolymer catalysts

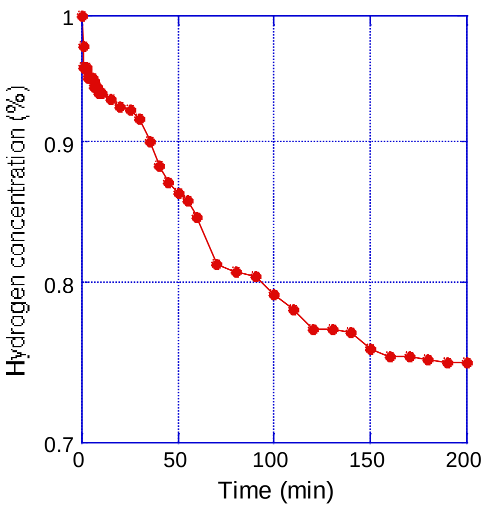


Fig. 3 Time change of hydrogen concentration in the cylindrical container with the geopolymer catalysts which were set under wet condition initially

3. Conclusion

The performance of the geopolymer catalyst under wet conditions was confirmed.

References

[1] J. Henrie, et al., GEND 051, 1985.

H03

Hydrogen safety technology utilizing the automotive catalyst

T. Kita[1], H. Ono[1], K. Takenaka[1], S. Kishimoto[1], M. Taniguchi[2], D. Matsumura[3], Y. Nishihata[3], R. Hino[3], E-A. Reinecke[4], K. Takase[5], H. Tanaka[1]

1. Kwansei Gakuin University, 2. Daihatsu Motor Co., Ltd. 3. Japan Atomic Energy Agency, 4. Forschungszentrum Jülich GmbH, 5. Nagaoka University of Technology

Abstract

We are aiming to establish hydrogen safety technology. It is a serious problem that water is decomposed by strong radiation to generate hydrogen in a container for long-term storage of radioactive waste collected by decontamination, as well as decommissioning nuclear reactors. An intelligent automotive catalyst showed excellent activity to recombine hydrogen and oxygen without external power supply. Moreover, it is found that the monolithic catalyst oxidized a hydrogen from -25 °C.

1. Introduction

It is necessary to store radioactive waste related to decontamination and decommissioning reactor safely and long term. In the container of nuclear radioactive waste isolated, water is decomposed by radiation to produce hydrogen. The passive autocatalytic recombiner (PAR) which needs no electric power supply and heating sources has been developed. We are aiming to apply the intelligent automotive catalyst to PAR.

2. Research and Development

2.1. Catalyst preparation

Three types of monolith catalysts (900, 100 and 30 cpsi) were prepared using a mass-produced automotive "Intelligent Catalyst". The catalysts consisted of Pt, Pd and Rh-perovskites.

2.2. Laboratory scale catalytic reaction (forced flow)

The catalytic activities were observed from -25 °C (Fig. 1). Higher density cell catalyst showed better activity. When the activity of the catalyst arose, the temperature rose, with no electric power supply and heating sources, due to hydrogen oxidation and the hydrogen conversion efficiency exceeded 90% rapidly.

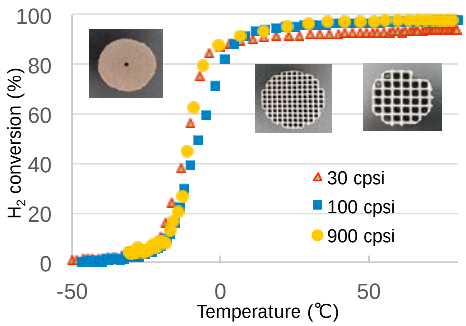


Fig. 1. Cryogenic catalytic activity.

2.3. Large scale catalytic reaction (natural convection)

One piece of the monolithic catalyst (Φ70 x L10 mm = 38.5 mL), shown in Table. 1, was set in REKO-4 reactor (5,330 L) and exposed the mixture gas (6%-H_2, Balance-Air) at a room temperature (Fig. 2). Hydrogen reaction and natural convection velocity were improved 3 times by the use of a chimney. When the cell density decreased, the amount of hydrogen treatment doubled. In the cell structure with small gas flow resistance, natural convection and the hydrogen oxygen recombination reaction accelerated.

Table. 1. Experimental conditions.

Catalyst	Condition	H2 throughput [g/h]
30 cpsi	w/o Chimney	4.16
30 cpsi	with Chimney	11.7
100 cpsi	with Chimney	5.79

Fig. 2. REKO-4 reactor.

3. Conclusion

The intelligent automotive catalyst showed high hydrogen and oxygen recombination activity.

- In the forced flow, the catalytic activity increased with increasing cell density.
- The catalytic activities were observed from -25 °C under the forced flow.
- Quite contrary, in natural convection, the catalytic activity increased as the cell density decreased.
- In a container for storing radioactive waste, low density cell catalyst showed better activity.

Our laboratory continues the study more in order to apply our catalyst into the practical PAR.

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 (MEXT).

K01

Challenge for the Decommissioning of Fukushima based on the Developed Technologies

Susumu Tosha[1], Shohei Sato[1], Masaki Jimba[1] and Toshiki Fukui[1]
[1]IHI Corporation

Abstract

To contribute to the decommissioning of Fukushima Daiichi, IHI develops the technologies of decontamination, remote dismantling and vitirification based on 60 year experience in the nuclear field.

1. Introduction

IHI achieves the experience in the nuclear field over 60 years. IHI has supplied not only the primary equipment such as the reactor pressure vessel and contaminant vessel, but also the construction and maintenance to the nuclear power plant. In the nuclear fuel cycle and radwaste management field, IHI supplied the service of design, fabrication, installation, and trial operation. Based on these experiences, the technologies of decontamination, remote dismantling and vitirification are being developed for the contribution to the decommissioning of Fukushima Daiichi.

2. Experiences in the Nuclear Field

IHI accumulated the experiences of the design, fabrication, installation and maintenance for the primary equipment of the nuclear power plant through the supply of reactor pressure vessel and contaminant vessel to the BWR all over the world including the Fukushima Daiichi unit 1 to 3. IHI is Tier-1 contractor of the highly active liquid waste vitrificaiton facility and vitrified waste storage facilities in Rokkasho Reprocessing Plant. Due to extremely high dose rate of vitrification cell, all works inside the cell, such as operation, maintenance, repair, replacement must be performed by remote handling. Based on these experiences, IHI could contribute to the decommissioning of Fukushima Daiichi.

3. Technical Development

IHI develops the state-of-art technologies based on the experience indicated in the chapter two in this paper. For the decontamination technology, Ultra-High Pressure Liquid Nitrogen technology “NitroJet®” which can eliminates the cross contamination and secondary waste streams are developed. For the remote dismantling system, the laser cutting technology and remote handling simulation technology with 3D-CAD, which were installed in the dismantling facilities for vitrification melter in Rokkasho Reprocessing Facility are developed to apply to the decommissioning of Fukushima Daiichi. The cold crucible induction melter technologies are also developed to stabilize the variety of radioactive waste including organic substances based on the experience of the development of Liquid Fed Ceramic Melter in the Rokkasho Reprocessing Facility.

4. Conclusion

IHI develops the state-of-art technologies of decontamination, remote dismantling and vitirification based on the 60 year experience in the nuclear field to contribute for the earliest possible decommissioning of Fukushima Daiichi.

K02

Basic Research Programs of Vitrification Technology for Waste Volume Reduction (1)Development of Vitrification Technology for Low Level Radioactive Waste(LLW)

Yasutomo Tajiri[1], Oniki Toshiro[1] Toshiaki Kakihara and Toyonobu Nabemoto[1]

[1]IHI Corporation

Abstract

The vitrification tests for the LLW generated from nuclear power plant and reprocessing facility were performed. The simulated wastes of resin, ash and nitrate sodium liquid were vitrified in the alumina crucible, and the homogeneity and durability of vitrified glass was confirmed. Furthermore, the continuous vitrification with the simulated wastes which simulated the physical forms of above wastes was tested in the small furnace, and evaluated the molten condition, environmental applicability of off gas and volume reduction capability.

1. Introduction

The vitrificaion technology is one of the processing methods of radioactive waste for stable disposal. To confirm the applicability of vitrification technology to LLW, the stability of vitrified glass and vitirification operation must be confirmed. In this research, the homogeneity and stability of glass were evaluated by the test in the crucible. Furthermore, the molten glass condition and off gas of the simulated wastes were evaluated by continuous vitrification in the furnace.

2. Evaluation of glasses

Firstly, the glass composition of resin, ash and nitrate sodium liquid were set based on the glass database, the state diagrams of slags and so on, and the additives of each wastes for vitrification were calculated based on the glass composition of each waste. Secondly, the simulated waste and its additives were melted in the crucible at 1100 degrees Celsius(Fig.1). As a result of vitrificaion, the homogeneity was confirmed for each glass according to the SEM. The stable results were obtained according to the PCT-leaching test. The glass compositions that didn’t satisfy the evaluations were regulated again. The PCT-leaching rate of the sodium silicate glass of nitrate sodium liquid was improved by adding Al or Ti to glass. This mechanism has been investigated with some universities.

Fig.1. produced glass

3. Evaluation of operations

To evaluate the stable operation, the simulated wastes which simulated the physical forms were vitrified continuously in the furnace. As a result, the stable operations were confirmed by observing molten conditions that the simulated wastes were mixed in glasses continuously(Fig.2). Also, the high volume reduction was observed for resin glass compared to cement solidification. This is because the resin was decomposed in the process of vitrification. On the other hand, the 96% of sulfur components of resin was volatilized. It is assumed that the treatment system is necessary to remove gas of sulfur in the vitrification process of resin.

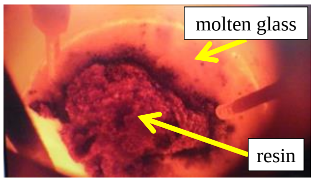


Fig.2. Molten condition

4. Conclusion

The applicability of vitrification for resin, ash and nitrate sodium liquid was confirmed by the vitrification tests. The electric conductivity and viscosity of molten glass of LLW must be obtained to evaluate the applicability to various type of glass melting furnace after this.

5. Acknowledgements

This study was carried out as a part of the basic research programs of vitrification technology for waste volume reduction supported by the Ministry of Economy, Trade and Industry, Japan.

K03

Study on Vitrification Technology for Radioactive Waste at Fukushima Daiich NPS

Toshiro Oniki[1],Yasutomo Tajiri[1], Toyonobu Nabemoto[1], Toshiki Fukui[1],
Seok-Ju Hwang[2], Cheon-Woo Kim[2]
[1] IHI Corporation, [2] Central Research Institute, Korea Hydro & Nuclear Power Co.,Ltd

Abstract

The process of various types of radioactive wastes generated in Fukushima Daiichi Nuclear Power Station (NPS) site is one of the issues to be solved. The purpose of this study is to stabilize these wastes by vitrification technology using CCIM, especially wastes with high level of radioactivity such as secondary wastes generated from contaminated water treatment systems.

1. Introduction

The process of various types of radioactive wastes generated in Fukushima Daiichi Nuclear Power Station (NPS) site is one of the issues to be solved. IHI started research on the treatment of waste from the Fukushima Daiichi NPS, and a study of the application of vitrification technology to such waste after the Great East Japan Earthquake of 2011. The purpose of this study is to stabilize these wastes by vitrification technology, especially wastes with high level of radioactivity such as secondary wastes generated from contaminated water treatment systems. Cold Crucible Induction Melter(CCIM) is selected for vitrification process of these wastes by IHI. CCIM is developed in collaboration with KHNP in this study, zeolite, slurry and sludge are selected as target wastes.

2. Experiment and Preliminary Conceptual Design

Firstly, laboratory scale tests were performed to consider appropriate glass compositions for each waste in order to evaluate the waste loading and durability. Physical properties, such as PCT leaching rate, viscosity and electrical conductivity of candidate glass compositions were evaluated in order to optimize the glass compositions. Secondly, demonstration tests using CCIM mock-up facility were performed to evaluate the applicability for each waste. The main purpose of demonstration test is to confirm the throughput and operation condition during feeding of simulant and glass frit. The behavior of cesium into the CCIM system was evaluated during feeding period. In addition, homogeneity of product glass by SEM/EDS analysis and PCT leaching rate were confirmed.

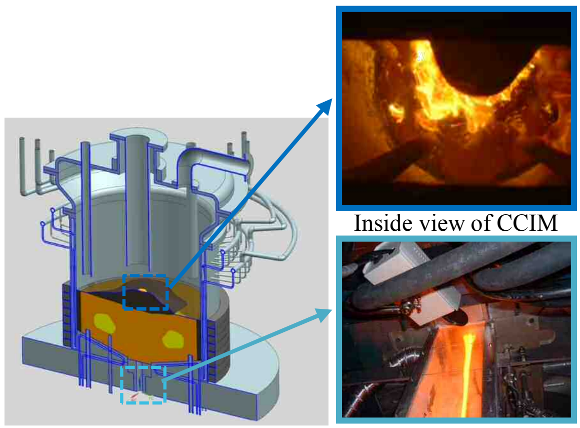


Figure 1. Schematic diagram of CCIM

Furthermore, preliminary concept of process flow of CCIM system and arrangement of equipment were studied for processing the slurry waste.

3. Conclusion

IHI has assessed the applicability of vitrification technology to waste from the Fukushima Daiichi NPS, and it has selected a CCIM as the glass melter and confirmed that the melter is capable of processing those wastes such as zeolite, slurry and sludge on mock-up test.

K04 Basic research programs of vitrification technology for waste volume reduction (2) Development of high waste loading glasses

Yoshiyuki Miura[1], Tetsuji Yano[2], Jun Matsuoka[3], Toru Sugawara[4] and Atsunobu Masuno[5]
[1]Japan Nuclear Fuel Limited, [2]Tokyo Institute of Technology, [3]The University of Shiga Prefecture, [4]Akita University, [5]Hirosaki University

Abstract

In japan, the advanced vitrification technologies which could more stably immobilize the more low level wastes and high level liquid wastes (HLLW) for waste volume reduction will be needed in near future. The technologies for HLLW have been developed from 2014. As a result, we could develop the new glass formulations having higher waste loading. This paper describes progress of the development.

1. Introduction

In this research, the developmental works such as the high waste loading glasses, the alternate glasses of current borosilicate glasses including glass-ceramics and the minor actinide adsorbent glasses have been entrusted with some organizations. JNFL is primarily concerned with the high waste loading glass focused on solubility of molybdenum which may cause the formation of undesirable secondary phase (yellow phase) in the case of HLLW vitrification. JNFL have collaborated with various universities to study the above programs. Furthermore, JNFL have uniquely investigated the effects of several constituents to keep a good chemical durability with high waste loading, which are conflicting with each other, of simulated HLLW glasses.

2. Experimental

There are some development approaches for investigation of high waste loading glasses. The one of the approaches is to modify the composition of the borosilicate glasses which currently used at Rokkasho Reprocessing Plant without any further additives. In this approach, the effects of concentration of network formers, intermediates and network modifiers which are fundamental elements of glasses were investigated.

Through this investigation, JNFL had obtained candidate glass composition ranges until FY2016. Currently, JNFL are undergoing a narrowing the candidate glass composition ranges from the point of view of physical properties, chemical durability and melter operation.

And furthermore, the mechanisms of improvement of molybdenum solubility have been investigated from the point of view of not only chemical reaction in the cold cap but also glass network structure.

3. Conclusion

In this research, the candidate glass composition ranges for high waste loading were obtained. In the future, it will be farther optimized by modifying the compositions of formers, intermediates and modifiers from the point of view of chemical durability, MoO_3 solubility and physical properties.

This work was carried out as a part of the basic research programs of vitrification technology for waste volume reduction supported by the Ministry of Economy, Trade and Industry, Japan.

K05

Volume reduction and recycling of contaminated metal by melting

Satoshi Karigome[1], Kentaro Shibata[1],
Yasuhiro Miyagoshi[2] and Norifumi Onogi[2]
[1]The Japan Atomic Power Company (JAPC), [2]JFE Engineering Corporation (JFEE)

Abstract

JAPC has an extensive record of nuclear power plant (NPP) decommissioning and assessments for clearance of materials from NPPs. JFEE contributed to disposal of disaster wastes in Fukushima Pref. following the Great East Japan Earthquake. Utilizing this experience, we will contribute to volume reduction and recycling of radioactively contaminated metals.

1. Background of JAPC

- First decommissioning of a commercial NPP (Tokai NPP) in Japan. (Figure 1)
- Assessment of contaminated wastes from decommissioned NPPs for clearance and recycling. (Figure 2)

2. Background of JFEE

- Construction and operation of temporary incinerators for disaster waste, including radioactive substances, and design of volume reduction facilities for incineration ash from the temporary incinerators in Fukushima Pref. (Figure 3)
- Design and supply of ash melting furnaces for general waste incineration ash.

3. Melting of radioactive contaminated metal

- Volume reduction of scrap metal contaminated with radioactive cesium (Cs) by melting, and separation of radioactive Cs from the metal.
- Assessment of decontaminated metal for clearance and recycling.

Figure 1. Dismantling of Fuel handling Machine

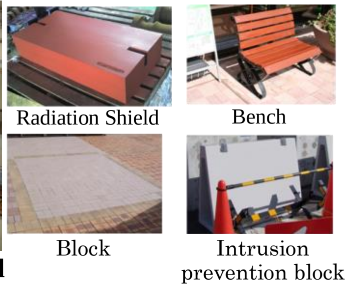


Figure 2. Recycled products

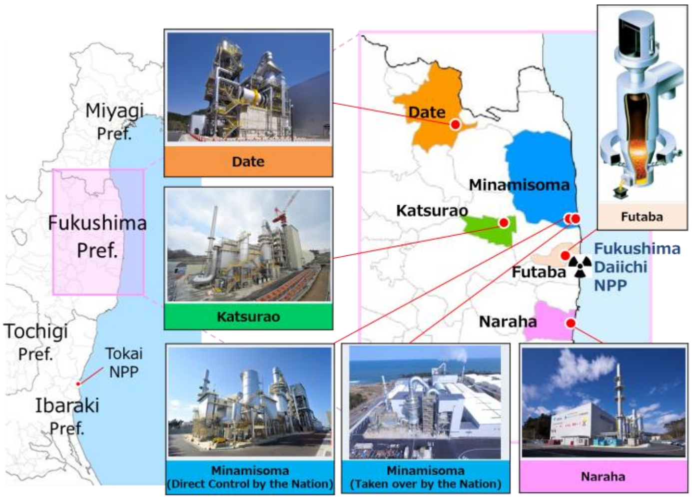


Figure 3. Incineration plants for radioactive disaster wastes in Fukushima Pref.

- Radioactive Cs in metals is distributed to slag and dust
- Slag separation using difference in specific gravity
- Contaminated scrap metal with radiation level of less than approx. 1 mSv/h can be used as clearance metal

- Enclosed structure to prevent dispersal of radioactive Cs
- Production of ingots for recycling use

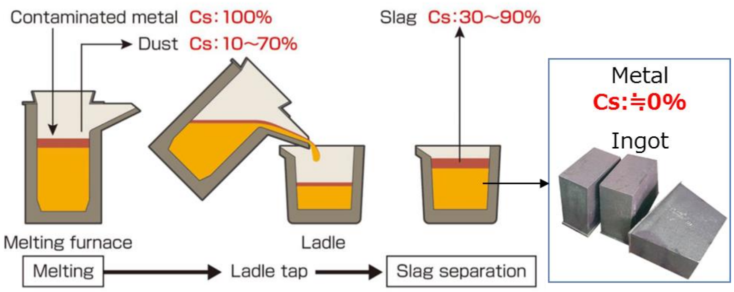


Figure 4. Process flow of radioactive Cs separation

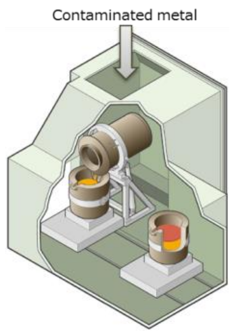


Figure 5. Conceptual design of the equipment

K06

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 ~~four~~ 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).

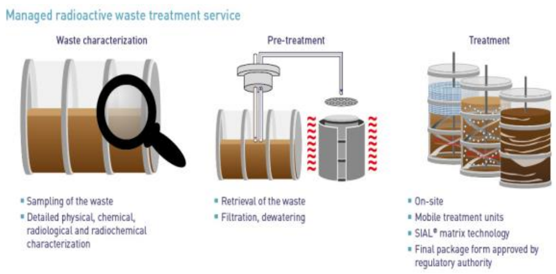


Figure 1 SIAL® solidification treatment process

Figure 2 SIAL® solidified samples

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 were treated and large capacity tanks were emptied and cleaned. The purpose of the project was to retrieve the waste streams from operational tanks and its solidification in 200 liter drums.

K07

Contribution to Environmental Remediation in Fukushima (2)

Takao Ikeda, Yasuhiro Suzuki, Takahiro Kikuchi
JGC Corporation

Abstract

To reduce the final disposal volume of removed soil and incineration ash from decontamination work, we have demonstrated the dry pyro-treatment technology for removal of persistently adhering radioactive cesium from the object materials. Subsequently, we have been developing, with Kyoto University and Taiheiyo Cement, a geopolymer treatment method to safely dispose of the secondary waste generated from pyro-treatment.

1. Dry Pyro-Treatment Technology

The key technology of dry pyro-treatment is to vaporize and remove radioactive cesium in the object materials using some reaction accelerant in a rotary furnace at high temperatures of around 1,350 °C. Through this treatment, the cesium content of the object materials is reduced to levels that allow unconditional recycling. In contrast, fly ash with concentrated cesium is generated as a by-product. The process flow diagram is shown in Figure 1.

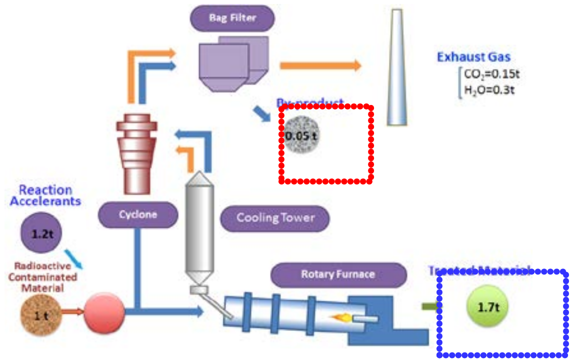


Figure 1 Process flow of dry pyro-treatment process

2. Geopolymer Solidification of Fly Ash

Since fly ash is in the form of chloride, cesium is easily soluble and cannot be expected to remain confined even if it is cement solidified. We have been developing a geopolymer solidification process as an optional technique to solidify fly ash.

Figure 2 Experimental equipment and solidified geopolymer

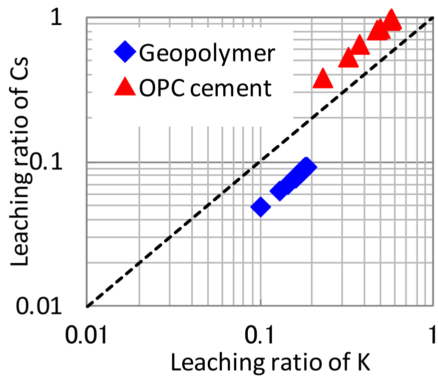


Figure 3 Comparison of leaching ratio between K and Cs

3. Conclusions

The solidified geopolymer of simulated incineration ash was produced, and a compression strength of 20 N/mm^2 or more was obtained. It was confirmed that the G value is lower than that of the cement solidified body because the moisture content is reduced by vacuum heating treatment.

A leaching test in accordance with the IAEA standard leaching test method shows that the leaching ratio of Cs was selectively low as compared with Na and K, and it was observed by XRD and XAFS analysis that cesium was immobilized in the pollucite structure ($CsAlSi_2O_6$).

K08

Radiological separation of contaminated materials -experiences with a high performance mobile belt conveyor system-

F. Langer[1], Dr. M. Sokcic-Kostic[1]

[1]NUKEM Technologies Engineering Services GmbH

Abstract

The existing, containerized, mobile belt conveyor system offers radiological characterization of soil and possible other materials of approximately similar size with potential contamination. Based on radiological characterization, the material is segregated and directed to respective containers, thus reducing the volume for subsequent handling. The system also obtains information and generates documentation to accompany the characterized material in accordance with regulatory requirements.

Positive experiences with a mobile belt conveyor scanning system (FREMES)

Photos: courtesy of FBFC International

1. Purpose of the belt conveyor scanning system

- Radiological characterization of bulk material
- Classification of contaminated bulk material
- Segregation and filling of the material
- Volume reduction for subsequent handling
- Generation of sophisticated documentation per package
- Comparison with regulatory limits

Figure 1. On-site installation of FREMES

2. Achieved benefits

- Volume reduction regarding expansive disposal routes
- No secondary waste generated
- No need for process media
- Detection of Alpha-, Beta- and Gamma- emitters
- Evaluation of spatial activity distribution
- Proven toughness for construction site application
- Filling in containers according to customer’s requirements

Figure 2. Feeding via trucks

3. Technical Data

Throughput up to	100 tons per hour
Sorting size	150 kg
Detection limit	< 0.005 mSv/h
Installation size	3x 40’ transport containers

Figure 3. Removal of free release material

3. Conclusion

This mobile belt conveyor system can automatically segregate materials into three different material streams (free release, limited release/utilization on site, radioactive waste) and can be optimized to customer’s needs.

References: www.nukemtechnologies.com; jochen.petermann@nukemtechnologies-si.de

K09

Radioactive Waste Management
- treatment, storage and disposal of LILW and HLW -

K. Büttner[1], R. Slametschka[1]

[1]NUKEM Technologies Engineering Services GmbH

Abstract

Appropriate conditioning of all kinds of waste reduces waste volumes thereby saving storage space and helping to limit the costs of interim and long-term storage, making waste less hazardous and ensuring a safe final disposal. NUKEM offers customized solutions for all waste problems which rank amongst the world's leading technologies for the treatment of radioactive waste, e.g.:

- Waste treatment concepts and systems, as well as turnkey waste treatment centers;
- Interim storage and final disposal facilities.

1. Waste treatment technologies and systems

- ➢ Evaporation
- ➢ Concentration
- ➢ Microwave drying
- ➢ Nuclide separation
- ➢ Ultrafiltration / Reverse osmosis
- ➢ Biological water treatment
- ➢ Cementation
- ➢ Vitrification
- ➢ High-force compaction
- ➢ Incineration
- ➢ Pyrolysis / Pyrohydrolysis
- ➢ Sorting / Segregation

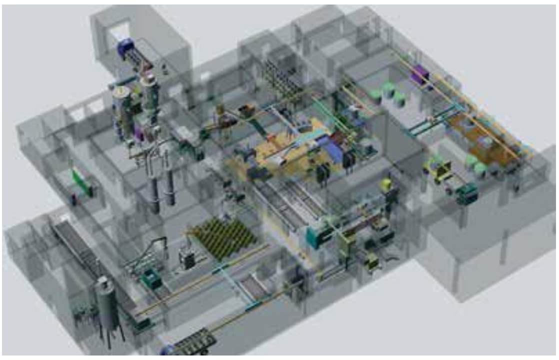


Figure 1: Waste Treatment Center

Reference Project:

Industrial Complex for Radwaste Management at Chernobyl NPP, Ukraine (throughput over 3,500 m^3 waste per year).

2. Waste storage and disposal facilities

- ➢ Temporary storage for Low- and Intermediate Level Waste
- ➢ Near-surface disposal facility

3. Conclusion

Depending on customers requirements and the types and quantities of waste NUKEM Technologies offers already proved waste treatment concepts up to turn-key waste treatment centers and storage and final disposal facilities.

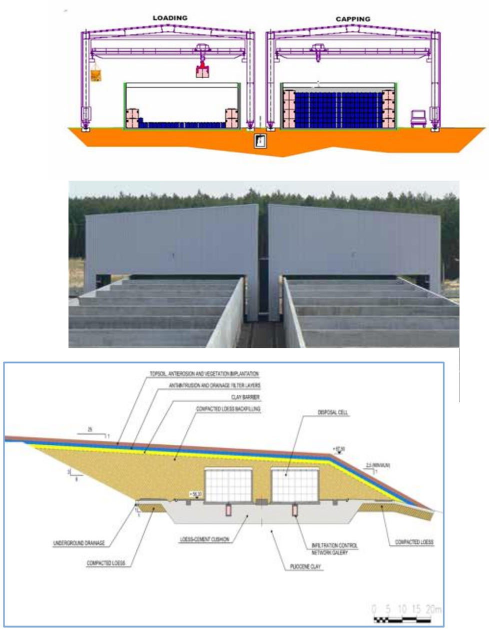


Figure 2: Waste Storage and Disposal Facility (schematic)

References: www.nukemtechnologies.com ; jochen.petermann@nukemtechnologies-si.de

K10 Tailor-made solutions with proven Technology to Waste Management

Motoki Nishibori
KOBELCO STUDSVIK Co., Ltd.

KOBELCO STUDSVIK Co., Ltd. was formed on July, 2016, jointly by Kobe Steel Ltd. and Studsvik AB, focusing on delivering design, engineering to provide innovative radioactive waste management solutions to the Japanese nuclear industry. KOBELCO STUDSVIK will provide technologies and processes to reduce radioactive waste safely and recycle metals generated from the decommissioning effectively for the Fukushima Daiichi Nuclear Power Station.

1. Metal Recycling

In EU, where the effective reuse of resources has socially taken root, metal materials of nuclear power equipment are safely recycled. Studsvik is the organization who developed the metal recycle model in 1980's. The technology can be used for the increasing demand for metal recycling in Japan. In Fig. 1, metal recycle system flow and, in Fig. 2, photos of track record in Berkeley are shown respectively.

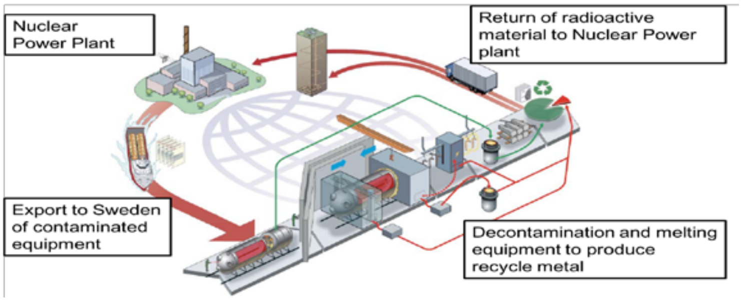


Fig. 1 Metal Recycle system flow

Fig. 2 Photo of 1,550t Boilers in Berkeley (UK)

2. THOR technology

THOR (THermal Organic Reduction) is a proven thermal treatment technology for LLW and ILW, with pyrolysis and steam reforming mechanism.
Fig. 3 shows THOR process flow.

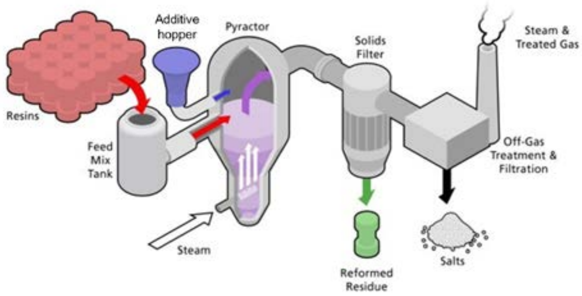


Fig. 3 THOR process

(1) Wastes
Ion exchange resins, rubbers, activated carbon, plastics etc. are treatable wastes.

(2) Reformed Residue
Radioactive wastes are converted into dry and stable solid residue product (Reformed residue).
The reformed residue in which radioactive nuclides are captured is insoluble and leach-resistant.

(3) Volume reduction
When spent ion exchange resin is treated, its volume is reduced up to 1/15, depending on waste type and final waste form.
Fig.4 shows the Sample resins before-and-after treatment.

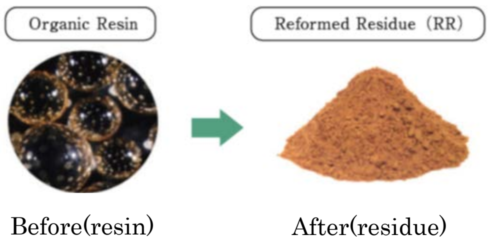


Fig. 4 Reducing up to 1/15

KOBELCO Studsvik

J01 Current Status of the JAEA's Fukushima R&D Bases for the Decommissioning of the Fukushima Daiichi Nuclear Power Station

Nobuhiro Shobu, Makoto Ooka, Toshiro Nakai and Shinichi Nakayama
Japan Atomic Energy Agency

Abstract

The Japan Atomic Energy Agency (JAEA) is operating and constructing R&D bases at the coastal area of the Fukushima Prefecture for the decommissioning of the Fukushima Daiichi Nuclear Power Station (NPS) of the Tokyo Electric Power Company Holdings, Inc. (TEPCO).

1. Introduction

R&D for the environmental restoration of Fukushima and the decommissioning of the Fukushima Daiichi NPS is a prioritized mission of JAEA. JAEA is carrying out the R&D activities at the Collaborative Laboratories for Advanced Decommissioning Science (CLADS) in Tomioka Town and the Naraha Center for Remote Control Technology Development in Naraha Town (Naraha Center), and constructing the Radioactive Material Analysis and Research Facility in Okuma Town (Okuma Analysis and Research Center: Okuma Center).

2. JAEA's R&D bases for the decommissioning of the Fukushima Daiichi NPS

2-1. Collaborative Laboratories for Advanced Decommissioning Science

CLADS is the core of JAEA's R&D for the decommissioning of the Fukushima Daiichi NPS, strongly linked with each infrastructural base of Naraha Center and Okuma Center. CLADS began the operation in April 2017 of the Main Building in Tomioka Town, where the evacuation order was lifted in April 2017. CLADS is expected to act as both an international research hub and a symbol of the shared international interest in decommissioning the Fukushima Daiichi NPS.

2-2. Naraha Center for Remote Control Technology Development

JAEA is working on remote control technology development. The Naraha Center for Remote Control Technology Development, which began operation in April 2016, are available for external users who are interested in R&D on remote-control devices (e.g., robots) necessary for decommissioning work. The external users can conduct demonstration/element-test and operation training here. The center is not only for decommissioning related work, but also for developing robots and their demonstration tests for disaster responses.

2-3. Okuma Analysis and Research Center

The Okuma Analysis and Research Center is supposed to analyze and characterize the radioactive wastes and nuclear fuel debris for development of long-term waste management. The center consists of three buildings: an Administration building, Laboratory-1 and Laboratory-2. Administration building was opened in March 2018. Laboratory-1 is under construction, and Laboratory-2 is at the designing phase.

J02 Operation Training System for Decommissioning Work Using Immersive Virtual Reality Technology

Yoshihiro Tsuchida[1], Takahiro Ohno[1], Nobuhisa Nosaki[1] and Mitsugu Kato
[1]Naraha Center for Remote Control Technology Development, Japan Atomic Energy Agency

Abstract

The Naraha Center for Remote Control Technology Development is a shared facility to conduct development/demonstration tests of remote controlled devices such as robots used for the decommissioning of Fukushima Daiichi Nuclear Power Station. In addition to the test equipment for such devices, the center provides operation training system for decommissioning work using immersive virtual reality technology.

1. Introduction

In order to carry out safely and reliably the decommissioning work, it is important to prepare a work procedure after sufficiently understanding of the site condition in advance and pre-training. However, it is difficult to do that on site due to high radiation dose rate. Therefore, we developed and installed the operation training system using immersive virtual reality technology in the center.

2. Specifications and functions of the system

The system is composed of a computer cluster system to visualize the 3D model data, device for measuring trainee's eyes, projectors, screens and a console for a supervisor. The immersive area configured with CAVE style screens has a height and a width of 3.6 meters and a depth of 2.25 meters, and makes it possible to display structures in a full size. Consequently, trainee can look up at the ceiling direction and obtain a high immersive feeling.

Fig. 1 View of the system

Data on the first and the basement floors of the reactor buildings for unit 1 - 3 have been equipped in the system thus far. The trainee can freely move within the virtual space and train decommissioning work by combining with functions such as indication and calculation of air radiation dose rate, lighting simulation, distance measurement of arbitrary two points, import of 3D CAD data of equipment.

Fig. 2 Training in the system

3. Summary

Utilizing this system enables trainee to experience the realistic work environment repeatedly, consider and verify work plans such as a traffic line of a human, a loading/unloading route and an installation site of equipment, and so on. As a result, we believe that this system contributes toward carrying out safely and reliably the decommissioning work, and improving the decommissioning working efficiency. Our future works are to expand the data based on needs of candidate users, and to develop a mechanism for applying ever-changing on-site situation to the system.

J03 Outline of JAEA Okuma Analysis and Research Center

Akihiro Ishimi, Toshihiko Inoue, Ichiro Kosaka, Shinichi Koyama, Tomozo Koyama
Japan Atomic Energy Agency

Abstract

Japan Atomic Energy Agency (JAEA) is constructing Okuma Analysis and Research Center near Tokyo Electric Power Company (TEPCO), Fukushima Daiichi Nuclear Power Station (1F) site. Outlines of the Okuma Analysis and Research Center are introduced at the poster.

1. Outline of the Okuma Analysis and Research Center

JAEA works on various missions for decommissioning of the 1F, and the research and development facility “Okuma Analysis and Research Center” is designed and has been constructed near 1F. As shown in Figure-1, the Okuma Analysis and Research Center consists of three buildings; an administrative building, Laboratory-1 and Laboratory-2. The Okuma Analysis and Research Center will be used for research and development to ascertain characteristics of radioactive wastes, and technological development to treat and dispose fuel debris.

2. Administrative building

The Administrative building provides office space and meeting rooms for the researchers, and an apparatus mock-up space “Workshop”. The simulated steel hot cell, Glove Box, hoods and some apparatus such as master-slave manipulators were installed at the workshop, and the mock-up test and training of the analysis will be performed with these apparatus. The construction of this building began in October 2016, and an operation started in March of 2018.

3. Laboratory-1

Laboratory-1 will perform radioactive analysis, chemical and mechanical characterization of low and medium level of radioactive wastes such as radioactive rubble and secondary wastes to establish strategy and methodology for treatment and disposal of them. The schematic layout is shown in Figure-2. Laboratory-1 will be equipped with 10 glove boxes, and 56 hoods for low level radiochemical analysis. For the handling of medium level samples, 4 steel cells will be installed. The construction of Laboratory-1 began in April 2017 and plans an operational start in fiscal year 2020.

4. Conclusion

JAEA started an operation of Administrative building, and is moving forward with the construction of Laboratory-1 and the design of Laboratory-2 for the analysis of the fuel debris.

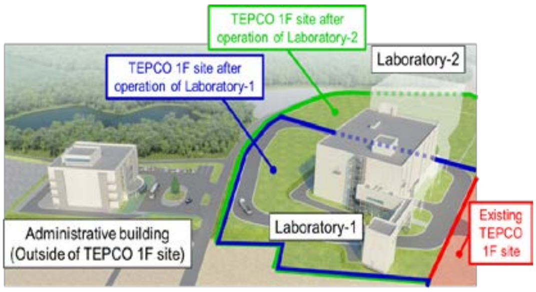


Figure-1 Okuma Analysis and Research Center

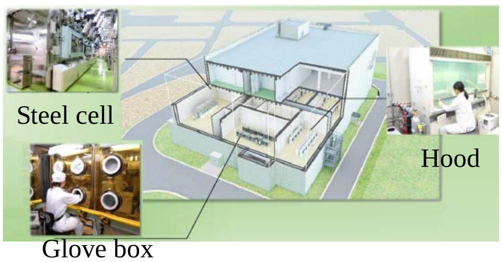


Figure-2 Laboratory-1

J04

Plasma heating test on fuel assembly degradation

Takuya Yamashita[1,2], Ikken Sato[1,2]
[1]JAEA, [2]IRID

Abstract

A large uncertainty is present in understanding of BWR accident progression behavior. A series of tests were conducted to provide experimental data that address uncertainties. As a results, useful information on core state just before slumping was obtained. This information will help us to comprehend core degradation in BWRs like those in the Fukushima Dai-ichi nuclear power plant (1F).

1. Introduction

Core degradation in a BWR like the 1F has not been comprehensively understood. This uncertainty, which was clarified by the MAAP-MELCOR Crosswalk [1], cannot be resolved with existing experimental data and knowledge. In this study, a core-material melting and relocation (CMMR) experiments were conducted to provide information to answer the uncertainties related to BWR-design conditions so that understanding on accident progression behavior in 1F can be improved.

2. CMMR experiment design

The CMMR test equipment mainly consists of the plasma heating system, and the simulated fuel assembly. The plasma heating system used for the tests were a Phoenix Solutions Company commercial-grade non-transferred torch (model PT200). The simulated fuel assembly was composed of simulated a fuel rod bundle within a channel-box of BWR. In this test, ZrO_2 pellets were installed instead of UO_2 pellets.

3. Test conditions

The test conditions were selected with intention to simulate the state of 1F Unit 2 just before slumping. Oxygen concentration kept less than 0.1%. Maximum temperature was close to the melting point ZrO_2 pellets.

3. Conclusion

(a) Macroscopic gas permeability of the core approaching ceramic-fuel melting was confirmed under the condition simulating the 1F Unit2 condition (See Fig.1).

(b) The hot core fuel remained as columns in these tests suggesting possible actual BWR scenario where weight of core upper part would result in collapse of fuel columns basically keeping axial profile of fuel temperature (though shortened) rather than pellet-wise relocation from the hottest part (See Fig.1).

(c) Fuel debris without melting would keep its permeability even after collapse and relocation to lower plenum allowing enhanced cooling capability when heat-transferring fluid and heat sink is maintained.

Fig.1 Post-test appearance of the CMMR-4

Acknowledgements

The content of this material includes output of the METI subsidy for Project of Decommissioning and contaminated Water Management (Upgrading for Identifying Comprehensive Conditions inside the Reactor)

Reference

[1] EPRI,"Modular Accident Analysis Program (MAAP) – MELCOR Crosswalk, Phase 1 Study", Nov.2014

J05

Characterization of the Large-scale MCCI Test Products for Fuel Debris Removal from the Fukushima Daiichi Nuclear Power Plant

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[1]JAEA/IRID, [2]CEA

Abstract

A large-scale test of the molten core-concrete interaction (MCCI) was performed to investigate heterogeneity of characteristics of the MCCI products. The results of the post-test analysis confirmed importance of appropriate selection of defueling tools according to the location and the characteristics of MCCI products.

1. Introduction

In the Unit 1 of the Fukushima Daiichi Nuclear Power Plant (1F), the molten core materials are expected to have flowed into the bottom of the primary containment vessel, causing the molten core-concrete interaction (MCCI) at the pedestal floor. Clarifying the characteristics of the MCCI products is important for the removal of those materials from the damaged floor. Especially, characteristics such as solid phases and mechanical properties could be locally changed due to spatial variation of the material composition and temperature history, making appropriate selection of defueling tools much more complicated. To achieve insights on this "heterogeneity" of the MCCI products, a large-scale MCCI test was performed in collaboration with CEA, France.

2. Method

Conditions for the large-scale test were constructed specifically taking the average composition of core materials and heating conditions for the 1F Unit 1. Approximately 50 kg of the simulant core materials (mixture of UO_2, Zr, ZrO_2, and stainless steel) were melted in the crucible made of the basaltic concrete. Post-test analyses were performed on the solidified samples taken from various representative zones, e.g., bulk region, interface between molten cores and concrete, etc, to characterize the solid phases and micro-hardness.

3. Conclusions

The post-test analysis of the samples revealed stratified structure of the solidified melt which mainly consisted of hard porous oxide (Si-rich), hard dense oxide (Si-rich), and soft metallic block (Fe-rich), as shown in Figure 1. The difference in characteristics will necessitate the appropriate selection of defueling tools according to the location and the characteristics of MCCI products.

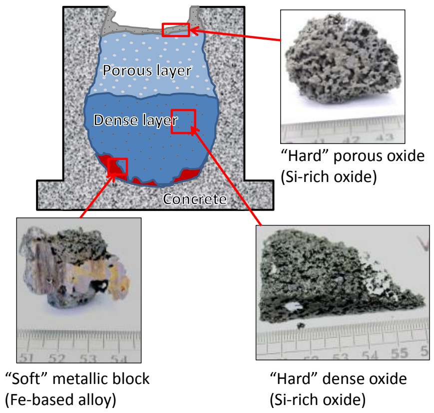

Figure 1. Typical appearances of the solidified melt after the large-scale MCCI test

Acknowledgment

This paper includes the results obtained under the research program entrusted to the International Research Institute for Nuclear Decommissioning (IRID) by the Agency for Natural Resources and Energy, Ministry of Economy, Trade and Industry (METI) of Japan.

J06

Advanced study on remote and in-situ elemental analysis for molten fuel debris in damaged core by innovative optical spectroscopy

Ikuo Wakaida[1], Hironori Ohba[1,2], Yuji Ikeda[3], Tetsuo Sakka[4], Takunori Taira[5]

[1]Japan Atomic Energy Agency/CLADS, [2]QST/QuBS, [3]Imagineeing Inc., [4]Kyoto Univ., [5]NINS/IMS

Abstract

Radiation resistant optical fiber based laser induced breakdown spectroscopy (Fiber LIBS) is developed for in-core and in-situ elemental analysis of debris in damaged core, and its activity is examined under high radiation field of about 10kGy/h and under water. For higher sensitivity, long pulse laser, microwave assisted LIBS are performed, and additionally, Microchip laser for longer light delivery and flexible use is introduced in high radiation condition.

1. Introduction

For the decommissioning of “Fukushima Daiich Nuclear Power Station(F1NPS)” which contained damaged or melt downed core, 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 these requirements, 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 technical development based on LIBS technology.

2. Current activities of LIBS

2-1. Conventional fiber LIBS

As for R&D of onsite, in-situ monitoring under the severe environments, we have been developing the Optical Fiber LIBS Probe using with radiation resistant optical fiber for laser and plasma emission light delivery. The concept of Fiber LIBS and its application for in-situ monitoring are shown in Fig.1. The transmission of the optical fiber in near infrared range will be confirmed to have no damage by radiation rate of 10kGy/h and total dose of more than 2MGy by the ^{60}Co irradiation test. Characteristics of plasma generation, emission under strong radiation have been checked and no influence in specific emission line under strong radiation will be confirmed.

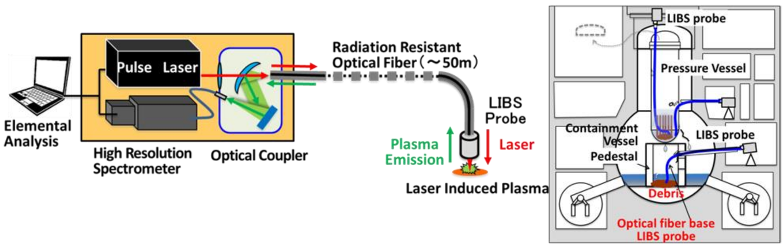

Fig.1 Schematics of Fiber LIBS system and its application for F1NPS

2-1. Microchip laser LIBS

As for the requirement of longer application in remote analysis, we are now developing “Microchip laser LIBS system” in which the laser source will be installed at the top of the LIBS Probe Head as shown in Fig.2. Unlike the Fiber LIBS system in which breakdown laser delivered though optical fiber, laser power and focusing characteristics will be improved, and it may have better performance such as longer focusing length and possibility of laser processing to the surface of debris. A good performance in laser emission under strong radiation of 10kGy/h will be just now demonstrated.

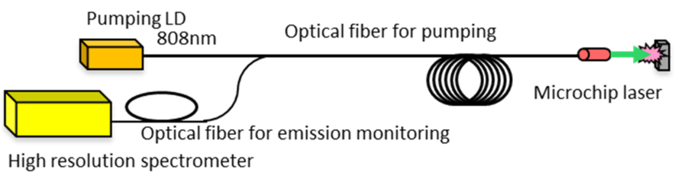

Fig.2 Basic concept of Microchip Laser Fiber LIBS system

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).

J07

Application of ICP-MS to analysis of samples from 1F site at Radioactive Material Analysis and Research Facility

Van-Khoai Do[1,2], Takuma Horita[1,2], Katsue Miura[1,2], Maho Iwasaki[1,2]
[1] Japan Atomic Energy Agency (JAEA)
[2] International Research Institute of Nuclear Decommissioning (IRID)*

Radioactive Material Analysis and Research Facility has been recently established to support the analysis of samples including fuel debris and radioactive wastes from the decommissioning of Fukushima Daiichi Nuclear Power Station (1F). Conventional radiometry, which is laborious and time consuming, ICP-MS is considered as a potential alternative for long-lived nuclides in fuel debris because of its superior sensitivity. Moreover, the use of ICP-MS is aimed to increase the analysis capacity, which make possible to repeat measurements in order to obtain precise results and to reduce exposed dose of analysts because ICP-MS is much more rapid than radiometric methods. However, isobaric interference from the same mass nuclides is likely to limit the application of ICP-MS in our field. Employing latest advanced technologies, a triple quadrupole ICP-MS, Agilent 8900 (ICP-QQQ-MS), enables to suppress the interference. In this presentation, we are reporting some demonstrative separations of difficult-to-measure nuclides using the ICP-QQQ-MS system.

Figure 1 illustrates the schematic structure of the ICP-QQQ-MS.The ICP-QQQ-MS is equipped with two quadrupole mass filters (Q1 and Q2) that are tandemly arranged, and an octopole collision/reaction system (ORS) is set between the filters [1]. Mechanism of interference suppression is described as follows: samples are first ionized in the Ar plasma. Then, the resulting ions are led to the first quadrupole filter (Q1), which allows to pass through only ions with m/z of interest including the same mass interferents. Then these selected ions are led to react with reaction gases to form different complex ions with different mass-shifts. The second quadrupole (Q2) functions as the second mass filter that allows the ions of interest to be detected by an electron multiplier detector (EM detector).

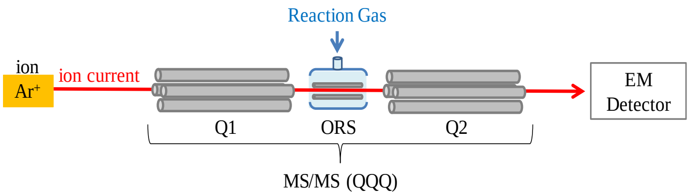

Fig. 1: Illustration of ICP-QQQ-MS

For demonstration of selective measurement, interfering ion pairs were selected aiming at future applications, including (Ni-59, Co-59), (Ni-63, Cu-63), (Se-79, Br-79), (Sr-90, Zr-90, Y-90), (Zr-93, Nb-93, Mo-93), (Pd-107, Ag-107), (I-129, Xe-129), (Cs-135, Ba-135), and (Sm-151, Eu-151). Presently, non-radioactive solution standards are using to evaluate the separation. Four gases including He, H_2, NH_3, and O_2 were tested as collision and reaction gases. Table 1 summarizes the selected reaction gases that are possible to separate nuclides of interest. LOD (part per trillion-ppt or ng/L) is limit of detection of the most abundant natural isotopes that are used as tracers. The results indicated that it is possible to eliminate the interference by using a suitable reaction gas. ICP-QQQ-MS promises an alternative method for rapid analysis of fuel debris as well as radioactive samples from 1F.

Nuclide	Interfering nuclide	Reaction gas	LOD (ppt)
Ni-63	Cu-63	NH_3	13.7
Se-79	Br-79	O_2	21.0
Sr-90	Zr-90	O_2	0.21
	Y-90		
Zr-93	Nb-93	NH_3	1.1
	Mo-93		
Pd-107	Ag-107	NH_3	
I-129	Xe-129	O_2	6.5
Sm-151	Eu-151	NH_3	0.24

Table 1: Result of reaction gas selection

Reference

[1] ICP-QQQ-MS Agilent 8900 brochure, available online at https://www.agilent.com/en/products/icp-ms/icp-ms-systems/8900-triple-quadrupole-icp-ms

* This presentation includes results obtained under the research program entrusted to IRID by METI.

J08 Estimation of radioactive nuclide distribution at NPS site

Developing model based on analytical data of on- and off-site

Kazuki Iijima[1], Yoshikazu Koma[1], Tadafumi Niizato[1], Terumi Dohi[1], Yoshihito Ohmura[2], Takayuki Sasaki[3] and Kenji Nanba[4]

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Abstract

Distribution of radionuclides around the site was investigated, and the model of radioactive contamination was developed to describe inside and outside of the Fukushima Daiichi Nuclear Power Station (NPS).

1. Introduction

To establish method of radioactive waste management, inventory of radioactivity contained in waste is essential information. Distribution of contamination is also important to estimate inventory of the wastes including those will be generated in the future decommissioning. Environmental contamination around the site is useful to obtain information to make a mathematical model of radioactivity distribution for the on-site. Contamination of topsoil, lichen and tree was investigated, and a model was made by considering the data as well as reported on-site data.

2. Analysis of environmental samples and investigation on contamination model

Environmental samples were collected with considering on the traces of radioactive plumes and analyzed. With utilizing the data a contamination model was proposed.

2-1. Topsoil

According to concentration depth profile, Cs was retained at the surface, while Sr was slowly transported, suggesting correlation with distribution coefficient. Model constituted from three processes well reproduced the profiles.

2-2. Lichen

Lichens can accumulate radionuclides for a long period. Captured small particles on the thalli were found to be different in chemical composition. The ratio of $^{134}Cs/^{137}Cs$ in lichens and their particles were also found to be different. These findings suggest multiple sources from reactors and variable states in transport.

2-3. Tree

For modeling, it was assumed that cedar retains initial contamination on its bark, while Cs was also uniformly distributed inside due to nutrient translocation. Any α emitters were not detected.

2-4. Modeling on-site contamination

By expressing a plume as Gaussian dispersion distribution and nuclide concentrations as relative ratio of transport, on-site distribution of radioactive was estimated as shown in Figure.

3. Conclusion

The derived radioactivity distribution will help analytical investigation including planning. And knowledge of nuclide composition of plumes and each reactor contamination will bring better estimation of contamination inside the buildings.

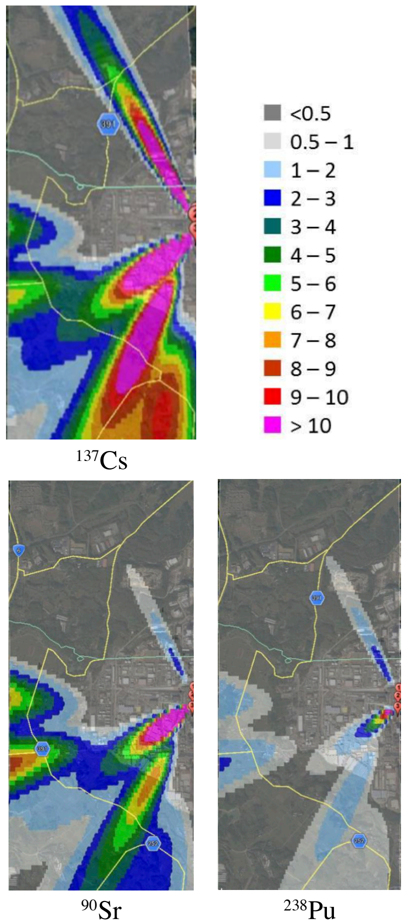

Figure. Estimated radionuclide on-site distribution. Unit in MBq/m^2.

Note: This work is the result from the work entrusted to the parties of the authors from MEXT.

J09

Development of Corrosion Database under Radiation Environment

Tomonori Sato[1], Kuniki Hata[1], Yoshiyuki Kaji[1], Hiroyuki Inoue[2], Kousei 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.

2. Research plan and results

2-1. Mechanism of corrosion in irradiated condition

To build a database for estimating the degradation of corrosion in irradiated condition, corrosion and radiolysis data will be accumulated as shown in Fig. 1. Concerning development of data set and database for radiolysis, arrangement/estimation of radiolysis existed data and preliminary radiolysis analysis were carried out. Corrosion test environment in gamma-ray irradiation was constructed in Takasaki Advanced Radiation Research Institute of National Institute for Quantum and Radiological Science and Technology (QST). In this facility, gamma-ray irradiation can be carried out in the range from ten to thousands of dose rate. Regarding mechanism of corrosion in irradiated condition, preliminary corrosion tests for influence of gamma irradiation on both de-passivation of carbon steel and corrosion in wet/dry environment have been started.

2-2. Evaluation of corrosion risk in each equipment

Corrosion risk for individual equipment during decommissioning process is analyzed by using investigation sheet. Concerning estimation method for potential corrosion risk, basic concept was constructed with overlooking overall decommissioning process of 1F by existed knowledge and literatures.

This project is conducted as a part of “Analysis of Corrosion Mechanism in Specific Environment” supported by the Ministry of Education, Culture, Sports, Science and Technology (MEXT) of Japan.

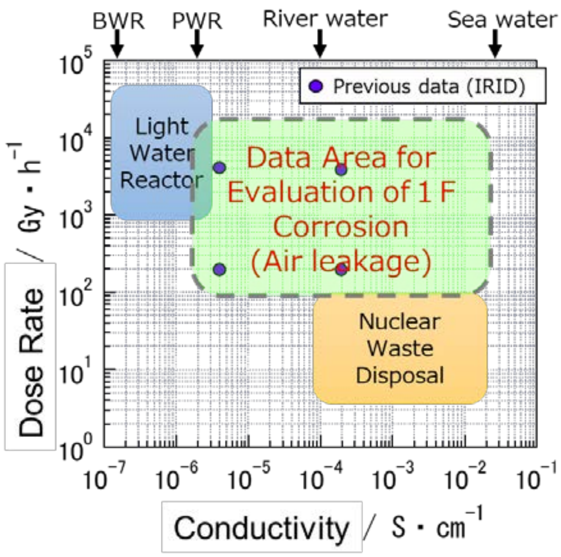

Fig. 1 Data area for evaluation of 1F corrosion

J10

A Prediction of the Dose Rate Distribution in the Primary Containment Vessel of the Fukushima Daiichi Nuclear Power Station

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

Abstract

We have predicted the dose rate distribution inside PCV of Unit 1 and Unit 2 in 2021 by combination of Monte Carlo calculation with local dose rates measurements obtained so far by PCV internal investigations.

1. Method

The dose rate distribution in PCV $D(\vec{r})$ is expressed as follows:

$$D(\vec{r}) = \sum_i S_i d_i(\vec{r})$$

where, *i* indicates the radiation source such as Cs contamination, fuel debris and activation materials, which are estimated based on the accident progression analysis, burnup calculation, and activation calculation, respectively, $d_i(\vec{r})$ is a Green function obtained by the particle transport Monte Carlo calculation code PHITS [1] for the *i*-th unit radiation source in PCV, and S_i is source intensity of the *i*-th radiation source. Although the uncertainty of S_i is large, if local dose rate measurements were obtained by internal investigation, we can modify the source intensities to fit the measured dose rate.

2. Predicted result

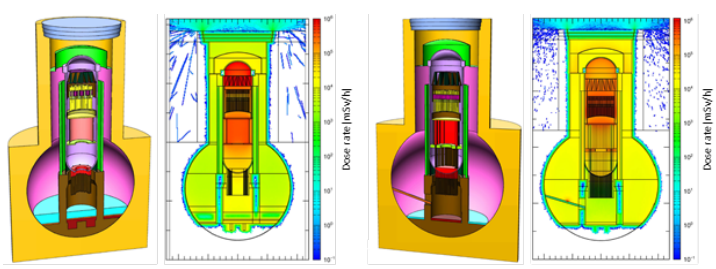
The above method was applied to the Unit 1 and Unit 2 of the Fukushima Daiich NPS. Table 1 shows the internal investigations in which dose rates have been measured. The measured dose rates were considered to modify the Cs contamination sources sensitive to measured dose rate. The results of modified dose rate distributions in 2021 are shown in Fig.1.

Table 1 Internal investigations in Unit1 and Unit 2

Unit 1			Unit 2		
Name	Date	Data	Name	Date	Data
1st entry	Oct.2012	8	1st entry	Mar. 2012	8
B1 investigation	Apr. 2015	16	A2 investigation	Jan.-Feb. 2017	3
B2 investigation	Mar. 2017	20	A2' investigation	Jan. 2018	8

3. Conclusion

We have successfully predicted the dose rate distribution consistent with dose rate measurements so far. This result is valuable to develop detection technologies of fuel debris in PCV and to optimize fuel debris retrieval method.



(a) Unit 1 **(b) Unit 2**

Figure 1. Model and predicted dose rate distribution in PCV

References

[1] T. Sato, Y. Iwamoto, S. Hashimoto, et al., "Features of Particle and Heavy Ion Transport code System (PHITS) version 3.02, J. Nucl. Sci. Technol. 55, 684-690 (2018).

J11

Study on Wet Waste Storage - Irradiation Behavior of Carbonate Slurry and Wet Zeolite -

Isao Yamagishi[1] and Takafumi Motooka[1]
[1]Japan Atomic Energy Agency

Abstract

Carbonate slurry or wet zeolite as a simulated waste was irradiated by gamma-ray. Radiolytic gas generation resulted in the slurry expansion and supernatant separation from the slurry.. The supernatant was observed during irradiation of the zeolite immersed fully in seawater but not in case of the partial immersion.

1. Introduction

Overflow of supernatant from storage containers of radioactive carbonate slurry generated from ALPS(MRRS^{TM}) was discovered at Fukushima Daiichi Nuclear Power Station in 2015. Hydrogen gas was also detected in the containers. We have conducted irradiation experiments of the simulated slurry by Co-60 gannma-rays to elucidate the mechanism originated from radiolytic gas generation. Irradiation behavior of the zeolite, which is supplied for Cs adsorption vessel of SARRY^{TM} was investigated using its mixture of seawater.

2. Irradiation experiments and results

The simulated carbonate slurry provided by Kurita Water Industries Ltd was poured in a quartz tube (ID 20 mm) to 100 mm height, and irradiated with PMMA dosimeter at a dose rate of 8.5 kGy/h as shown in Figure 1 [1]. Radiolytic gas generation and its bubble formation resulted in the slurry expansion and supernatant separation from the slurry. The increase of dose led to the decrease of slurry expansion due to the partial bubble release and the increase of supernatant volume.

The mixture of the zeolite (UOP IE96) and artificial seawater (ASW) was poured into a quartz tube (ID 40 mm), and the ASW volume was adjusted to two levels; (1) top of zeolite bed (full immersion) and (2) a quarter bed (1/4 immersion). Two tubes were irradiated by Co-60 gamma-ray at 7.2 kGy/h for 27 h as shown in Figure 2 [2]. In full case, the ASW level rose from 402 mm to 438 mm with the supernatant formation. However, the supernatant was not formed in 1/4 case, which simulates a drained Cs adsorption vessel.

3. Conclusion

Radiolytic gas generation is a key issue for the safe storage of highly radioactive and wet wastes. We observed its influence during experiments but need further investigation to elucidate the mechanism.

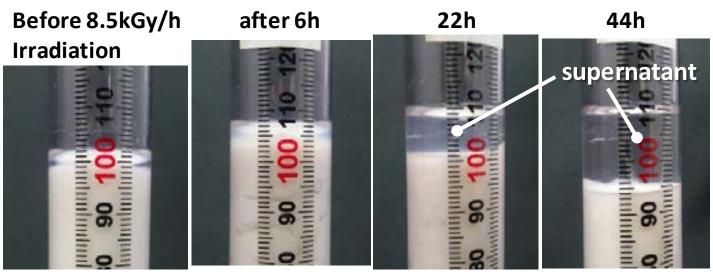


Figure 1. Irradiation of carbonate slurry (SS: 95 g/L)

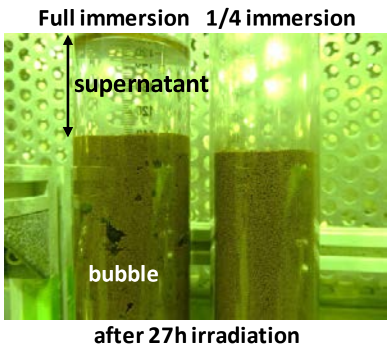


Figure 2. Irradiation of mixtures of zeolite and seawater

References

[1] T.Motooka, R.Nagaishi and I.Yamagishi, Gas retention behavior of carbonate slurry under γ-ray irradiation” , QST Takasaki Annual Report 2015 (2017).

[2] T.Motooka and I,Yamagishi, “Behavior of Radiolytic Gas in Zeolite Bed under Gamma-ray Irradiation”, Proc. of Fall Meetings of Atomic Energy Society of Japan, 2A03 (2017).

J12 Visualization of radioactive substances by integrating radiation measurement, remote technology, and 3-D optical measurement

Yuki Sato[1], Yuta Terasaka[1], Masaaki Kaburagi[1], Yuta Tanifuji[1] and Tatuso Torii[1]
[1]Japan Atomic Energy Agency

Abstract

We visualized a hotspot from the sky over the waste storage space inside the Fukushima Daiichi Nuclear Power Station (FDNPS) using a remote radiation imaging system consisting of a drone equipped with a Compton camera. By superimposing the images of the hotspots on the 3D building model of the measurement area, we succeeded in drawing the map that can easily recognize the hotspots in the actual working environment.

1. Introduction

The measurement of a distribution of radioactive substances in the FDNPS is essential in order to execute the appropriate decommissioning tasks. Last year, we drew a 3D radiation distribution map by integrating the radiation image measured with the Compton camera into the point cloud data of inside the turbine building of unit 3 acquired using a scanning laser range finder [1]; we fabricated a compact Compton camera based on the technology developed by the joint team from Waseda University and Hamamatsu Photonics K.K. [2]. In addition, we are also developing the remote measurement technology of radioactive substances using a drone equipped with the Compton camera. In this work, the measurement result using these devices are introduced.

2. Radiation imaging experiment inside the FDNPS

We visualized a hotspot from the sky over the waste storage space in the FDNPS using a remote radiation imaging system consisting of a drone equipped with a Compton camera [3]. Figure 1 is a 3D model of the waste storage space created using a photogrammetry technic. The hotspots are displayed in red on the 3D building model. The areas where the survey meter showed a higher dose rate than the surroundings roughly agree with the red areas detected by the Compton camera. We succeeded in drawing the map that indicates the positions of the hotspots in the working environment.

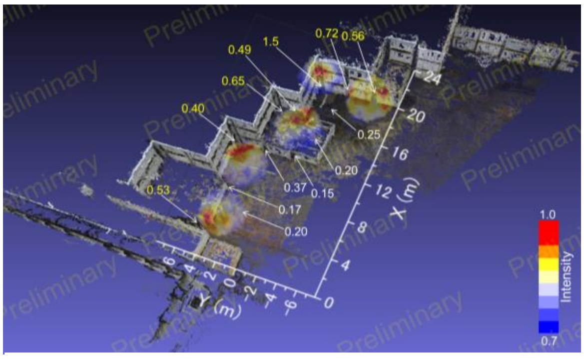


Fig. 1. Radiation images are drawn on the 3D building model of the waste storage space in the FDNPS. The areas shown in red are the hotspots detected by the Compton camera. Values are ground survey results at 1 m height (mSv/h).

3. Conclusion

The images of the hotspots obtained by the Compton camera mounted on the drone are drawn on the 3D building model of the working environment created using the photogrammetry. It is a helpful method for workers to easily recognize the hotspot at the work environment inside the FDNPS.

References

[1] Y. Sato et al., Radiation imaging using a compact Compton camera inside the Fukushima Daiichi Nuclear Power Station building, *J. Nucl. Sci. Technol*., (in press), doi.org/10.1080/00223131.2018.1473171.

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J13

Development of simulation function of remotely operated vehicle for underwater reconnaissance tasks

Kenta Suzuki, Shunji Mukaida and Kuniaki Kawabata
Japan Atomic Energy Agency

Abstract

This paper describes the development of the ROV (Remotely Operated Vehicle) simulation function implemented on our robot simulator system. The function provides virtual thruster device to generate thrusting forces and torques and also, it calculates the effect of fluid dynamics, buoyancy, lifting force and so on. In this paper, we introduce developed function and test result with ROV model utilized for underwater reconnaissance tasks.

1. Introduction

Currently, we develop a robot simulator system for supporting effective decommissioning tasks execution by utilizeing remotely operated robots [1]. We also continue to extend its functions by plug-in software. In some of survey tasks at Fukushima Daiichi Nuclear Power Station (FDNPS), ROVs were deployed and a ROV was utilized for the task to investigate the situation inside PCV. Therefore, we decided to implement a new function to simulate underwater behavior of robot to our system.

2. ROV simulation function implemented on robot simulator system

In order to simulate behaviors of ROV, the calculations for applying the effect fluid dynamics and some visual properties must be done and simulated device generates thrusting forces and torques is also required. Therefore, we developed a plug-in software and virtual device to implement such functions to the robot simulator.

For adjusting the effect of fluid dynamics, the buoyancy, lifting force, fluid resistance force, and giving the effect of additional mass on the ROV's body can be set on developed plug-in by the user. Simulated a thruster device can be added on the body part where the user wishes at. By these functions, the behavior of ROV in underwater environment are realized. Fig.1 shows an example view of ROV simulation. In this test run, we use a simple plant-like model that referring to TOSHIBA's ROV that was deployed into the PCV of FDNPS [2].

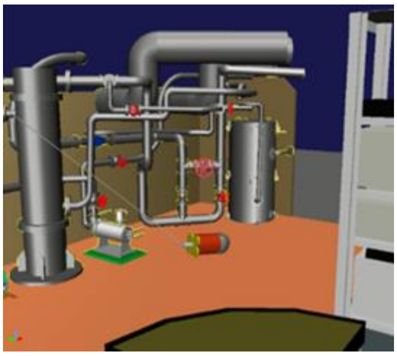

Figure 1. Example view of ROV simulation

3. Conclusion and future works

Developed ROV simulation function provides thruster device to add thrusting forces and torques and some fluid dynamics effects to ROV's behavior in underwater environment. By test run, we could confirm possible to simulate the behavior of remotely operated robot for underwater survey tasks.

In our future work, we would to develop the other plug-in functions to extend our robot simulator system for comitting robot development and the operator proficiency training.

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[2] Handouts at press conference 2017 | Archives "Progress of Unit 3 PCV internal investigation (Preliminary report of July 19 investigation)", Tokyo Electric Power Company, http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2017/index-e.html, (June, 2018)

J14

Information Gathering of Working Environment Based on 3D Reconstruction from Plural Images for Decommissioning

Toshihide Hanari[1] and Kuniaki Kawabata[1]

[1]Japan Atomic Energy Agency

Abstract

This paper describes our research on an environmental information gathering method of working environment based on 3D reconstruction from plural images for decommissioning work. In this paper, we report to apply 3D reconstruction method and post-processing for improving visibility of the environment model to the images from an underwater investigation task inside Primary Containment Vessel (PCV).

1. Introduction

For decommissioning work of Fukushima Daiichi Nuclear Power Station (FDNPS) of Tokyo Electric Power Company Holdings, Inc. (TEPCO), task execution by remotely operated machine is one of the key issues. For such remote operation, structural data of the working environment is useful for sharing information and planning next or future approach. Currently, we consider to utilize a 3D modeling method from images that collected by past operation. Structure from Motion (SfM) is a method for estimating 3D structure of objects and camera's trajectory, simultaneously. In previous work, we generated 3D point cloud model of some targets by using SfM and evaluated the generated model[1]. From the results, we consider that the improvement of visibility is required. In this paper, we describe to improve visibility of the obtained 3D model by SfM.

2. 3D reconstruction experiment and result

For improving visibility of the generated model, we applied dense reconstruction, mesh generation and texturing as post-processing. These processes generate texture model from 3D point cloud data by SfM. In this case, we used OpenMVS software[2] for post-processing. In this experiment, we used the images captured from a video by the investigation task using remotely operated vehicle inside PCV of Unit 3 of FDNPS in 2017[3] as shown Figure 1 (Source: TEPCO). The video was recorded around the bottom of the Control Rod Drive (CRD) housing.

Figure 2 shows 3D reconstruction result of the CRD housing and supporting clamps using SfM and the above-mentioned post-processing. As shown in Figure 2, the reconstruction was almost done and a part of the supporting clamp was not generated. We could recognize 3D structure of the CRD housing and supporting clamps from the result.

4. Conclusion

In this paper, we described the result of 3D structure reconstruction based on SfM and post-processing from the images recorded in the PCV of FDNPS. As the result, we could confirm that 3D structural reconstruction was partly done in this experiment.

In the future work, we will attempt to improve the reconstruction accuracy.

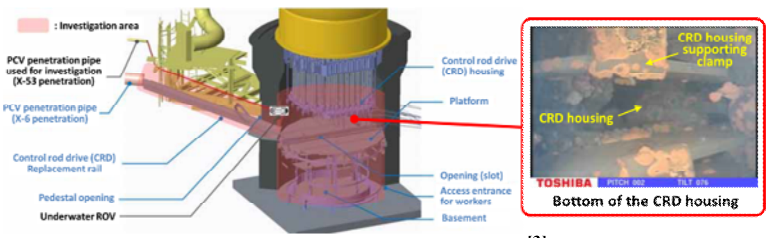

Figure 1. Outline view of investigation in PCV[3] (Source:TEPCO)

Figure 2. Result of 3D reconstruction

References

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[3] Tokyo Electric Power Company Holdings, Inc.: Photos and Videos Library, Progress of Unit 3 PCV internal investigation (Preliminary report of July 21 investigation), http://photo.tepco.co.jp/en/date/2017/201707-e/170721-01e.html

J15

Sensor Database for Research of SLAM in Reactor Buildings

Taichi Yamada[1] and Kuniaki Kawabata[1]

[1]Japan Atomic Energy Agency

Abstract

In reactor buildings of Fukushima Daiichi (hereinafter referred as 1F), there are still uninvestigated areas due to the accident on March 11th, 2011. Hence the reconnaissance by using remotely operated robot in reactor buildings is important for planning the decommissioning. Therefore, Simultaneous Localization And Mapping (SLAM) is a key technology in robotics. This research aims to construct the sensor database for the SLAM research base.

1. Introduction

There are still uninvestigated areas in reactor buildings of 1F. For such investigation, SLAM is one of the key technology developed in robotics. SLAM technology commits to provide the information on where the data such as image and dose rate was token. Thus, in order to enhance SLAM research, we construct the sensor database for SLAM research like face image database for face recognition research base. This paper discusses the required specification of the sensor database for such purpose and applying decommissioning works.

2. Sensor database for SLAM research

SLAM computes the location of the system, from only sensor data such as Light Detection and Ranging (LiDAR), camera and so on. If we have sensor data about the place for investigation, it is not necessary that the SLAM works online. Therefore, the sensor database makes possible to test SLAM method easily and repeatedly. Requirements for the database as a research base for 1F decommissioning and these resolutions are listed below.

1. Logging sensor data of the database are performed in a realistic workspace for assuming the workspace in 1F. In this reason we use mockup fields in Naraha Center for Remote Control Technology Development [1].
2. Reconstructability of mockup fields for logging is required to update the mockup field with 1F investigation results. Because of this, we record a mockup field configuration with visual markers.
3. Ground-truth of robot trajectory is necessary for quantification of test results to compare between SLAM algorithms. Since, we regard the robot position and heading direction measured by motion capture as the ground-truth. The quantified result is provided by comparing the robot trajectory estimated by SLAM with the ground-truth.

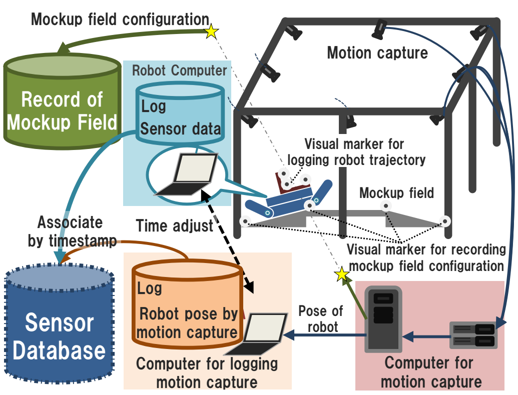

Figure 1 Logging System Summary

3. Conclusion

This paper describes the sensor database for testing SLAM algorithms to apply them to the 1F decommissioning task. In the future works, we will examine the specification of the robot and the mockup field and to log more sensor data with various type of mockups to construct a sensor database.

References

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

J16

3-D sensing technologies of radioactive substances and environmental structure inside the Fukushima Daiichi Nuclear Power Station

Yuta Tanifuji, Yuki Sato, Yuta Terasaka, Hiroshi Usami, Masaaki Kaburagi,
Kuniaki Kawabata, and Tatsuo Torii
Japan Atomic Energy Agency

Abstract

For promoting the decommissioning works, collecting 3D structure information of the working environment is important to understand detail status like distribution of the obstacles and radioactive substances. In this report, we introduce experimental setup for collecting 3D point cloud data and measurement result in the experiment in the turbine building of Unit3 of Fukushima Daiichi Nuclear Power station (FDNPS).

1. Experiment inside the turbine building of the turbine building of Unit 3

Point cloud produced by LiDAR (Light Detection and Ranging) sensor is a typical data to express and visualize the structure of the object. Therefore, we utilize a LiDAR sensor to collect 3D point cloud data for the investigation in the FDNPS. We utilized a portable LiDAR (YVT-X002 made by HOKUYO AUTOMATIC CO., LTD.) for measuring the structure of the environment. This LiDAR can measure 10360 three-dimensional points by one scan (5frames/sec.). The LiDAR is connected to PC and ROS is utilized for collecting point cloud data. In order to avoid occlusion problem, we attempt to compose 3D structure data by merging the measurement data at plural locations.

Fig.1 shows integrated result with 3D point cloud data obtained by the LiDAR and radiation data by the compact Compton camera developed by Waseda University and Hamamatsu Photonics K.K. [1]. In this figure, point cloud data appears as gray points and we can recognize 2.2m width of the narrow corridor, the existence of a step at 8m from the entrance position and no obstacle in the air. Also, measured data by the LiDAR indicates that highly contaminated spot is located at approximately 10m from the entrance of the corridor. At that place, there were the hoses and it is considered that concentrated contamination existed around for these hoses.

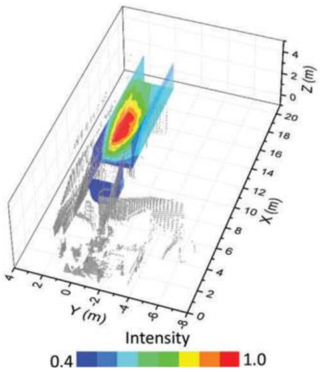

Fig.1 Radiation distribution map superimposed 3D point cloud data [2]

2. Conclusion

In this report, we describes our attempt on 3D structure measurement in FDNPS by using a portable 3D LiDAR. We also introduce measurement result of 3D point cloud at the inside of the turbine building of Unit 3 of the FDNPS and obtained 3D point cloud data was integrated with the measured radiation data to generate 3D distribution map of radioactive substances. Such 3D visualization gives good effort to recognize the state of the environment. In our future works, we would aim to implement sensory data collection and integration system to remote control system.

References

[1] J. Kataoka et al., Handy Compton camera using 3D position-sensitive scintillators coupled with large-area monolithic MPPC arrays, *Nucl. Instr. and Meth. A*, 732 (2013) 403-407.

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ON01 NEA's role in addressing International Challenges in Radioactive Waste Management

Ichiro Otsuka and Gloria Kwong

OECD Nuclear Energy Agency

The nuclear community has long recognized that long-term management of nuclear waste, in particular the disposition of high-level radioactive waste and spent nuclear fuel, is a socio-technical issue and a Decide-Announce-Defend (DAD) approach cannot be successful. The NEA acknowledged in the 1990s that "an inform societal judgement is necessary" while technical expertise and expert confidence in the safety of geologic disposal are insufficient, on their own, to justify to a wider audience geologic disposal. The decisions, whether, when and how to implement it will need a thorough public examination and involvement of all relevant stakeholders. Noting the increased stakeholder involvement in the consensus decision-making process, the NEA Radioactive Waste Management Committee (RWMC) is assisting member countries to develop safe, sustainable and broadly acceptable strategies for the long-term management of all types of radioactive waste. In this role, the RWMC also provides member governments and other relevant stakeholders with authoritative, reliable information on the political, strategic and regulatory aspects of managing radioactive waste and decommissioning. In 2016, the RWMC issued a collective statement emphasizing the importance of implementing a holistic and sustainable management strategy for radioactive waste and spent fuel at an early stage. In particular, this RWMC Statement presents a methodology for transforming the vision of the Committee into tangible work activities. Following the guidance of this Statement, future activities of the RWMC, i.e.2019 and beyond, will focus on three pillars, namely (i) environmental and operational safety; (ii) societal aspects and (iii) economics. It is envisaged that a work program organized in this manner will benefit from a strongly unified vision that fosters deeper global collaboration while maintaining a broad outlook for existing and future work project. The Committee, in implementing its new programme of work, maintains its fundamental purposes:

- fosters a shared and broad-based understanding of the state of the art and emerging issues;
- facilitates the elaboration of waste management strategies that respect societal requirements;
- helps to provide common bases to the national regulatory frameworks;
- enables the management of radioactive waste and materials to benefit from the progress of scientific and technical knowledge, e.g., through joint projects and specialist meetings;
- contributes to knowledge consolidation and transfer, e.g., through the publication of technical reports, consensus statements and short flyers; and
- helps to advance best practice, e.g., by supporting international peer reviews.

For more information on this NEA committee, visit http://www.oecd-nea.org/rwm/

ON02 The NEA Committee on Decommissioning of Nuclear Installations and Legacy Management (CDLM): a new platform for international cooperation

Ichiro Otsuka and Gloria Kwong
OECD Nuclear Energy Agency

As commercial nuclear power continues into its sixth decade, a growing number of nuclear reactors are at or reaching the decommissioning phase. Efficient decommissioning of a nuclear facility can be a complex process which requires careful planning and skillful selection of decommissioning strategies and dismantling techniques and processes. In addition to the increasing demand in nuclear decommissioning, many NEA countries are also facing challenges in managing their legacy waste and sites as a result of historic nuclear activities. To assist member countries to develop safe and cost-effective practices in managing these challenges, the NEA is creating a new technical committee on Decommissioning of Nuclear Installations and Legacy Management (CDLM).

The key objectives of the CDLM include the following:

i) provide a forum to facilitate the exchange of experience and information on policies and practices in the areas of decommissioning and legacy management between NEA member countries, and to advance the state of the art on technical, environmental, policy, financial and societal aspects in these areas;

ii) advance and develop international understanding and guidance, in support of national authorities, to address questions of common concern regarding concepts, policies, regulatory frameworks, and strategies for decommissioning and legacy management.;

iii) share and further develop concepts, policies and approaches for reaching accepted and sustainable decisions on decommissioning and legacy site end-state aspects aiming at reducing the burden on current and future generations from facility decommissioning and remediation; in particular to keep under review the developments in the optimisation of decommissioning strategies bringing in balance technical, economic, safety and ethical aspects;

iv) enhance worker and public safety, and environmental protection by developing practical radiation protection guidance to manage nuclear legacy sites and decommissioning;

v) further develop approaches for the estimation of decommissioning project costs by improving understanding of the risks associated with financial consequences in regards to cost estimating and financing; and decommissioning planning for uncertainties.

ON03

NEA's Support to the Fukushima Daiichi Decommissioning and Addressing Public Concerns

Ichiro Otsuka, Edward Lazo and Kentaro Funaki
OECD Nuclear Energy Agency

Seven years after the Fukushima Daiichi nuclear power plant accident, the Japanese government and Tokyo Electric Power Holdings, Inc. (TEPCO) are shifting their focus to strategic planning for long-term challenges related to the decommissioning of the damaged reactors. The international community has been helping to address the unprecedented challenges of managing the accident facilities. The NEA is playing a key supporting and coordinating role in the international community, in particular in the area of radioactive waste management and the evaluation of the conditions and location of fuel debris.

The NEA Working Party on Decommissioning and Dismantling (WPDD), under the aegis of the NEA Radioactive Waste Management Committee (RWMC), provides a good knowledge base for the mid- to long-term planning of decommissioning the Fukushima Daiichi NPP. In 2017, TEPCO joined the NEA Co-operative Programme on Decommissioning (CPD), a joint undertaking gathering and sharing experience gained from recent decommissioning projects among NEA member countries, to learn and exchange practical decommissioning experience in order to achieve safe, economical and environmentally sound decommissioning.

The NEA Committee on the Safety of Nuclear Installations (CSNI) has been assisting member countries in developing the required scientific and technical knowledge base for maintaining the safety of nuclear reactors and fuel cycle facilities. The NEA Joint Project Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station (BSAF) assists member countries to improve severe accident codes and knowledge of the reactor core status so as to prepare for future decommissioning. The CSNI Senior Expert Group on Safety Research Opportunities Post Fukushima (SAREF), using information gained from post-accident examinations of Fukushima Daiichi, identified safety research knowledge gaps to advance safety. The CSNI Working Group on Analysis and Management of Accidents (WGAMA) has accumulated knowledge and expertise relevant to decommissioning strategy planning for the Fukushima Daiichi NPP.

With respect to radioactive waste management, the NEA Expert Group on Fukushima Waste Management and Decommissioning R&D (EGFWMD) s provided strategic waste management advice to Japan in managing post-accident waste of Fukushima Daiichi. In 2018, the RWMC is also establishing an Expert Group on Characterisation Methodology on Unconventional and Legacy Waste (EGCUL) whose aim is to focus on the development of an integrated methodology for characterizing unconventional radioactive waste.

The Committee on Radiological Protection and Public Health (CRPPH) continues its risk communication through dialogues with the International Commission of Radiological Protection (ICRP) and the NEA International Workshop on Post-Accident Food Safety Science Workshop.

Recent activities of these groups are further explained in separate posters in this conference.

ON04

Strategic Approach to Managing Fukushima Daiichi Waste

Ichiro Otsuka and Gloria Kwong
OECD Nuclear Energy Agency

In response to the urgent need to safely manage radioactive waste as a result of the Fukushima accident, the NEA Radioactive Waste Management Committee (RWMC) established an expert group to advise the Japanese Government on strategic management of large quantities of post-accident waste that has complex or unknown properties. The NEA Expert Group on Fukushima Waste Management and Decommissioning R&D (EGFWMD) undertook this three-year initiative in 2014 to assess the technical challenges faced by the waste managers, regulators and other stakeholders. In this project, the EGFWMD examined the physical and chemical nature as well as the radiological characteristics of the waste. Specific issues related to feasible waste conditioning and volume reduction to facilitate interim storage and future disposal, decontamination of the facilities were also assessed. In December 2016, the expert group concluded its work and published a project report that provides technical opinions and recommendations on efficient post-accident waste management strategy and R&D at the Fukushima Daiichi site. The report concluded that characterisation and categorisation of post-accident radioactive waste are among the most difficult challenges for waste management at the Fukushima Daiichi NPP in the near term. A strategic approach which includes a sampling and characterization plan based on statistical approaches, calculation methods, and a review and evaluation process for the data obtained should be developed to manage the complex characterisation process of the post-accident waste. The group also acknowledged that international knowledge and experience in managing legacy waste, accident waste and other pertinent examples would be valuable in address challenges in managing the Fukushima Daiichi waste.

In continuing its technical support to Japan, the RWMC is creating a new Expert Group on Characterisation Methodology of Unconventional and Legacy Waste (EGCUL) whose aim is to share knowledge and experience in characterisinga large amount of unknown waste. A key deliverable of the EGCUL is to develop an integrated methodology for the characterization of unconventional and legacy waste of Japan. The EGCUL is anticipated to include specialists from Japanese organisations such as the Japan Atomic Energy Agency (JAEA), the Nuclear Regulation Authority (NRA), TEPCO and the NDF, alongside experts from around the world with experience in radioactive waste characterisation. More detailed outcomes of the EGFWMD and the work-plan of the EGCUL are presented in this poster session.

F01

IN CAN MELTER for the conditioning of high and medium-level and long-lived waste from D&D operations

Christophe GIROLD [1], Christine GEORGES [1], Stéphane CATHERIN[2], Thierry PREVOST [3], Guillaume LECOMTE [4]

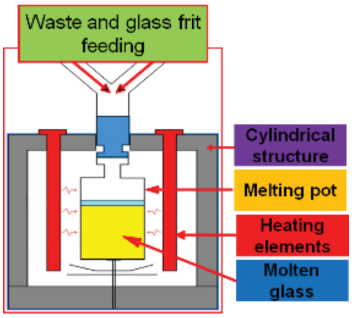
[1] CEA, Nuclear Energy Division, [2] ANDRA, [3] AREVA NC, [4]ECM Technologies

Abstract

"IN CAN MELTER" process is being developed by CEA for its own waste coming from D&D operations, when "glass-like" encapsulation is required for further disposal but when cost reduction is an aim. Low temperature solidification with few additives would be of great interest is the encapsulation of mixed effluent waste such as zeolite, co precipitation sludge or powder of fuel debris.

• A robust, simple and versatile in situ vitrification process adapted to D&D waste

- Adapted to high or medium level waste,
- The container is used as a crucible ("in can"), renewed with each batch: no need to drain the glass and corrosion is limited by the batch limited time,
- Waste package complying with existing routes and/or on-site storage facilities (CSD),
- Heated by a simple and robust resistance heated furnace, inexpensive, easy to maintain, easy to implement and offering a great homogeneity to the melted mixture. This allows a metallic fraction in the canister and an excellent control of temperature.
- Flexible enough to accommodate uncertainties on waste composition and however produce a confinement with quality description, with source term calculation methodology for long term storage.
- Small amount of secondary waste (no components with molten glass remains after use) and minimum investment and operating cost.



• Tested for High Level Cs deposits from Marcoule UP1 shop under decommissioning

• Application to effluent treatment waste (zeolites, sludges, …) on going

Lab scale glass formulation adapted to the in can process with a 70 - 80 % waste loading of zeolites, carbonated and ferric sludge, sand, silicotitanate.

• In the frame of a French founded program (DEM&MELT project), a new prototype is under building to develop a vitrification tool adapted to D&D

- **All the process being include in a small footprint frame to be easily forward-deployable on an existing D&D site**
- **Capacity : With a 300 kg (x3 / in can) CSD canister**
- **Producing a small amount of secondary waste**
- **Designed to cost as aimed to be dismantled after treatment operation**

F02 Development of Laser Cutting and Dust & Fumes Collection technologies for Fuel Debris Retrieval

D. Roulet[1], C. Georges[2], C. Chagnot[2], C. Journeau[2], E. Porcheron[3]

[1] ONET Technologies, France, [2] CEA, French Alternative Energies and Atomic, [3] IRSN

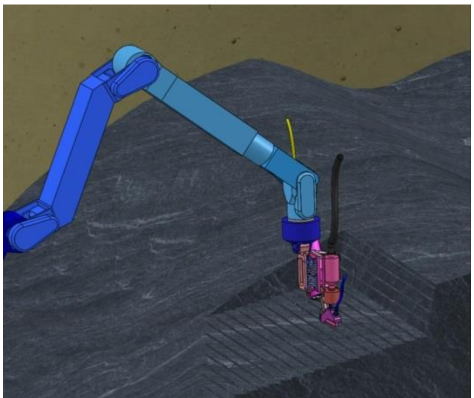
Abstract

Among the numerous challenges towards the decommissioning of Fukushima Dai-ichi reactors, removing fuel debris is for sure on of the tougher ones. A French consortium “ONET Technologies, CEA and IRSN” is developing laser cutting technologies for fuel debris with the implementation experience in UP1 French reprocessing plant. In addition to laser cutting development in air and underwater, dust and fumes collection technologies are being developed to capture the aerosols released during the debris laser cutting and their collection.

1. Cutting Technologies

The variety of fuel debris location, material and shape will need various cutting techniques to be developed. Laser cutting technologies are capable to overcome many configurations with proven performance, and great benefit with regards to remote application. Research for application to Fukushima environment is on-going with promising results.

Several technologies are being developed (in air, underwater, emerging or non-emerging) with good results and performance fo future application to Fukushima Fuel Debris.



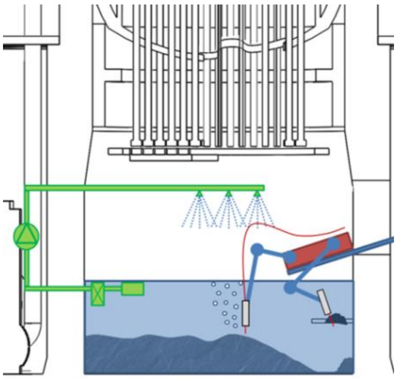
Simulation of fuel debris non-emerging cutting by laser

2. Knowledge of dust and fumes generated during cutting

Laser cutting being a thermal cutting method, aerosols, gases and particles are emitted during the process and the characterization of such secondary outlets is of paramount importance. The conducted trials and associated numerical simulation enable estimation of released radioactive elements, aerosols size and gases, which are an important input data for collection and treatment systems definition and estimation of release to environment.

3. Collection Technologies

In parallel to cutting technologies, collection systems are being developed to capture the dust and fumes generated during the cutting. Such technologies are to be deployed remotely, and in case of underwater cutting spray systems for capturing aerosols are under development.



Conclusions

Laser cutting technologies associated with collection technologies are key processes for fuel debris retrieval, but could be also used for other systems cutting as part of Fukushima-Daiichi decommissioning as CRD or RPV internal structures.

F03 Potentialities offered by New Binders in conditioning of problematic low- and intermediate-level radioactive waste

David Lambertin, Christine Georges, Fabien Frizon, Céline Cau dit Coumes,
CEA, Nuclear Energy Division

Abstract

Some waste produced by decommissioning operations may chemically react with cement phases or mixing water, affecting hydration and eventually resulting in the deterioration of the waste form in storage or disposal, for instance by swelling and cracking. Three kinds of inorganic binders offer better chemical compatibility, thus avoiding pre-treatment or increased volumes of waste: calcium aluminate and sulfo-aluminate cements, magnesium and calcium phosphate cements, alkali-activated binders and more specifically geopolymers.

Calcium Sulphoaluminate Cements (CSAC)

CSAC clinker are produced by firing mixtures of limestone, gypsum and bauxite. Foreseen applications of CSAC for waste conditioning ranges from Cementation of borate or heavy metals such as lead, cadmium, zinc, and trivalent chromium or to immobilization of Aluminum. Due to the different cement chemistries, the rate of hydration of CSAC may be less affected by strong retarders of OPC such as heavy metals or borate ions and its mineralogical structure is favorable for waste immobilization, offering possibilities of ion substitutions.

Magnesium Phosphate Cements (MKP)

Phosphate cements are the main representatives of acid-base cements. Waste immobilization results from two processes: precipitation of many contaminants, particularly actinides, as phosphates with very low solubiphysical encapsulation in a dense phosphate matrix. Good results have been reported for several types of wastes, e.g. low-level debris wastes contaminated by 137Cs, highly saline effluents or metallic aluminum by decreasing hydrogen corrosion evolution (Figure 1).

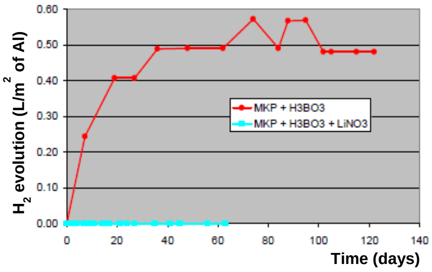


Figure 1: Impact of addition of LiNO3 (2 wt.%)

Geopolymers: alkali-activated binders

Alkali-activated binders are made by mixing solid aluminosilicates, such as fly ash, metakaolin, various clays usually activated by heat or blast furnace slag, with an activating solution comprising high concentrations of alkali hydroxide and / or polysilicate. The reaction product, formed according to a dissolution/ polycondensation process, exhibits an X-ray amorphous network structure.

It is possible to vary the properties of the produced alkali-activated binders over a wide range, and to tailor them to specific requirements like high compressive strength, low permeability, low shrinkage.

Figure 2: Encapsulation of Mg cladding in geopolymer

Figure 3 : Oil/ geopolymer composite

Geopolymers offer great potential to stabilize/solidify hazardous waste like heavy metals, alkali-earth, alkali ions and magnesium alloy encapsulation (Figure 2).

Geopolymers present a new way for direct organic liquid waste encapsulation by stabilizing emulsion of oil in an alkali silicate solution and being directly solidified by addition of metakaolin (Figure 3).

Understanding the chemistry of cement – waste interactions, and their consequences on the physical properties of the solidified waste forms, including their long-term evolution, has to be further considered for their acceptance.

F04

Foams and gels for decontamination of structure and concrete walls

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Abstract

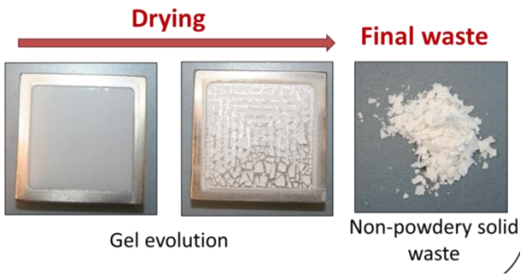
The specificity of gels processes developed by CEA and ORANO for the decontamination of structure are that they are sprayable and vacuumable. The main advantage of these gels is that the final secondary waste is a non-powdery solid waste that can be recovered by brushing or vacuuming. This self-drying technology is well adapted to treat large surface as walls or floor but in case of complex geometry or of vessel, CEA and ORANO recommend the use of gellified foam. These both technologies give unique tools for treatment of structure with the objective to minimize to production of secondary waste. CEA develops new self-drying gel since a couple of years designed to remove bitumen stain contamination, or specific to the decontamination of concrete walls.

1. Introduction

The decommissioning of nuclear facilities for end-life management is a major issue in present and future years. However, their prior decontamination is a necessity to avoid the generation of high volume of contaminated waste. These operations will conduct to the de-categorization of structures in order to valorize them in traditional deconstruction sectors or to rehabilitate them for a new utilization. Among the available decontamination techniques, we must chose technologies that minimize the secondary waste while being highly efficient.

2. Specification

In this context, self-drying vacuumable gels and gellified foams technologies are mature processes developed a few years ago.



The gels are formulated by mineral colloidal particles (alumina, silica or titanium oxide) dispersed in an aqueous solution containing the reactants (acid, base or oxidant). This reactant is chosen to remove the contamination by attacking the substrate and then transferring the radio-contaminant from the surface to the gel flakes after drying. The specific rheological properties of these gels make them sprayable on the contaminated surface and their thixotropic behavior allows them to recover their viscosity immediately after spraying and adhere to the surface. After a few hours of drying, the gel has turned onto a non-powdery solid which can be easily brushed or vacuumed. The main advantage of this process is that no liquid effluents are produced. These gels are commercialized by the FEVDI company with a large range of specific gels for different applications depending on the nature of the surface to be treated and the nature of the contamination. FEVDI can also purchase specific sprayer for this application.

A gellified foam has a long lifetime and then allows the decontamination of complex shape (vessel containing tubes, complex geometries…) by maintaining during a few hours the contact between the contaminated surface and the aqueous reactant containing into the foam. These kinds of foam are formulated by mixing into an aqueous phase a very small amount of surfactant, a viscous polymer gel (the gelling reactant) and the reactant (acid). This process produces lower amount liquid effluents than batch processes with a volume reduction by a factor of ten.

3. New developments

To spread off these technologies beyond their traditional use, we develop two kinds of new formulations: Hybrid gel well designed for the treatment of organic layers and specifically the removal of bitumen smudges [1]. The replacement of aqueous solution classically used in vacuumable gel by an organic bio-solvent leads to the definition of a self-drying gel able to dissolve fast and efficiently contaminated organic layer and specifically bitumen; Wet pastes specifically designed for the decontamination in depth of porous surface, and then can be applied for concrete walls decontamination. This is a two steps treatment. First water contained into the paste diffuses to the porosity of the contaminated substrate (concrete) to solubilize contaminants. Then during the drying phase, water and solubilized contaminants flows back from the solid to the drying paste allowing the extraction of the contaminants by advection.

4. Conclusion

Gellified foam and self-drying gel processes are mature processes for the decontamination of solid waste. They can be used for the treatment of structure, or complex shape as vessels, tube…These technologies still go on by the improvement of their implementation, and by the development of new kind of formulations.

References

[1] Gossard A., Frances, F. "Vacuumable gel and method for eliminating contamination contained in an organic layer on the surface of a solid substrate" WO2018024990A1

[2] Faure S., Fournel B, Fuentes P. "Composition, foam and method for surface decontamination" WO2004008463

F05 Fission Products, Uranium and Plutonium measurements in gases and corium after a PWR severe accident

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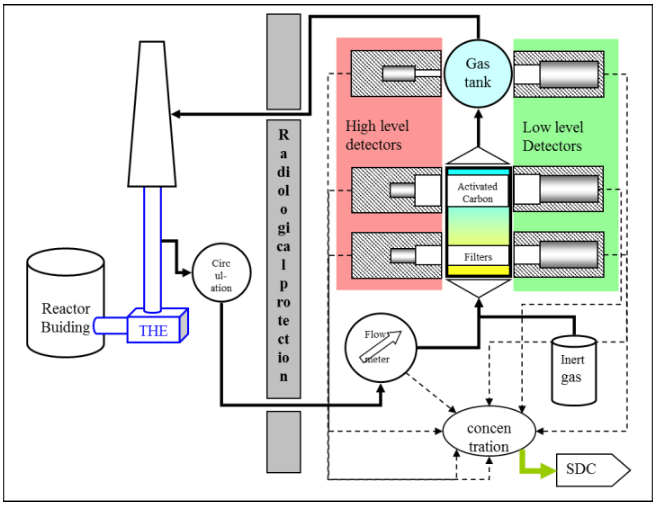
Following the Fukushima accident, a lot of recommendations were drawn by international organizations (IAEA, OECD, NUGENIA,etc.) in order to improve the safety in such accidental conditions and limit consequences. One of these recommendations was to improve the robustness of the instrumentation, as well as to better determine the Source Term involved in a nuclear accident.

In this context, CEA designed two systems: one to assess the reactor building gas releases in case of severe accident and one to measure Uranium and Plutonium contained in spent fuel hulls.

The first system, designed in collaboration with Institut Max von Laue-Paul Langevin (ILL) consist in a gamma spectrometry system devoted to the measurement of Fission Products gases. It implements a high sensitivity and high energy resolution Hyper-Pure Germanium detector (HPGe). It is robust and able to operate under the severe environment encountered after accidental situations.



The gamma spectrometry measurement system allows the measurement of the numerous gamma ray lines emitted by the short live fission products and then to detect a situation of re-criticality

The second system, called “PACCMAN” is a transportable system allowing an active neutron measurement of U and Pu fissile isotopes in a 25 liter waste drum. The measuring setup is designed to be docked to a hot cell (PADIRAC system). It is made on the basis of a RD10-type castle around which measurement instrumentation is implemented. It uses a neutron generator and ^{3}He counters to reach a detection limit mass of about 100 mg ^{239}Pu equivalent in 15 minutes. Under several conditions, this system could be efficiently used to measure a fissile content in small quantities of corium to assess the criticality risk.

PACCMAN System

The Nuclear Measurement Laboratory is specialized in non-destructive methods, such as gamma-ray spectrometry, passive neutron measurement, active neutron interrogation, neutron activation analysis, high energy X-ray imaging systems for radiography and tomography applications. It concerns expertise, R&D, experimental assays, modelling, perfecting in laboratory and in-situ measurements. The applications are for example nuclear fuel, radioactive waste, dismantling and decommissioning, activity and fissile masses characterization, NRBCE security

F06

Sorbmatech sorbent for Cesium removal from contaminated liquid effluents

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[1] CEA Nuclear Energy Division, [2] CTI,[3] ORANO

Abstract
CEA, ORANO and CTI collaborate since a couple of years for the development of a new family of inorganic sorbent, named Sorbmatech®Cs, highly efficient for the decontamination of liquid effluents and suitable as final waste form. A new sorbent developed in this context was able to extract 137Cs selectively (K_d equal to 10^5 to 10^6 ml.g^{-1} depending on the salinity of the effluent), with a very fast kinetics (equilibrium reached in a few minutes), either in stirred reactors or in column processes.

1. Introduction

Permanent needs to reduce radioactive species release into the environment requires continuous technology improvement in liquid effluent treatment process, minimizing secondary solid waste production. Not only accidental situations (such as in Fukushima Daichii), but also nuclear energy production industry, nuclear fuel recycling plants and end-of-life nuclear facilities decommissioning, are at the origin of contaminated aqueous liquid waste production. Among all radionuclides that need to be treated, 137Cs is considered as the most abundant and hazardous element due to its presence in many types of waste and its relatively long half-life (30 years).

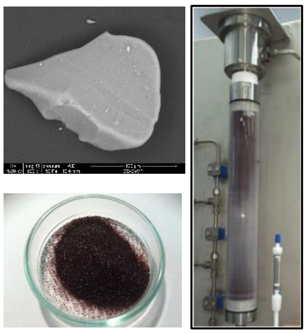
2. Specification

In this context, we have develop an efficient sorbent based on porous silica support (pores size a few to a few ten nm) loaded with hexacyanoferrate nanoparticles (HCF). The general formula of HCF is KxMy[Fe(CN)6]z where M, a bivalent transition metal ion, can be Cu, Ni or Co. The selectivity of these various HCF varies with the pH of the effluent versus the nature of the transition metal, so we can simply adjust the efficiency of the sorbent depending on the HCF inserted into the porous silica support. This fabrication approach can be applied for various support, from porous grain to filtration membranes.

To date the partnership between CEA-CTI-ORANO leads to the development of a simple and low cost synthetic route allowing the fabrication of a large-scale manufacturing of this product. Two products have been developed so far: (1) a 200-500µm grain size (S250g) well adapted for column processes (low pressure-drop) and (2) a <100µm grain size (S250p) well adapted for stirred reactor uses. The evaluation of the efficiency of these pre-industrial products, from pre-industrial mockup (1L column, 3L reactor) to lab-scale with 137Cs doped effluents show very high selectivity even in saline solution (KdCs from 105 to 106 mL/g). A wide range of effluents can then be treated: high, medium and low activity, pH from 2 to 10, high salinity (saline, mineral or sea water). But the most interesting property of this product is its really high sorption kinetics even with large grain size suitable (200-500µm) for column process. This allows the use of high flow rates of treatment (up to 10m/h) with low pressure drop and deep breakthrough curve. Smaller granulometry silica grain size (<100µm grain size) enables both good dispersion and easy sedimentation in stirred reactor.

3. Developments

100Kg of SORBMATECH®Cs for column treatment (200-500µm grain size) was produced and the pre-industrial manufacturing of an equivalent product (Grains <100µm), usable in stirred reactor, is also being evaluated up to 10Kg production.
The outlook for SORBMATECH® sorbent series is not only the widening of the pH range of the potentially treated effluents (by modifying the exchanger for increasing performances in acidic medium, and by modifying the support for holding in basic medium), but also the widening of radionuclide range that can be treated, with the first line of sight of 90Sr.
Offering customers a complete solution, for effluent treatment using dimensioning model for implementing column process at industrial scale, and also for long-term stabilization of generated waste, directly suitable with vitrification process for example, are under progress.



4. Conclusion

The insertion of nanoparticles HCF into porous silica support leads to a high selective Cs-sorbent with fast kinetics of adsorption suitable for industrial effluent treatment, whatever the technology, stirred reactor or fix bed. These products are in a pre-industrial manufacturing step and will be complete to treat other RN such as Sr.

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[2] Caroline Michel, Yves Barré, Laurent De Windt, Caroline de Dieuleveult, Emmanuelle Brackx, Agnès Grandjean, Journal of Environmental Chemical Engineering, 5 (2017), 810-817.

[3] Agnès Grandjean, Yves Barré, « procédé de préparation d’un matériau solide nanocomposite à base d’hexa- et octacyanométallates de métaux alcalins », patent WO2016038206 A1

F07 Virtual Reality: a way to prepare and optimize operations in decommissioning projects

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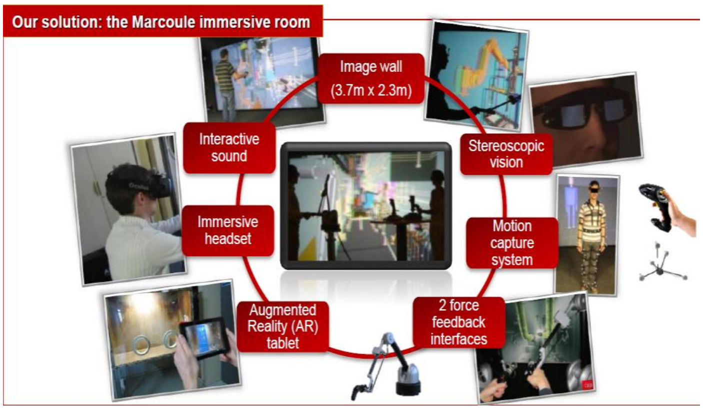
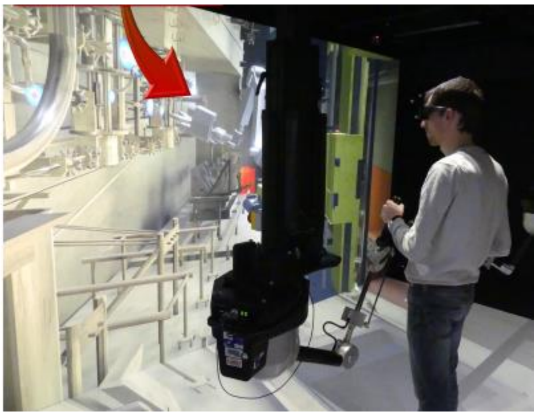
CEA must manage the end of life of its nuclear fuel cycle facilities built in the sixties.
These high priority actions led to the creation of a dismantling R&D division which provides innovative tools, including in-situ radiological characterization, remote handling and cutting, and intervention scenario simulation.
Simulation is an ideal means of visualizing and therefore better knowing highly radioactive environments where humans cannot enter, of testing different technical alternatives, and of training workers prior to interventions.
A VR simulation can be defined as an interactive and immersive simulation that enables the user to interact with a computer-simulated environment. VR environments, mostly based on visual immersion displayed through stereoscopic devices, can also include additional sensory information, such as sound or touch.
New PRESAGE immersive room in Marcoule, based on audio, tactile and visual immersion, provides a useful support to verify pre-defined scenarios and to design alternative solutions if necessary.

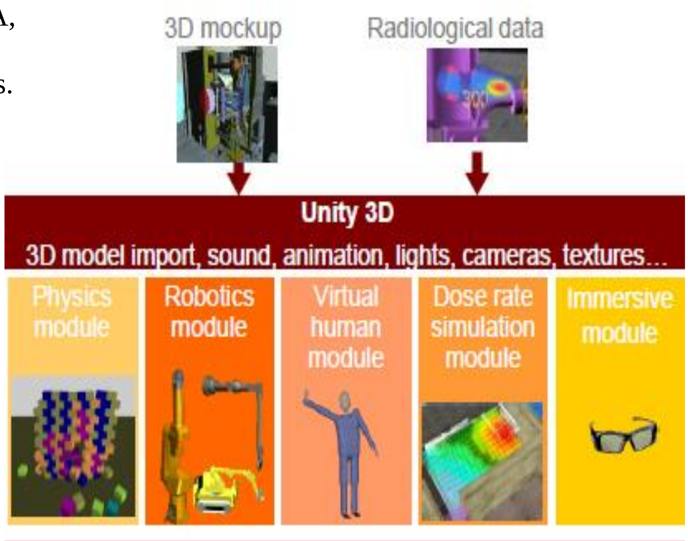




iDROP, A VR SOLUTION FOR NUCLEAR OPERATIONS

iDROP is a new software developed by the CEA, and offers a global approach taking into account dose rate, remote handling and human operations.
iDROP offers 5 modules:

- The physics module calculates forces and collisions and avoids object penetration.
- The robotics module includes the main industrial robots (about 40) which can operate in nuclear facilities.
- The virtual human module enables the simulation of manual operations taking into account the human degrees of freedom and features.
- The dose rate module integrates MERCURE calculation code and estimates dose rate received at any point taking into account the radiological data.
- The immersive module enables the simulation to be connected to immersive VR and AR devices.



F08 Supply Chain for Decommissioning in France

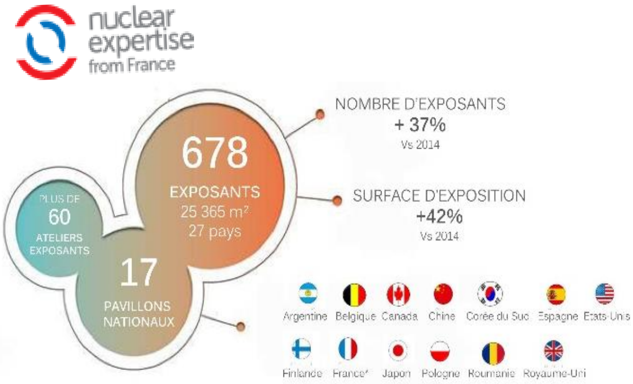
Christine GEORGES*, Anne Sophie DEFAY**, Serge PEREZ *, Serge BOUFFARD***, Claudia ROCHE****

(*): CEA, DEN, France, (**): AIFEN; (***): Nucléopolis; (****): Nuclear Valley

Decommissioning benefits from France's nuclear supply chain that was constantly in action since the sixties, thanks to its uninterrupted nuclear programs. Nuclear industry is France's third-largest industry, with over 2,500 French companies, nearly 220,000 employees, a yearly turnover of over €46 billions and added value reaching €15 billion, R&D investment ranking 4th amongst innovative industries.

CSFN (French nuclear governmental Steering Committee) coordinates the actions of the French nuclear stakeholders in the unifying topics: innovation, training, exports, etc.

In 2013, CSFN created AIFEN to support the French nuclear team abroad and specifically its Small and Medium-sized Entreprises (SMEs). AIFEN designed the Nuclear Expertise from France logo, which is a tailored brand illustrating the French nuclear team and organizes, every two years, the World Nuclear Exhibition (WNE), meeting point of the global nuclear community, gathering around 10 000 visitors, from more than 70 countries.



CSFN aims to strengthen ties and partnerships between the different bodies, including Nuclear Safety Authority, public authorities and related governmental Departments, Certification and qualification bodies, Nuclear operators and research institutions (EDF, CEA, AREVA, and ANDRA), Suppliers of nuclear technologies, AIFEN and regional groups (NUCLEAR VALLEY, PVSI, Nucleopolis, etc.), Trade groups and labour unions (GIIN).

Set up in 1959, GIIN - French Nuclear Suppliers Association is an independent and non-governmental organization acting to support the French supply chain. GIIN represents over 450 SMEs and mid-caps companies with expertise in new build, plant operation, fuel cycle, dismantling & decommissioning, and research reactors.

Nuclear Valley, founded in 2004, national cluster dedicated to civil nuclear in France, with more than 230 companies, half SMEs and Mid-Caps, operating in civil nuclear. It provides support innovation and training through collaborative innovation projects in the field of metallurgy, big components, non-destructive testing, maintenance and decommissioning and civil work.

PVSI, created in 2014 around Marcoule CEA site, combines 24 companies from industry, research and education, aiming at forging a French sector of excellence in the field of Decommissioning, through increased ETIs to large account resources, training and professional retraining of the operators.

Nucleopolis, nuclear cluster of Normandie Energies, created in 2010, federates about 100 companies: major industrial groups, high-performance SMEs, renowned research and training institutions (CEA, GANIL, Universities, etc.) and care centers.

1. The NDA: **Supporting Innovation** 革新を支える

The NDA ensures the safe and efficient clean-up of the UK's nuclear legacy from the UK's pioneering post-war programme. Covering 17 sites, the NDA's mission includes taking all the structures apart, dealing with the waste and ultimately restoring sites.

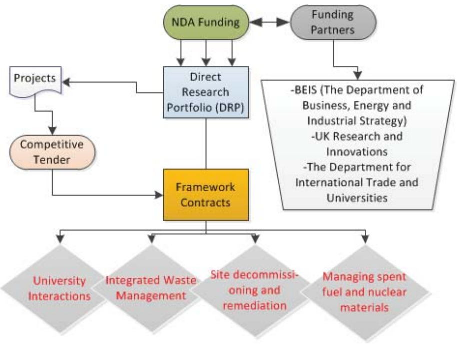
One of NDA's responsibilities is to ensure the adequate amount of R&D is carried out to deliver the full decommissioning programme. Many 'never-done-before' projects require significant innovation and novel engineering approaches.

R&D is required to ensure technical challenges are understood, to encourage development of innovative ideas and enable the demonstration of new technologies followed by successful on-site implementation.

Funding

NDA fund R&D through two main routes:-

- Directly through projects commissioned by the NDA to support strategy (DRP)
- Via Site Licence Companies on specific site projects where spend is part of main annual budget



The DRP is the main mechanism by which the NDA directly funds R&D. R&D proposals are developed with three key drivers in mind: Inform Strategy, Deliver Innovation and Maintain and Develop Skill.

Governance

Advice and guidance for co-ordinating nuclear decommissioning R&D is provided by the NDA's Research Board. Independently chaired, membership comprises senior representatives from UK government, UK industry, NDA Estate, regulators and from overseas.

Support is provided on a working level by the Nuclear Waste and Decommissioning Research Forum (NWDRF). NWDRF promotes collaboration across the NDA estate and the UK whilst ensuring innovation is delivered efficiently, cost effectively, quickly and with less duplication.

Case Studies: Innovative Remote Technologies

In 2017, NDA and Innovate UK launched a competition named "Integrated Innovation in Nuclear Decommissioning" inviting proposals for projects that could deliver solutions to decommissioning challenges at Sellafield. Submissions were invited from all industrial sectors.

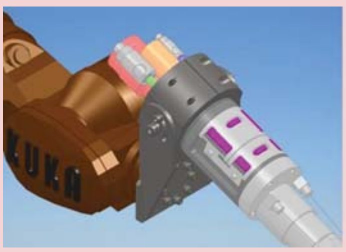
The response from suppliers and academic institutions was so promising that additional funding was made available by BEIS. Suggestions came from consortia made up of SMEs, large corporations and universities, belonging to industries as diverse as sea-fishing and medical imaging.

Five promising ideas have been shortlisted for the final stages. Each will receive up to £1.5m to build prototype demonstrators for testing in a mock up radioactive environment.

Robot Laser Cutting

Laser cutting minimises spread of contamination during operations. A commercially available robot was fitted with laser technology and tested in a radioactive environment. To allow laser cutting to be deployed remotely in a safe and efficient way, trials and development work included metal cutting and testing of control systems and ventilation.

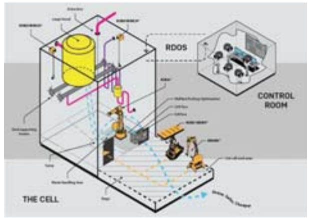
The potential to save money and reduce risk has been seen and the learning is now being used for alpha cutting.



Robotics: Nu-Decom

Plant Operations using remotely-operated tools: Utilisation of the BROKK 150C with a Pecker Attachment for Decommissioning activities.

Elephants to Ants: Innovation in Integration

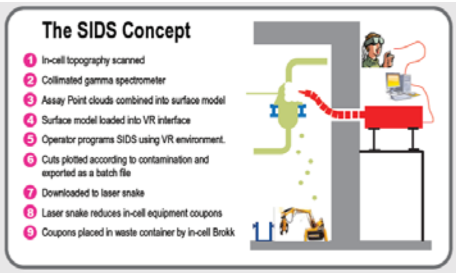


Elephants to Ants is creating a real-time sensed, hardware-in-the-loop VR Operating System for decommissioning, to enable engineers to control teams off-the-shelf robots with confidence, speed and precision.

Integrated Innovation for Nuclear Decommissioning

A nuclear-centred integration approach, which innovates a new modular control & automation strategy that can be proven & validated within the nuclear environment.

Sellafield In-Cell Decommissioning System (SIDS)



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INNOVATION IS GREAT BRITAIN & NORTHERN IRELAND

2. Remote Operations: **Under the Water** 水中技術

This poster focuses on the use of remote handling technology in the decommissioning and demolition of nuclear facilities in the UK. Remote handling solutions are providing effective, more efficient solutions to challenges faced across the NDA estate. Underwater-environment technologies range from novel navigation and sampling techniques to commercially available Remote Operated Vehicle (ROV) platforms. These technologies enable access to areas that would be otherwise impossible to work within without huge investments of time and money required to drain down and shield to allow traditional decommissioning techniques to be used safely.

A. INTEGRATE | 統合

Underwater surveys at Sellafield - FORTIS

FORTIS sonar technology has been deployed in legacy ponds at Sellafield, where it would be too difficult or dangerous for human entry. The system uses point cloud technology to generate 3D CAD models that can be interrogated to allow better understanding of the geometries within a facility and has enabled detailed planning of future decommissioning works.

B. MODIFY | 修正

ROVs have proved to be efficient in taking on a wide range of activities, from pond liquor sampling and characterisation to underwater inspections, surveys and maintenance tasks.

ROV platforms at Sellafield's Legacy facilities - Rovtech

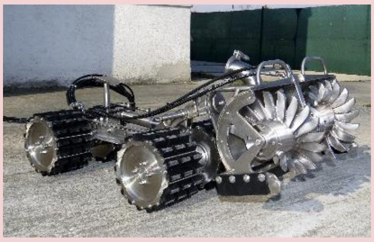
The Nano platform is a modular system designed to allow fast disassembly and cleaning while wearing multi-layer protective clothing. Variable buoyancy systems are designed to operate in areas that have large quantities of silt and sludge that traditional depth control systems can disturb, spreading contamination and obscuring vision. They were deployed at

In-pond operations at Sellafield – James Fisher Nuclear (JFN)

JFN have built up a vast experience base of operating ROV systems within the UK's nuclear facilities.

Larger 'Work Class' ROVs regularly carry out successful size reduction and remediation tasks at Sellafield's FGMSP. They have successfully employed a series of tools developed by JFN, ranging from manipulator arms, saws, cutters through to scoops and dredgers.

C. DESIGN & DEVELOP | 設計と開発



Underwater sampling at Sellafield - NNL

Sampling sludge and sediments left over from nuclear waste while under water is a difficult task due to the fact that disruption of the environment in and around the sludge can change its properties providing false data. NNL has developed a series of custom tooling that minimises this distubance and prevent the uptake of additional liquor.

Underwater sampling at Bohunice NPP - Wood

The submersible Crawler Systems utilise simple but robust hydraulic and pneumatic drive systems to minimise the need for hardened electronics when operating in high dose rate environments therefore greatly increasing the systems' operating lifetime. These systems have been used in decommissioning activities across Europe.

Conclusion

The ability to access, monitor, investigate and operate equipment and plant under water within a nuclear facility can be critical for successful and safe decommissioning. Remote operation technologies have been used in the UK to save money and reduce the risk of working in these environments while also allowing physical operations to be carried out in areas which have not been easily accessible. The UK continues to benefit from these technologies and is proud to share their experience with Japan.

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3. Remote Operations: **On the Ground** 地上で

This poster focuses on the use of remote handling technology in the decommissioning and demolition of nuclear facilities in the UK. Remote handling solutions are providing effective, more efficient solutions to challenges faced across the NDA estate. Remote technology provides a safer, controlled barrier between the workforce and many hazards present on a decommissioning site including radiation, contamination and health and safety issues. The technology can be integrated across a wider waste retrieval system to ensure that safe, and appropriate tools are being used and reduces the lifetime of these retrievals projects.

A. INTEGRATE | 統合

Commercially available robotic technologies are reviewed for use and tested under a range of functions including teleoperation and automatic modes for standard tasks. This promotes safety benefits and supporting evidence to underpin the safety case for the operation of the equipment.

NNL Kuka Robots for Sorting & Segregating

Brokk ROV at Trawsfynydd NPP

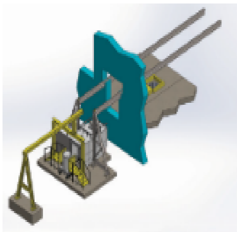
B. MODIFY | 修正

Projects with commercially available robotic technologies have adapted specially designed components to be fit for purpose within various hostile, unpredicted environments.

This provides highly optimised and engineered solutions that meet all the client requirements, de-risk the design from a functional aspect and reduce the full life-cycle decommissioning costs.

Dounreay shaft intervention platform – Veolia Nuclear Solutions
This project is unlike anything undertaken in the UK before, possibly the world.

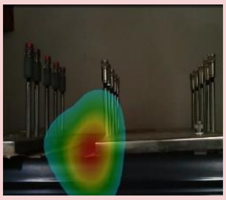
Berkeley Chute Silo – Aquila
In-silo size reduction, retrieval and transfer of waste to shielded containers for export to on-site storage facility.



Harwell & Hunterston HiRAM & GEM assay - Nuvia
High Resolution Assay Monitor (HiRAM) is a trailer-mounted high-res gamma spectroscopy semi-automatic assay system. Gamma Excavation Monitor (GEM) system is a remotely operated gross gamma system. Both used successfully to assay contaminated land and rubble.

C. DESIGN & DEVELOP | 設計と開発

MHI Fuel Debris Retrieval - Veolia Nuclear Solutions
This bespoke remotely-controlled robotic arm system is desgined to investigate fuel debris in Fukushima Daiichi Unit 2 Reactor. The solution combines knowledge gained through work in the nuclear fusion section and demanding environments at Dounreay and Sellafield and was developed with UK supply chain including NIS and Createc for enclosure,



Hot Spot locators at Fukushima Daiichi – Innovative Physics Ltd (IPL)

Radiation imaging systems can map contamination in hazardous environments. Integrating a pixelated sensor and photomultiplier tube with an optical camera, these systems have been proven to be effective for decontamination and decommissioning purposes, for example aiding the post event clean-up operations around the Fukushima Daiichi Power Plant.

Conclusion

With experience from a range of sites and projects, the UK industry has gathered experience in remote handing, sort and segregation by looking at each individual situation and creating solutions with either commercial off the shelf equipment, modified industry robotics or bespoke systems. UK industry is pleased to be working in Fukushima and transferring the experience and knowledge gained from these projects.

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DBD
James Fisher Nuclear
CREATEC
PCUBED
British Embassy Tokyo
NIA Nuclear Industry Association
gleeds
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MOTT MACDONALD
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英国は原子力廃止措置の最適なパートナー

4. Remote Operations: **In the Air** 気中技術

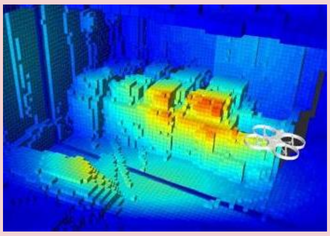
This poster focuses on the use of remote handling technology in the decommissioning and demolition of nuclear facilities in the UK. Remote handling solutions are providing effective, more efficient solutions to challenges faced across the NDA estate. Access to high places to inspect and to do work is a recurring challenge in nuclear decommissioning. Traditionally, this could be done using scaffolding or mobile work platforms, but these have several limitations: they expose workers to hazards (both radiological and working at height); they can take up a lot of space, where space is precious, and they can take time to set up and operate making them costly. The UK has developed remotely-operated technology that can address some of these limitations.

A. MODIFY | 修正

UAV at Sellafield – James Fisher Nuclear (JFN)

JFN has adapted and deployed commercially available internal and external UAVs and sensors to collect critical data quickly, more cost effectively and with much greater safety than by conventional means. UAVs have delivered a range of instruments and tools including high-resolution digital and thermal cameras, radiation probes, LASER scanners and air samplers into restricted access or high hazard areas. This flexibility and versatility has enabled easy access to vital data to give decision-makers a more complete picture. All data has high level encryption for complete security in any eventuality including complete loss of the UAV.

B. DESIGN & DEVELOP | 設計と開発

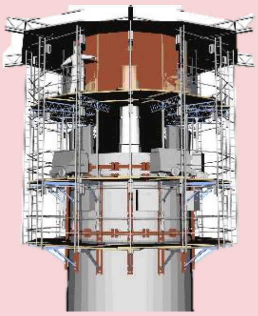


RISER UAV at Fukushima Daiichi – Createc

COTS UAVs use GPS for control, which means they can only be used outside where they have a view of satellites. To extend the use of UAVs to internal structures, Createc developed a lidar navigation system for UAVs. The system was used to make 'RISER' in collaboration with UAV development specialist Bluebear Systems. RISER has been operated at Fukushima Daiichi since March 2017, and has investigated many hard-to-access areas, including inside damaged turbine halls and reactor buildings. RISER is integrated with Createc's N-Visage 3D radiation imaging technology to provide maps of radioactive contamination which help with planning decommissioning.

Self-Climbing Platform (SCP) at Sellafield – Nuvia

Sometimes it is also necessary to carry out decommissioning works at height without good access. A current example is the SCP deployed to decommission one of the highest stacks at Sellafield. This stack is 60m high and projects from the top of a 60m tall building. Access around the stack is very limited due to other high hazard buildings. The SCP climbs up the stack and then dismantles the stack from the top down. The 3-tier SCP was installed in 2017, climbed to the top of the stack and is currently in demolition operations. It has total of 52 descend cycles ~1m each with demolition activities taking place at each descend cycle to remove the liner, concrete wall and rebar. The SCP is currently at the 13th descend cycle with 20% of the stack already demolished.



Conclusion

The use of UAVs is beginning to become a standard technique for inspection of building at nuclear sites in the UK. The use of drones was pioneered at Sellafield, where Sellafield recognised the potential to save money, reduce risk and actively engage with its suppliers to set up standards for operating UAVs at its site. Since then, other sites, including operating sites, have followed a similar model.

Innovative solutions continue to be developed in the UK to tackle the issues around access, space and proximity to radiological and conventional safety hazards to make decommissioning and dismantling of nuclear facilities safer, quicker and more cost effective. UK industry is pleased to be working in Fukushima and transferring the experience and knowledge gained from these projects.

NDA Nuclear Decommissioning Authority · cavendish nuclear · DBD · James Fisher Nuclear · CREATEC · PCU3ED · British Embassy Tokyo · NIA · gleeds · NIS · MOTT MACDONALD · NUCLEAR AMRC · NUVIA · RLB Rider Levett Bucknall · UK Atomic Energy Authority · innovative physics · Sellafield Ltd · Rolls-Royce · wood. · NATIONAL NUCLEAR LABORATORY · INS INTERNATIONAL NUCLEAR SERVICES · ATKINS · VEOLIA

R01

Polyfunctional Adsorbent LHT-9: “All-in-One” Platform for the Development of Multi-Nuclide Removal Systems

Sergey Britvin[1], Boris Burakov[2], Valery Mararitsa[1], Yury Demidov[1], Yury Petrov[2], Bella Zubekhina[2] and Victoria Gribova[1]

[1]Socium Ltd., [2]V.G. Khlopin Radium Institute

Abstract

LHT-9 is a multi-nuclide sorbent applicable for both HLW and LLW decontamination. Granulated LHT-9 is suitable for use in flow columns, submersible filters and even exhaust gas cleaning systems. Besides, the sorbent can be readily modified into derivatives, like selective iodine scavengers. Altogether, LHT-9 can be regarded as a perspective platform for “all-in-one” implementation at the ALPS-type decontamination facilities.

1. Introduction

The distinctive feature of nuclear waste is multi-nuclide composition. Consequently, its decontamination requires multistep sorption by a series of selective sorbents. The development of single “all-in-one” sorbent is still a technological challenge. We herein present LHT-9, a promising candidate for multi-nuclide removal.

2. Layered Hydrazinium Titanate

LHT-9 is a layered titanate containing chemically bound hydrazine [1,2]. It is capable of simultaneous sorption of more than 50 *chemical elements*, including 62 *radinuclides* known at the Fukushima site. It possesses specific surface area up to 500 m^2/g and available in granulated form. LHT-9 can be used for decontamination of both HLW and LLW. The difficult-to-clean *volatiles*, like elemental iodine and methyl iodide can be readily removed from the exhaust gases. Besides, the modified LHT-9 sorbents are effective halogen-selective scavengers of both iodide (I^-) and iodate (IO_3^-) from the liquid waste.

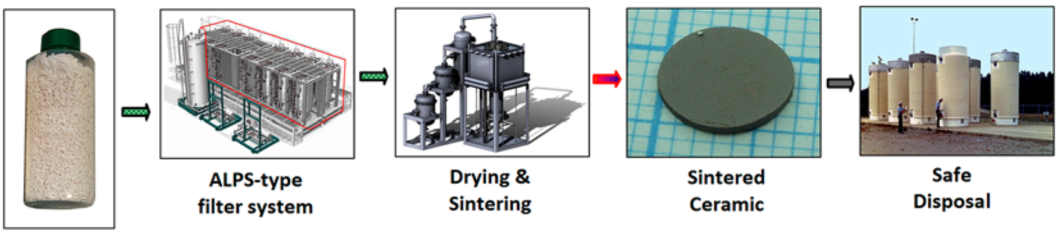


Figure 1. The flowchart illustrating implementation of LHT-9 for nuclear waste disposal

3. Conclusion

Based on the properties of LHT-9, this sorbent can be successfully implemented into the multi-nuclide removal systems at the Fukushima site. It can be also used for decontamination of ordinary technological waste produced during normal operation of atomic power stations.

References

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[2] PCT WO2011116788A1 / RU2564339C2

[3] Zubekhina, B.Yu. *et al.* (2018): MRS Advances; DOI: 10.1557/adv.2018.208

R02

Development of prediction model of the F1's fuel debris accelerated aging

Main approaches for estimation of fuel debris properties changing.

Oleg Bagryanov, Senior Project manager, TENEX*, Sergei Poglyad, Deputy head of radiochemical technologies department PhD, RIAR*, Boris Burakov, Head of laboratory DSc KRI*

Abstract

While sampling of F1 fuel debris (FD) is still to be done in the future, projects for developing systems, technologies and equipment of FD retrieval, storage and transportation are ongoing and require input data on the FD properties. In the absence of real sampling such data may be obtained through research of overseas accidents and modeling of samples simulating FD of F1. Rosatom is carrying out project to study ageing properties of F1 FD by researching Chernobyl samples and modeling simulated samples imitating FD of F1. This will allow to estimate the mechanisms of fuel debris self-destruction, to determine the factor most affecting changes in properties, to develop the prediction model of F1's fuel debris accelerated aging (based on the results of researches of Chernobyl's fuel-containing materials).

1. Introduction

Rosatom develops a prediction model that will allow to estimate the changes of the main FD properties in time. Such prediction model will provide researchers (working on the developing of fuel debris retrieval, packaging, storage and stansportation) with a possibilities to verify their solutions.

2. Main steps in development of prediction model

2-1. Investigation on the characteristics of fuel debris in overseas accident reactor and their influences on the system to retrieve fuel debris

The main purpose of this stage is to summarize prior research of Chernobyl fuel containing materials (FCM), make the assumptions on major factors contributing to degradation due to ageing, and develop research programme based on those investigations.

2-2. Acquisition of data on the changes by aging of fuel debris and its migration into water

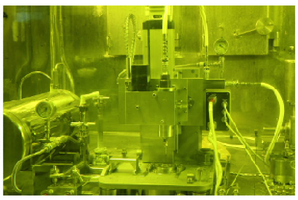
Based on the initial Chernobyl research and programme of research model active samples will be prepared and tests with simulated fuel debris will be performed to investigate degradation processes due to ageing.

2-3. Comprehensive prediction of changes by aging

Based on the results of Stage 2, comprehensive prediction of changes by ageing should be made to estimate changes after 10, 20, 30 and 50 years since generation of fuel debris.

3. Conclusion

The tests with simulated active samples is the one of the possible scheme to approve (by regulatory body) solutions developed for the FD management.



*Companies of Rosatom group

R03 Mobile robotic manipulator of anthropomorphic design

Sergei Floria[1], Dmitry Kucherov[1]
[1]RosRAO, FSUE

Abstract

RosRAO has developed and conducted an experimental operation of mobile robotic manipulator (MRM) of anthropomorphic design to conduct work within radiation fields.

The manipulator is built on hydraulic circuit with replaceable electromechanical grippers for precise operations. The robot copies the movement of operator through a special copying suit and allows to conduct works at a distance up to 100 meters.

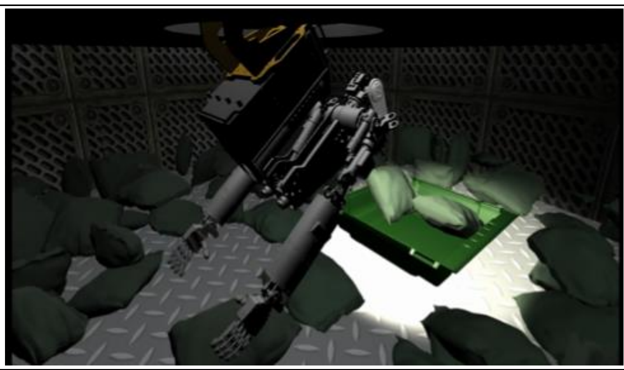
1. Introduction

The MRM ensures manipulations with stuff under 15 kilos with the help of one hand, there are possible operations with any common industrial tools. Such a configuration allows careful segregation of non-standard or fragile radioactive wastes as well as applying common manual tools to dismantle equipment in high radiation fields. Robot’s hands are manufactured with additive technologies.

Pic.1 MRM mounted on the industrial manipulator

Pic.2 Control suit

Pic.3 SRW storage



Pic.4 Screenshot: operator’s training program

SK01

A Tritium Measurement Technology of Underwater around Decommissioning Site

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

Tritium is a pure-beta emitter. When tritium is absorbed into the body, it displaces protium of water molecule to form Organically Bound Tritium (OBT) and causes ionization damage of Deoxyribonucleic Acid (DNA). When the nuclear power plants were dismantled, it is necessary to develop a device for measuring the tritium concentration in underwater around the decommissioning site.

1. Introduction

In order to measure the radioactivity concentration of tritium in water, tritium used in this study is concentrated tritium gas sample. Tritium calibrator is currently in development. When tritium calibrator is developed in the future, to compare the experimental results with commercial tritium monitor calibrator and to verify the validity of the experimental results by the developed system, the radioactivity concentration was measured by commercial tritium monitor calibrator. In order to confirm the validity of the measurement results by commercial tritium monitor calibrator, the expected theoretical results of the radioactivity concentration got by measuring of the calibrator volume, pressure, tritium gas decay factor, and tritium gas initial concentration should be compared with the actual experimental results.

2. Method and Result

2-1. The Theoretical Calculation

The theoretical calculation was carried out by the equation derived by measuring the volume and the pressure of calibrator [1]. Figure 1 shows activity concentration of tritium according to pressure in metering volume. the range of pressure in this study was varied from 11 psi to 20 psi. The theoretical calculation results are represented as black closed square of figure 1. The slope of the theoretical calculations was estimated by 0.567.

2-2. The Experimental Measurement

The experimentally measured results are represented as red open square on the figure 1. The slope of the experimentally measured results was estimated by 0.565 using linear fitting.

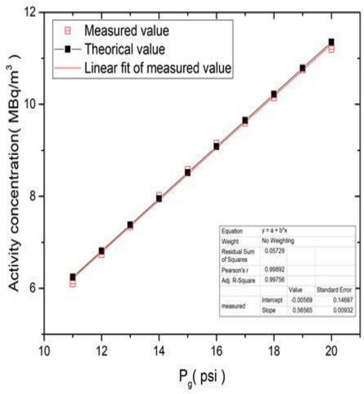


Figure 1. radioactivity concentration according to pressure.

3. Conclusion

Both the theoretical calculation and the experimentally measured results were linearly increased. The slope of each results was highly similar with relative error of 0.35 %. These results can be a reference for the performance of device to be developed.

References

[1] OVERHOFF TECHNOLOGY CORPORATION, CALIBRATION PROCEDURE TRITIUM GAS CALIBRATOR MODEL 10017

SK02 Detection Technique for Real-time Monitoring of Tritium in Underwater

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 detection technique for real-time monitoring of tritium in water was proposed. The method was based on the electrolysis. Experimental set-up was carried out to measure tritiated water samples continuously using electrolysis. The detection efficiency for hydrogen with tritium was estimated as 27%.

1. Introduction

The decommissioning of the Fukushima Daiichi Nuclear Power Plant is a pending issue not just for Japan but also for the international community. During the D&D process radioactivity monitoring is essential to prevent secondary contamination and dispersion of contaminated material [2]. Monitoring of tritium in the laboratory is time-consuming work. In this study, a detection technique for real-time monitoring of tritiated water was proposed based on the electrolysis.

2. Methods and Results

Figure 1 shows overall features of real-time tritium monitoring system. When the sample (tritiated water) is poured in the sample container, the pump pushes the sample to the electrolysis cell. The sample is electrolyzed and hydrogen gas with tritium is released. Overflowed water is trapped and only hydrogen gas goes through the plastic scintillator based detection chamber. The detection efficiency of the tritiated hydrogen gas was estimated as 27%. The detection limit of the system was 17.8 kBq/L for 15,000 s of measurement time.

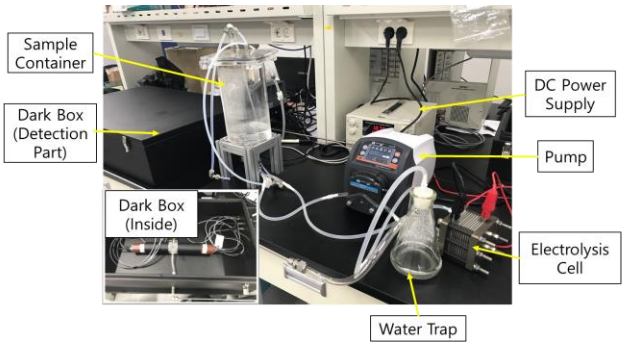


Figure 1. Real-time tritium monitoring system

3. Conclusion

A detection technique for real-time tritium monitoring was proposed. The method was based on the electrolysis and experimentally characterized. The technique was applicable for high-radioactive tritiated water which can be released during decommissioning process.

References

[1] Mori, Koji, et al. "Continuous improvement of fitness-for-duty management programs for workers engaging in stabilizing and decommissioning work at the Fukushima Daiichi Nuclear Power Plant." *Journal of occupational health*, 60.2:196-201 (2018).

[2] Kang, Houjun, Dong Zhang, and Zhengkun Shi. "Measurement of tissue free water tritium in plant." *Atomic Energy Science and Technology*, 43.3:221-223 (2009).

SK03 The Efficiency Calculation of Plastic Scintillator for in-situ Beta Measurement System in Groundwater

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

Abstract

Develop in-situ groundwater beta measurement system of which detection part comes into direct contact with the matter. The performance of plastic scintillator for in-situ measurement system was analyzed using simulation and experimental methods.

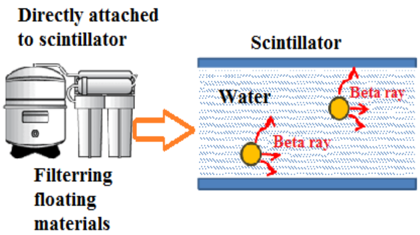


Fig. 1 Concept of in-situ system

1. Introduction

Decommissioning site can be contaminated with various radionuclides, this site characterization is needed. Conventional analysis method of beta-emitting radionuclides is usually costly and result in long delay between sampling and obtaining results [1]. Therefore, in-situ monitoring technology is requested like Fig. 1.

2. Method and Result

2-1. Plastic scintillator and beta source

Plastic scintillator is used as it is non-hygroscopic and can be fabricated into various shapes with large sensitive area. ^{3}H, ^{14}C, ^{32}P, $^{90}Sr/^{90}Y$ were pure beta emitting radionuclides used in the simulation and experiment.

2-2. MCNP simulation and experiment

The efficiency of the plate was determined instead by the energy deposition of beta particles using MCNP6 F8 (e.p) tally. For experiment radioactive aqueous samples were prepared and counted for 600 seconds. The dimension of vial was similar to simulation one. The thickness of plastic scintillator was chosen as 1 mm.

2-3. Result

The counts of ^{3}H and ^{14}C are similar to that of the blank sample. But ^{32}P and $^{90}Sr/^{90}Y$ showed good agreement between simulation and experiment. The relative difference values of efficiency were 1.8 % for ^{32}P and 0.4 % for $^{90}Sr/^{90}Y$ as shown in Table 1.

Table 1. Experiment results compared to simulation

Source	Experiment efficiency	Simulation efficiency	Relative difference
^{32}P	5.55 ± 0.08 %	5.650 ± 0.002 %	1.8 %
$^{90}Sr/^{90}Y$	4.60 ± 0.01 %	4.620 ± 0.001 %	0.4 %

3. Conclusion

The efficiency of plastic scintillator was calculated based on MCNP simulation and experiment. 1 mm of plastic scintillator could be used in in-situ system. Thin plastic layer (0.5 mm) would degrade accuracy due to noise and background.

References

[1] In-situ radiation detection demonstration, Stephen W. Duce, Amir H. Mohagheighi, Mark L. Miller, Robert R. Reese and David R. Miller. WM'00 Conference, February 27 – March 2, 2000, Tucson, AZ.

SK04 Radiation Dose Rate Distribution Monitoring System with Portable Gamma Detector for Decommissioning Site

Rapid and broad scale of site monitoring

UkJae Lee[1], Jun Woo Bae[1], Hee Reyoung Kim[1]

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

Abstract

In order to secure the radiological safety of the decommissioning site, a system for monitoring the radiation dose rate which can apply to a wide area was established.

1. Introduction

It is essential to secure the radiological safety of radiation workers and surrounding residents within the decommissioning site. Based on this need, the system which can be applicable to a wide range of areas, and rapid site monitoring technology is suggested.

2. Method and Result

2-1. System components

The system is based on portable 3 in by 3 in NaI(Tl) scintillator detector which has energy range of 80 keV ~ 3 MeV and 7.5 % of detection resolution. The contour-mapping program, implemented with MATLAB software, enables the 2D or 3D display of radiation level distributions with one click of a mouse immediately after the corresponding area is scanned by the portable gamma radiation detector.

2-2. Monitoring methods

Scanning around site was performed in a vehicle with the installed system. The information on latitude, longitude, and radiation dose rate were obtained through actual measurements and used as data sets for the radiation contour mapping. The surrounding area of the monitoring point was scanned by a vehicle and its speed was approximately 30 km/h. The distance between the ground and the detector was 1 m.

2-3. Results

Figure 1 shows contour maps of radiation distributions. It verifies that the radiation distributions were shown in the 2D and 3D contour maps with geographical information distributions.

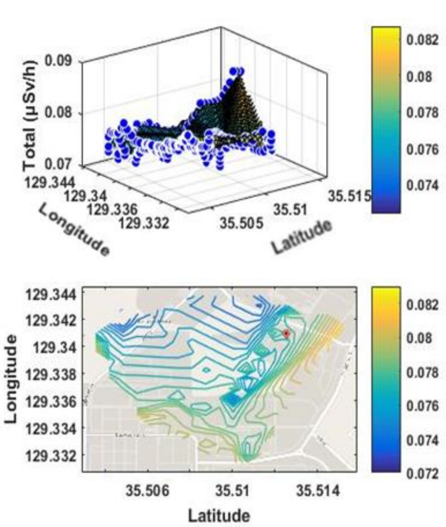


Figure 1 The Result of Monitoring System for Radiation Dose Rate Distribution

3. Conclusion

Real measurement-based quick radiation distribution monitoring system was established. The system could be directly applied to in-situ finding of hot spots in the vast area subject to decommissioning of nuclear power plants.

References

[1] Kim, H. R., et al. (2008). "The Establishment of an In-Situ Real Time Radiation Contour Mapping Technique." Journal of Nuclear Science and Technology 45(sup5): 662-665.

SK05 Criteria Analysis of Autonomous Aerial and Marine Radiation Monitoring System for Safe NPP Decommissioning

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

Abstract

A new aerial and marine radioactivity monitoring system is developed based on application of autonomous vehicles. Specifically, by assuming several radiological accident scenarios during nuclear decommissioning, requirements for MDA (Minimum Detectable Activity) and deployment and motion pattern are determined.

1. Introduction

Decommissioning of nuclear power plants requires a lot of workforce and cost. Autonomous vehicle can be suitable candidates for radioactivity monitoring system, reducing additional workforce and cost. There are types of autonomous vehicles for both aerial and marine environmental survey, such as fixed wing UAV (Unmanned Aerial Vehicle), rotary wing UAV, autonomous underwater glider, and autonomous float. In order to construct efficient system by using above vehicles, requirements including MDA quantities, deployment/motion pattern should be determined appropriately.

2. Requirements for autonomous radioactivity monitoring system

2-1. Accident scenarios during nuclear decommissioning

IAEA suggested 6 example list of incidents and accidents during decommissioning of nuclear facilities (e.g., Spill of decontamination fluid, failure of ventilation system) [1]. Such scenarios can affect not only workers, but also neighboring residents. Thus, long-range aerial and marine monitoring activities should be conducted to protect nearby people.

2-2. MDA requirements

Aerial MDA quantities were set up based on AMS (Aerial Measuring System) of U.S. DOE [2], which uses helicopter and airplane for airborne radiation measurement. Marine MDA quantities were determined by 10 years radioactivity survey data of KINS (Korea Institute of Nuclear Safety) over the Far Eastern Seas [3].

2-3. Deployment and motion pattern for autonomous tracking

Autonomous vehicles would be deployed in and around nuclear power plants to monitor aerial and marine radioactivity year-round. When radiological accident occurs during nuclear decommissioning, motion for finding higher radioactivity level, serpentine motion pattern, and concentric motion pattern will be beneficial to assess radiological dispersion efficiently.

References

[1] IAEA, 2013, Safety Assessment for Decommissioning, Safety Report Series No. 77, IAEA.

[2] U.S. DOE, 2000, Overview of FRMAC Operations, U.S. DOE.

[3] Yoon et al., 2015, Marine Environmental Radioactivity Survey, KINS.

US01

AVANTech's Advanced Polymer Solidification for Radioactive Waste Stabilization: Case Studies

Tracy Barker, Gary Benda, Jim Braun, Raja Beereddy and Dennis Brunsell
AVANTech, Inc., 2050 American Italian Way, Columbia, SC 29209

This poster describes recent developments in polymer technology, as well as summarizes several applications and the history of process improvements that simplify the process and improve the efficiency for stabilizing low and intermediate level radioactive and mixed radioactive/hazardous wastes.

Polymer technology has been used successfully since the early 1990s to solidify and stabilize radioactive and mixed hazardous/radioactive waste at United States (US) Department of Energy (DOE), US Department of Defense (DOD), United Kingdom government sites, and at commercial nuclear power plants. It is an effective, efficient, and economical means of preparing waste for storage and disposal.

In 1992, Dow Chemical effected a transfer of the polymer solidification technology to Diversified Technologies Services, Inc. (DTS) of Knoxville, TN, (a firm acquired in 2014 by AVANTech, Inc. of Columbia, SC). DTS undertook to expand the Dow polymer solidification initiative to applications beyond liquids and resins from chemical cleaning solutions to in-situ solidification of granular waste such as ion exchange resin and activated carbon.

Based on extensive waste form testing, the Advanced Polymer Solidification (APS™) was shown to be a viable method of generating stable US NRC Class B & C resins and filters for on-site storage pending the opening of a US NRC Class B & C disposal facility. A key advantage of APS™ is that it renders a product that is compliant with NRC 10 CFR 61.56(b) Branch Technical Position (BTP).

The AVANTech APS™ processes use in-container mixing, high-speed mixing, and in-container In-Situ solidification to address a wide variety of radioactive and mixed waste materials, including liquids, sludges, bead resin, granular media, and powders. The resulting polymer-solidified product forms a rock-hard monolith and meets or exceeds the US NRC 10 CFR 61.56(b) BTP requirements for a Class B & C waste form whose integrity is assured for long-term storage, transportation, and land burial environments.

Recently, polymer solidification has been combined with other waste pretreatment technologies, such as evaporator concentrate drying, to further enhance volume reduction and packaging efficiency.

This poster reviews applications of the technology for use at Fukushima and identifies waste streams where polymers have advantages over cement, vitrification, and other stabilization processes.

APS™ Mobile Metered Injection System - Primary Resin Solidification at the US Diablo Canyon Nuclear Power Plant

US02

Monitoring Internal Conditions of Radioactive Waste Storage Containers

Gregory Allan, Pentek

The cleanup process at the Fukushima Daiichi site presents many technical challenges. One of the more important tasks involves the packaging, short-term storage, transport, and long-term storage of radioactive waste.

The waste will consist of traditional and non-traditional forms. An example of a traditional form-factor are the spent fuel assemblies loaded in fuel storage containers. Non-traditional form factors include the contaminated steel, concrete and other building materials being removed by robotic systems and other decontamination and demolition (D&D) activities. In fact, Pentek supplied an WIFI controlled robotic system, known as the Moose equipment, to TEPCO and Kajima to remove rubble from the refueling floor in U3 at Fukushima. Custom, fork-truck compatible waste containers were used for the storage of large rubber (< 10 cm in diameter) and standard 200-liter drums were used for small debris and dust.

In 2016-2017, Pentek, and its sister company CM Technologies, completed a research program for the US Department of Energy to develop a technology that could pass data and power through the thick steel containers used for spent fuel storage in the USA without any wires or penetration holes in the container.

This presentation explains this technology and how it could be applied in radioactive waste packaging at Fukushima.

US03 UK National Waste Program: A Successful Transformation in Radioactive Waste Management

Tim Carraway, Dennis Thompson, and Max Ehrhardt

AECOM

Abstract

This Poster illustrates AECOM's experience in developing and executing a National Waste Management Program for a country with very limited radioactive waste disposal capacity. Similar to Japan, the United Kingdom's radioactive waste management approach 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.

1. Implementing 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, AECOM has developed and implemented the UK National Waste Program that accomplishes the goals of the Policy. Before AECOM developed the UK National Waste Program, it was projected that the LLWR site's disposal capacity would be fully exhausted by the year 2023. As a result of AECOM's 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. AECOM Approach to UK Low-Level Waste Management

The success of the UK's LLW Policy has contributed to the 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 as well as other waste generators including Defense Projects, Medical Facilities, Educational Facilities and other industrial generators of radioactive waste. A waste-informed decommissioning process is applied such that limited disposal waste volumes are incorporated into the decommissioning strategy to provide the best overall solution for the country.

The National Waste Program includes individual services contracts with all UK waste generators. AECOM's 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

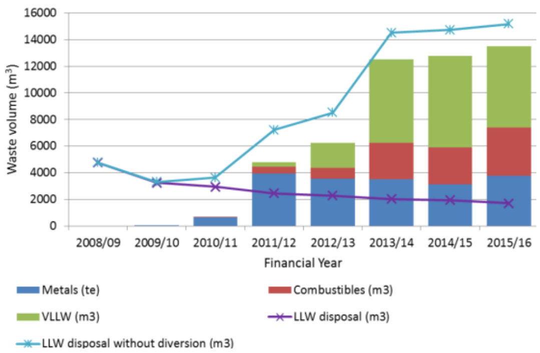


Figure 1. Reduced LLW Disposal under the NWP

Avoidance of Radioactive Waste generation is the best option as it has extended the life of the LLWR site and has greatly reduced the cost of disposal, transportation, and treatment requirements in the UK. Avoidance has best been achieved by AECOM working directly with the decommissioning sites to ensure that the correct Decommissioning and Dismantling processes are applied to minimize the generation of as much radioactive waste as possible. Applying decommissioning and dismantling methods that avoid co-mingling of radioactive waste and non-radioactive waste is a primary approach to avoid radioactive waste generation. Other methods used by AECOM to minimize disposal at LLWR include Metal Size Reduction, Decontamination via Shot Blasting, Metal Melting, Incineration of Combustible Waste, and Shredding and Compaction.

3. Conclusion

Even though waste volumes in the UK have increased as shown in Figure 1, the risks of depleting the disposal capacity at LLWR and the newly opened disposal vaults at the Dounreay site in Scotland are being successfully mitigated. By utilizing the Waste Management Program, waste generators are avoiding more than 90% of their waste from requiring disposal at LLWR. Waste treatment and avoidance methods are proving to be either cost neutral or cheaper than disposal, resulting in sustainable savings for the UK's decommissioning programs. The existing LLWR site will be able to service the needs of UK waste generators to the year 2130 and potentially beyond.

US04

EnergySolutions Decommissioning Experience

Colin Austin[1] and Makoto Kikuchi[2]

[1] Energy*Solutions*, [2] MKC Consulting

Abstract

EnergySolutions (ES) is an international leader in commercial nuclear power plant (NPP) decommissioning. It has developed this experience over a period of 30 years performing NPP decommissioning and related nuclear waste management services in the USA and Europe. This presentation will provide some specific examples of its decommissioning experience in the USA and identify benefits that this experience brings to NPP decommissioning projects.

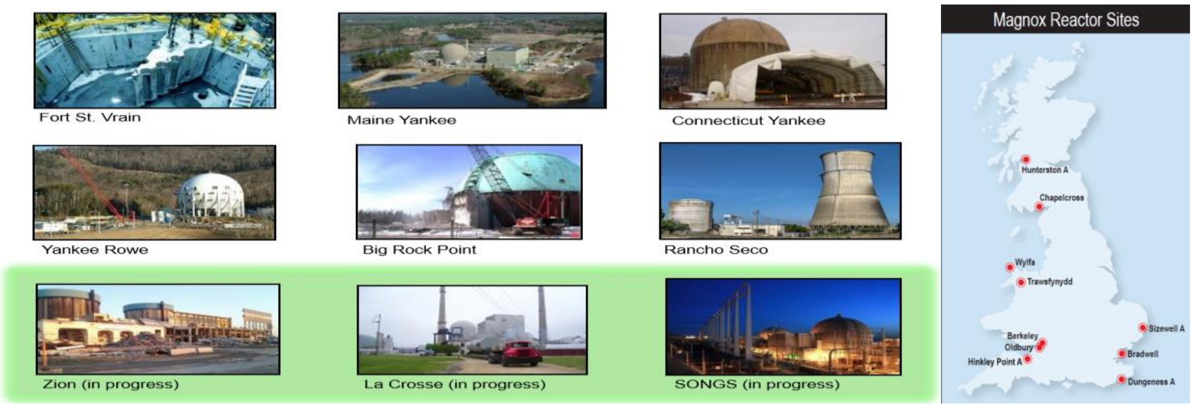
1. Decommissioning Life Cycle Experience

ES was formed 12 years ago and is now made up of over 10 different companies that have been integrated to provide the complete value chain of services necessary to successfully perform NPP decommissioning projects. The scope of work in successful NPP decommissioning that ES has been involved in includes:

- Regulator compliant estimates for funding,
- Detailed decommissioning planning,
- Spent fuel management and storage,
- Decontamination and dismantling,
- Equipment, facility and site characterization,
- Waste management (treatment, disposal packaging, transportation and logistics), and,
- NPP site and waste management facility operating license holder.

This experience allows ES to understand and optimize the NPP decommissioning strategy to ensure projects are performed safely, in full compliance with all regulatory and permitting requirements and at the lowest cost.

2. Extensive Experience in USA and Europe



3. Japanese Experience (Technology Transfer, Services & Equipment)

JAPCO	D&D Collaboration
TEPCO	ALPS Water Treatment
	1F Project Management Support
JNFL(IHI)	Vitrification Technology Support

EnergySolutions' experience is transferable to Japan.

US05

EnergySolutions Waste Management Experience

Colin Austin[1] and Makoto Kikuchi[2]

[1] Energy*Solutions*, [2] MKC Consulting

Abstract

The key to success in decommissioning is the effective management of the radioactive waste that it generates. An effective waste management strategy is focused on the optimum approach to complying with the final approach for dispositioning or disposing of the waste. ES developed a broad range of waste processing (decontamination and volume minimization) techniques to optimize the waste strategy to meet different national and international disposal criteria. Examples of these techniques include:

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.

This experience allows ES to develop optimized waste management strategies for operating NPPs and NPP D&D projects that can be delivered safely, in full compliance with all regulatory and permitting requirements and at the lowest cost.

1. Acquisition and Integration of Companies to Realize full Support Waste Management and D&D

ENVIROCARE of Utah, Inc.
(Bulk Class A LLW disposal/MW, Feb 2005)

SCIENTECH D&D Division
(D&D planning and Characterization, Sept 2005)

BNG AMERICA
(D&D Contractor/Big Rock Point, Feb 2006)

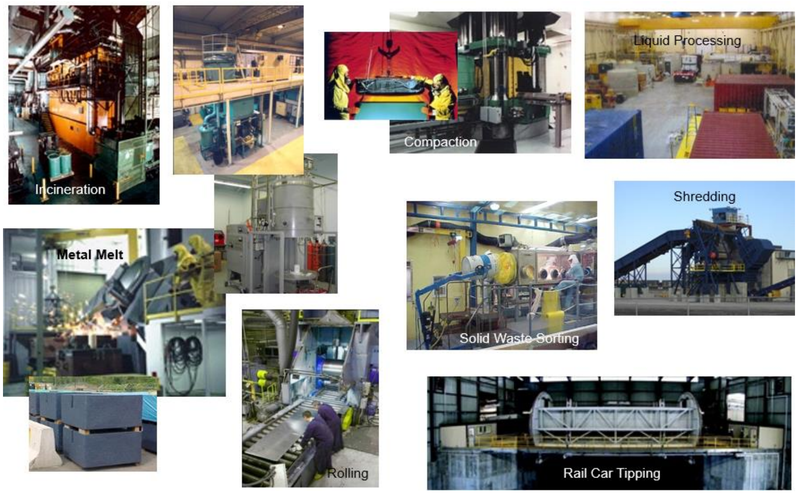
Duratek / HITTMAN TRANSPORT SERVICES, INC.
(Waste processing, large components, D&D logistics; liquid waste; B/C casks; transport services, disposal, June 2006)

NUKEM
(Casks, liquid waste, June 2007)

Studsvik
(Resin processing, March 2014)

MHF
(Rail logistics, containers, fleet management, March 2015)

2. Licensed disposal Facilities and Processing Facilities in US



3. Japanese Experience

J-PARC	2500-ton shield block using recycled metal
TEPCO	Bunker processing equipment
NDF	Waste management guidelines
NFT	Used cask recycle study

EnergySolutions' experience is transferable to Japan.

List of Organizations

Organization	ID No.
ADVAN ENG. Co., Ltd.	H01, H02
AECOM	US03
AIFEN	F08
Akita University	K04
AMRC (Nuclear Advanced Manufacturing Research Centre)	GB01, GB02, GB03, GB04
ANADEC (Orano Atox D&D Solutions Co., Ltd.)	B07
ANDRA	F01
AREVA NC	F01
Atkins	GB01, GB02, GB03, GB04
ATOX Co., Ltd.	B06
AVANTech, Inc.	US01
British Embassy Tokyo	GB01, GB02, GB03, GB04
Cavendish Nuclear Ltd.	GB01, GB02, GB03, GB04
CEA (French Alternative Energies and Atomic Energy Commission)	C07, J05, F01, F02, F03, F04, F05, F06, F07, F08
CHUBU Electric Power Co., Inc.	D05
CORNES Technology Limited	B09
Createc	GB01, GB02, GB03, GB04
CRIEPI (Central Research Institute of Electric Power Industry)	A03, C03, C07, D07
CTI	F06
DAIHATSU Co., Ltd.	H01
Daihatsu Motor Co., Ltd.	H03
DBD (DBD International)	GB01, GB02, GB03, GB04
EBARA Corporation	D01, D02, D03
ECM Technologies	F01
ELyTMax	C07
EnergySolutions	A06, US04, US05
FEVDI	F04
Forschungszentrum Jülich GmbH	H03
Fuji Electric Co., Ltd.	K06
Fujikura Ltd.	B10
Fukushima Consortium of Robotics Research for Decommissioning and Disaster Response	B01, B02

Fukushima Midori Anzen Inc.	B13
Fukushima University	J08
Gleeds	GB01, GB02, GB03, GB04
Goodwin PLC	GB01, GB02, GB03, GB04
Gunma University	C07
Hirosaki University	K04
Hitachi, Ltd.	B08, C01
Hitachi-GE Nuclear Energy, Ltd.	C01
Hosei University	A02
IAE (Institute of Applied Energy)	A03
IHI Corporation	K01, K02, K03
Imagineeing Inc.	J06
INS (International Nuclear Services)	GB01, GB02, GB03, GB04
INSA (National Institute of Applied Sciences)	C07
IPL (Innovative Physics Limited)	GB01, GB02, GB03, GB04
IRID (International Research Institute for Nuclear Decommissioning)	B03, B04, B06, C02, J04, J05, J07
IRSN (Institut de Radioprotection et de Sûreté Nucléaire)	F02
JAEA (Japan Atomic Energy Agency)	A03, C02, C06, H01, H03, J01, J02, J03, J04, J05, J06, J07, J08, J09, J10, J11, J12, J13, J14, J15, J16
James Fisher Nuclear Ltd.	GB01, GB02, GB03, GB04
Japan Radioisotope Association	A03
JAPC (The Japan Atomic Power Company)	A06, K05
JFEE (JFE Engineering Corporation)	K05
JGC Corporation	K07
JNFL (Japan Nuclear Fuel Limited)	K04
Kajima Corporation	E01, E02
KHNP (Korea Hydro & Nuclear Power Co., Ltd.)	K03
KOBELCO STUDSVIK Co., Ltd.	K10
KRI (V.G. Khlopin Radium Institute)	R01, R02
Kurion Japan K.K	B05
Kwansei Gakuin University	H01, H03
Kyoto University	C03, J06, J08
Mirion Technologies (Canberra) K. K.	C04, C05

Mitsubishi Research Institute, Inc.	A04
MKC Consulting	US04, US05
MNF (Mitsubishi Nuclear Fuel Co., Ltd.)	D04
Mott MacDonald	GB01, GB02, GB03, GB04
Nagaoka University of Technology	H01, H02, H03
Nagoya University	C06
National Museum of Nature and Science	J08
NDA (Nuclear Decommissioning Authority)	GB01, GB02, GB03, GB04
NDF (Nuclear Damage Compensation and Decommissioning Facilitation Corporation)	A01, D07
NFD (Nippon Nuclear Fuel Development Co., Ltd.)	C02
NIA (Nuclear Industry Association)	GB01, GB02, GB03, GB04
NINS (National Institutes of Natural Sciences)	J06
NIS Ltd.	GB01, GB02, GB03, GB04
NNL (National Nuclear Laboratory)	GB01, GB02, GB03, GB04
Nuclear Valley	F08
Nucléopolis	F08
NUKEM Technologies Engineering Services GmbH	K08, K09
Nuvia	GB01, GB02, GB03, GB04
Obayashi Corporation	D06
OECD/NEA (Organisation for Economic Co-operation and Development/Nuclear Energy Agency)	ON01, ON02, ON03, ON04
ONET Technologies	F02
ORANO	F04, F06
Osaka Prefecture University	J09
OSMOS Association	C10
OSMOS Group SA.	C10
PCubed	GB01, GB02, GB03, GB04
Pentek	US02
QST (National Institutes for Quantum and Radiological Science and Technology)	J06, J09
RIAR (Research Institute of Atomic Reactors)	R02
RLB (Rider Levett Bucknall)	GB01, GB02, GB03, GB04
Rolls Royce	GB01, GB02, GB03, GB04
RosRAO	R03
Sellafield Ltd.	GB01, GB02, GB03, GB04
Shibaura Institute of Technology	A05

Socium Ltd.	R01
TAISEI CORPORATION	E03
TAIYO KOGYO CORPORATION	E03
TENEX	R02
TEPCO (Tokyo Electric Power Company Holdings, Inc.)	E03, E04, E05
The Graduate School for the Creation of New Photonics Industries	D05
The University of Shiga Prefecture	K04
The University of Tokyo	C06, C08, C09, J09
Tigers Polymer Corporation	B12
Tohoku Institute of Technology	C06
Tohoku University	C06, C07, J09
Tokai University	A02, A05
Tokyo City University	A05
Tokyo Institute of Technology	A05, K04, J09
Tokyo Medical and Dental University	A05
Toshiba Corporation	A02
Toshiba Energy Systems & Solutions Corporation	A03
Toyokoh Co., Ltd.	D05
Toyo Rubber Chip Co, Ltd.	B11
TOYO UNION Company Limited	E03
UKAEA (UK Atomic Energy Authority)	GB01, GB02, GB03, GB04
UNIST (Ulsan National Institute of Science and Technology)	SK01, SK02, SK03, SK04, SK05
Utsunomiya University	H01
Veolia Nuclear Solutions, Inc.	B05, GB01, GB02, GB03, GB04
Wood	D06, K06
Wood PLC	GB01, GB02, GB03, GB04

Nuclear Damage Compensation and
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http://www.dd.ndf.go.jp/eindex.html
July 2018