ABSTRACTS of the Technical Poster Session

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

1FD7

The 7th International Forum on the Decommissioning of the Fukushima Daiichi Nuclear Power Station

Mon, August 28, 2023

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

NDF Nuclear Damage Compensation and Decommissioning Facilitation Corporation (NDF)

ひとつ、ひとつ、実現する ふくしま

Session セッション		
A	Research and Development related to Decommissioning	廃炉関連研究全体
B	Fuel Debris Retrieval Technologies	燃料デブリ取り出し技術
C	State Inside the Reactors and Fuel Debris Properties	炉内状況と燃料デブリ性状
D	Spent Fuel Removal	プール燃料取り出し
E	Radiation Measurement Technologies and Radiation Durability	放射線計測技術と放射線耐性
I	Investigation of Integrity	健全性確認
W	Waste Management	廃棄物対策
H	Analysis	分析
F	France	フランス
GB	Great Britain	英国
J	JAEA	日本原子力研究開発機構
O	Okayama University	岡山大学

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A01 Research and Development of the Project of Decommissioning, Contaminated Water and Treated Water Management and Connection to Engineering

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廃炉・汚染水・処理水対策事業による研究開発とエンジニアリングへの連携

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平成 25 年度より経済産業省は、技術的難易度の高い研究開発の支援を目的とし、廃炉・汚染水・処理水対策基金（旧：廃炉・汚染水対策基金）を設立している。当該基金を活用し、公募による補助事業として「廃炉・汚染水・処理水対策事業」が今日に至るまで実施されており、本講演では、本事業における各種補助事業間の関連性、また本事業の福島第一原子力発電所廃炉への貢献について紹介する。

A02 Organizational Profile of IRID

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技術研究組合　国際廃炉研究開発機構（IRID）の概要

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塚田 明広[2]
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技術研究組合　国際廃炉研究開発機構（IRID）は、2013 年 8 月の設立以来、廃炉技術の基盤強化を視野に、当面の緊急課題である福島第一原子力発電所廃炉作業に必要な研究開発に取り組んできた。ここでは IRID の活動方針、福島第一原子力発電所の廃炉に関する役割分担、研究開発の取り組みについて紹介する。

A03 Overview of IRID R&D Projects

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技術研究組合　国際廃炉研究開発機構（IRID）の研究開発状況

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技術研究組合　国際廃炉研究開発機構（IRID）は、2013 年 8 月の設立以来、廃炉技術の基盤強化を視野に、当面の緊急課題である福島第一原子力発電所廃炉作業に必要な研究開発に取り組んできた。福島第一原子力発電所廃炉の最大の課題は溶融した燃料が冷えて固まった燃料デブリの取り出しである。ここでは、IRID が取り組んできた研究開発及び、現在進めている研究開発について紹介する。

B01 Navigation and Control of a Novel Shock-resistant Mechanical Manipulator for Fuel Debris Retrieval

Shinsuke Nakashima[1]
[1]The University of Tokyo

Ren Komatsu[2], Alessandro Moro[3], Angela Faragasso[2], Hanwool Woo[4], Nobuto Matsuhira[2], Kuniaki Kawabata[5], Hajime Asama[2]
[2]The University of Tokyo, [3]RITECS Inc., [4]Kogakuin University, [5]Japan Atomic Energy Agency (JAEA)

燃料デブリ取り出しのための機械式マニピュレータのナビゲーションおよび制御

中島 慎介[1]
[1]東京大学

小松 廉[2], Alessandro Moro[3], Faragasso Angela[2], 禹 ハンウル[4], 松日楽 信人[2], 川端 邦明[5], 淺間 一[2]
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本研究は、未知環境での衝突対応の為の機械的可変インピーダンスアクチュエータを用いたロボットマニピュレータの開発および効率的な探査・廃止措置のための人工知能を使った制御手法の構築に取り組む。従来調査では困難だった開口部から奥の領域における調査を行う他、先端部のグリッパーで、ペデスタルの底部に存在する小石状の燃料デブリの回収を目指す。ペデスタル内部の環境制約に対応する為のマニピュレータ機構と遠隔操作システムの開発に取り組む。

B02 Work Robot with Flexible Structures

Satoshi Ohno[1]
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Yuya Yoshimura[2], Takahiro Nagai[2], Ren Morinaka[3], Katsuhiko Hirano[2], Koichi Kurosawa[2]
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柔構造作業ロボット

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吉村 祐哉[2], 長井 隆浩[2], 守中 廉[3], 平野 克彦[2], 黒澤 孝一[2]
[2]日立GEニュークリア･エナジー（株）, [3]日立GEニュークリア･エナジー（株）／技術研究組合国際廃炉研究開発機構

柔構造作業ロボットは、福島第一原子力発電所のような高放射線環境下で、人間が行う「細かな作業」（例：機器の組み立て、吊り荷の玉掛作業等）を実現可能な遠隔作業ロボットである。本ロボットは、水圧シリンダを始めとした水圧駆動要素やバネ要素を用いることで、「柔らかい」動作を特徴として有している。また電子部品を使用していないために、高線量下でも長期間の作業が可能となっている。ロボット本体は、水圧駆動で動作し、その制御用の水圧ユニットは比較的良好な環境（原子炉建屋内の低線量エリア等）に設置して使用する。この水圧ユニットは、ロボット本体の構成が変わった場合でも流用が可能である。柔構造作業ロボットにおいては、水圧駆動の各関節をモジュール単位で組み合わせてロボット本体を構成するため、短期間で適用環境に合わせた形態での現地導入が可能である。本稿では、この柔構造作業ロボットの特徴や適用例について紹介する。

C01 **Remote Control Technology for Monitoring Inside RPV Pedestal during Retrieval of Fuel Debris: Prototype Experiments**

Nobuto Matsuhira[1]
[1]The University of Tokyo

Ren Komatsu[2], Shinsuke Nakashima[2], Atsushi Yamashita[2], Rui Fukui[2], Hiroyuki Takahashi[2], Kenji Shimazoe[2], Takayuki Takahashi[3], Yasuyoshi Yokokohji[4], Kuniaki Kawabata[5], Hajime Asama[2]
[2]The University of Tokyo, [3]Fukushima University, [4]Kobe University, [5]Japan Atomic Energy Agency (JAEA)

燃料デブリ取り出し時における炉内状況把握のための遠隔技術：プロトタイプ実験

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[1]東京大学

小松 廉[2], 中島 慎介[2], 山下 淳[2], 福井 類[2], 髙橋 浩之[2], 島添 健次[2], 高橋 隆行[3], 横小路 泰義[4], 川端 邦明[5], 淺間 一[2]
[2]東京大学, [3]福島大学, [4]神戸大学, [5]日本原子力研究開発機構

本研究では、福島第一原発の廃炉に向けて、遠隔技術分野を中心とした研究人材の育成を行う。燃料デブリ取り出し時における炉内状況把握のためのモニタリングプラットフォームの構築、およびプラットフォーム上を移動するセンサによる計測・可視化についての研究を行う。このような研究課題に参画することによる研究教育、講義等の座学、施設見学の３つの柱で研究人材を育成することを目的とする。本稿ではプロトタイプ実験について述べる。

C02 **Microbial recover of nuclear fuel debris components**

Toshihiko Ohnuki[1]
[1]Tokyo Institute of Technology

微生物機能を活用した燃料デブリ組成の溶解分離回収

大貫 敏彦[1]
[1]東京工業大学

損傷した福島第一原子力発電所の廃炉作業において、回収されたデブリの処理は、廃棄物の保管と処分のために重要である。デブリ成分の分離溶解は、デブリの処理を簡素化できる。そのため、デブリ成分の微生物処理を提案する。シデロフォア放出微生物（SB）の燃料デブリ類似物に対する溶解を調べた。CeO_2 または UO_2-ZrO_2 固溶体と金属鉄を含む燃料デブリ類似物（FDA）を寒天培地あるいは液体培地中で接触させた。溶液中の溶存元素の経時変化や固形分の SEM-EDX 分析から、SB は微生物細胞を含まない場合よりも多量の Fe と UO_2 が溶存していることがわかった。一方、Zr 酸化物は溶解しなかった。これらの結果は、SB が燃料デブリ中の Fe と UO_2 の溶解を選択的に促進することを示している。回収された燃料デブリの分離処理への微生物活性の適用性について議論する。

C03 **Preliminary verification of criticality impact analysis code applicable to weakly coupled systems including fuel debris particles**

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Toru Obara[2]
[2]Tokyo Institute of Technology

燃料デブリ多粒子体系を含む弱結合炉に適用可能な臨界影響解析コードの予備検証

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小原 徹[2]
[2]東京工業大学

福島第一原子力発電所の燃料デブリ取出し作業を対象とした臨界影響解析は、現場作業員の安全確保方策を検討・確立するために重要である。燃料デブリは損傷した原子炉格納容器内部に広く分布していることが予想される。また、燃料デブリの形状として粒子状または塊状が想定される。これらの燃料デブリの特徴は、通常の原子炉とは大きく異なっている。このような背景のもと、本研究は遅発中性子による核分裂にも対応した多領域積分型動特性解析コード MIK2.0 を開発している。MIK コードは積分型動特性解析モデルに基づいているため、複数の核燃料が互いに離れて分布している弱結合炉体系や、上記のような特徴を有する燃料デブリ体系にも適用することができる。今回の開発では、動く燃料デブリ多粒子体系を含む弱結合炉体系が解析対象となる可能性も考慮している。MIK2.0 コードの予備検証として、GODIVA 炉の超臨界実験を対象とした再現解析の進捗について報告する。

C04 Progress and Prospects of the Internal Investigation of Unit 1 Primary Containment Vessel

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1 号機 PCV 内部調査の進捗と展望について

中島 悟[1]
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福島第一原子力発電所の 1～3 号機は、2011 年に発生した事故の影響により、燃料と炉内構造物を含んだ燃料デブリが PCV 内に溶け落ちている。事故後、燃料デブリの特定等を目的に PCV 内部調査を実施しており、最近では 2022～2023 年に 1 号機の地下階について、水中 ROV を用いた調査を実施した。調査の結果、PCV 地下階の南側およびペデスタル内の情報を取得し、堆積物や既設構造物の状態について確認した。堆積物はペデスタル内だけでなく、S/C に繋がるジェットデフレクターの裏側にも確認されており、ベント管内に流出している可能性がある事が判明した。また、事故当時、1 号機は他号機よりも高温状態が続いていたと推定されており、調査結果からも、ペデスタル内の 1 部の構造物が消失している事、ペデスタル基部の配筋が露出している事が確認された。今後は、ベント管への堆積物の流出状況の確認や、燃料デブリのサンプリングについて計画していく。

D01 Engineering efforts for spent fuels removal from Fukushima Daiichi nuclear power station unit 1

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[1]Kajima Corporation

Akio Hirata[2], Tomochika Gyobu[2], Takamasa Nishioka[2], Akira Takahashi[2], Ryota Mizutani[2]
[2]Kajima Corporation

福島第一原子力発電所 1 号機　プール燃料取出し工事における設計面および施工面での取組み

豊島 憲治[1]
[1]鹿島建設株式会社

平田 明生[2], 刑部 知周[2], 西岡 聖雅[2], 高橋 彬[2], 水谷 亮太[2]
[2]鹿島建設株式会社

福島第一原子力発電所 1 号機のプール燃料取出しにおいて、サイト周辺の帰還住人に配慮した放射性物質の拡散抑制および現地作業者の被ばく線量低減を目的に、当社が採用した設計面および施工面での取組みについて紹介する。

E01 Development of Optical Materials and Its Evaluation for the Radiation Dose-Rate Monitor under Ultra-High Dose-Rate

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[1]Tohoku University/Osaka University

Chihaya Fujiwara[2], Shohei Kodama[3], Daisuke Matsukura[4], Ai Kaminaka[4], Maki Ohno[4], Takushi Takata[5], Hiroki Tanaka[5], Akihiro Yamaji[4]
[2]TIRI, [3]Saitama University, [4]Tohoku University, [5]Kyoto University

高線量率場でのリアルタイムの線量計測に向けた材料研究とその評価

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藤原 千隼[2], 小玉 翔平[3], 松倉 大輔[4], 上中 あい[4], 大野 真紀[4] 髙田 卓志[5], 田中 浩基[5], 山路 晃広[4]
[2]東京都立産業技術研究センター, [3]埼玉大学, [4]東北大学, [5]京都大学

高線量率場でのリアルタイムの線量計測に向けた赤色・近赤外発光シンチレータの開発を行い、これらを線量計に搭載させたときの線量応答を実施したので、これらを紹介する。

E02 Neutron detection system without radiation protection for criticality approach monitoring based on diamond sensors and radiation-resistive integrated-circuits

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Junichi Kaneko[2], Hitoshi Umezawa[3], Tomohiro Endo[4], Yoshihiko Tanimura[5], Kenichi Watanabe[6], Masaya Miyahara[7], Tetsuichi Kishishita[7], Takahiro Shimaoka[3], Masayoshi Shoji[7], Kengo Oda[2]
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遮蔽不要な臨界近接監視システム用ダイヤモンド中性子検出器の開発

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金子 純一[2], 梅沢 仁[3], 遠藤 知弘[4], 谷村 嘉彦[5], 渡辺 健一[6], 宮原 正也[7], 岸下 徹一[7], 嶋岡 毅紘[3], 庄子 正剛[7], 織田 堅吾[2]
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1F の炉内状況の把握、燃料デブリ取出しの早期実現、臨界リスク管理に資することを目的とし、遮蔽不要な臨界近接監視システム用中性子検出器の要素技術を開発した。中性子検出器は、軽量かつ最大 1kGy/h の高γ線環境下で数 cps/nv の高い中性子検出感度を満たす仕様とし、これを実現するために炉雑音解析法を使用し数値仕様を決定したのち、ダイヤモンド中性子検出素子と耐放射線性集積回路技術を応用した信号処理・データ転送用集積回路群を試作し、1 kGy/h での安定動作と中性子感度の目標値達成を確認した。更に 6cm 径ドライチューブ挿入可能な実機モックアップを開発し、実機開発に必要なデータを取得した。並行して、ダイヤモンド検出素子量産化に向けたベンチャー企業を立ち上げた。また実機の使用を想定した臨界近接評価手法の検討も並行して行い、体系情報が不明かつ低い中性子計数率でも適用可能な逆動特性法を開発した。

E03 The World's Highest Radiation Tolerant Performance Camera

Mikio Katsura[1]
[1]CORNES Technologies Ltd.

Hazuki Nakazawa[2], Yudai Suzuki[2], Naoki Kajihara[2]
[2]CORNES Technologies Ltd.

世界中の原子力施設で活躍する耐放射線カメラシステム

桂 幹夫[1]
[1]コーンズテクノロジー株式会社

中澤 葉月[2], 鈴木 雄大[2], 梶原 尚樹[2]
[2]コーンズテクノロジー株式会社

コーンズテクノロジー株式会社は、原子力施設や研究所に耐放射線性カメラを供給する英国 Mirion Technologies（IST）社の国内代理店です。Mirion 社は 40 年近くにわたり耐放射線カメラの供給と技術開発を続けており、原子力施設内の監視用途（CCTV）や炉内調査・メンテナンス、ロボットやアーム装置の操作や監視等、世界中の原子力施設で幅広く活躍する耐放射線性カメラをご提供しております。
最近では CMOS 技術を採用した高い耐放射線性（1MGy 対応）を持つカラーシステムカメラも開発され、線量が高く厳しい環境下でもカラー映像の取得が可能になりました。小型軽量のカラーカメラやパンチルト機構や照明オプションを持つシステムカメラ等、ユーザ様のご要望を満足する幅広いカメララインナップをご紹介しております。

E04 The world's highest radiation-resistant lubricants supporting decommissioning of the nuclear reactor

Yoshikazu Hayashi[1]
[1]MORESCO Corporation

廃炉を支える世界最高水準の耐放射線性潤滑剤

林 義和[1]
[1]株式会社MORESCO

世界最高峰の耐放射線性潤滑剤「MORESCO-HIRAD」は、廃炉、原発、加速器、放射線医療、そして来る核融合などの用途にまで、世界で広く搭載が進んでいる。
廃炉では、現在の試験的デブリ取り出し用機器への搭載に留まらず、次世代のデブリ拡大取り出し用機器にも拡大しつつある。我々は、廃炉を含む過酷な放射線下の機器の安定的な稼働に貢献するべく、その潤滑剤（オイルやグリース）のコアな学術的研究を、欧州や日本の主要学術機関と進め続けている。これらは、既存構造の潤滑剤への中性子やγ線などの照射評価を通じた放射線ダメージやその機構解明に加え、更に高度な耐放射線性潤滑剤の候補品となる新規構造体の独自合成・創出、それらへの評価拡大などにも及ぶ。こうして示された、既存構造潤滑剤の圧倒的な耐放射線性データや、その特性を活かした廃炉への展開実例、更に既存品と新規構造体の分子構造図の一例などを詳細に示した発表を行う。

E05 Development of Gamma and Neutron Dosimeters Based on Solar Cells

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[2]RIKEN, [3]Sanjo City University, [4]National Institute of Advanced Industrial Science and Technology (AIST), [5]Tohoku University, [6]Japan Aerospace Exploration Agency (JAXA), [7]National Institute of Technology, Kisarazu College (NITKC)

太陽電池を応用したガンマ線・中性子線線量計の開発

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福島第一原子力発電所の原子炉格納容器（PCV）から燃料デブリを取り出し、1F の廃炉を効率的に進めるためには、PCV 内の放射線源および線量率の分布を知り、適切な除染や放射線遮蔽措置を行って、作業者の安全性を確保するとともに、廃炉に使用する装置類の耐放射線性等の最適化をする必要がある。太陽電池型線量計は、宇宙用太陽電池として開発されてきた高放射線耐性を有する半導体素子を利用した自立駆動形の省電力・小型センサーとして開発を進めており、フレキシブルシート化、多接続化マッピングモニタリングシステムおよびガンマ線・中性子検出構造など多彩なアプリケーションの構成が可能である。本発表では、太陽電池による放射線検出システムについて解説するとともに、PCV や建屋内部における適応可能性について報告する。

I01 Establishment of 3-D dose dispersion forecasting method and development of in-structure survey using the transparency difference of each line gamma-ray

Shinya Sonoda[1]
[1]Kyoto University

Atsushi Takada[2], Tohru Tanimori[2], Minoru Tanigaki[3], Akihiro Taniguchi[3], Haruyasu Nagai[4], Hiromasa Nakayama[4], Tetsuya Mizumoto[5], Shotaro Komura[5]
[2]Kyoto University, [3]KURNS, [4]Japan Atomic Energy Agency (JAEA), [5]FSiC Inc.

3 次元線量拡散予測法の確立とγ線透過率差を利用した構造体内調査法の開発

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[1]京都大学

高田 淳史[2], 谷森 達[2], 谷垣 実[3], 谷口 秋洋[3], 永井 晴康[4], 中山 浩成[4], 水本 哲矢[5], 古村 翔太郎[5]
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核ガンマ線の方向を完全に決定、ガンマ線全単射画像が測定できる電子飛跡検出型コンプトンカメラを実現した。JAEA が開発した WSPEEDI と組み合わせてガンマ線画像モニタリングに基づく拡散予想システムが可能であることを実証した。これまでの成果に基づき、1F 内のサブ mSv/h 環境下で使用可能な 3 次元汚染物質飛散検知・予測システムの実用化を行う。また、134-Cs ガンマ線を利用した建屋内の 3 次元 Cs 分布測定法の開発、1.5MeV 以上のデブリガンマ線探査を行いデブリの全容把握を目指す。ETCC で使用する MPPC は増幅率の温度依存性のため 1F 内のように温度が不安定な場所でも検出器温度を一定に保つことが課題であるが、温度制御機能を組み込み安定化させることに成功した。また、高線量環境下での偶発事象の低減のためシンチレータを厚さ 1cm の鉛板で遮蔽し、JAEA が所有する放射線標準施設棟の校正用線源を用いて 1F 内の高線量場に近い環境で動作試験を実施した。

W01 Quantitative Evaluation of Long-Term State Changes of Concrete Contaminated with Radioactive Nuclides Considering Actual Ageing and Deteriorating Factors

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Kazuo Yamada[2], Kazutoshi Shibuya[3], Yoshifumi Hosokawa[4], Yasumasa Tojo[5], Go Igarashi[6], Yo Hibino[6], Yoshikazu Koma[7]
[2]National Institute for Environmental Studies, [3]Taiheiyo Consultant Co. Ltd., [4]Taiheiyo Cement Corp., [5]Hokkaido University, [6]Nagoya University, [7]Japan Atomic Energy Agency (JAEA)

実際の経年劣化要因を考慮した放射性核種汚染コンクリートの長期状態変化の定量評価

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福島第一原子力発電所におけるコンクリート構造物の廃止措置では、放射性核種で汚染されたコンクリートの量や濃度を推定することが重要である。本研究では、コンクリート中の汚染濃度分布を定量的に予測するために、実環境を考慮した浸透実験に基づく放射性核種の浸透挙動予測手法を検討した。

W02 Challenge of Novel Hybrid-waste-solidification of Mobile Nuclei Generated in Fukushima Nuclear Power Station and Establishment of Rational Disposal Concept and its Safety Assessment - Summary of the 3-year-project and toward the next step

Masahiko Nakase[1]
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Ryosuke Maki[2], Hidetoshi Kikunaga[3], Tomofumi Sakuragi[5], Miki Harigai[5], Hidekazu Asano[5], Tohru Kobayashi[4], Satofumi Maruyama[6]
[2]Okayama University of Science, [3]Tohoku University, [4]Japan Atomic Energy Agency (JAEA), [5]Radioactive Waste Management Funding and Research Center, [6]Tokyo City University

福島原子力発電所事故由来の難固定核種の新規ハイブリッド固化への挑戦と合理的な処分概念の構築・安全評価 - 3 年間のプロジェクト成果と次の展開について

中瀬 正彦[1]
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牧 涼介[2], 菊永 英寿[3], 桜木 智史[5], 針貝 美樹[5], 朝野 英一[5], 小林 徹[4], 丸山 恵史[6]
[2]岡山理科大学, [3]東北大学, [4]日本原子力研究開発機構, [5]原子力環境整備促進・資金管理センター, [6]東京都市大学

令和 3 年度に採択された英知を結集した原子力科学技術・人材育成推進事業における廃棄物研究が研究が大きく進捗している。1F で発生する多様な廃棄物について、その処分まで結節した検討が必要な段階に達しつつある。そこで我々は長期安定性が良く研究され、既に安全評価実績のある Zircaloy や SUS に廃棄物（1 次固化体）を単分散させマトリクスの腐食速度により廃棄体寿命を設定することで、多様な廃棄物を単一の概念で安全評価までを可能とする考え方、「ハイブリッド固化体」の概念を提唱した。その成立性について、多様な実験、計算機科学、処分概念設計、安全評価により検討を進めた。系統的な検討の結果、例えば難固定性かつ長期被爆の主因となるヨウ素廃棄物においては SUS マトリクスが適していること等を明らかにした。多様な実験、計算、評価結果を総括するとともに、今後の展開についてまで展望する。

W03 Application for removal of radionuclide in radioactive wastewater by the natural zeolite and reused of used diapers

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Shin-Chin Tsai[2,3], Pei-Tung Hsueh[4], Neng-Chuan Tien[2,3], Ren-Yu Pan[2]
[2]Radioactive Waste Disposal Technology Research and Development Center, National Tsing Hua University, Hsinchu 30044, Taiwan (R.O.C), [3]Nuclear Science and Technology Development Center, National Tsing Hua University, Hsinchu 30044, Taiwan (R.O.C), [4]Institute Nuclear Science and Technology Development Center, National Tsing Hua University, Hsinchu 30044, Taiwan (R.O.C)

Zeolite, a natural mineral ($Na_8\ (AlO_2)_8(SiO_2)_{40}\cdot 20H_2O$), and used diapers including fine fibers having high selectivity toward Cs and Co for removal radioactive water from decommissioning NPP in Taiwan. The sorption of Cs and Co were investigated by ASTM batch methods and distribution coefficients (Kd) were

obtained. In conclusion, there is a high effective and good performance (fibers showed high volume reduction) for decommissioning NPP by applying natural zeolite and used diapers to achieve UN Sustainable Development Goals (SDGs) to replace a lot of chemical resin and solvent.

W04 Waste Management Symposia: The Annual Phoenix Conference Exchanging Knowledge from Around the World

Kazuhiro Suzuki[1]

[1]Waste Management Symposia, Inc.

Gary Benda[2], Takashi Mitsui[3]

[2]Waste Management Symposia, Inc., [3]Tousou Mirai Technology Co.Ltd.

WM2024：世界中からの知識を交換する年次フェニックス会議

鈴木 一弘[1]

[1]Waste Management Symposia, Inc.

Gary Benda[2], 三井 崇[3]

[2]Waste Management Symposia, Inc., [3]東双みらいテクノロジー株式会社

WM Symposia が主催する WM 国際会議は、毎年 3 月上旬にアリゾナ州フェニックス市で開催し、第 50 回となる WM2024 は、3 月 10 日（日）～14 日（木）に開催します。世界中の政府機関、産業界、研究機関、学界、地方自治体、そして国際機関等から 2,500 名以上の参加が見込まれ、プロジェクト、地域対応、技術、人材開発などの幅広い分野の計画から実施経験にいたる最新の情報の共有とネットワーク作りの場です。放射性廃棄物と関連するタイムリーなバックエンド課題に焦点を当て、福島第一廃炉も取上げています。WM2024 では、ロボット・遠隔操作に加え、三次元シミュレーション、AI 技術、革新炉のバックエンド等も取り上げます。会議は、DOE 長官がメッセージを寄せるプレナリーセッション、技術セッション、展示会場で構成されます。人材開発支援として、学生向けプログラム、STEM 教育支援ワークショップも用意します。

W05 Development of Metal Matrix Waste form for Immobilization of Spent ALPS Adsorbents with Powder Metallurgy Hot Isostatic Pressing.

Tomofumi Sakuragi[1]

[1]Radioactive Waste Management Funding and Research Center

ALPS ヨウ素吸着材の金属マトリクス固化による安定固化技術の開発

桜木 智史[1]

[1]原子力環境整備促進･資金管理センター

多核種除去設備（ALPS）で汚染水の処理として用いられた吸着材の処理・処分が課題となる。特に、放射性ヨウ素（I-129）は、高温での安定固化処理が難しく、長半減期で低収着性のため処分後の安全性（被ばく線量）に懸念がある。そこで、廃棄物を耐食性に優れた金属マトリクスで固化するハイブリッド固化体を提案している。熱間等方圧加圧法（HIP）により焼結固化することで、金属マトリクスは緻密化・インゴット化し、さらにヨウ素の揮発を防止することができる。
本研究では、ALPS 吸着材（酸化セリウム及び銀ゼオライト）をステンレス鋼（SUS）で HIP 固化した。固化体中の ALPS 吸着材は均質に分散し、インゴット化した SUS マトリクスに確固に閉じ込められた。また、腐食速度に基づいた解析から十分な固化体寿命や処分後の長期安全性が見込めることから、ハイブリッド HIP 固化の技術適用性に見通しを得ることができた。

W06 SIAL®:Geopolymer solidification technology approved by Slovak / Czech Nuclear Authority

Hisashi Mikami[1]

[1]Fuji Electric Co.,Ltd.

Milena Prazska[2], Marcela Brazsekova[2], Tatsumi Kurogi[3], Nobuyuki Sekine[3]

[2]Jacobs (Slovakia), [3]Fuji Electric Co.,Ltd.

スロバキアとチェコの規制当局から認可されているSIAL® ジオポリマー固型化技術

見上 寿[1]
[1]富士電機株式会社
ミレーナ プラツカ[2], マルセラ ブラツエコワ[2], 黒木 竜海[2], 関根 伸行[2]
[2]Jacobs (Slovkia), [3]富士電機株式会社

スロバキアとチェコの規制当局から認可されているSIAL® ジオポリマー固型化技術の特徴と性能の実績を紹介します。この技術は約20年に亘り運用されています。
近年、ジオポリマーは、温暖化ガス削減に効果のある固型化剤であるとともに、核種閉じ込め性能や耐熱・耐酸性能に着目され、セメントでは固型化困難な廃棄物への適用がEU PREDIS Pj等で検討されています。この特性は、福島第一原子力発電所の汚染水二次廃棄物に対しても、スラッジや各種スラリー等の模擬廃棄物固型化試験で示されています。
また、最近では、安全なデブリ取り出しを目的に、炉内のデブリの安定化のため温水中での固型化、高温表面への接触環境での固型化維持、残存炉内構造物への充填性等の検討を実施しています。

W07 Jacobs CMS Decommissioning & Regenerative Solutions (D&RS) Products & Services

Toshi Yamanouchi[1]
[1]Jacobs CMS
Satoshi Nakasaku[2]
[2]Jacobs CMS

ジェイコブス　廃止措置及び地域再生ソリューションにおける製品およびサービス

山之内 壽彦[1]
[1]ジェイコブス　CMS
中作 敏之[2]
[2]ジェイコブス　CMS

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ConnectFlow
亀裂や多孔質岩石を通る流体や溶質の移動をシミュレートするためのソフトウェアスイートであり、構造地質学、エンジニアリング、ハイドロジオロジー、水文地球化学、汚染物質輸送プロセスの統合評価を可能にします。
ANSWERS
主に原子炉物理学、放射線遮蔽と線量測定、臨界度などの分野を対象とした、世界クラスの原子炉および放射線モデリングおよび分析を行うためのソフトウェア及びコンサルテーションサービスです。
TANICS
高活性廃棄物容器シーリング用自動溶接および検査ソリューションです。原子力産業用途向けに特別に設計されており、ステンレス鋼の放射性物質容器を遠隔で密閉できる、再現性の高い整合性の高い突合せ溶接を実現します。

H01 Material Identification Method using Multivariate Analysis for Chemical Analysis and Application Possibility to Fragment Ion of Fuel Debris Dissolved in Wastewater

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[1]SSS. Fukushima University
Makoto Furukawa[2], Katsushige Fujimoto[3], Yoshitaka Takagai[4]
[2]SSS. Fukushima University / PerkinElmer G.K., [3]SSS. Fukushima University, [4]SSS. Fukushima University / IER. Fukushima University

化学分析における多変量解析を用いた材料特定手法と汚染水に溶出した燃料デブリ由来のフラグメントイオンへの適用における可能性

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古川 真[2], 藤本 勝成[3], 高貝 慶隆[4]
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燃料デブリのように複数の材料の溶融物の材料組成を特定することは困難である。従来技術では、複合材料の化学分析において、定量値と別な材料の認証値が一致した場合にその識別をすることはできない。本研究では試料に含まれる元の材料を識別するため、不完全に溶解させた溶解液を ICP-MS 計測することで、それぞれの材料に含まれる元素のわずかな溶解度の差異を利用し、多変量解析に導入することで元材料の特定を行った。不完全な溶解を行うことで、優先的に溶解した成分が元材料を特徴づけたデータとして得られた。このデータを多変量解析フローに適用した。その結果、その不完全溶出の最適化を検討し、溶解成分から元の材料を廃液の化学成分データから特定することができる方法を開発した。

H02 Development of ICP-MS analytical method for rapid analysis of radioactive ^{94}Nb in waste

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廃棄物中の放射性ニオブの迅速分析を目的とする ICP-MS 分析法の開発

長沼 和希[1]
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松枝 誠[2], 柳澤 華代[3], 及川 博史[4], 橋本 淳一[4], 高貝 慶隆[5]
[2]日本原子力研究開発機構, [3]福島大学, [4]ジーエルサイエンス, [5]福島大学

放射性ニオブ（Nb-94）は、β線およびγ線を放出する長半減期（2.03 × 10^4 年）である。東京電力福島第一原子力発電所における放射性廃棄物の長期保管において、長半減期核種である 94Nb が支配的な核種の一つになることから、廃棄物中の 94Nb の分析が非常に重要になる。誘導結合プラズマ質量分析法（ICP-MS）は、長半減期核種であるため放射能計測法と比較すると高感度に測定することができる。しかしながら、同重体である Zr-94（天然同位体比 17.4%）および Mo-94（天然同位体比 9.2%）の存在が ICP-MS 測定を妨害する。本研究では、シリカゲルによる固相抽出と ICP-MS 内でのダイナミックリアクションセル（DRC）による同重体の分離を行い、カスケード型によるオンライン分析を実施した。

F01 Methodology to address the liquid radionuclides source term from corium leaching after severe accident.

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The quantification of radionuclides release into the water for Severe Accident Management (SAM) is essential for carrying out safety analyses. These aim to assess the impact of their releases on the environment at short, mid and long term. This work presents the methodology developed at CEA, based on the synthesis, characterization and leaching of prototypical corium materials, i.e. with UO_2, in order to quantify the liquid source term from corium leaching after a Severe Accident (SA). The PLINIUS platform (CEA-IRESNE) at Cadarache site is dedicated to SA studies and is composed of five facilities (KROTOS, VULCANO, VITI, MERELAVA and FUJISAN). This platform allow to perform various studies such as the corium thermophysical properties measurements, interactions between the fuel and the coolant or corium spreading on concrete. In addition, the PLINIUS facilities can perform tests simulating different SA conditions using prototypical corium. Previous tests on COLIMA facility (simulating in-vessel and ex-vessel corium) have been used for leaching studies under irradiation in the ATALANTE facility (CEA-ISEC) at Marcoule site as part of a collaboration with JAEA. A leaching methodology initially developed and proven for conventional spent fuels has been successfully implemented for these prototypical corium samples. The radionuclides released fractions were determined and the surfaces of the samples characterized after alteration. Geochemical modeling was also performed to interpret the results. The Fractional Release Rates of the (U, Zr)O_2 phase for these samples were found to be close to or one order of magnitude lower than that of conventional spent fuels leached under oxidizing conditions, but the radionuclides release mechanisms seem different. Further studies are needed to better understand the differences with spent fuel and get closer to more relevant accident scenarios. A methodology for studying the interactions between prototypical corium and water is now available at CEA. It relies on the synthesis of samples at Plinius platform and peforming leaching experiments at ATALANTE facility.

F02 Development of Innovative Plasma Treatment Processes for the Management of Radioactive Organic Liquid Solid Waste

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The purpose of this poster is to present the results of the bench development of two complementary processes for the mineralization of radioactive organic liquid wastes using plasma thermal treatment. The operation of nuclear industry facilities and the dismantling sites of end-of-life facilities produce large quantities of radioactive wastes. Their management constitutes a major economic, environmental, societal and industrial challenge. While a large number of these wastes have a management route in existing or future disposal facility, some give rise to complex scientific and technical issues. This is the case for several types of radioactive organic liquid waste (ROLW) whose either chemical or radiological characteristics are not compatible with the incinerator operating rules (high level of 14C and 3H or halogen content, corrosive gas production etc.). Therefore, their treatment process remains to be defined in order to make them compatible with existing or future waste management coordination. Their volume declared through the French national inventory of radioactive waste is a few hundred cubic meters and consists of oils, various organic liquids and scintillating cocktails. In order to optimize decommissioning waste management and better anticipate related issues, French nuclear waste producers CEA and the French national radioactive waste management agency (Andra) have decided to develop innovative thermal treatment processes based on plasma technologies to operate these wastes for disposal. Thermal

treatments can provide interesting benefits such as volume reduction or chemical stabilization, helping waste producer to master costs, schedule, or to optimize final volume and repository safety.
The project was supported by the French government program “Programme d’Investissements d’Avenir” whose management has been entrusted to Andra. The two complementary processes assessed are using plasma technology: the IDOHL process (French acronym for “Installation de Destruction d’OrganoHalogenes Liquides”), an aerial induction plasma treatment process, and the ELIPSE process (French acronym for “Elimination des Liquides par Plasma sous Eau”), a very innovative process implementing a submerged non-transferred arc plasma.

F03 French laser - cutting R&D facilities in support of dismantling

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For more than 30 years, CEA has been developing laser–cutting technology. It designs and industrializes laser tool prototypes, integrates and implements the technology in various remote handling equipment and carries out safety and scenarios studies for nuclear dismantling application. As a result of its R&D programs, the technology has been qualified to cut up to 200mm thick stainless steel, and air–cooled and deep–gouging laser tools have been developed to tackle the challenges of 1F dismantling. Several facilities dedicated to laser–cutting R&D studies are available in France and presented in this paper. CELENA and DELIA facilities, located at CEA Saclay are dedicated to in air, nitrogen atmosphere and underwater experimentations as well as aerosols sampling and monitoring, while Hera facility at CEA Marcoule is used for advanced robotic decommissioning application, digital twin development and dismantling scenarios assessment using VR simulation in the PRES@GE[2] immersive room. Since 2014, many decommissioning studies for Fukushima Daiichi have been performed in these facilities. More recently, ONET Technologies started commissioning the TECHNOCENTRE, a new facility for industrial scale studies of in air and underwater laser–cutting application with representative test configurations. It will be inaugurated at the end of 2023.

F04 Development of Control Rod Drive Housing Cutting and Removal Technologies

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The aim of the project was to demonstrate the existence of a solution to safely and efficiently remove the Control Rod Drive Housing of the damaged Units of the Fukushima Daiichi NPS. This has been achieved through selection and testing of the best candidate technologies and the proposal of a concept of dismantling scenario considering the safety requirements, including falling objects prevention.

F05 Fuel Debris Dust Dispersion Suppression System Technologies

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This study has demonstrated that systems can be implemented inside the pedestals of the damaged units of Fukushima Daiichi NPS to suppress the Fuel Debris dust dispersion risk. Application of coatings with remote controlled means and consideration for different aeras (metallic frames, under dripping water, underwater, etc.), has been proven to be efficient. Impacts of use has been assessed.

F06 Solutions for D&D Legacy Waste Stabilization: Dem&Melt, Cementation, Solidification with Resins

Daphne Ogawa[1]
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As the D&D of UP2-400 progresses, various types of Legacy Waste are retrieved for treatment in existing or newly built workshops. Among the retrieved waste are considered: sludge with widely varying chemical properties, very different types of solid waste that must be immobilized, and many others.
Based on its strong experience in waste treatment, Orano has already developed, or is currently developing and qualifying different flexible stabilization processes, many of which could be applied to waste in 1F.
Three of those processes are described : Dem&Melt, Cementation, and Solidification with Resins.

F07 Reactor D&D: Recent successes and achievements in the U.S.

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With over 10 projects to date between the U.S. and Germany, Orano has gained a comprehensive expertise in successfully dismantling the full spectrum of BWR and PWR reactors, including the segmentation, packaging, transportation, and disposal of the Reactor Vessel (RV) and Reactor Vessel Internals (RVI). In the US, Orano recently completed the first full segmentation of a commercial BWR (Vermont Yankee) and is currently dismantling the Crystal River 3 PWR implementing an innovative segmentation solution. This experience could be relevant for upcoming D&D operations in Fukushima-Daiichi.

F08 Innovative solutions for Full-scale Fuel Debris Retrieval (FFDR): New Lateral Opening for RPV access and hot cells conceptual design for Full-Scale FDR through PCV side access

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To pursue the full-fledged (or full-scale) fuel debris retrieval at Fukushima Daiichi, several methods to access RPV are under development. On the one hand, a lateral opening toward the Reactor Pressure Vessel is made possible by integrating water jet cutting and demolition tools combined with a ferro-concrete debris suction system. On the other hand, hot cells conceptual design for Full-Scale FDR through PCV side access is completed. Both solutions are designed for 1F needs based on French technology.

F09 Anemone retrieval tool -An innovative gripping technology developed for the recovery of fuel debris in 1F reactors

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Orano develops various tools for investigation and sampling, designed to be remotely operated and maintained in harsh environment, for its needs and for its customers'. The Anemone tool is the new generation of these tools.
The Anemone tool is inspired from the sea anemone behavior. It can catch, imprison, and recover any kind of solid element no matter its shape, material, or density.
-Anemone tool has been used for the first time in nuclear industry on Orano La Hague plant to collect highly radiating graphite samples, in April 2022.
-Anemone tool has been selected to recover 1F reactors fuel debris. A specific development for this application is in progress.

F10 Managing large scale retrieval and decommissioning programs as a Nuclear Operator and Site Owner

Daphne Ogawa[1]
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Jean Michel Chabeuf[2]
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Very few nuclear operators face the triple challenge of managing high active nuclear material retrieval and full-scale decommissioning on constrained nuclear sites. TEPCO is one, Orano is another one.

Managing such an endeavour requires to establish and reassess constantly a global strategy which combines and balances multiple factors which can be summarized as follows:
- Hazard reduction and site Safety in normal and abnormal conditions
- Funding mechanisms and best uses of available funds
- Technology development
- Waste and effluent management
- Internal and external resources

- Stakeholder management

Weaknesses or failures in any of those factors inevitably impacts the program delivery and requires a re-evaluation of the overall strategy.

Orano has been managing such program for nearly two decades now, most notably on the site of La Hague in North-Western France, and has thus, over the years, developed specific processes, mechanisms, and competences to ensure the individual and collective progress of these factors.

F11 Hot Cells and remote Equipment for D&D Legacy Waste Retrieval, and similarities with Fuel Debris Retrieval in 1F

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[2]Orano

As the D&D of La Hague UP2-400 progresses, Orano has designed, built, qualified, and started operating several facilities for Legacy Waste Retrieval. These projects are carried out over long period of time and involve the handling of highly radioactive waste in hot cells with remote equipment means. This requires the development and qualification of dedicated retrieval, sorting and treatment processes, taking into consideration the waste flow from early design stage. The significant design constraints such as technical challenges due to the high activity, adaptation to peculiar waste specifications, installation in old facilities, etc. bear strong similarities with ongoing requirements for FDR facilities in 1F.

F12 NUHOMS® MATRIX system :Horizontal Storage System for Used Nuclear Fuels, Fuel Debris and Nuclear wastes applied to Fukushima

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The NUHOMS®MATRIX Horizontal Storage Module (HSM) is a two-tiered concrete storage overpack system for dry shielded canisters (DSC) to store used nuclear fuels, fuel debris, and nuclear wastes. It is an evolutionary design of the existing NUHOMS®HSM. MATRIX is characterized by its small footprint, storage efficiency, seismic resistance, and self-shielding.Combined with Orano's TMI-2 fuel debris canister design knowledge and the recent successes in U.S. reactor D&D projects, the NUHOMS®MATRIX is an optimal and cost-effective solution being considered for the Fukushima nuclear fuel and fuel debris safe long storage.

F13 Sustainable treatment, clearance and safe recycling of retired large components

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Jerome Altounian[2]
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Cyclife Sweden AB is the experienced supplier in characterisation, transportation, and treatment of large contaminated components. Its approach aims to reduce decommissioning schedule, minimise waste volume, and optimise metal clearance for recycling. 100+ large components have been treated and several are contracted for treatment the coming years. Cyclife provides decontamination and melting services for pre-treated metals delivered in ISO containers.

F14 Methodology for an optimised design of a waste treatment facility

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Cyclife Engineering is dedicated to managing nuclear decommissioning projects. Its knowledge of PWR key components and of the associated dismantling activities allow to understand the complete environment and to define an approach for the design of a waste facility into four stages: data collection, process design and waste routes, HVAC, fluids, I&C, CW; and preparation of next steps.

GB01 Retrieval and management of radioactive materials and wastes supporting facility decommissioning

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英キャベンディッシュ社の取り出し関連技術

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キャベンディッシュは英国の原子力関連企業であり、日本でも廃炉支援を行っている。今回は、英国におけるレガシーポンド／レガシーサイロなどの施設からの廃棄物／スラリーなどの取出しに関する知見／経験についてご説明する。

GB02 Decontamination & Decommissioning Technologies for the Civil Nuclear Industry

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Abstract:
The nuclear industry faces numerous challenges in decommissioning, including improving efficiency and safety while reducing costs. Advanced technologies, including robotics, sensors, and artificial intelligence, have the potential to address these challenges by automating hazardous tasks, improving accuracy, and reducing human error.

Introduction:
Below outlines some examples of technologies that Innovative Physics Limited (IPL) have developed.

Sort & Segregation of Nuclear Waste
Utilising advanced computer vision, machine learning, and robotic technology to automate the sorting and segregation of nuclear waste. The system accurately identifies and sorts different types of debris while improving efficiency, reducing costs, and enhancing safety. Notably, the automated system eliminates the need to expose workers to harmful radiation. The future potential of this technology is significant as it can be applied to other industries, such as waste management, mining, and construction.

Gamma Imaging Systems
Decontaminating an area containing nuclear waste is difficult due to the intangible nature of radiation. Working closely with Japanese partners, IPL designed and developed a gamma imaging system capable of showing “hot spots” of radioactivity. The systems provide an image/video of a large area, allowing workers to quickly and remotely observe where radiation hot spots are located and determine, within minutes, the radioisotope being emitted.

Neutron Detection
Decommissioning planning requires a comprehensive mapping of the radiological environment. Importantly, the location of the fissile material is required, i.e. the neutron field. This allows path planning for removing such material while avoiding criticality events. Custom solid-state neutron detectors using a semi-conductor deposited with Boron-10 (B10), which show a high gamma radiation tolerance and gamma rejection ratio (Co-60, Cs-137up to > 1/106cps/cm2s) to enable monitoring of neutron flux in highly radioactive environments, such as criticality monitoring,emergency management, core monitoring. The novel neutron detector architecture is modular and thus can be integrated into many applications.

Very Low-Level and Low-Level Waste Management
To measure the surface contamination (gamma dose rate) and the isotope of radioactivity. Large scintillators and the relative movement of sensors and objects are used to identify the location of radioactivity. The system uses Time Delay Integration(TDI)techniques to provide additional security and variable speed detection.

GB03 Efficient encapsulation of hazardous orphan wastes using geopolymer matrices

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Encapsulation of active wastes are a important stage in producing long term storage options for active wastes.Typically, cementitious grouts are used as an encapsulation matrix which can present processing challenges with some waste forms. Lucideon have developed an alternative geopolymer matrices for encapsulation of a number of problematic waste streams, in particular oils, graphite and magnesium hydroxide-based sludges.Greater waste loading has been demonstrated along with additional benefits in the processing of waste streams.

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J01 BWR lower plenum damage by reaction between metallic debris and structural materials

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金属デブリによる BWR 炉下部プレナムの破損挙動

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福島第一原子力発電所事故では、下部プレナム領域にて溶融金属プールが形成され、これが圧力容器の初期破損に寄与したと考えられている。この破損挙動を理解するため、下部プレナムに存在する制御棒駆動機構構造材体を模擬した試験体との模擬金属デブリの高温反応試験を実施した。

J02 Development of an image clarification method using deep learning for improving the operator's spatial awareness

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操作者の空間認識向上のためのディープラーニングを用いた画像鮮明化手法の開発

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本発表では、遠隔操作時における環境把握支援を目的として深層学習を用いたカメラ画像鮮明化手法の研究開発について報告する。作業現場で取得される画像はノイズや転送速度などの影響を受け不鮮明な画像が得られる場合がある。従来の画像鮮明化手法では被写体の色彩が大きく異なる場合にパラメータを都度最適化しなければならず汎用性が低下する課題がある。この課題を解決するため、我々は深層学習を用いた学習ベースの画像鮮明化手法について研究開発に取り組んでいる。提案手法はU-netという畳み込みニューラルネットワークのフレームワークを用いて多様なデータを学習させておくことで精度の良い画像鮮明化処理を実現するものである。今回は深層学習を用いた画像鮮明化の効果について従来手法としてバイキュービック法及び機械学習を採用し、画像品質の評価手法であるBRISQUEにより定量的に比較検証を行ったのでその結果について報告する。

J03 Development of an "in-situ" alpha air monitor in a harsh environment

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過酷環境用「その場」α線用空気モニタの開発

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福島第一原子力発電所（1F）のPCV内で想定される過酷な環境（高湿度、高β／γ線バックグラウンド）におけるαエアロゾルの「その場」モニタリングシステムの検出器コンポーネントを設計・試作した。試作機の一部をMOX燃料施設のグローブボックス解体現場に設置し、実際のαエアロゾルに対する高速応答性能と長期運転性能を実証した。

J04 Application Example of Research Findings: Investigation inside Unit 2 Well

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研究成果の現場適用例

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東京電力福島第一原子力発電所2号機の原子炉ウエル内における空間線量率の調査において、原子力機構独自の研究成果と英知事業の研究成果を融合し、廃炉現場への適用を支援した。具体的には、作業前に空間線

量率の予測解析を行い、事前に安全対策を講じた。また、高線量でも測定可能な新開発のセンサーと従来の測定方法を組み合わせることで、測定精度を確保した。さらに、作業中に回収した物質をLIBS分析したところ、海水注入による微量の塩分が検出された。

J05 Development of handheld alpha/beta imaging detector for contamination measurement

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汚染測定のための可搬型アルファ／ベータイメージング検出器の開発

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我々は、福島第一原子力発電所の現場に持ち運び可能な可搬型アルファ・ベータ・イメージング検出器を開発した。この検出器は２つのシンチレータ（プラスチックシンチレータとCe:La-GPSセラミックシンチレータ）で構成されており、アルファ粒子とベータ粒子の識別が可能である。開発した検出器は、福島第一原子力発電所の現場におけるプルトニウム汚染の検出に使用できる。

J06 Research and Development of Digital Technologies to Explore Radiation Source Distributions for Exposure Reduction ~ Current Research and Development Progress ~

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被ばく低減のための線源分布探索に係るデジタル技術の研究開発 ～現在までの研究開発進捗～

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東京電力福島第一原子力発電所（以下、1Fという）での燃料デブリ取り出しの本格的実施に先立ち、線量率の高い原子炉建屋（以下、「R/B」という）内において、安全なアクセスルート構築を行うためには、高強度線源の除染や遮へい等の環境改善が必要である。JAEAは、現場の線量率観測データ等を基に高強度線源の逆推定を行い、サイバー空間（VR）だけでなく、物理空間（MR、AR）も交えて、除染や遮へい等の効果を検討するシステム開発を進めている。本報告では、現在までの研究開発成果を示すとともに、1Fへの現場適用を図るために必要となる高機能化の取組みの概要を紹介する。

J07 Simulation systems of individual external exposure dose using monitoring data of environmental radiation

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環境放射線モニタリングデータを活用した被ばく線量のシミュレーションシステム

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避難指示区域では、除染の進展に伴い避難指示が解除されつつある。避難指示の解除に際しては、被ばく線量を評価・把握すると共に、十分なリスクコミュニケーションが求められている。本研究では、個々のユーザーの行動パターンと空間線量率を基に、容易に外部被ばく線量を評価するシミュレーションシステムを、それぞれの目的に合わせて3種類開発した。一つ目はユーザーが入力した行動パターンについて予測的に外部被ばく線量を評価するシステム、二つ目はスマートフォン内臓のGPSで記録された行動パターンに沿い、ウェブサーバー上で遡及的に外部被ばく線量を評価するシステム、三つ目はスマートフォンによる行動パターンの記録と、PCによる外部被ばく線量の計算から成り、集団の放射線防護対策に適したシステムである。開発したシステムの一部は実際に自治体でリスクコミュニケーションのため導入されており、今後の更なる活用が望まれる。

J08 Chemical Researches Assisting in Decommissioning Fukushima Daiichi Nuclear Power Station Including Material Corrosion, Radioactive Nuclides Behavior and Waste Solidification

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Youko Takahatake[2], Azusa Ito[2], Fumiyoshi Ueno[2], Takashi Okada[2], Ryuji Nagaishi[2], Yoshikazu Koma[2]
[2]Japan Atomic Energy Agency (JAEA)

福島第一廃炉に向けた材料腐食、放射性核種挙動及び廃棄物固化等の化学的研究

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高畠 容子[2], 伊藤 あずさ[2], 上野 文義[2], 岡田 尚[2], 永石 隆二[2], 駒 義和[2]
[2]日本原子力研究開発機構

福島第一原子力発電所の廃炉には幅広い研究開発が必要である。日本原子力研究開発機構は、技術応用の支援だけでなく基礎研究にも貢献しており、材料及び廃棄物管理分野の基礎研究の最近の成果を簡単にレビューする。

J09 R&D on Waste Management for Decommissioning Fukushima Daiichi Nuclear Power Station

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Naoya Kaji[2]
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福島第一原子力発電所の廃止措置に向けた放射性廃棄物に関する研究開発

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鍛治 直也[2]
[2]日本原子力研究開発機構

福島第一原子力発電所の廃止措置に関して、放射性廃棄物を安全に管理するため、キャラクタリゼーション、処理及び処分の技術を具体化する手法を検討している。キャラクタリゼーションにおける放射化学分析は茨城地区での分析を継続しており、今後、分析・研究施設第1棟が主力となっていく。蓄積する分析データは、キャラクタリゼーションの方法論とともに、知識ベースとして構築していく。リスクを低減する観点から、汚染水処理に伴い発生する二次廃棄物を早期に安定化する手法を検討しており、効率的な物性推定、スクリーニングの手法を検討し、また、固化体の長期的な安定性に関する基礎データを収集している。処分すべき廃棄物の量や性状が不確実である前提において、有望な処分手法を検討するための手法を検討している。関係するステイクホルダーとの協力のもとに研究開発を進めていく。

J10 Interference-free determination for long-lived radionuclides based on solid-phase extraction combined with ICP-MS/MS

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Yuki Ohta[2], Mardongan-Banjarnaho Irvin[2], Takahiro Furuse[2]
[2]Japan Atomic Energy Agency (JAEA)

固相抽出法とICP-MS/MSを組み合わせた長半減期核種の高感度分析手法の検討

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福島第一原子力発電所（1F）由来の放射性廃棄物分析を行う大熊分析研究センターでは、2022 年 10 月より中・低線量の廃棄物試料を取り扱う第 1 棟のホット運用が開始され、1F 廃棄物の受入れ、分析が開始された。測定対象核種の中で長半減期核種の測定については、従来の放射能測定法では複雑な前処理操作と長時間の測定が必要となる一方、ICP-MS 法は感度及び測定時間の面でアドバンテージがある。特にハイエンドモデルである ICP-MS/MS は装置自体が高い分離性能を有しており効果的にスペクトル干渉を低減できることから、前処理における化学分離プロセスの簡略化が可能となり、更なる簡易・迅速化が見込める。本発表では、93Zr, 93Mo, 107Pd, 126Sn, などの長半減期核種について、1F 廃棄物試料への適用を目的とした ICP-MS/MS による分析法の研究開発に関する最近の成果を報告する。

J11 Measurement of Radioactive Materials Contained in the ALPS Treated Water by JAEA as a Third Party

Yoshihiro Tsuchida[1]
[1]Japan Atomic Energy Agency (JAEA)
Naoya Kaji[2], Ritsuro Tokumori[2]
[2]Japan Atomic Energy Agency (JAEA)

ALPS 処理水の第三者分析

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鍛治 直也[2], 徳森 律朗[2]
[2]日本原子力研究開発機構

ALPS 処理水に含まれる放射性物質の測定は、客観性及び透明性の高い測定を目的として、原子力機構が第三者機関として放射性物質分析・研究施設第 1 棟で実施している。

J12 Summary of JAEA's efforts for fuel debris retrieval

Tomohiro Tomitsuka[1]
[1]Japan Atomic Energy Agency (JAEA)
Shin-ichi Koyama[2], Naoya Kaji[2], Satomi Kakutani[2]
[2]Japan Atomic Energy Agency (JAEA)

燃料デブリ取出しに向けた機構の取組みの総括

富塚 知博[1]
[1]日本原子力研究開発機構
小山 真一[2], 鍛治 直也[2], 角谷 聡洋[2]
[2]日本原子力研究開発機構

日本原子力研究開発機構では、福島第一原子力発電所での燃料デブリの試験的取り出しに向けて、デブリ取り出し装置の実規模モックアップ試験、DX による放射線の可視化、解析手法の開発や燃料デブリの性状把握、

解析手法の開発等の準備を着実に進めています。大熊分析・研究センター第 2 棟の稼働に先立ち、茨城地区のホットラボで燃料デブリの分析を開始する予定です。

J13 JAEA's Efforts for Human Resource Development and Regional Commitment in Fukushima

Tomohiro Tomitsuka[1]
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Ryoichiro Kuroki[2], Satomi Kakutani[2]
[2]Japan Atomic Energy Agency (JAEA)

JAEA の福島における人材育成と地域貢献への取組み

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[1]日本原子力研究開発機構
黒木 亮一郎[2], 角谷 聡洋[2]
[2]日本原子力研究開発機構

日本原子力研究開発機構では、福島第一原子力発電所の廃炉と福島県の環境再生に向けた研究開発を行っています。これらの活動を中長期的に行うことを考慮すると、人材の確保・育成、地元企業の参加、地元企業への技術移転と現場実装が重要となります。

O01 Remote monitoring system used in a severe radiation environment

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Minoru Watanabe[2], Nobuya Watanabe[2]
[2]Okayama University

過酷な放射線環境下で使用するリモートモニタリングシステム

関岡 空己[1]
[1]岡山大学
渡邊 実[2], 渡邊 誠也[2]
[2]岡山大学

原子力発電所などの過酷な放射線環境で使用するデバイスをテストする際、処理結果を確認するためにリモートモニタリングシステムが必要である。そこで、我々は放射線耐性のあるデバイスである、光再構成型ゲートアレイの試験を行うためのリモートモニタリングシステムを DE1-SOC FPGA（Field Programmable Gate Array）の FPGA ブロックおよび ARM プロセッサを用いて開発した。このリモートモニタリングシステムは制御信号およびテストベクタを生成し、光再構成型ゲートアレイの出力信号をモニタリング可能である。

O02 Design example of a triple modular redundancy ALU, a register file, and a program counter for a processor

Masato Isobe[1]
[1]Okayama University

プロセッサの三重化 LU とレジスタファイルとプログラムカウンタの設計例

磯邉 雅人[1]
[1]岡山大学

宇宙空間や原子力発電所などの放射線の強い環境下では、プロセッサには放射線に対する耐性が求められる。特に原子力発電所の廃炉といった恒久故障の起こりやすい場所では、恒久故障に耐性のあるものでなければ、すぐに故障してしまうため、非常に短命となってしまう。ただ、既存のプロセッサは恒久故障に対しては非常に弱く、強い放射線環境下では急激に劣化し、故障に至ってしまう。実際、一般的に運用に放射線耐性が必要とされている宇宙で用いられているプロセッサであっても、廃炉での長期間運用を考えた際には、十分な恒久故障耐性を持っているとは言えない。そこで我々は、通常ソフトエラー対策として用いられる三重化をトータルドーズ耐性（恒久故障耐性）を上げるために用いたプロセッサの実装に取り組んでいる。今回は、ALU とレジスタとカウンタの三重化実装の例を示している。

O03 Photodiode current range measurement result of an optically reconfigurable gate array VLSI

Sae Goto[1]
[1]Okayama University

Minoru Watanabe[2], Nobuya Watanabe[2]
[2]Okayama University

光再構成型ゲートアレイ VLSI におけるフォトダイオードの電流範囲の測定

後藤 彩絵[1]
[1]岡山大学

渡邊 実[2], 渡邊 誠也[2]
[2]岡山大学

集積回路は放射線に対して脆弱であり、宇宙や原子力発電所のような強放射線環境下において使用される集積回路は、放射線への耐性を高める必要がある。我々が開発中の光再構成型ゲートアレイは、完全並列構成を使用する放射線耐性の高いデバイスである。本稿では、光再構成型ゲートアレイ VLSI において動作可能なフォトダイオードの電流範囲の測定結果を示す。

O04 Evaluation of low-voltage operations of an optically reconfigurable gate array VLSI

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[1]Okayama University

Minoru Watanabe[2], Nobuya Watanabe[2]
[2]Okayama University

光再構成型ゲートアレイの低電圧動作評価

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[1]岡山大学

渡邊 実[2], 渡邊 誠也[2]
[2]岡山大学

現在広く使用されている集積回路は放射線に対して極めて脆弱であり、原子力発電所の廃炉現場のような強放射線環境下では一時的なエラーや恒久的な故障が簡単に生じる。
そこで我々はトータルドーズ耐性の高い耐放射線光再構成型ゲートアレイの研究を行っている。
本稿では耐放射線光再構成型ゲートアレイ VLSI の低電圧における再構成動作評価結果について報告する。

O05 Total-ionizing-dose tolerance of an optically reconfigurable gate array VLSI

Kaho Yamada[1]
[1]Okayama University

Minoru Watanabe[2], Nobuya Watanabe[2]
[2]Okayama University

光再構成型ゲートアレイ VLSI におけるトータルドーズ耐性

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渡邊 実[2], 渡邊 誠也[2]
[2]岡山大学

再構成可能なプログラマブルデバイスであるFPGA（Field Programmable Gate Array）は、回路情報の転送を物理的にシリアルバスを通じて行う。放射線環境下においてシリアルバスに故障が発生すると回路構成は行えない。
そこで我々は回路転送に光学技術を使用した光再構成型ゲートアレイを提案している。本稿では、光再構成型ゲートアレイを構成する要素の１つである光再構成型ゲートアレイ VLSI に対して行ったトータルドーズ耐性試験の結果について報告する。

O06 An optically reconfigurable gate array driven byan unstabilized power supply unit

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[1]Okayama University
Minoru Watanabe[2], Nobuya Watanabe[2]
[2]Okayama University

非安定化電源で動作可能な光再構成型ゲートアレイ

辻野 将[1]
[1]岡山大学
渡邊 実[2], 渡邊 誠也[2]
[2]岡山大学

現在の集積回路は放射線に脆弱で、高放射線環境下では早期に故障してしまうため、高い放射耐性を持つ線集積回路が必要とされている。
我々は放射線に対して高い耐性を持つ光再構成型ゲートアレイ VLSI を開発している。
しかし、この光再構成型ゲートアレイを動作させるための安定化電源装置が必要になるが、安定化電源は放射線に弱く、光再構成型ゲートアレイ VLSI の寿命は安定化電源装置の寿命により制限されている。
そこで、本稿では、放射線に対して脆弱なトランジスタを取り除いた非安定化電源で光再構成可能ゲートアレイ VLSI を駆動する新しい運用方法を紹介する。

O07 A mono instruction set computer architecture on an optically reconfigurable gate array VLSI

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[1]Okayama University
Minoru Watanabe[2], Nobuya Watanabe[2]
[2]Okayama University

光再構成型ゲートアレイ VLSI へのモノ・インストラクション・セット・コンピュータの実装

今井 颯真[1]
[1]岡山大学
渡邊 実[2], 渡邊 誠也[2]
[2]岡山大学

我々は、数ナノ秒で再構成可能な光再構成ゲートアレイ（ORGA）VLSI を開発している。このような高速ダイナミックリコンフィギュレーションを利用し、最も簡単なアーキテクチャを導入することで、プログラマブルゲートアレイの性能を向上させることができる。本稿では、その一つである MISC（Mono Instruction Set Computer）アーキテクチャの評価を行った。今回は、その評価方法として、実装した各命令の処理時間を、現在多くのコンピュータで採用されている RISC（reduced instruction set computer）アーキテクチャと比較することによって行った。その結果、すべての命令において、RISC プロセッサより MISC プロセッサのほうが処理時間が短くなった。この結果をもって、MISC プロセッサは RISC プロセッサよりも高い性能を実現できることが分かった。

O08 Realization of a wafer-scale VLSI by using optically reconfigurable gate array architecture

Atsushi Takata[1]
[1]Okayama University

Minoru Watanabe[2], Nobuya Watanabe[2]
[2]Okayama University

光再構成型ゲートアレイによるウェハースケール VLSI の実現

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[1]岡山大学

渡邊 実[2], 渡邊 誠也[2]
[2]岡山大学

現在の VLSI（Very Large-Scale Integration）は、製造時にウェハ上に多くの欠陥が生じるため、常に小さなダイ上に作製される。しかし、光再構成型ゲートアレイのように完全並列に回路が構成できれば、欠陥を含む巨大なウェハをプログラマブルゲートアレイとして利用することができる。本論文では、光再構成型ゲートアレイアーキテクチャに基づくウェハスケール VLSI の実現方法について述べる。

O09 Sequential circuit implementation onto optically reconfigurable gate array VLSI using a ring oscillator

Shintaro Takatsuki[1]
[1]Okayama University

Minoru Watanabe[2], Nobuya Watanabe[2]
[2]Okayama University

リングオシレータを用いた光再構成型ゲートアレイ VLSI への順序回路実装

高月 信太朗[1]
[1]岡山大学

渡邊 実[2], 渡邊 誠也[2]
[2]岡山大学

高放射線環境で使用されるデバイスは恒久的な故障に至りやすい。順序回路に使われる水晶発振器も高放射線環境下においては壊れる可能性がある。そこで本論文では順序回路に対して水晶発振器の代わりにリングオシレータを用いる手法を紹介する。ここで紹介するリングオシレータ回路は光再構成可能なゲートアレイ VLSI に実装され、正しい動作が実験的に確認されている。

A01 Research and Development of the Project of Decommissioning, Contaminated Water and Treated Water Management and Connection to Engineering

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Abstract

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

1. Introduction

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

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

R&Ds for the decommissioning of the Fukushima Daiichi NPS have been subdivided and subsidized. Each subsidized project is being carried out by organizations in Japan and overseas. The subsidized projects are classified into "Internal Investigation", "Development of Retrieval Method", "Improvement of Work Environment", and "Processing of Solid Waste, etc.". The R&Ds of Fuel Debris Retrieval have been conducted based on the information obtained by Internal Investigation. In addition, the results of R&Ds such as Development of Fuel Debris Retrieval Methods are reflected to Improvement of Work Environment. The research projects of Processing of Solid Waste are also studied in cooperation with R&Ds for Fuel Debris Retrieval and Improvement of Work Environment. In this way, the current projects are closely related to each other and connected to engineering and the engineering has been utilizing wisdom in Japan and overseas. To obtain the information on R&Ds from organizations in Japan and overseas, the Management Office is also conducting RFI (Request for Information) each spring on the website. https://en.dccc-program.jp/

A02

Organizational Profile of IRID

Naoaki Okuzumi, Akihiro Tsukada
International Research Institute for Nuclear Decommissioning (IRID)

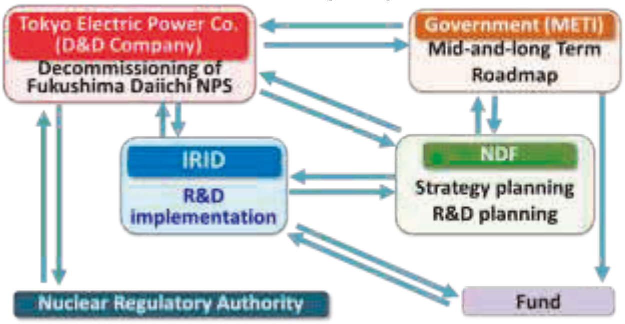
Abstract

Ever since the International Research Institute for Nuclear Decommissioning (IRID) was establishment in August 2013, IRID has engaged in research and development (R&D) of technologies necessary for the decommissioning of the Fukushima Daiichi Nuclear Power Station (NPS) which is an urgent issue. IRID focuses on strengthening the platform of decommissioning technology.

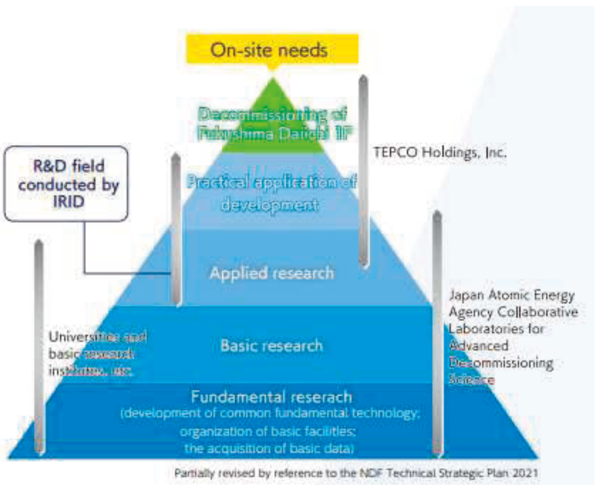
1. Scope of Work

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

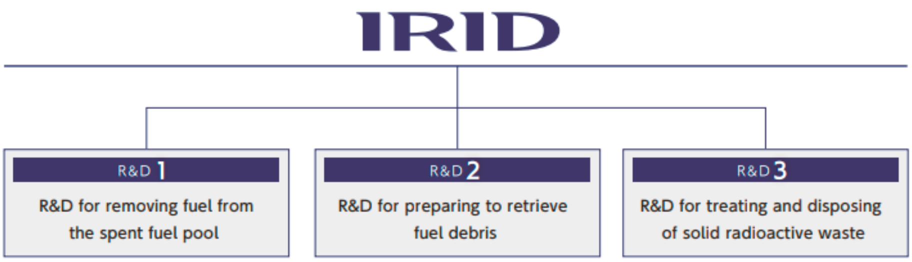
2. Roles of Organizations in the Decommissioning Projects of the Fukushima Daiichi NPS.



3. R&D Scope of IRID



4. Three major R&D for Nuclear Decommissioning



A03

Overview of IRID R&D Projects

Naoaki Okuzumi, Akihiro Tsukada
International Research Institute for Nuclear Decommissioning (IRID)

Abstract

For the decommissioning of the Fukushima Daiichi Nuclear Power Station (NPS), four key players including TEPCO, the Japanese government, NDF and IRID are closely working together. IRID is a complex entity consisted of nineteen organizations that have responsibility in research and development (R&D) for the decommissioning of the Fukushima Daiichi NPS. IRID engages in three major R&D projects: (1) Project of fuel removal from spent fuel pool, (2) Project of preparation for retrieving fuel debris and (3) Project of the treatment and disposal of solid radioactive waste. These R&D projects are being conducted under the Mid-and-Long-Term Roadmap issued by the government. The period until completion of the decommissioning is divided into three phases. Currently, the second phase, R&D for preparation for retrieving fuel debris is underway.

1. Progress of R&D

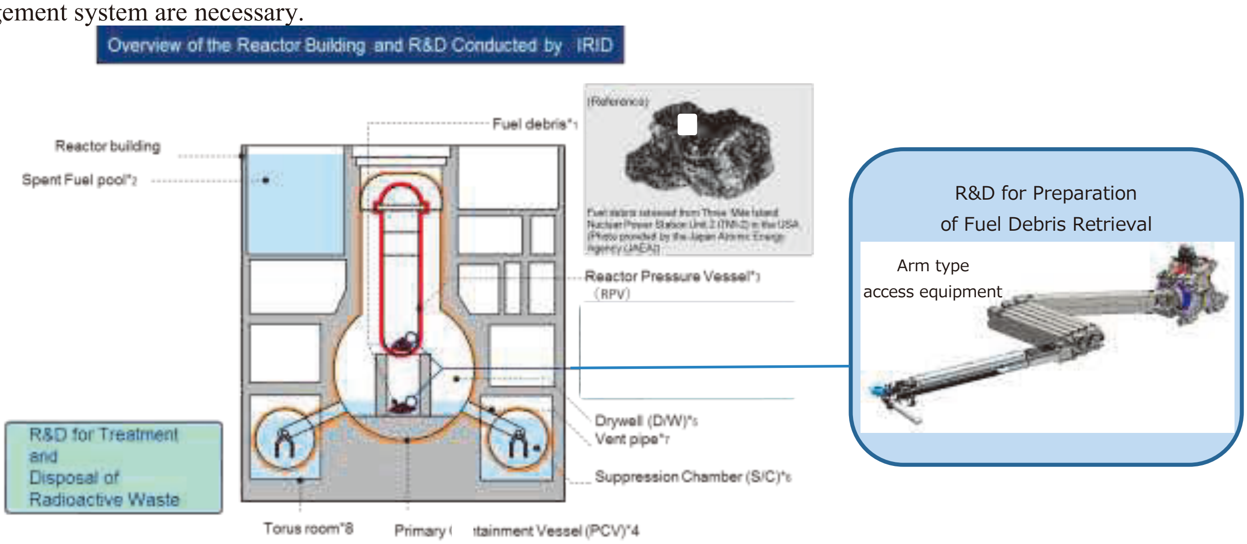
IRID proceeds with preparation of fuel debris retrieval based on a strategy indicated in the "*Technical Strategic Plan*" issued by the Nuclear Damage Compensation and Decommissioning Facilitation Corporation (NDF). IRID was developed element and robotic technologies for the Fukushima Daiichi.

First, detection technology to directly access fuel debris in the PCV was developed. In April 2015, a robot successfully entered the Unit 1 PCV. In FY 2016, a preparation for fuel debris investigation outside the pedestal started. At the same time, investigation robots for inside the pedestal, and remote-operated equipment to make an opening of the Unit 2 PCV penetration was developed to reduce worker's exposure. Additionally, fuel debris investigation inside the pedestal was conducted for Unit 3 by using an underwater swimming robot, and investigation equipment mounted a telescopic pipe and cameras for Unit 2. These robots can access fuel debris by remote operation and successfully obtained visual data of the PCV interiors. The Severe Accident Analysis Code was upgraded to identify fuel debris inside the reactor, and investigations through the cosmic-ray muon were performed. The distribution of fuel debris in the Unit 1 reactor was investigated from outside the reactor building by cosmic rays muon. The results of muon investigation revealed that almost no fuel remains in the reactor core. The muon transmission measurement was performed for Unit 2 from March to July 2016, and for Unit 3 from May to September 2017.

Furthermore, IRID developed the arm type access equipment to conduct a trial retrieval of fuel debris in the Fukushima Daiichi Unit 2 for more detailed investigation inside the PCV through the existing X-6 penetration. An original model of the investigation equipment is a robot arm used for maintenance of an experimental fusion reactor, placed in UK. The robot arm was redesigned to meet requirements for the Fukushima Daiichi.

2. Future Development

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



B01

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

S. Nakashima[1], R. Komatsu[1], A. Moro[2], A. Faragasso[1], H. Woo[3],
N. Matsuhira[1], K. Kawabata[4] and H. Asama[1]
[1]The Univ. of Tokyo, [2]RITECS Inc., [3]Kogakuin Univ., and [4]JAEA

Abstract

The aim of this research is to develop a novel manipulator for retrieving fuel debris on the Fukushima Daiichi Nuclear Power Plant (1F) as shown in Figure 1. We designed a shock-resistant CVT (Continuously Variable Transmission) robot in collaboration with the University of Sussex.

1. Introduction

Fuel debris retrieval at the bottom of the primary containment vessel (PCV) is one of the significant tasks for the decommissioning of the nuclear power plant and in particular for 1F. It is challenging for conventional manipulators to perform the retrieval process due to the presence of radiation, water leakage and poor lighting conditions. We tackle those problems with the design and fabrication of a novel mechanical manipulator and its control and navigation algorithm. CVT-based actuation improves the robot's shock resistance. AI-based navigation algorithm enables semiautonomous navigation and grasping in the cluttered environment inside the PCV.

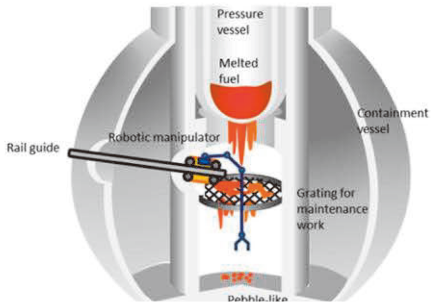


Figure 1. Robotic manipulator in PCV

2. Development Subjects

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

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

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

We simulated realistic environments to test our machine learning algorithms using PyBullet simulator. Here, a remote operator can set the gripper target position.

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

The system performance was validated on the life-sized robot arm.

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

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

3. Conclusion

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

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

B02

Work Robot with Flexible Structures

Satoshi Ohno[1,2], Yuya Yoshimura[1], Takahiro Nagai[1], Ren Morinaka[1,2],
Katsuhiko Hirano[1] and Koichi Kurosawa[1]

[1]Hitachi-GE Nuclear Energy, Ltd., [2]International Research Institute for Nuclear Decommissioning

Abstract

Hitachi-GE has developed a water hydraulically driven "Work Robot with Flexible Structures". The main feature of this robot is its ability to perform "flexible" movements, which enables it to perform detailed work such as that performed by humans in a high-radiation environment.

1. Introduction

Hitachi-GE has developed a "Work robot with flexible structures" that can perform "detailed work" (e.g., assembling equipment, slinging a suspended load, etc.) that can be performed by humans in a high-radiation environment. In this system, a robot system with a " flexible" structure can be assembled in a short period of time and in a manner that is suited to the application environment by combining each joint module. This paper introduces the features and application examples of the " Work robot with flexible structures".

2. Features

The "Work robot with flexible structures" is a robot system with the following features:

1) The combination of hydraulically driven joints enables a "flexible" structure, which allows remote work even in rough positioning. As a result, "detailed work", which was difficult with conventional motorized multi-axis robots, can be performed.

2) No electronic components are used, so the robot can be used for a long period of time even in a high-radiation environment.

3) As shown in Figure 1, the water hydraulic unit for control of the main body of the robot system is placed on a low-radiation environment and can be diverted even if the configuration of the main body is changed.

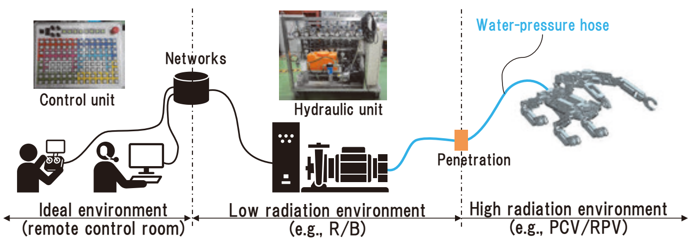


Figure 1. System configuration for a Work robot with flexible structures

3. Applied to remote work

This robot was utilized for elemental tests of various remote operations for fuel debris retrieval conducted at *International Research Institute for Nuclear Decommissioning*, and the feasibility of such operations was confirmed (Figure 2). Also, it was applied to the removal of interfering objects for the installation of the retained water transfer system in the turbine building of Unit 3 starting in October 2019.

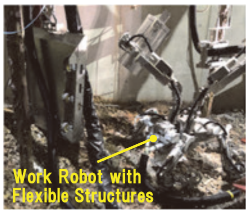


Figure 2. Remote operation by the robot[1]

References

[1] The final result of subsidy program "Project of Decommissioning and Contaminated Water Management (Development of Technologies for Retrieving Fuel Debris and Internal Structures (FY2020))" in the FY2017/18 Supplementary Budget, page 275.

C01

Remote Control Technology for Monitoring Inside RPV Pedestal during Retrieval of Fuel Debris: Prototype Experiments

N. Matsuhira[1], R. Komatsu[1], S. Nakashima[1], A. Yamashita[1], R. Fukui[1], H. Takahashi[1], K. Shimazoe[1], H. Woo[2], Y. Tamura[3], T. Takahashi[4], Y. Yokokohji[5], K. Nakamura[6], K. Naruse[7], T. Hanari[8], K. Kawabata[8], S. Suzuki[1] and H. Asama[1]

[1]The Univ. of Tokyo, [2]Kogakuin Univ., [3]Tohoku Univ., [4]Fukushima Univ., [5]Kobe Univ., [6]Sapporo Univ., [7]Univ. of Aizu, and [8]JAEA

Abstract

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

1. Introduction

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

2. Development Subjects

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

Technology for building modular, split-type platforms and highly rigid, lightweight arms that can be retracted compactly have been developed. The feasibility of the system was verified in deploying the track structure and moving the observation robot on the track. The design of the basic structure of the compact, highly rigid, lightweight arm was completed.

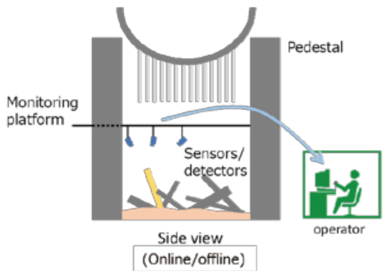


Figure 1. A monitoring platform concept

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

In a video presentation interface for operators and a highly realistic tele-operation system, a visualization target on a 3D environment model was specified and confirmed that the proposed method presented the unobstructed camera positions. An experimental platform was constructed that enables remote control using various multiple camera viewpoints placed in a real environment and virtual camera viewpoints in a simulation environment.

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

A neutron detection experiment was conducted in a high gamma-ray field using the developed neutron detector and confirmed that it can detect neutrons in the presence of 1 Gy/h of gamma radiation.

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

A system was developed that integrates the processes of camera image collection, storage, and three-dimensional reconstruction (SfM-MVS), and integrates the developed reconstruction image selection method into the system.

3. Conclusion

The research and development have been conducted on a monitoring platform for fuel debris removal. Some experiments were conducted using the mockup at the JAEA Naraha Center (NARREC). In addition, lectures with the current status and future issues of 1F were introduced for students, and the research on master/doctor theses and conference presentations were done by younger person as a human personnel development.

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

C02

Microbial recover of nuclear fuel debris components

Toshihiko Ohnuki
Tokyo Institute of Technology

Abstract

For decommissioning of the damaged Fukushima Daiichi NPP reactors, treatment of the retrieved debris is important for waste storage and disposal. Separative dissolution of the debris components probably simplifies treatment of the debris. Here we propose microbial treatment of the debris components. We have investigated the effect of a siderophore-releasing microorganism (SB) on fuel debris analogues. Fuel debris analogue (FDA) pellet samples and powder samples containing CeO_2 or UO_2-ZrO_2 solid solution and metallic iron were formed. FDA containing Ce (FDA-Ce) were contacted with SB on a membrane filter placed on agar medium for 50 days. To investigate the bacterial influence, the powder components of UO_2 and Fe were conducted SB in a Fe-deficient liquid medium under the aerobic condition. We found that SB dissolved higher amounts of Fe and UO_2 than those without microbe cells by time courses of dissolved elements in the solution, and SEM-EDX analyses of solids. On the contrary, Zr oxides were not dissolved. These results indicate that SB enhances selectively the dissolution of Fe and UO_2 in the fuel debris. We discuss the applicability of microbial activity to separative treatment of the retrieved fuel debris.

C03 Preliminary verification of criticality impact analysis code applicable to weakly coupled systems including fuel debris particles

Hiroki Takezawa[1] and Toru Obara[2]

[1]Nagaoka University of Technology, [2]Tokyo Institute of Technology

Abstract

For fuel debris removal work at the Fukushima Daiichi NPS (1F), it is also necessary to establish mitigation measures based on criticality impact evaluations. A space-dependent kinetic analysis code that is based on the integral kinetic model and applicable to weakly coupled systems including fuel debris moving particles has been under preliminary verification using GODIVA supercritical experimental results.

1. Introduction

It is important to evaluate the impact of a criticality that may occur in removing fuel debris from the damaged 1F reactors for ensuring safety of workers engaged in the removal operation [1]. The fuel debris located at the bottom of the 1F RCVs can be a fast-thermal weakly coupled reactor system including fuel debris particles that are possible to move (Fig. 1). So, a space-dependent kinetic analysis code applicable to weakly coupled reactors, Multi-region Integral Kinetic (MIK) code 2.0, has been under development based on the integral kinetic model (IKM) [2] and has been under preliminary verification.

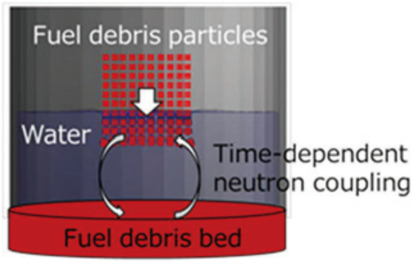


Figure 1. Example of weakly coupled system.

2. Methods

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

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

where $\alpha_{ij}^{\mathrm{p}}(\tau)$ and $\alpha_{ij}^{\mathrm{d}_I}(\tau)$ are the probability density functions of secondary fissions in region i at the present time t which are induced by prompt neutrons or delayed neutrons of the I^{th} precursor family that are generated from a source fission in region j at the past time t' with a time difference $\tau \equiv t - t'$ [secondary fissions@i/sec/source fission@j]. Delay of delayed neutrons emission will be introduced in Eq. (1) by the law of decay. Feedback effects can be reflected on Eq. (1) by updating the $\alpha_{ij}(\tau)$ functions during kinetic calculation.

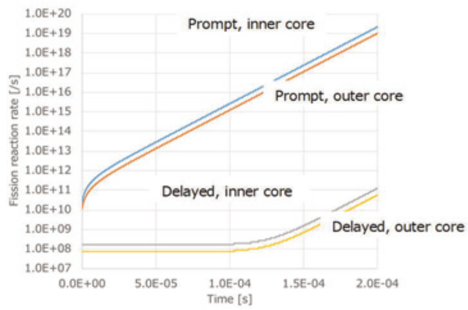


Figure 2. Example of GODIVA supercritical calculation by MIK2.0 code.

3. Progress and Future Plan

A result of preliminary verification of the MIK2.0 code using GODIVA supercritical experimental results is shown in Fig. 2 for an inserted reactivity of approximately 1.08 $. It was confirmed that the code can calculate fission reaction rates by prompt and delayed neutrons. After the verification, it will be coupled to an MPS code in order to include fuel debris particles movement in the analysis. This work is financially supported by the Nuclear Energy Science & Technology and Human Resource Development Project (through concentrating wisdom) from the Japan Atomic Energy Agency / Collaborative Laboratories for Advanced Decommissioning Science.

References

[1] https://clads.jaea.go.jp/en/rd/map/2023/issues/fdr/fdr-207.html, [2] Takezawa, Tuya, Obara, NSE195(2021)1236.

C04 Progress and Prospects of the Internal Investigation of Unit 1 Primary Containment Vessel

Satoru Nakashima[1], Nobuyuki Kumakawa [1], Shoichi Shinzawa[1], Tomoyuki Arai[1]
[1]Tokyo Electric Power Company Holdings Inc.

Abstract

During the Unit 1 Primary Containment Vessel (PCV) drywell (D/W) investigation conducted from 2022 to 2023, the state of the south side of the PCV and inside of the pedestal have been surveyed. Significant pedestal wall concrete degradation and damage to the structures within pedestal was confirmed, showing higher degree of damage in comparison with Units 2 and 3.

1. Introduction

Due to the accident in 2011, fuel materials together with core internals melted down and part of them relocated into the PCV at Units 1-3 of Fukushima Daiichi Nuclear Power Station. Ever since then, PCV internal investigations are underway with the aim of clarifying fuel debris distribution and properties. South part of the basement floor and pedestal area of Unit 1 D/W was investigated in 2022~2023 utilizing submersible ROVs to obtain visual and dosimetry data and samples.

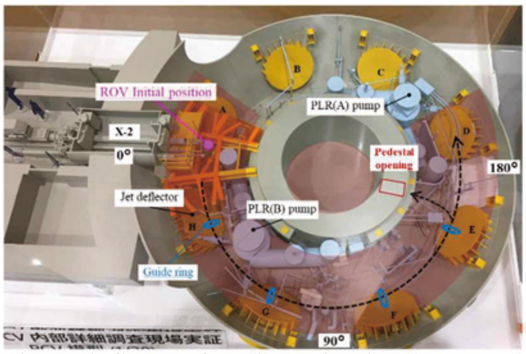


Figure 1. Scope of the investigation (Red area)

2. Investigation findings

Deposited materials were confirmed both inside and outside of the pedestal area, including back of jet deflectors of some of the vent pipes leading to wetwell. In the pedestal (Fig. 3), deposits, concrete degradation exposing rebar and structures assumed to be fallen CRD housings were confirmed. In case of Unit 1, practically no parts of the CRD exchange equipment originally installed in the pedestal area could be confirmed. On the other hand, in Unit 2, this equipment is mostly sound. Although this equipment was found collapsed in Unit 3, many of its parts could be identified. Significant damage of concrete wall has so far been found only in Unit 1.


Figure 2. Unit 1


Figure 3. Unit 2

3. Conclusion

Big degree of damage inside the pedestal area of Unit 1, and spread of deposits also outside of pedestal was confirmed in Unit 1. The different states of all 3 Units is the result of difference in accident progressions, with Unit 1 having the earliest onset and longest period of time without effective water cooling. For further clarification, investigation of the distribution of deposits in vent pipes and sampling of deposits will be conducted.

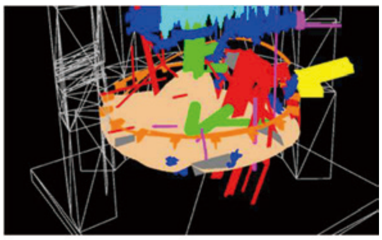


Figure 4. Unit 3

References

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

D01

Engineering efforts for spent fuels removal from Fukushima Daiichi nuclear power station unit 1

Kenji Toyoshima[1], Akio Hirata[1], Tomochika Gyobu[1],
Takamasa Nishioka[1], Akira Takahashi[1] and Ryota Mizutani[1]
[1]Kajima Corporation

Abstract

In consideration of the returning residents in the vicinity of the site, the following plan is adopted to minimize the dispersal of radioactive material as much as possible for spent fuels removal from unit 1.

- A large cover is constructed in advance, and high-dose debris is removed remotely inside the cover.
- After removing the debris to reduce the radiation dose, the pool fuels are removed by Fuel Handling Machine(FHM).

1. Introduction

This paper shows engineering efforts for minimizing the dispersal of radioactive material to environment and reducing radiation exposure of labor in spent fuels removal from unit 1.

2. Engineering efforts

2-1. Large covers supported from existing buildings

The following advantages are achieved by supporting the large cover from the existing building.

- Ensure seismic resistance without a newly constructed foundation → Reduce the amount of contaminated soil generated
- Debris removal inside the cover →Prevent dust scattering
- Prevent rainwater from contacting debris →Reduce the amount of contaminated water generated
- Work can be done under rough weather →Improved construction efficiency
- Retractable roof → Work also can be done with crawler cranes

2-2. 3D mobile W-trolley overhead crane

A 3D movable W-trolley overhead crane equipped with two trolleys is used to accurately grasp, cut, and remove irregularly collapsed rooftop steel frames, overhead cranes, FHM, etc.

2-3. Debris removal simulation

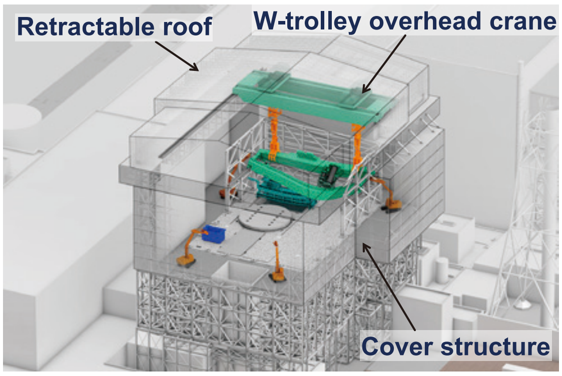
The simulation technology used for Unit 3 has been upgraded, and more accurate analysis has been conducted to "visualize" in advance demolition work that cannot be verified onsite.

2-4. Modular construction

In order to reduce radiation exposure, the steel frame is assembled in a low-dose area, and the assembled large unit is transported to the site by a self-propelled multi-axle truck and lifted by large lifting equipment.

2-5. Remote-controlled anchor drilling

Developed and applying remote-controlled drilling hole technology for anchors to secure large cover structure to outer wall surfaces.



Overview of a large cover for Unit1

E01 Development of Optical Materials and Its Evaluation for the Radiation Dose-Rate Monitor under Ultra-High Dose-Rate

Shunsuke Kurosawa[1,2], Chihaya Fujiwara[3], Shohei Kodama[4], Daisuke Matsukura[1], Ai Kaminaka[1], Maki Ohno[1], Takushi Takata[5], Hiroki Tanaka[5], Akihiro Yamaji[1]
[1]Tohoku Univ., [2]Osaka Univ., [3]TIRI, [4]Saitama Univ., [5]Kyoto Univ.

Abstract

We have developed a real-time monitoring system for high-dose-rate with a wide dynamic dose range using a novel red-emission scintillation material with a high light output of over 60,000 photons/MeV.

1. Introduction

Mapping of high-dose-rate distribution in Fukushima Daiichi Nuclear Power Station is one of the first steps for decommissioning. Since conventional techniques are hard to monitor the rate due to the high-dose-rate condition of over 0.1 Sv/h, we have developed a fiber-type dose-rate monitor consisting of a red-emission scintillator. Scintillation photons are read with a photo-detector under a lower dose-rate thorough an optical fiber. On the other hand, Cherenkov and scintillation photons excited by radiations in the glass are expected to be observed as noises and these noises have generally emission wavelengths of below 550 nm. To discriminate the noise, we have developed a red-emission scintillator, Cs_2HfI_6 with an emission band of over 600 nm and other materials.

2. Methods

Cs_2HfI_6 and other crystals were grown by the vertical Bridgman-Stockbarger method. Such crystal samples were installed into the fiber-type dose-rate monitor. This monitor was demonstrated with a 20-m-long optical fiber and CCD spectrometer. Also, a photo-multiplier tube was used as a photodetector, and as a gamma-ray source, we used a ~100 TBq ^{60}Co source at Institute for Integrated Radiation and Nuclear Science, Kyoto University.

3. Results

Cs_2HfI_6 had an emission wavelength of around 700 nm excited by X-ray, and this scintillator was estimated to have a light output of over 60,000 photons/MeV. The scintillation signal intensities were measured as a function of the dose ratio, then the monitorable dynamic range was determined as the order of 10^{-3} up to 100 kSv/h [1]. In addition, other materials were also demonstrated as gamma-ray detection scintillators with the same setup as the above at Kyoto University.

4. Conclusion

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

References

[1] S. Kodama and S. Kurosawa et al., APEX 13, (2020) 047002

E02 Neutron detection system without radiation protection for criticality approach monitoring based on diamond sensors and radiation-resistive integrated-circuits

M. M. Tanaka[1], J. Kaneko[2], H. Umezawa[3], T. Endo[4], Y. Tanimura[5], K. Watanabe[6], Y. Fujita[1], E. Hamada[1], H. Kawashima[3], T. Kishishita[1], Y. Kobayakawa[2], M. Miyahara[1], S. Nishino[5], K. Oda[2], M. Sakaguchi[1], H. Sendai[1], T. Shimaoka[3], M. Shoji[1], K. Tauchi[1] and H. Uchinoyae[1]

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

Abstract

This report summarizes the research results of the "Technology development of diamond-base neutron sensors and radiation-resistive integrated-circuits for shielding-free criticality approach monitoring system" conducted until FY2023.

1. Introduction

The study aims to develop key components(diamond sensors and radiation-resistive integrated circuits) of a neutron detector without a radiation shield for a criticality approach monitoring system shown in Figure 1. It is required high neutron detection efficiency(a few cps/nv) under 1 kGy/h and compact-light-weight to fit constraints of the penetration size and the payload. One of key issues is high radiation environment. The response function of the detector have to be optimized to minimize the γ-ray counts and to maximize the neutron detection efficiency. The fabrication process of the diamond neutron sensor is newly developed for the purpose, and the developed signal-processing integrated circuits and a high speed data-transfer integrated circuit have to work up to a few MGy.

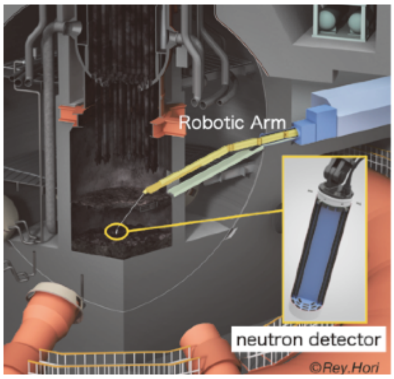


Figure 1. neutron detector for a criticality approach monitor

2. Status

The feasibility study clarified that the Feynman-α method for the subcriticality measurement can measure the prompt neutron decay constant if the ratio of γ-ray counts to neutron counts is less than 1.[1] After evaluations of sensors and integrated circuits using neutron sources and γ-ray sources, evidences show our proposed neutron detector satisfies the specification of subcriticality measurement at 1F. Proposed neutron detection system comprises a few thousand of diamond neutron sensors with LiF converters and several modules with rad-hard ASICs. We also developed a 6cm diameter mockup which can be inserted into dry tubes to evaluate the characteristics. The mockup shows expected neutron efficiency results shown in Figure2.

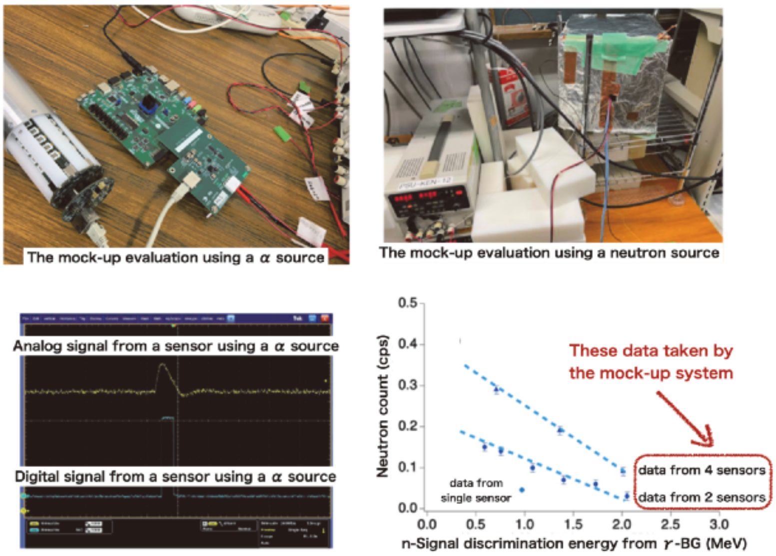


Figure 2. The mock-up evaluation results

3. Acknowledgement

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

References

[1] Reports of FY2021 and FY2022 Nuclear Energy Science & Technology and Human Resource Development Project. https://doi.org/10.11484/jaea-review-2021-038, https://doi.org/10.11484/jaea-review-2022-031 Report of FY2023 to be published.

E03

The World's Highest Radiation Tolerant Performance Camera

Mikio Katsura, Hazuki Nakazawa, Yudai Suzuki and Naoki Kajihara
CORNES Technologies Ltd.

Abstract
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 (Power Plant, Reprocessing and Decommissioning) 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.

Hyperion™ Compact Camera System

The Hyperion Compact Gen II Camera follows many years of extensive Research and Development and considerable investment in new technologies. Featuring all new Radiation Tolerant zoom lens and optics with integrated pan and tilt, this new version offers all the benefits of the earlier generation Hyperion camera but in a lighter and more compact outstation. The Hyperion camera has been independently tested to 1 MGy with Cobalt-60 sources.

In addition, Mirion is pleased to offer both monochrome and color variants without a pan/tilt unit for even greater deployment flexibility on Servo-Manipulators, custom deployment mechanisms, and tooling applications.

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

- 100 Mrad / 1 MGy total dose (gamma)
- Digital performance, high radiation tolerance
- Color or Monochrome options
- Pan/Tilt-free version available
- In air or underwater operation
- System-on-Chip flexibility
- Superior picture geometry
- High performance solid-state 1x megapixel sensor
- Low cost of ownership, driven by low maintenance and long life

Web Application operation

The camera video stream can be viewed, and the camera controlled using a web application via Ethernet.
The System provides a web application on the camera IP address on a port number as set by the camera configuration.
The web application can be accessed by a web browser connecting via HTTP.

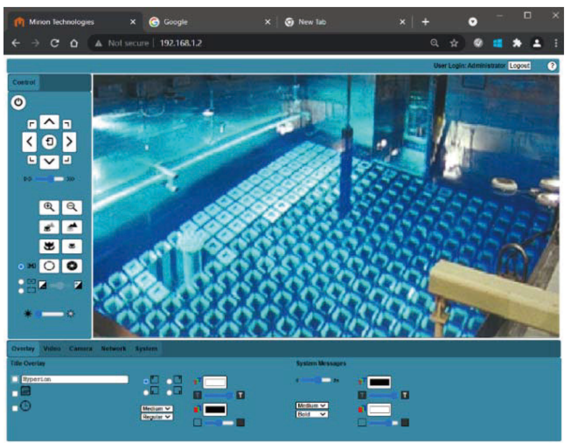


Figure1: Web application

Conclusion

The new Hyperion Compact high radiation tolerant digital cameras extends on our years of research and development into digital radiation tolerant electronics combined with our unique Mirion color processing algorithms to provide an unsurpassed user experience for high radiation tolerant imaging.

E04

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

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

Abstract

MORESCO-HIRADs, the world's most radiation resistant lubricants, are widely installed in applications such as decommissioning, accelerator, radiomedicine and upcoming nuclear fusion. In decommissioning, it has not only been installed in the recent "trial debris retrieval equipment", but is beginning extended to the upcoming "expanded debris retrieval equipment".

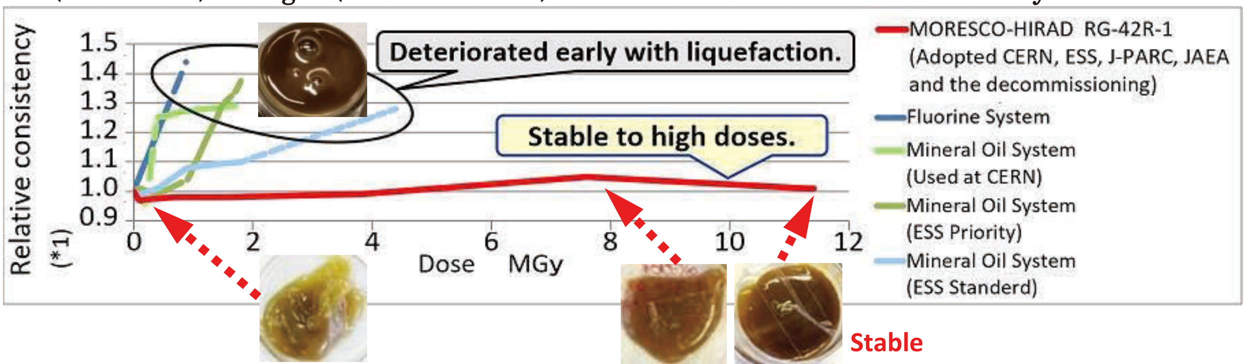
1. Introduction

We are conducting valuable studies on the lubricants (oils and greases) with international major academic institutions, so as to contribute to stable operations of worldwide equipment under harsh radiations, including decommissioning.

Some of these are presented below, with examples of the application for decommissioning.

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

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

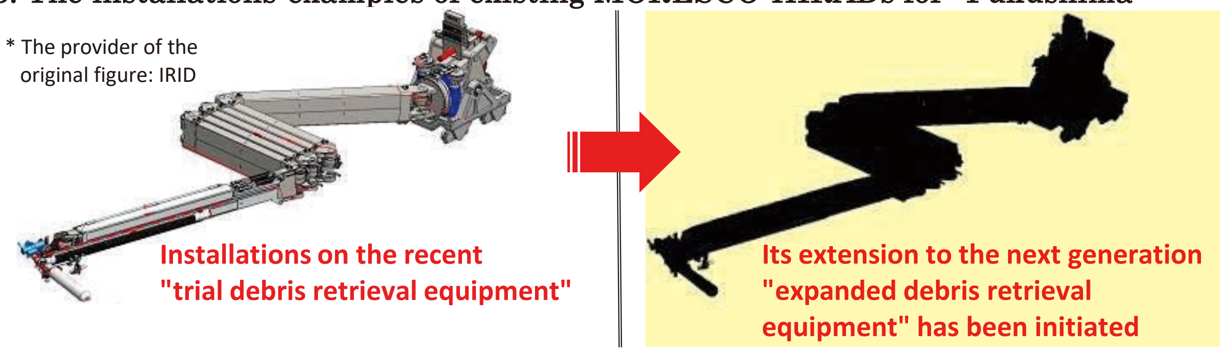


Stable

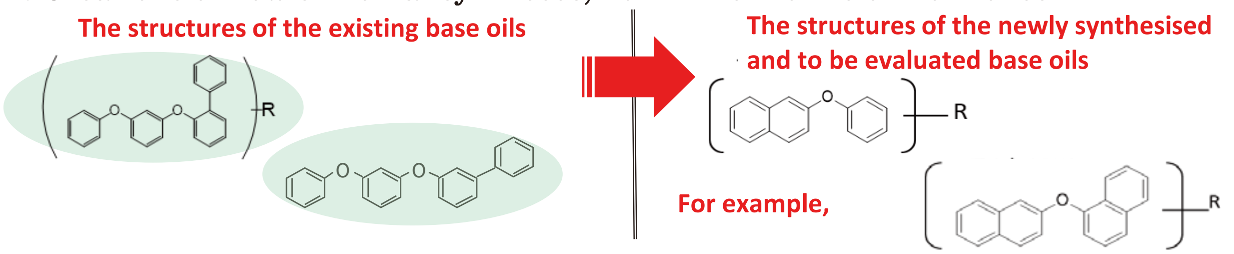
(* The appearances of RG-42R-1 for each dose)

(*1) is calculated with each sample's respectively value at the time of non-irradiation as "1.0".

3. The installations-examples of existing MORESCO-HIRADs for "Fukushima"



4. Creations of new structual syntheses, not limited to the current ones



Contact details for inquiries - Tel: +81 78 303 9010, e-mail : hayashi@moresco.co.jp

E05 Development of Gamma and Neutron Dosimeters Based on Solar Cells

Yasuki Okuno[1], Tomohiro Kobayashi[1], Mitsuru Imaizumi[2], Yukiko Kamikawa[3], Ryuta Kasada[4], Tetsuya Nakamura[5] and Tamotsu Okamoto[6]

[1]RIKEN, [2]Sanjo City Univ., [3]AIST, [4]Tohoku Univ., [5]JAXA, [6]NITKC

Abstract

Solar cell dosimeters can be configured for a wide variety of applications, including flexible sheeting, multi-connected mapping monitoring systems, and gamma/neutron detection structures. In this presentation, the radiation detection system using solar cells will be described and its applicability to PCVs, vertical and internal applications will be reported.

1. Introduction

In order to remove fuel debris from the containment vessel (PCV) of the Fukushima Daiichi Nuclear Power Plant and to efficiently decommission 1F, it is necessary to know the radiation sources and dose rate distribution in the PCV, to take appropriate decontamination and radiation shielding measures to ensure worker safety, and to optimize radiation resistance and other properties of equipment used for decommissioning. Optimization of radiation resistance of equipment used for decommissioning is also necessary. Radiation detection using solar cell elements has been developed as a self-driven dosimeter that outputs a radiation-induced current as a signal without applying an external voltage by means of a built-in potential drive. In this study, we will develop a system with high functionality based on the solar cell dosimeter for 1F mounting.

2. Experimental

The solar cells were prepared from silicon and CIGS solar cells. A source-measure unit (Keysight, B2901A) was used to measure signals in a radiation environment. As a neutron detection structure, a coated film such as boron was placed on the solar cell elements. A Co-60 source was utilized as the gamma source. RANS was used as the neutron source. A multiplexer was used to construct the mapping system.

3. Results and Discussion

Figure 1 shows a prototype gamma-ray sensor using CIGS solar cells.The system consists of a ammeter, cable, and solar cell detector. In order to understand dosimeter characteristics, it is important to clarify detection sensitivity, resolution, dynamic range, direction dependence, response time, calibration method, and radiation tolerance, and experimental results were obtained on those characteristics of a solar cell dosimeter based on CIGS solar cells in gamma and neutron environments.

4. Conclusion

By developing a system that can map radiation fields by applying a solar cell device-based sensor that is self-supporting and remotely driven, we obtained knowledge to implement a system that can guarantee safety for workers and residents by obtaining comprehensive and real-time radiation information in the PCV and monitoring leakage of gamma and neutron radiation, which are very penetrating and can cause accidents, in a real environment.

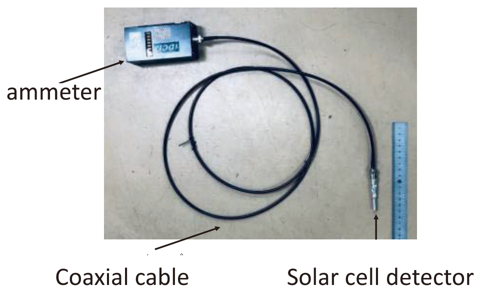


Figure 1. Solar cell-based dosimeter system

I01 Establishment of 3-D dose dispersion forecasting method and development of in-structure survey using the transparency difference of each line gamma-ray

Shinya Sonoda[1], Atsushi Takada[1], Tohru Tanimori[1], Minoru Tanigaki[2], Akihiro Taniguchi[2], Haruyasu Nagai[3], Hiromasa Nakayama[3], Daiki Satoh[3], Tetsuya Mizumoto[4], Shotaro Komura[4]
[1]Kyoto Univ., [2]KURNS, [3]JAEA, [4]FSiC Inc.

Abstract
We have developed an Electron Tracking Compton Camera (ETCC) that can provide the bijection imaging of gamma-rays. We measured gamma-rays from the whole area of 1F site (~1km square) including three reactor buildings at once and obtained the quantitative dose distributions. We will measure the 3D dose distribution on 1F site and combine ETCC with WSPEEDI to construct a practical radiation detection and prediction system.

1. Introduction

Nuclear gamma-ray imaging is considered to be essential technique for radiation science. However, since the bijection imaging similar to that of an optical camera has not been realized, the quantitative measurements are not possible, and then the applications regulated by law is limited. We have developed an ETCC, which can uniquely determine the nuclear gamma-ray direction by measuring the recoil electron direction. Thus, ETCC provides the bijection gamma-ray image, which enable us to obtain the quantitative 3D dose distribution based on the stereo method as same as optical camera. By analyzing the distribution with WSPEEDI, a system for predicting the radioactive diffusion is realized. In order to use it inside the reactor building, the radiation tolerance of the ETCC will be improved to operate under sub-milli Sv/h environment.

2. Instruments & Measurement

By using the 3D dose reconstruction method developed by JAEA, the 3D distribution of gamma radiation emitted from the operating reactor at Nuclear Science Research Institute, Kyoto University has been successfully obtained. Also, the diffusion of 41-Ar was measured on a movie of gamma-ray images. Combining ETCC with WSPEEDI, we demonstrated that a diffusion prediction system is feasible. Since the MPPC gain for the scintillator in ETCC is dependent on temperature, it is necessary to stabilize the device temperature. As the temperature control system of the MPPC was installed. Also, we shielded the Scintillators of ETCC with a 1 cm thick lead plate for the operation under sub-m Sv/h condition as a first step. Imaging performance evaluation under high radiation intensity environment was conducted using gamma radiation sources (Cs-137 and Co-60) for calibration at the Facility of Radiation Standards owned by JAEA.

3. Conclusion

Based on these results, we will develop a practical 3D radiation contaminant dispersion detection and prediction system in sub-mSv/h environment that can be used in outer part of reactor buildings of 1F site. We will also develop a 3D Cs distribution measurement method inside the reactor buildings by concentrating to highly penetrating 134-Cs gamma-rays, taking advantage of the MeV or higher gamma-ray imaging capability of ETCC Furthermore, we will investigate gamma-rays penetrating the reactor wall at 1.5 MeV or higher emitted from the debris under several mSv/h environment to realize clarification of the debris state.

W01 Quantitative Evaluation of Long-Term State Changes of Concrete Contaminated with Radioactive Nuclides Considering Actual Ageing and Deteriorating Factors

Ippei Maruyama[1, 6], Kazuo Yamada[2], Kazutoshi Shibuya[3], Yoshifumi Hosokawa[4], Yasumasa Tojo[5], Go Igarashi[6], Yo Hibino[6] and Yoshikazu Koma[7]

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

Abstract

In the decommissioning of concrete structures at the Fukushima Daiichi NPP, it is important to estimate the amounts and concentrations of concrete contaminated with radioactive nuclides. In this study, a prediction method of radionuclide penetration behavior based on penetration experiments considering the actual environment is investigated for quantitative prediction of contamination concentration distribution in concrete.

1. Introduction

Various factors of concrete such as materials used (cement type, aggregate), changes in condition (drying/carbonation, dissolution, cracks), and contacting conditions with radionuclides (contaminated water from the reactor mixed with added seawater for cooling) affect penetration behaviors of radionuclides into concrete. Based on the issues to be considered in the quantitative physical quantity prediction of contaminated concrete, we set the problem shown in Figure 1 and studied a method for predicting the penetration behavior of radionuclides that takes into account the actual environment and contamination history.

2. Results

Based on the experimental results, models were developed for cracking and pore structure changes associated with moisture migration, thermodynamic equilibrium considering the interaction of Cs and Sr with C-A-S-H (cement hydrates), and evaluation of the effects of cracking using apparent diffusion coefficients. We also attempted to estimate the contamination profile and predict the amount of contaminated concrete as shown in Figure 2.

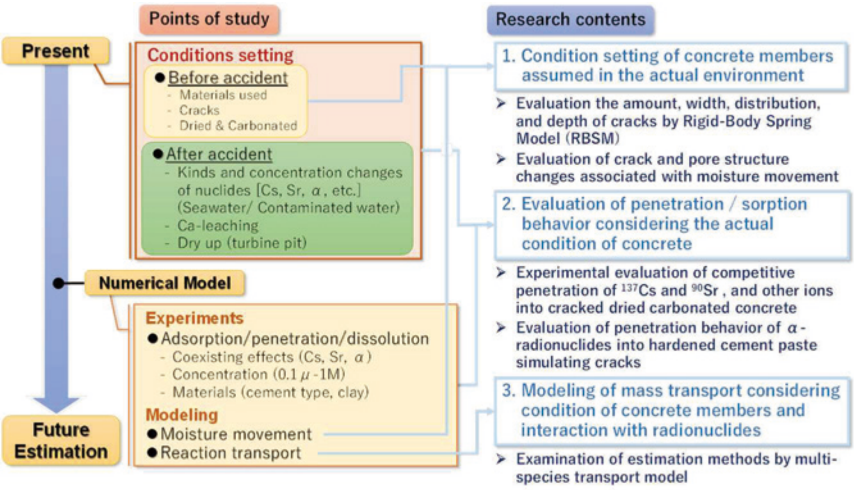


Figure 1. Framework of the study

The study is related to concrete in a saturated condition in the underground structure of a turbine pit, but further refinement of the model based on comparison with the actual contamination situation is needed, and the application of the model to concrete in different contamination situations should be expanded.

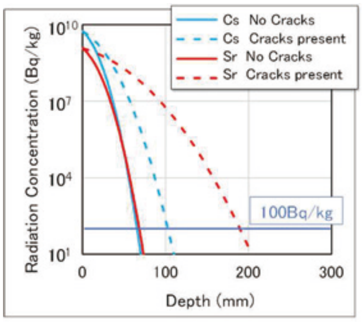


Figure 2. Estimation of radioactivity concentration profiles

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

W02

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

- Summary of the 3-year-project and toward the next step

Masahiko Nakase[1], Ryosuke Maki[2], Hidetoshi Kikunaga[3], Tomofumi Sakuragi[4], Miki Harigai[4], Hidekazu Asano[4] ,Tohru Kobayashi[5], Satofumi Maruyama[6]

[1]Tokyo Institute of Technology, [2]Okayama University of Science, [3]Tohoku University, [4]Japan Atomic Energy Agency, [5]Radioactive Waste Management Funding and Research Center, [6]Tokyo City University

Abstract We are developing the rational waste disposal concept of various wastes generated in the Fukushima Daiichi Nuclear Power Station (1F) by the “hybrid-waste-solidification” concept (**Fig.1**). The primary wastes containing radioisotope were mixed with matrix materials and solidified by Spark Plasma Sintering (SPS) or Hot Isostatic Pressing (HIP), depending on the nuclei (**Fig.2**). The well-characterized matrix materials, such as SUS and Zircalloy, make the long-term stability evaluation and safety assessment possible. In the final year, we combine our knowledge and experience in this project to evaluate waste solidification and disposal scenarios.

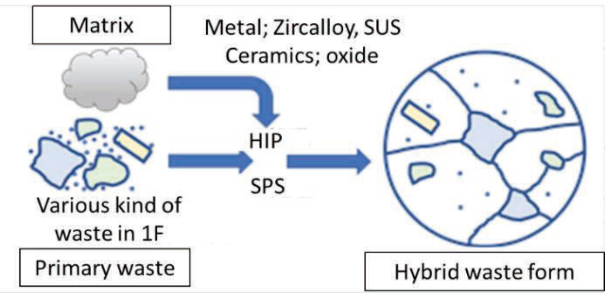


Fig.1 Schematics of HIP and SPS treatment

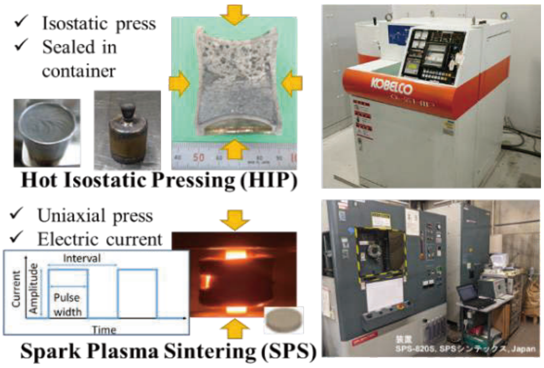


Fig.2 Schematics of HIP and SPS treatment

1. Summary of the current achievement Some essential elements for disposal, such as Iodine, a mobile element, and Actinide, a toxic alpha emitter, were targeted in the project. The combination of the primary wastes and the matrix materials was systematically tested and characterized by many analytical techniques, such as SEM, TEM, and XAFS, as well as DFT calculations. The radiation effects were tested using LINAC, AVF cyclotron, and Co-60 facilities, and leaching behavior was also studied. Finally, SUS turned out to be appropriate for the matrix. Then, a further disposal scenario study based on the reference HIPed waste with some assumptions was implemented. The inventory of the 1F was calculated, and the amount of the hybrid waste for iodine was estimated. The lifetime of the hybrid waste was set based on the known corrosion rate of the matrix, and the dose calculation after disposal was implemented based on the migration. Finally, the waste fabrication and disposal were connected, and the preliminary scenario study became possible. In the final year, all the experimental, calculation, and scenario studies are compiled, and the applicability of the new concept, “hybrid waste solidification” is surveyed.

2. Future works The 3-year project on hybrid waste solidification progressed steadily. The further studies are compiled as followings: (1) Deepen the computational approach for assessing long-term stability and radiation effects. (2) Design study on the hybrid waste that reduces the waste amount considering the disposal concept and safety assessment. (3) Scale-up, engineering demonstration for cost evaluation, and (4) Application to fuel debris.

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

W03 Application for removal of radionuclide in radioactive wastewater by the natural zeolite and reuse of used diapers

Chuan-Pin Lee[1], Shih-Chin Tsai[1,2] , Pei-Tung Hsueh[3], Neng-Chuan Tien[1,2], and Ren-Yu Pan[1]

[1] Radioactive Waste Disposal Technology Research and Development Center, National Tsing Hua University, Hsinchu 30044, Taiwan (R.O.C)

[2] Nuclear Science and Technology Development Center, National Tsing Hua University, Hsinchu 30044, Taiwan (R.O.C),

[3]Institute Nuclear Science and Technology Development Center, National Tsing Hua University, Hsinchu 30044, Taiwan (R.O.C)

Abstract

Zeolite, a natural mineral ($Na_8(AlO_2)_8(SiO_2)_{40}\cdot 20H_2O$), and used diapers including fine fibers having high selectivity toward Cs and Co for removal radioactive water from decommissioning NPP in Taiwan. The sorption of Cs and Co were investigated by ASTM batch methods and distribution coefficients (K_d) were obtained. In conclusion, there is a high effective and good performance (fibers showed high volume reduction) for decommissioning NPP by applying natural zeolite and used diapers to achieve UN Sustainable Development Goals (SDGs) to replace a lot of chemical resin and solvent.

W04

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

Kazuhiro Suzuki[1], Gary Benda[1] and Takashi Mitsui[2]
[1] WM Symposia, Inc. [2] Tousou Mirai Technology Co. Ltd.

Abstract

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

1. Introduction

The WM2024 Conference, the 50th version, will be held March 10 - 14, 2024 at the Phoenix Convention Center in Phoenix, Arizona. Conference theme is "Marking 50 Years: Proud of Our Past, Poised for the Future" with a strong Conference focus on promoting the next generation of students and radwaste management professionals, encouraging exchange of expertise to young professionals around the world. WM2024 will feature over 500 papers and more than 80 panel discussions in over 160 technical sessions, complemented by nearly 175 exhibiting companies, the industry's largest.

2. Poster

The poster provides Conference details and describes the Technical Panel, Poster and Oral Sessions, Exhibitor, Student and Sponsorship program, as well as the opportunity to network with over 2,800 industry specialists and managers from more than 35 countries and learn of trends and developments from the most senior industry managers around the world. WM2024 has special programs aimed at encouraging participation of the world's leading companies in the exhibit hall, including World Pavilion and a program for live product demonstration.

3. Conclusion

The Conference promotes, among Japanese and professionals from around the world, a broad exchange of knowledge in 12 Technical Tracks, including technologies, operations, safety, security & safeguards, waste management issues, decommissioning and dismantling, environmental remediation, advanced nuclear reactors and STEM education.

The deadline for submittal of Abstracts for WM2024 is August 25, 2023. Details are shown on the poster.

References: www.wmsym.org
www.t-g-consulting.com

W05

Development of Metal Matrix Waste form for Immobilization of Spent ALPS Adsorbents with Powder Metallurgy Hot Isostatic Pressing.

Tomofumi Sakuragi[1], Ryosuke Maki[2], Ryo Hamada[1], Miki Harigai[1], Shingo Tanaka[1], Hidekazu Asano[1,3], Masahiko Nakase[3], Hidetoshi Kikunaga[4], Sinta Watanabe[3], Toru Kobayashi[5] and Kenji Takeshita[3]

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

Abstract

Metallic matrix waste forms were synthesized to confirm immobilization ALPS adsorbents for long-term after disposal safety. The adsorbents were well confined in the stainless steel matrix without defects or pores using power metallurgy hot isostatic pressing (PM-HIP).

1. Introduction

The decommissioning of the Fukushima Daiichi nuclear power plant presents a great challenge in terms of waste management, including large amounts of adsorbents for the advanced liquid processing system (ALPS). Owing to the extremely long-half life and mobility through the repository barrier systems, the durability of the waste form especially radioiodine-bearing wastes is crucial for the safety of final disposal [1,2]. The previous work successfully synthesized the slurry waste from ALPS by PM-HIP [3]. Then, this study focuses on the immobilization of ALPS iodine adsorbents.

2. Waste Form Synthesis and Results

The ALPS adsorbents for iodine, cerium oxide and silver zeolite, were mixed with stainless steel powder at a volume ratio of 1:9. They were HIPed under 1000 °C, 175 MPa, and 3 hours. As shown in Figure 1, the adsorbents after HIPing were monodispersed homogeneously and confined in well sintered metal matrices without any defects or pores. Therefore, the PM-HIP waste form immobilized ALPS adsorbents in stainless steel matrix was successfully synthesized. The corrosion rate of stainless steel can predict a long enough lifetime of the waste form after disposal. The detailed microscopy and elemental distribution analyses will be presented in the poster.

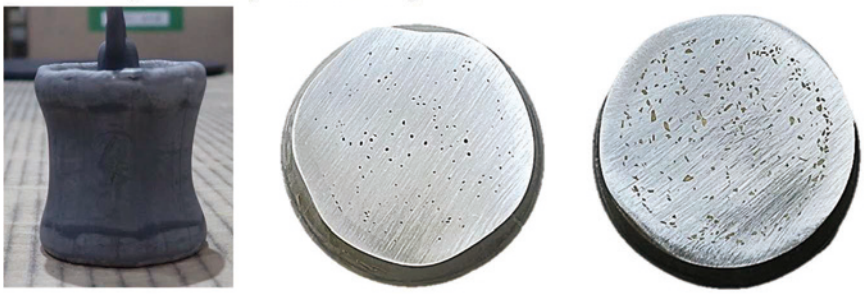


Figure 1. External appearance of HIPed waste form (left) and the cross section confining the ALPS cerium oxide (center) and silver zeolite (right) in SS matrix.

Acknowledgment

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

References

[1] M. Nakase et al., Proc. International Topical Workshop on Fukushima Decommissioning Research, 2022 (FDR2022-1038).

[2] T. Sakuragi et al., Proc. International Topical Workshop on Fukushima Decommissioning Research, 2022 (FDR2022-1024).

[3] T. Sakuragi et al., Proc. the 2023, 30th International Conference on Nuclear Engineering, ICONE30 (ICONE30-1078).

W06

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

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

Abstract

We present the features and the performance records of SIAL® geopolymer solidification technology is licensed by both the Slovak (ÚJD SR) and Czech Nuclear (SUJB) regulators, and the technology has been used successfully for 20 years. Recently, geopolymer has been noted as an immobilization technology and which shows potential of immobilizing sludge and slurry generated by treatments of contaminated water at Fukushima Daiichi Accident. More recent research on the advanced SIAL® geopolymer has focused on a flodding application implemented for combined resistance under hot water and direct contact with hot surfaces.

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® geopolymer can provide efficient and practical on-site treatment of radioactive waste streams at room temperature. 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).

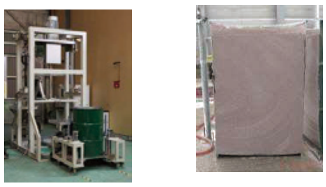


Figure 1 200L Indrum kneading machine and SIAL® solidified body of resin

Figure 2 Areva sludge and Fe and Carbonate slurry waste streams cross section observation

3. Performance Record Example

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

4. Flodding application

More recent research on the advanced SIAL® geopolymer has focused on a flodding application. Laboratory-scale trials have been proven that it is possible to apply the grout under water at 90℃ and can mature to the required strength even in contact with a surface at around 300℃.

Figure 3 SIAL® matrix surface after 1hr at 300℃

W07

Jacobs CMS Decommissioning & Regenerative Solutions (D&RS) Products & Services

Toshi Yamanouchi & Satoshi NAKASAKU

Jacobs CMS

Abstract

Jacobs CMS have developed ConnectFlow, ANSWERS & TANICS. ConnectFlow provides a flexible, multi-scale modelling environment for simulating the migration of fluids and solutes. ANSWERS is a software suite containing world class reactor & radiation modelling & analysis tools. TANICS is an automated welding and inspection solution currently in development for higher activity waste container sealing.

Introduction

Jacobs CMS provides a range of products and services globally including in countries such as Japan, Sweden, Finland, UK & USA. Three such products developed by Jacobs CMS, ConnectFlow, ANSWERS and TANICS, all of which have capability for use within the nuclear industry are to be highlighted. Both ConnectFlow and ANSWERS are used in industry globally. TANICS is currently in final development, however, has great potential benefit for the nuclear industry and packaging of higher activity materials.

ConnectFlow

ConnectFlow is a software suite that provides a flexible, multi-scale modelling environment for simulating the migration of fluids and solutes through fractures and porous rocks, enabling the integrated assessment of structural geology, engineering, hydrogeology, hydrogeochemisry and contaminant transport processes. ConnectFlow allows for the use of both the Discrete Fracture Network (DFN) and Continuous Porous Medium (CPM) models. Uniquely among subsurface flow modelling software, ConnectFlow also offers the option to construct embedded models that integrate one or more sub-models of different types. ConnectFlow is used by a range of international customers globally. ConnectFlow has been used for site-descriptive modelling for the Japanese radioactive waste disposal siting programme as well as numerous other project applications.

ANSWERS

ANSWERS is a software service utilising a suite of world class reactor & radiation modelling & analysis tools, predominantly aimed at fields such as Reactor Physics (WIMS), Radiation Shielding and Dosimetry (MCBEND and RANKERN) and Criticality (MONK). ANSWERS has been supporting customers in the nuclear industry for almost 40 years with high quality consultancy and software services, including the provision of comprehensive documentation, training courses and a detailed helpline for users of ANSWERS as part of the software service. ANSWERS codes are used for studies on a range of reactor types including AGR, Magnox, PWR, BWR, CANDU, VVER, RBMK, PBMR, SMRs, SFRs, other Advanced Reactors (AMRs) and experimental reactors.

TANICS

TANICS is an automated welding and inspection solution currently being finalised for higher activity waste container sealing. Automated welding robots are commonplace in other industries but are not normally suitable for use in the nuclear industry due to the requirement for high integrity welding. TANICS is designed specifically for nuclear industry applications delivering repeatable high integrity butt welds that are capable of remotely sealing stainless steel radioactive material containers. TANICS allows for the welding of a variety of container diameters and can perform full and partial penetration butt and fillet welds in both the vertical and horizontal orientations.

Conclusion

Jacobs CMS have developed three products to highlight, ConnectFlow, ANSWERS & TANICS, all of which can have benefits within the nuclear industry. ConnectFlow and ANSWERS have already seen high usage globally and continue to provide support to industry. TANICS is in final development stages however has great potential benefit for nuclear industry applications.

H01 Material Identification Method using Multivariate Analysis for Chemical Analysis and Application Possibility to Fragment Ion of Fuel Debris Dissolved in Wastewater

Tamao Tanji[1], Makoto Furukawa[1,2], Katsushige Fujimoto[1] and Yoshitaka Takagai[1,3]
[1]SSS. Fukushima Univ., [2]ParkinElmer Japan G.K., [3]IER. Fukushima Univ.

Abstract

This method can identify original material from wastewater in which composite solid materials samples are dissolved. It is difficult to identify the composite material such as fuel debris which is composed of melt by multiple materials. In most case of composite materials, conventional chemical analysis cannot identify the materials, even if the quantitative value agrees with the certified value of another material. In this study, stepwise multivariate analysis was utilized to identify the original material in the composite materials sample by using the slight differences of the solubility of elements contained in the material which incompletely dissolved in solutions, and then measured by inductively coupled plasma mass spectrometry (ICP-MS).

1. Introduction

Identifying origin material of multi-material melts, such as fuel debris generated by Fukushima Daiichi Nuclear Power Plants (1F) accident has been challenging in conventional chemical analysis. Thus, it was difficult to distinguish each mother materials using the chemical component values of multiple materials. In this study, multivariate analysis was utilized to distinguish between quantitative values of metal ions attributed to multiple materials. While it is very difficult to deal with a large volume of chemical data and finding something new in human senses, the small differences are recognized by multivariate analysis. In this study, a cascade multivariate analysis (stepwise analysis) composing hierarchical cluster analysis (HCA) and principal component analysis (PCA) were used for the identification.

2. Materials and method

2-1. Sample preparation

To incompletely dissolute, some series of acid solutions (2M nitric acid, 2M hydrochloric acid, and 0.4M acid mixture solutions which were contained acetic acid, boric acid, and phosphoric acid, prepared with 2M NaOH) was used as eluate. Appropriate weight of the material was dissolved in these eluates, and then the solutions were used as sample solutions.

2-2. Quantitative analysis

The 64 elements in the sample solutions were quantified by inductivity coupled plasma mass spectrometry (ICP-MS). Among of the element measured, 48 significant elements were used as data-set. The lower values than limit of quantitation (LOQ) were regarded as limit of detection (LOD), when the date was imputed into data set. The data set was then standardized by summing the values of the 48 elements.

3. Conclusion

In this study, the surface of the composite material was partially dissolved in eluate, and trace amounts of elements was quantified by ICP-MS. Then quantification data was analyzed by stepwise multivariate analysis based on HCA and PCA. The method successfully identified composite material.

H02

Development of ICP-MS analytical method for rapid analysis of radioactive ^{94}Nb in waste

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[1]Fukushima Univ., [2]Japan Atomic Energy Agency, [3]GL Sciences

Abstract

Long half-life radioactive ^{94}Nb which emits beta particle and gamma-ray is one of typical difficult-to-measure nuclide. In mass spectrometric analysis of ^{94}Nb, the separation of isobaric interferences from Zr and Mo are essential. This study succeeded in the separation of Zr and Mo using the combination of solid phase extraction (SPE) and dynamic reaction cell (DRC) in inductively coupled plasma mass spectrometry (ICP-MS). The application to cascade analysis was conducted as online analysis.

1.Introduction

In the decommissioning of Fukushima Daiich Nuclear Power Plant, long-term storage of radioactive waste is required. Then, the monitoring long half-life radioactive ^{94}Nb (2.03×10^4 y) is very important. ICP-MS has advantages rather than radioactivity measurement in the point of sensitivity. However, isobaric interference from ^{94}Zr and ^{94}Mo (natural abundance 17.4% and 9.2%) were caused in the measurement. In this study, separation from isobaric was conducted by SPE with silica gel and DRC in ICP-MS.

2. Experiment

2-1. Solid Phase Extraction

A 68-element mixture solution was passed through a column packed with silica gel (SPE column). Then, some elements selectively adsorbed on the SPE column. The concentrations before and after SPE were measured by ICP-MS. The SPE column was washed, and then Nb adsorbed on SPE column was eluted using eluate.

2-2. Isobaric separation using DRC in ICP-MS

In DRC in ICP-MS, NO, CO_2, O_2, and, NH_3 gas were used. As the result of the separation using mass shift method, Zr and Mo were separated from Nb in the use of NH_3.gas The flow rate and other conditions were optimized in this study.

2-3. Online Analysis

For automatic analysis, the online analysis was constructed by connecting SPE and ICP-MS via switching-valve.

3. Conclusion

In this study, the online cascade analysis of ^{94}Nb based on ICP-MS was conducted by combination of SPE using silica gel and DRC with NH_3 gas. The cascade separation was effectively succeed in the removal of Zr an Mo as isobaric interference nuclides .

F01

Methodology to address the liquid radionuclides source term from corium leaching after severe accident

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Abstract

The quantification of radionuclides release into the water for Severe Accident Management (SAM) is essential for carrying out safety analyses. These aim to assess the impact of their releases on the environment at short, mid and long term. This work presents the methodology developed at CEA, based on the synthesis, characterization and leaching of prototypical corium materials, *i.e. with* UO_2, in order to quantify the liquid source term from corium leaching after a Severe Accident (SA).

1. Introduction

In order to quantify and understand the radionuclides release mechanisms into the water during a SA, like at Fukushima Daiichi Nuclear Power Station, it is essential to develop a methodology based on different kinds of materials representative of corium progression. Given the difficulty of collecting and using highly radioactive samples coming from the accident site itself, the study of prototypical corium remains the best accessible route requiring:

1) **A technological platform for corium synthesis** applying conditions close to those of SA;
2) **Characterization tools for corium to specify the nature of the phases involved** (in terms of structure and microstructure, chemical composition, etc…) as well as the distribution of radionuclides into these phases;
3) And finally, **high activity laboratories to perform leaching experiments under realistic conditions** by including intense γ irradiation fields.

The analysis of leaching solutions and the characterization of altered surfaces coupled with geochemical modeling tools then makes it possible to aidentify the mechanisms accounting for the radionuclides release in solution. These tools and this approach are now accessible at CEA and are presented here through illustrations and examples.

2. Illustrations and examples of the methodology developed at CEA for prototypical corium study

The PLINIUS platform (CEA-IRESNE) at Cadarache site is dedicated to SA studies and is composed of five facilities (KROTOS, VULCANO, VITI, MERELAVA and FUJISAN). This platform allow to perform various studies such as the corium thermophysical properties measurements, interactions between the fuel and the coolant or corium spreading on concrete. In addition, the PLINIUS facilities can perform tests simulating different SA conditions using prototypical corium. Previous tests on COLIMA facility (simulating in-vessel and ex-vessel corium) have been used for leaching studies under irradiation in the ATALANTE facility (CEA-ISEC) at Marcoule site as part of a collaboration with JAEA [1]. A leaching methodology initially developed and proven for conventional spent fuels has been successfully implemented for these prototypical corium samples. The radionuclides released fractions were determined and the surfaces of the samples characterized after alteration. Geochemical modeling was also performed to interpret the results. The Fractional Release Rates of the $(U, Zr)O_2$ phase for these samples were found to be close to or one order of magnitude lower than that of conventional spent fuels leached under oxidizing conditions, but the radionuclides release mechanisms seem different. Further studies are needed to better understand the differences with spent fuel and get closer to more relevant accident scenarios.

3. Conclusion

A methodology for studying the interactions between prototypical corium and water is now available at CEA. It relies on the synthesis of samples at Plinius platform and peforming leaching experiments at ATALANTE facility.

[1] Nakayoshi, A., C. Jegou, L. De Windt, S. Perrin, T. Washyia (2020). "Leaching behavior of prototypical corium samples: A step to understand the interactions between the fuel debris and water at the Fukushima Daiichi reactors" Nuclear Engineering and Design 360: 1-18.

F02

Development of Innovative Plasma Treatment Processes for the Management of Radioactive Organic Liquid Solid Waste

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Abstract

The purpose of this poster is to present the results of the bench development of two complementary processes for the mineralization of radioactive organic liquid wastes using plasma thermal treatment.

Introduction

The operation of nuclear industry facilities and the dismantling sites of end-of-life facilities produce large quantities of radioactive wastes. Their management constitutes a major economic, environmental, societal and industrial challenge. While a large number of these wastes have a management route in existing or future disposal facility, some give rise to complex scientific and technical issues. This is the case for several types of radioactive organic liquid waste (ROLW) whose either chemical or radiological characteristics are not compatible with the incinerator operating rules (high level of ^{14}C and ^{3}H or halogen content, corrosive gas production etc.). Therefore, their treatment process remains to be defined in order to make them compatible with existing or future waste management coordination. Their volume declared through the French national inventory of radioactive waste is a few hundred cubic meters and consists of oils, various organic liquids and scintillating cocktails.

Aim of the project

In order to optimize decommissioning waste management and better anticipate related issues, French nuclear waste producers CEA and the French national radioactive waste management agency (Andra) have decided to develop innovative thermal treatment processes based on plasma technologies to operate these wastes for disposal. Thermal treatments can provide interesting benefits such as volume reduction or chemical stabilization, helping waste producer to master costs, schedule, or to optimize final volume and repository safety.

The project was supported by the French government program “Programme d’Investissements d’Avenir” whose management has been entrusted to Andra.

Processes Description

The two complementary processes assessed are using plasma technology: the IDOHL process (French acronym for “Installation de Destruction d’OrganoHalogénés Liquides”), an aerial induction plasma treatment process, and the ELIPSE process (French acronym for “Elimination des Liquides par Plasma sous Eau”), a very innovative process implementing a submerged non-transferred arc plasma.

F03

French laser – cutting R&D facilities in support of dismantling

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For more than 30 years, CEA has been developing laser–cutting technology. It designs and industrializes laser tool prototypes, integrates and implements the technology in various remote handling equipment and carries out safety and scenarios studies for nuclear dismantling application. As a result of its R&D programs, the technology has been qualified to cut up to 200mm thick stainless steel, and air–cooled and deep–gouging laser tools have been developed to tackle the challenges of 1F dismantling. Several facilities dedicated to laser–cutting R&D studies are available in France and presented in this paper. CELENA and DELIA facilities, located at CEA Saclay are dedicated to in air, nitrogen atmosphere and underwater experimentations as well as aerosols sampling and monitoring, while Hera facility at CEA Marcoule is used for advanced robotic decommissioning application, digital twin development and dismantling scenarios assessment using VR simulation in the PRES@GE^2 immersive room. Since 2014, many decommissioning studies for Fukushima Daiichi have been performed in these facilities. More recently, ONET Technologies started commissioning the TECHNOCENTRE, a new facility for industrial scale studies of in air and underwater laser–cutting application with representative test configurations. It will be inaugurated at the end of 2023.

F04

Development of Control Rod Drive Housing Cutting and Removal Technologies

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The aim of the project was to demonstrate the existence of a solution to safely and efficiently remove the Control Rod Drive Housing of the damaged Units of the Fukushima Daiichi NPS. This has been achieved through selection and testing of the best candidate technologies and the proposal of a concept of dismantling scenario considering the safety requirements, including falling objects prevention.

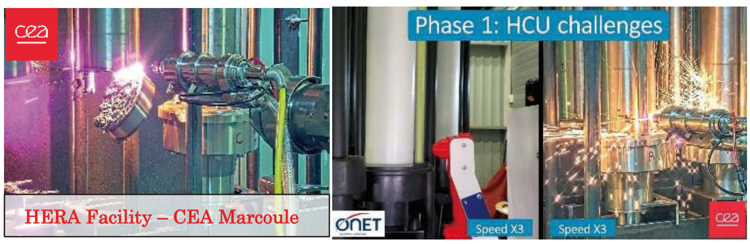
Starting from a panel of candidate cutting technologies, three technologies have been confirmed as the only suitable ones and have been tested:

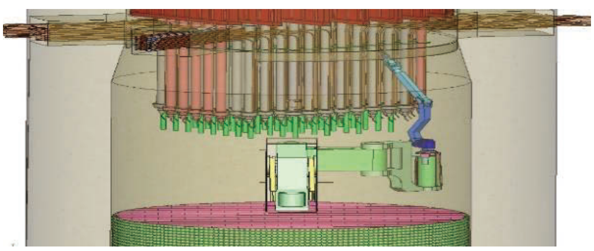
- Laser Cutting
- Grinding
- Abrasive Water Jet Cutting

Furthermore, laser cutting was identified as the best suitable technology overall and the only suitable one for some configurations.

Results of cutting capability tests with Laser and with disk cutting
Similar tests were performed with Abrasive Water Jet Cutting

Operability tests were carried out on a representative mock-up: laser cutting with straight laser tools and accessibility with angled laser tools. Remote implementation of fall-prevention systems has been carried out as well.





The feasibility of a cutting & removal scenario was established, with a segmentation plan and assessment of the duration of operations.

F05

Fuel Debris Dust Dispersion Suppression System Technologies

Rémi Delalez[1], Gaétan Bigoin[1], Thomas Da Silva[1], Romain Berlemont[1], Antonin Bouland[1], Emmanuel Porcheron[2], Thomas Gelain[2], Yohan Leblois[2], Viviane Bouyer[3], Christophe Journeau[3], Ioana Doyen[4], Lucas Brizzi[4], Romain Garnier[5], Henri-Noël De Grande[5], Franck Peysson[6], Jean-Patrick Thomé[6]

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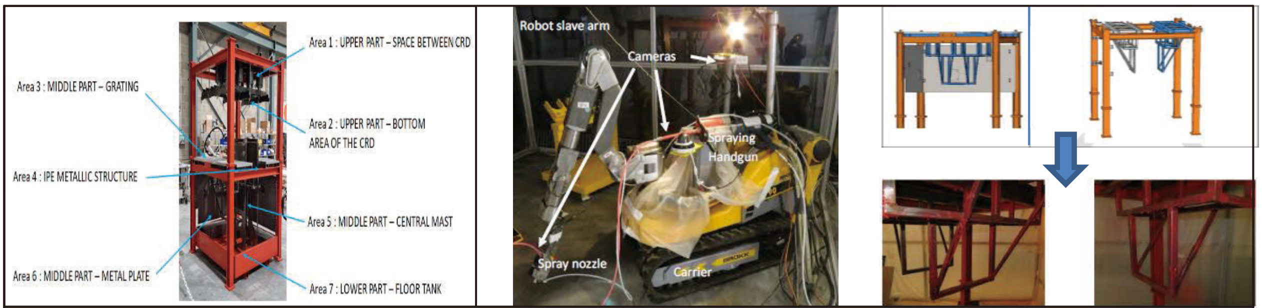
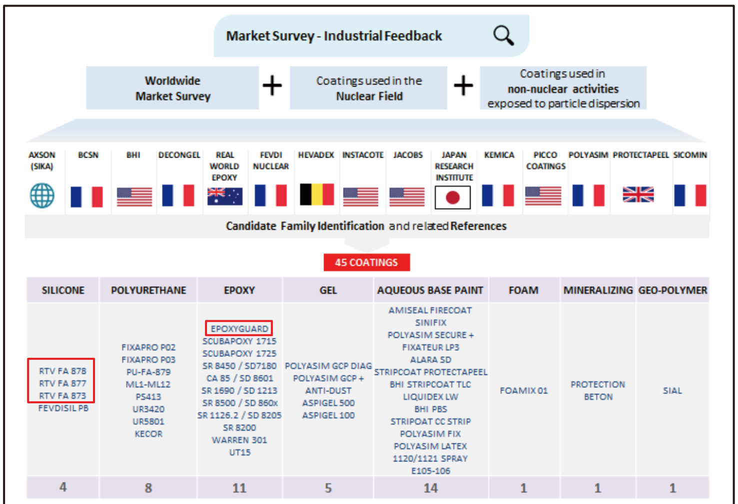
6BCEN, Driection Technique, Pôle Résines, 208 avenue du 8 mai 1945, 69142, Rillieux la Pape Cedex, France

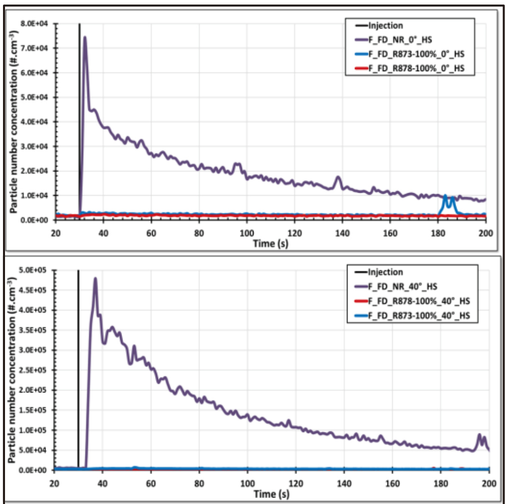
This study has demonstrated that systems can be implemented inside the pedestals of the damaged units of Fukushima Daiichi NPS to suppress the Fuel Debris dust dispersion risk. Application of coatings with remote controlled means and consideration for different areas (metallic frames, under dripping water, underwater, etc.), has been proven to be efficient. Impacts of use has been assessed.



A specific mock-up representative of the inner parts of the pedestal has been manufactured. Coating tests in every area, including tests underwater, have been carried out. Applicability and efficiency have been proven.



Benchmarking over 45 coating solutions has been carried out. Only the 4 best candidates were finally completely tested.



Measurement of coating efficiency: mass concentration of airborne particles resuspended with (blue and red) and without (purple) coating.

F06

Solutions for D&D Legacy Waste Stabilization: Dem&Melt, Cementation, Solidification with Resins

D. OGAWA[1], N. BRETON[1], V. JANIN[1], R. DIDIERLAURENT[1], F. PRUDHOMME[1], I. GIBOIRE[2]
[1] **ORANO,** [2] **CEA**

Abstract
As the D&D of UP2-400 progresses, various types of Legacy Waste are retrieved for treatment in existing or newly built workshops. Among the retrieved waste are considered: sludge with widely varying chemical properties, very different types of solid waste that must be immobilized, and many others.
Based on its strong experience in waste treatment, **Orano has already developed,** or is currently developing and qualifying **different flexible stabilization processes**, many of which could be applied to **waste in 1F**. Three of those processes are described below: **Dem&Melt**, **Cementation**, and **Solidification with Resins**.

1. Dem&Melt (In-can Melting)

Dem&Melt is a robust, simple, and versatile technology (codeveloped by Orano, CEA and ECMT) for the treatment and conditioning of a wide range of waste, based on LFE from 40 years of **High Activity Vitrification**. It implements a modular design adapted to nuclear requirements, with a canister directly used as a melter, improving durability while achieving **high rates of waste incorporation** and **good performance of the wasteform**. Low volatility of radioelements is also achieved.

Dem&Melt Principles

Full scale pilot tests, with challenging waste of various types (alpha liquid, fission products, zeolites, sludges, etc.) have already been performed, confirming its adaptability to existing secondary waste in 1F. The DEM&MELT technology would also be a relevant solution to process FDR secondary waste as it is a **very high-activity-compatible equipment** and as it provides durable containment of radionuclides & waste stabilization avoiding among other production of H2 by radiolysis.

2. Cementation


HRB Lost-Paddle Mixer (1000L)

Slurry/Sludge Cementation is a flexible low-temperature process, largely used in nuclear industry, and adapted to various types of waste. **Orano's Research and Development Center (HRB)**, has significant experience in developing various **hydraulic binders**, adapted to a **wide range of waste** (resins, sludge, fine particles) to strict final disposal regulations. It also designs and **qualify the industrial process** to cement large amount of waste in **hot cells**.

Cementation processes for Legacy Waste Retrieval have already been developed and are on-going commissioning tests in La Hague. Processes can be adapted (hydraulic binder) for **slurry/sludge in 1F**, directly or after centrifugation, to safely and durably stabilize the waste.

3. Solidification with Resins

Epoxy resin solidification is a flexible low-temperature process that can be implemented for various waste typologies. Orano has recently been developing a **highly resistant, hydrophobic** and self-placing blocking **epoxy matrix**, compatible with most types of waste (**including reactive metals and some chemically reactive species**, that can interact with regular cement mortar.


200L Epoxy Resin Solidified Package

200L scale tests have already been successfully performed with epoxy resin on Aluminum waste. Full-scale tests would need to be performed for adaptation to 1F, as well as industrial process design and qualification.

Conclusion

Orano has significant experience in developing and qualifying innovative solutions for the stabilization of variable radioactive waste to strict regulatory requirements. In view of the **solid waste (incl. slurry/sludge) treatment challenges faced in 1F** in the next few years, **Orano could provide technical solutions and support to TEPCO**.

F07

Reactor D&D: Recent successes and achievements in the U.S.

D.OGAWA, S. KRAMER, M. LUCAS, S. GUILLOT

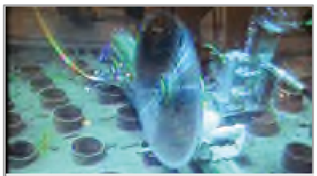
ORANO

Abstract

With over 10 projects to date between the U.S. and Germany, Orano has gained a comprehensive expertise in successfully dismantling the full spectrum of BWR and PWR reactors, including the segmentation, packaging, transportation, and disposal of the Reactor Vessel (RV) and Reactor Vessel Internals (RVI). In the US, Orano recently completed the first full segmentation of a commercial BWR (Vermont Yankee) and is currently dismantling the Crystal River 3 PWR implementing an innovative segmentation solution. This experience could be relevant for upcoming D&D operations in Fukushima-Daiichi.

Vermont Yankee (620 MWe BWR) – RV / RVI

The accelerated dismantling started immediately after the spent fuel was off-loaded and transferred to dry storage, and Orano completed the segmentation, packaging and disposal of the RV and RVI in less than four years.



Circular Saw with Custom Clamp / Feed System

Orano deployed field-proven segmentation systems and robust mechanical and thermal cutting technology tailored to the complex BWR components geometrical constraints. The steam dryer was segmented in air without grouting, which was a first for a BWR in the U.S.

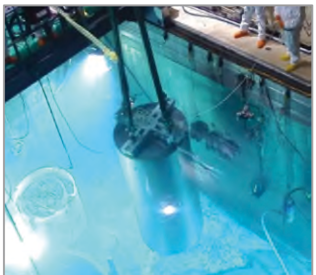
MP-197HB

Orano repurposed and licensed spent fuel transportation and dry storage systems to package, transport, store, or dispose of reactor irradiated materials. The increased package payload of the Orano TN MP197HB transport cask reduced the number of LLW packages to 38 for the entire Reactor Vessel and Internals.

Crystal River 3 (860 MWe PWR) – RV / RVI / Reactor Coolant System

In October 2020, NorthStar Group Services and Orano partnered to take over the plant license and operations from the utility Duke Energy, to perform the full dismantlement of the Crystal River Unit 3 Nuclear Power Plant.

Orano is implementing its patented “Optimized Segmentation” solution, consisting of using the RV to receive the RVI segments from which the Greater-Than-Class-C (GTCC) components were extracted and packaged in two Orano TN NUHOMS Rad Waste Containers (RWC) for onsite storage. The non-GTCC internals components are consolidated and encapsulated within the RV and the resulting monolith is segmented and conditioned in three custom individual packages for off-site transportation and disposal. Orano innovative “Optimized Segmentation” and packaging plan removes and packages the entire RV and RVI components in a total of 6 containers versus an average of 50-80 packages for the traditional approach of full segmentation.



Loaded RWC liner ready for packaging

Conclusion

To pursue the full-fledged fuel debris retrieval at Fukushima Daiichi in the decade to come, the operator TEPCO will have to develop an expertise in the full spectrum of dismantling activities, combining Reactor D&D and Fuel Cycle D&D, integrating continuous improvement process in its methods.

Building on proven technologies from past experience and extensive lessons learned, Orano continuously innovates to expand its toolbox and solutions offering to resolve the industry’s challenges with uncompromised safety performance, and is ready to support the Fukushima Daiichi cleanup and dismantling efforts.

F08

Innovative solutions for Full-scale Fuel Debris Retrieval (FFDR): New Lateral Opening for RPV access and hot cells conceptual design for Full-Scale FDR through PCV side access

D. OGAWA, V. JANIN, L. DAVID, A. ROLLET [1]
V. BESSIRON, G. BERGER, P. DE VITO, C. BERGER, S. MENOU, M. GIRAUD [2]
[1] Orano [2] Framatome

Abstract

To pursue the full-fledged (or full-scale) fuel debris retrieval at Fukushima Daiichi, several methods to access RPV are under development. On the one hand, a lateral opening toward the Reactor Pressure Vessel is made possible by integrating water jet cutting and demolition tools combined with a ferro-concrete debris suction system. On the other hand, hot cells conceptual design for Full-Scale FDR through PCV side access is completed. Both solutions are designed for 1F needs based on French technology.

1. New Lateral Opening for RPV access

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

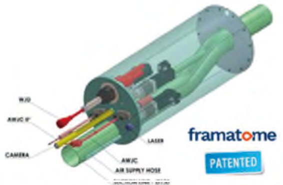


Fig 1. Framatome combined tool

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

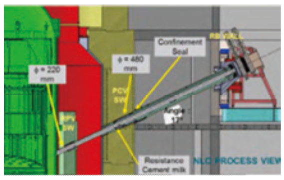


Fig 2. Orano-Framatome facility

Tests were conducted to confirm the feasibility of the system.

By optimizing the functions, a miniaturization of the system could be achieved, making the site implementation easier. This is the next R&D step considered by our team.

2. Cells conceptual design for Full-scale FDR through PCV side

The conceptual design includes hot cells (see figure 3) ensuring Full-Scale FDR robot arm access to PCV, maintenance of the robot arm, extraction, sorting and packing of and Fuel Debris (High Level Wastes) into shielded cask, and of Intermediate/Low-Level Waste into dedicated containers. Contamination control, decontamination systems, maintenance solutions including emergency situations are anticipated, based on proven solutions with operating experience feedback from French and foreign facilities.

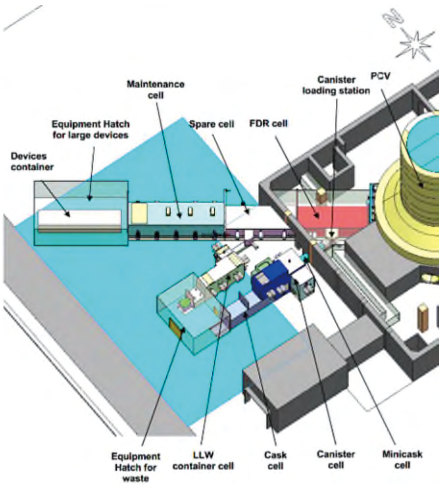


Fig 3. Fuel debris retrieval facilities overview

Conclusion

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

Those studies were performed in the frame of IRID "Development of Technology for Gradually Increasing Retrieval Scale of Fuel Debris" and "Development of Technology for Investigation inside Reactor Pressure Vessel (RPV)" by the subsidy of the projects of METI (Ministry of Economy Trade and Industry).

F09

Anemone retrieval tool - An innovative gripping technology developed for the recovery of fuel debris in 1F reactors

D. OGAWA, K. LE FLANCHEC, M. NAULIER, G. LEVAVASSEUR, F. PRUDHOMME [1]

[1] Orano

Abstract

Orano develops various tools for **investigation and sampling**, designed to be remotely operated and maintained in harsh environment, for its needs and for its customers'. The **Anemone tool** is the new generation of these tools. The Anemone tool is inspired from the sea anemone behavior. It can catch, imprison, and recover any kind of solid element no matter its shape, material, or density.

- Anemone tool has been used for the first time in nuclear industry on Orano La Hague plant to collect highly radiating graphite samples, in April 2022.
- Anemone tool has been selected to recover 1F reactors fuel debris. A specific development for this application is in progress.

1. How it works

- The tool is made from a rigid body, and a flexible head equipped with tentacles designed to grip and trap any type of object or material.
- The gripping action is provided by retraction of the anemone head.
- The anemone is supplied by low-voltage power.
- It is made for remote use either plugged on a pole, mounted on a remotely controlled arm or even equipped on a ROV.

Anemone tool head retraction principle

QR code to demo video

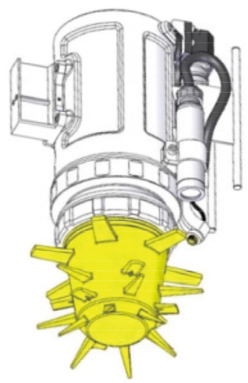
2. Many Advantages by conception

Anemone can be deployed in air or underwater.
It is very efficient even if the target to recover is covered with sand, rubbles or sludge.
The material resists cuts, tears, and high irradiation.
The dimensions and characteristics of the tool can be adapted to needs to fit in any situation.

3. Anemone tool first deployment

The Anemone solution has been used for the first time in the tank #41 at the silo 115 in Orano La Hague plant in 2022.
The Anemone was plugged on a pole and allowed the recovery of 9 irradiating graphite samples from the pit.
This was a high-risk operation due to safety conditions, performed successfully thank to the Anemone tool.

Graphite debris recovery in La Hague



4. Development to be used on 1F fuel debris retrieval programme

Since 2018, Orano is working with Mitsubishi Heavy Industries on Fukushima-Daiichi fuel-debris recovery program. The Anemone tool has been selected and is being specifically adapted to be equipped at the end of an access arm to recover fuel debris samples. The current adaptation are the following :

- upscaling of the Anemone head to meet 1F criteria,
- adaptation of the connection between the access arm and Anemone tool,
- implementation of laser pointers and rad-hardened camera for monitoring,
- making the Anemone maintainable in full remote operation.

The detailed design of the tool is on going with the objective of manufacturing and testing in the upcoming years towards the application on-site.

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

F10 Managing large scale retrieval and decommissioning programs as a Nuclear Operator and Site Owner

D. OGAWA, JM. CHABEUF [1]

[1] **Orano**

Abstract

Very few nuclear operators face the triple challenge of managing **high active nuclear material retrieval** and **full-scale decommissioning** on **constrained nuclear sites**. TEPCO is one, Orano is another one.

Managing such an endeavour requires to establish and reassess constantly a **global strategy** which combines and balances multiple factors which can be summarized as follows:

- Hazard reduction and site Safety in normal and abnormal conditions
- Funding mechanisms and best uses of available funds
- Technology development
- Waste and effluent management
- Internal and external resources
- Stakeholder management

Weaknesses or failures in any of those factors inevitably impacts the program delivery and requires a re-evaluation of the overall strategy.

Orano has been managing such program for nearly two decades now, most notably on the site of La Hague in North-Western France, and has thus, over the years, developed specific processes, mechanisms, and competences to ensure the individual and collective progress of these factors.

● Regarding **Safety and Hazard reduction**, the French regulator and Orano developed over time a specific regulatory framework which ensures maximum Safety while allowing flexibility to account for the specific challenges of decommissioning and legacy waste retrieval. Orano has prioritized the retrieval of its most hazardous legacy waste, the emblematic illustration being silo 130 on La Hague UP2-400 (presented on another abstract).

● To make the best use of the available funding, Orano has developed a **full cost approach to decommissioning**, where cost estimates are integrated with detailed schedules and risk analysis, allowing to evaluate the progress on a real time basis, and compare options or alternative scenarios in terms of impact on the overall programme.

● Regarding **Technological development**, Orano has a long experience of designing, constructing and operating complex facilities, which constant improvement are are integrated in the development of solutions for decommissioning and legacy waste retrieval, especially for their operability.

● As for **Waste management**, Orano has established a comprehensive roadmap to ensure that all the waste present on its sites will eventually be conditioned in a safe and economically viable manner. Such solutions range from the elaboration of specific glass matrices for specific high active effluents, to the creation of a recycling route for clearance metal, and the development of conditioning solutions for asbestos in old plants.

● Regarding **resources**, Orano has developed over the years comprehensive training programmes for the entire range of competences needed in decommissioning and waste retrieval activities. Orano has also established robust and proven mechanisms to ensure that solutions delivered by the supply chain meet the projects' objectives and will successfully operate once commissioned in challenging operation environment.

● Finally, regarding **stakeholders**, Orano is engaged in constant dialogue through the "local information committees" (CLI) in place on our nuclear sites, and has successfully brought to green-field or brown-field end state several dozens of former mines as well as former front-end nuclear sites (SICN in France).

Conclusion

The comprehensive experience gained by Orano as owner-operator-decommissioner of its nuclear sites is unique and the goals and challenges are quite relatable to TEPCO and Fukushima-Daiichi situation.

F11

Hot Cells and Remote Equipment for D&D Legacy Waste Retrieval, and similarities with Fuel Debris Retrieval in 1F

D. OGAWA, N. BRETON, V. JANIN [1]

[1] **Orano**

Abstract

As the D&D of La Hague UP2-400 progresses, Orano has **designed, built, qualified, and started operating several facilities for Legacy Waste Retrieval**. These projects are carried out over long period of time and involve the handling of highly radioactive waste in hot cells with remote equipment means. This requires the development and qualification of dedicated retrieval, sorting and treatment processes, taking into consideration the waste flow from early design stage. The **significant design constraints** such as technical challenges due to the high activity, adaptation to peculiar waste specifications, installation in old facilities, etc. **bear strong similarities with ongoing requirements for FDR** facilities in 1F.

1. Silo HAO

The "Silo HAO" is a pit in which has been temporarily stored legacy waste resulting from reprocessing of spent LWR fuels between 1976 and 1990, principally hulls and end-pieces. The **design constraints** of the new retrieval workshop (RCD HAO) **are particularly stringent** and similar to fuel debris retrieval workshop requirements:

RCD HAO Hot Cell Layout

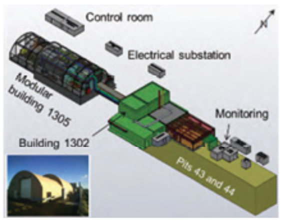
RCD HAO Hot Cell

- **The hot cell needed to be built in an operating nuclear facility** (ventilation, access limits), on top of the existing silo (over 1700 tons added), to modern seismic standards, and in a very dense nuclear environment.
- The waste flow generated from retrieval operations must **reintegrate the current treatment flow** of La Hague plant for hulls and end-pieces (compaction).
- A **dedicated lost-paddle cementation** unit was added during the design to accommodate fine particles from shearing or dissolution of spent fuel (**sludges**).
- **Retrieval of waste is performed remotely** from workstations with cameras or leaded-windows. Waste is sorted by operators with remote manipulators, and **fissile matter is accounted for with an ANI station**.

The RCD HAO Workshop is currently in inactive testing, and operation is expected to start in the next few years.

2. Silo 130

The "Silo 130" is a pit in which has been temporarily stored legacy waste resulting from stripped UNGG (Graphite GCR) spent fuel between 1969 and 1984. The purpose of the workshop, implemented in a **modular metallic/textile structure**, is to retrieve and condition waste from the silo in ECE drums (**over 700 drums will be produced in a decade**).



Silo 130 Hot Cell Layout

Silo 130 Sorting Station

Legacy waste are retrieved and crushed with a grapple, and then transferred for sorting. An **automated laser sorting station** has been developed to optimize the filling of drums, **following requirements on hydrogen release and magnesium content** regarding limits per drum (a Sileane-Orano development). Drums are then transported for interim storage.

The Silo 130 Retrieval Workshop has been successfully operating since 2019.

Conclusion

Orano has significant experience in remotely operated hot cells for waste retrieval, with its Silo 130 Project (already operating), and Silo HAO Project (in pre-commissioning tests). In view of the technical challenges faced by TEPCO for FDR, and similarities with Legacy Waste Retrieval projects, Orano can provide valuable assistance to TEPCO in the future from conception of the solution to design and implementation.

F12

NUHOMS® MATRIX system: Horizontal Storage System for Used Nuclear Fuels, Fuel Debris and Nuclear wastes applied to Fukushima

D.OGAWA [i], P. NARAYANAN[i], H. AKAMATSU[ii] , Y.YAMADA[ii]
[i] Orano, [ii] Transnuclear, Ltd

Abstract

The **NUHOMS® MATRIX Horizontal Storage Module** (HSM) is a two-tiered **concrete storage overpack system for dry shielded canisters** (DSC) **to store used nuclear fuels, fuel debris, and nuclear wastes**. It is an evolutionary design of the existing NUHOMS® HSM. MATRIX is characterized by its small footprint, storage efficiency, seismic resistance, and self-shielding. Combined with Orano's TMI-2 fuel debris canister design knowledge and the recent successes in U.S reactor D&D projects, the NUHOMS® MATRIX is an **optimal and cost-effective solution being considered for the Fukushima nuclear fuel and fuel debris safe long storage**.

Dual level horizontal concrete module

1. The NUHOMS® experience with fuel debris

The only example of dry storage of fuel debris took place in the USA for the TMI-2 reactor. Debris is stored at the Idaho National Laboratory (INL) within the NUHOMS® system which consists of a metal canister and a horizontal concrete storage overpack. Apart from this technology, all other storage method to be used for 1F fuel debris would be a First Of A Kind (FOAK) to be achieved. The NUHOMS® horizontal concrete storage system went through several developments and its performance was significantly enhanced since its adoption for TMI-2 melted fuel debris. For instance, the use of a dual level structure for the MATRIX would reduce considerably the footprint compared to the system used in TMI (45% reduction).

2. The NUHOMS® experience with used fuel

Orano TN's NUHOMS® systems have securely stored used nuclear fuel in the United States for more than two decades, with installations at over 30 sites around the country representing more than 45,000 stored used fuel assemblies. The NUHOMS® MATRIX HSM system has been approved by US NRC and was selected by the US Utility, Wolf Creek Nuclear Operating Company (WCNOC), for loading in 2020.

3. The NUHOMS® experience with wastes

The NUHOMS® MATRIX System is also capable of storing the NUHOMS® Radwaste Canister (RWC), a canister that is used to store dry irradiated and/or contaminated non-fuel-bearing components. The NUHOMS® system dry shielded canisters can be adapted internally to safely store many forms of nuclear waste (from Greater than Class C (GTCC) wastes to encapsulated fuel debris), all of which could be stored in the NUHOMS® MATRIX HSM array. This technology has been applied in recent Vermont Yankee and Crystal River NPP decommissioning projects with standard NUHOMS®, and its flexible design makes it a universal overpack that can meet nuclear operators' needs and requirements.

Conclusion

More than 1,200 systems were loaded in the USA, all associated hardware, auxiliary systems, procedures for loading/unloading, transfer, and monitoring are well developed. The small footprint and multiple uses makes the NUHOMS® MATRIX perfectly suited for decommissioning activities. The "Japanization" of the whole system, involving local companies, operators and manufacturers is considered as one of the key success for its implementation in 1F, together with the completion of the safety demonstration regarding seismic resistance.

Loading operation in horizontal

F13

Sustainable treatment, clearance and safe recycling of retired large components

Arne Larsson[1]

[1]Cyclife Sweden AB Groupe EDF

Abstract

Cyclife Sweden AB is the experienced supplier in characterisation, transportation, and treatment of large contaminated components. Its approach aims to reduce decommissioning schedule, minimise waste volume, and optimise metal clearance for recycling. 100+ large components have been treated and several are contracted for treatment the coming years. Cyclife provides decontamination and melting services for pre-treated metals delivered in ISO containers.

The EDF Group company Cyclife Sweden AB has for more than 35 years provided metal treatment services to the domestic and international nuclear industry with the aim to reduce the volume to be disposed as radioactive metals and waste, and at the same qualify up to 95% of the treated metal for safe recycling. Retired large components, qualified for shipment and treatment, are size-reduced and decontaminated in an efficient way without the generation of residues with complex properties.

The approach implementing a safe and fast evacuation and an efficient waste management is highly evaluated by most licensees. Examples of retired large components shipped for treatment the last decades are BWR turbines, BWR heat exchangers, Magnox boilers and ducts as well as PWR steam generators. In the coming decade, additional dozens of such components will be shipped to the Swedish facility for volume reduction and safe recycling of the produced metal ingots. The ingots are recycled to the conventional industry for the manufacturing of new products.

Retired large components may be decayed or chemically decontaminated, if needed to comply with the acceptance criteria for treatment, prior to delivery. In addition, the material arising from the segmentation of the retired large components are mechanically decontaminated by abrasive blasting, to secure clearance.

The following will be covered in the poster:

- Radiological characterisation, planning and transport
- Acceptance criteria for treatment
- Pre-treatment and melting
- Post treatment of residues
- Clearance and safe recycling
- Expected results

F14 Methodology for an optimised design of a waste treatment facility

Claire Vasseur[1] and Jean-Luc Falcone[1]

[1]Cyclife Engineering Groupe EDF

Abstract

Cyclife Engineering is dedicated to managing nuclear decommissioning projects. Its knowledge of PWR key components and of the associated dismantling activities allow to understand the complete environment and to define an approach for the design of a waste facility into four stages: data collection, process design and waste routes, HVAC, fluids, I&C, CW; and preparation of next steps.

Cyclife Engineering is an engineering company within Cyclife EDF group specialised in nuclear decommissioning and dismantling, and radioactive waste management that benefits from the experience of the decades of EDF's skills and assets in nuclear. Created in late 2019, the company performs preliminary or detailed design studies especially in complex waste storage and/or waste processing facilities.

The designing of a new waste treatment installation can have various purposes: cutting and sealing on site for further transport to another treatment facility, cutting and melting for volume reduction at final repository, and sorting, cutting, and melting to free release.

The first step of a new design project is to understand clients' needs and define all missing inputs that are of particular importance for the design: aim of the new facility, the type of objects to be treated, the time constraints, the implementation constraints (extension of an existing facility, green field, etc.), the regulatory constraints and the budget. It is also key to integrate previous work at earlier stages of design and understand what is mandatory and what is subject to potential optimisation. Depending on the level of detail (basic or detailed design), a functional analysis can also be performed.

Once this frame is set, the actual design work can begin. The process of treatment is designed by Cyclife's mechanical engineers, supported by radioprotection and layout engineers. Cutting is almost always implemented. Key parameters to choose cutting techniques are the size of component, the thickness and type of metal, its radioactive inventory, and the time constraints. Very quickly and when required, and upstream of the methodology, the hot cutting can only be implemented on slightly contaminated or uncontaminated pieces, to avoid contamination spread. Once the process and layout are defined, shielding needs and airborne contamination are calculated. Thus, civil work, HVAC, fluid systems and electricity can be designed. A 3D model of the facility is drawn and updated along the design. It serves to verify the interfaces between equipment, generate guide plans, but also bill of quantities for cost estimate. Cost estimate and schedule are finalised at the end of the project but need to be updated regularly during the project to fit and get a cost-efficiency.

Cyclife Engineering plans to develop its activity towards procurement and supervising of on-site construction. The design aims to reach the right level of details to build the facilities, and make sure the future facility will perform as required, by integrating the client constraints and the design/operating experience as Cyclife group. The proposed allotment of future contracts and a good risk analysis are key to determine what need to be detailed and secured and what can be designed directly by a sub-contractor.

The poster will present Cyclife Engineering's design methodology based on its clients 'needs as well as a few examples.

GB01

Retrieval and management of radioactive materials and wastes supporting facility decommissioning

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

Cavendish Nuclear Japan KK [1], Cavendish Nuclear Ltd [2].

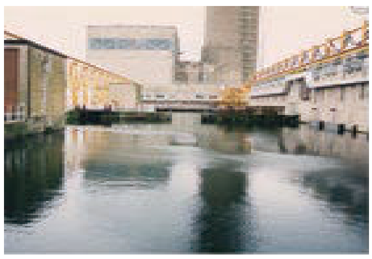
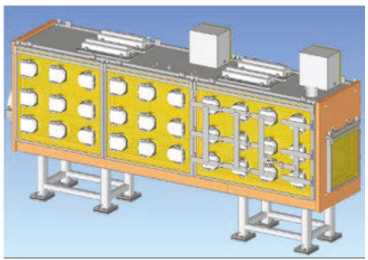
Abstract

As decommissioning and waste management challenges grow in complexity, the innovate application of robust and proven technology is employed to create reliable, remote operating solutions that move humans away from harm.

1. Introduction

The poster shows examples of recent projects that have been undertaken by Cavendish Nuclear, in support of decommissioning and material conditioning for customers in the UK.

2. Projects requiring the removal, handling and treatment of radioactive materials and wastes.

Pile Fuel Cladding Silo (PFCS) – Cavendish Nuclear was tasked with delivering plant and equipment capable of accessing historic Intermediate Level Waste in legacy PFCS at Sellafield, removing it and transferring it into containers for safe interim storage. The project covered detailed design through to manufacture, works testing at our Rosyth testing facility, transportation to Sellafield, installation, and commissioning.	
Berkeley Active Waste Vaults - The retrieval programme comprised several discrete projects for the design and installation of mechanical handling and processing equipment required to retrieve legacy Intermediate Level Waste from three subterranean vaults, at a Magnox site. Cavendish Nuclear was tasked to deliver plant and equipment capable of retrieving, processing, and packaging the waste for safe interim storage.	
Sellafield B29 - B29 is a redundant fuel receipt, storage and decanning facility situated between the Windscale piles on the Sellafield site in Cumbria. Cavendish Nuclear designed, supplied, manufactured, constructed, installed and commissioned a suite of three projects to facilitate the removal of the pond sludges including a Local Effluent Treatment Plant.	
Sellafield Retreatment Plant Project – The mission is to provide a facility that will receive special nuclear material from existing stores on the Sellafield site and process it into a form suitable for safe and apposite storage until 2120.	

GB02

Decontamination & Decommissioning Technologies for the Civil Nuclear Industry

Michael J. Anderson, Victoria E. Anderson-Matthew,

Kazushi Watanabe, Sarah A. Peirce

Innovative Physics Limited

Abstract:
The nuclear industry faces numerous challenges in decommissioning, including improving efficiency and safety while reducing costs. Advanced technologies, including robotics, sensors, and artificial intelligence, have the potential to address these challenges by automating hazardous tasks, improving accuracy, and reducing human error.

Introduction:
Below outlines some examples of technologies that Innovative Physics Limited (IPL) have developed.

Sort & Segregation of Nuclear Waste
Utilising advanced computer vision, machine learning, and robotic technology to automate the sorting and segregation of nuclear waste. The system accurately identifies and sorts different types of debris while improving efficiency, reducing costs, and enhancing safety. Notably, the automated system eliminates the need to expose workers to harmful radiation. The future potential of this technology is significant as it can be applied to other industries, such as waste management, mining, and construction.

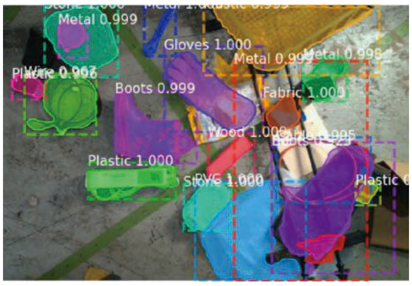


Figure 1: Object and Material Identification

Gamma Imaging Systems

Decontaminating an area containing nuclear waste is difficult due to the intangible nature of radiation. Working closely with Japanese partners, IPL designed and developed a gamma imaging system capable of showing “hot spots” of radioactivity. The systems provide an image/video of a large area, allowing workers to quickly and remotely observe where radiation hot spots are located and determine, within minutes, the radioisotope being emitted.

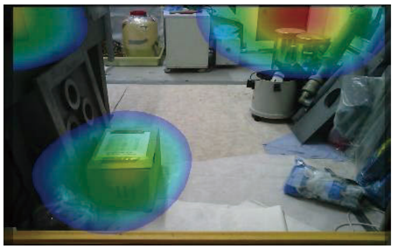


Figure 2: Gamma Imaging

Neutron Detection
Decommissioning planning requires a comprehensive mapping of the radiological environment. Importantly, the location of the fissile material is required, i.e. the neutron field. This allows path planning for removing such material while avoiding criticality events. Custom solid-state neutron detectors using a semi-conductor deposited with Boron-10 (B10), which show a high gamma radiation tolerance and gamma rejection ratio (Co-60, Cs-137up to > 1/106cps/cm2s) to enable monitoring of neutron flux in highly radioactive environments, such as criticality monitoring, emergency management, core monitoring. The novel neutron detector architecture is modular and thus can be integrated into many applications.

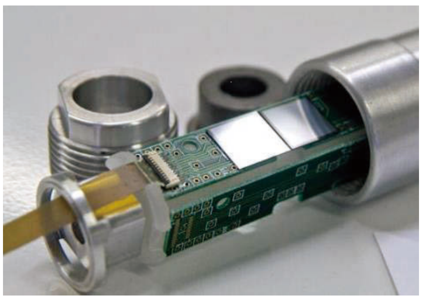


Figure 3: Semi-Conductor Neutron Detector

Very Low-Level and Low-Level Waste Management
To measure the surface contamination (gamma dose rate) and the isotope of radioactivity. Large scintillators and the relative movement of sensors and objects are used to identify the location of radioactivity. The system uses Time Delay Integration (TDI) techniques to provide additional security and variable speed detection.

GB03 Efficient encapsulation of hazardous orphan wastes using geopolymer matrices

Richard White[1], Stuart Maclachlan[1], Marta Fedorciuc-Onisa[1]

[1] Lucideon, Queens Road, Penkhull, Stoke on Trent, ST4 7LQ, UK

Abstract

Intermediate and high-level active wastes are generally encapsulated in containers using a solidification process that often involves cement grouting for long term storage. Optimal loading of waste can have a significant impact on the volume of final material to store. Cement grout is widely used as a flexible system for encapsulation, although some materials are particularly difficult to effectively handle.

1 Introduction

At Lucideon a number of geopolymer based formulations for incorporating high loadings of some of the more problematic waste streams have been established. These materials have been selected to include ones that have proven harder to handle. Selected waste streams were mineral oil, graphite and magnesium hydroxide sludges.

2 Results

To date work has focussed on developing robust formulations using non-active but representative stimulant wastes. Characterisation has focused on evolution of mechanical properties over periods of up to 3 months, along with leaching rates of potential deleterious components.

Formulations have been developed which have the potential of incorporating significantly higher waste loading than current cementitious systems have been developed in each case. The rheology of the mixes differs significantly from traditional cement-based solutions, which enables some significant performance improvements and ensures better and more uniform and completed encapsulation.

Preliminary irradiation work has been conducted showing suitable performance of the systems.

3 Next Steps

Next stages of the work will be further performance characterisation, increase scale of process before moving to active trials.

J01

BWR lower plenum damage by reaction between metallic debris and structural materials

Takumi Sato[1], Kenta Shimomura[1] and Yuji Nagae[1]
[1]Japan Atomic Energy Agency

Abstract

During the severe accident at the Fukushima Daiichi Nuclear Power Station, a molten metal pool formed by an analytical approach might contribute to the initial failure of the lower plenum. To understand this failure behavior, high-temperature reaction tests were conducted using simulated metal debris and a test bundle that simulated the structure of the control rod drive mechanism in the lower plenum.

1. Introduction

In the Fukushima Daiichi nuclear power station accident, core materials melted and fell into the lower plenum. In units 2 and 3, it is considered that the molten metal pool formed. The metal pool could react with structural materials such as the control rod drive (CRD), causing the initial failure of the lower plenum. To understand this failure behavior, we conducted the ELSA (Experiment on Late In-vessel Severe Accident Phenomena) test series[1], which focuses on the damage caused by the eutectic melting of the liquid metal pool and CRD structures.

2. Methods

A test bundle simulating the CRD structure was fabricated and loaded with SUS304-Zr alloy as the simulated metal debris. Two types of eutectic compositions, Zr-rich and stainless steel-rich, were used for the simlated metal debris. We performed two tests using these two sim-debris, respectively. The sample was gradually heated up to about 1400°C using the LIESAN (Large-scale Equipment for Investigation of Severe Accidents in Nuclear reactors) test facility, and in-situ observation was performed using a video camera.

3. Results and Discussion

In the Zr-rich metal debris condition, CRD failure progressed at a relatively low temperature of around 1200°C. In contrast, in the stainless-rich condition, the failure occurred when the temperature reached above the melting point of the structural material (stainless steel). The failure mode of the CRD differs depending on the amount of the metallic Zr in metallic debris; (1) melting due to eutectic reactions and (2) melting due to a temperature rise above the melting point.

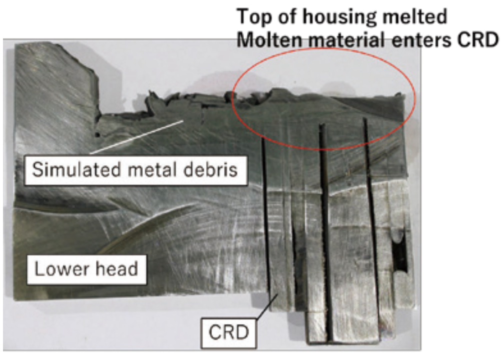


Fig. 1 Cross section of test bundle using stainless steel rich sim-metal debris conditions

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J02

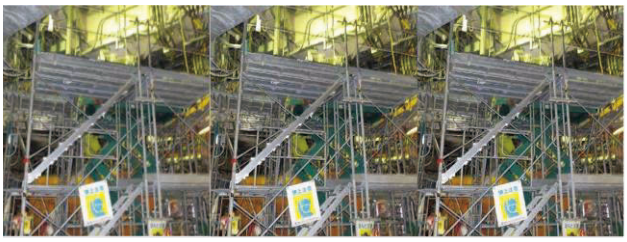
Development of an image clarification method using deep learning for improving the operator's spatial awareness

Yuta Tanifuji, Kuniaki Kawabata
Japan Atomic Energy Agency

Abstract

This paper reports on the development of a camera image clarifying method using deep learning to assist in recognizing the status of the workspace when executing the task remotely. By the result of the experiments, it was confirmed that the quality of images of actual decommissioning workspace clarified by proposed method is better than that of conventional methods.

Our motivation is to develop a method to clarify camera images obtained in the decommissioning work of Fukushima Daiichi Nuclear Power Station to recognize the workspace by the operators when executing the task remotely. Deep learning is a modern computational tool for image processing applications, and it acquires a complex input/output relationship with the design parameters which must be optimized and adjusted in conventional image clarifying method, by learning process. In our research, we utilized a 4-layer U-net[1], which is a framework of Convolutional Neural Network(CNN) as a framework and the training data are 800 images of the DIV2K[2] dataset that were resized to 720 x 720 pixels. Batch learning with 200 training cycles for 800 images was applied to train U-net framework, thus total number of training cycles was 160000. BRISQUE[3] that is an objective evaluation index for image quality was used to evaluate the quality of the processed images. In order to confirm the BRISQUE value of processed result by the proposed method compared with the one processed bicubic method which is a typical interoperation method for clarifying images, and the one processed by RAISR[4], which is non-CNN-based a machine learning method. Fig.1 shows obtained the image which was taken around the X-100B penetration in the Unit 1 reactor building[5] and the example of processed images by each method. The BRISQUE value which takes lower value is high image quality showed that the result using U-net had the highest quality of image.



BRISQUE: 51.16 BRISQUE: 45.20 BRISQUE: 41.34

Fig.1 Results of image clarification processing (Left: Bicubic, Center: RAISR, Right: U-net)

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J03

Development of an "in-situ" alpha air monitor in a harsh environment

Youichi Tsubota[1], Yasuhisa Kimura[1], Yuya Nagai[1], Sho Kojima[2], Shinji Tokonami[3] and Takahiro Nakagawa[1]

[1]Japan Atomic Energy Agency, [2]Ascend Co.,ltd., [3]Hirosaki University

Abstract

A detector component of in-situ monitoring system for α-aerosol in the harsh (high humidity, high β/γ-ray background) environment expected inside the 1F-PCV was designed and prototyped. A part of the system was installed at the glovebox dismantling site of a MOX fuel facility, and its fast response performance and long-term operation capability were demonstrated.

1. Introduction

The decommissioning of the Fukushima Daiichi Nuclear Power Station (1F) will involve large-scale retrieval of nuclear fuel debris (NFD) from the damaged reactor in the future. During the processing of the NFD, the fine particles originating from NFD will be scattered into the primary containment vessel (PCV). In particular, particles containing alpha nuclides (α-aerosols) can result in a drastically high effective dose upon inhalation. Therefore, it is important to monitor their concentration on a real-time basis inside the 1F-PCVs. Aiming for the in-situ monitoring of highly concentrated α-aerosols in a high-humidity, high-dose environment, we have prototyped the *in-situ* Alpha Air Monitor (IAAM) as a detector component of the monitoring system. In this report, we will give an overview of the IAAM and a preliminary report on a field test of the IAAM at the glovebox dismantling site in the MOX fuel facility at JAEA-Tokai.

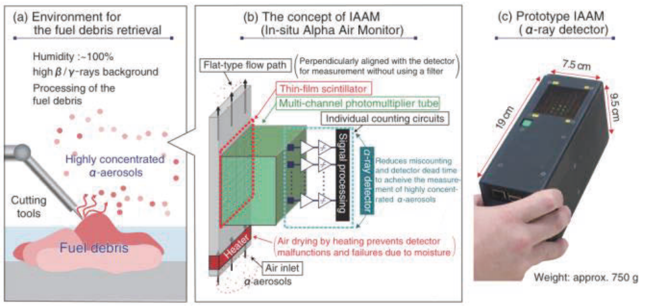


Figure 1. α-aerosol monitoring inside 1F primary containment vessels (PCVs) by the in-situ Alpha Air Monitor (IAAM).

2. Overview of the IAAM and preliminary results of field test

To achieve α-aerosol monitoring inside 1F-PCVs, we defined four requirements: (1) reliable operation under high humidity, (2) α-aerosol monitoring without using a filter, (3) high concentration α-aerosol measurement, and (4) selective measurement of α-aerosols in a high β/γ-ray environment. The IAAM was designed to satisfy these requirements. The key designs of the system are as folows,

(1) Air drying in the flow path prevents detector malfunctions due to moisture in the air.

(2) The α-aerosol measurement can be performed without using a filter by placing the α-ray detector perpendicular to a “flat-type flow path” whose width is sufficiently smaller than the range of the α-rays.

(3) By the direct bonding of a thin-film scintillator with a multi-anode photomultiplier tube, the scintillation light is separated into 64 groups for counting, thereby reducing signal miscounting. By adopting this method, α-aerosol concentrations of up to 3.2×10^2 Bq/cm^3 (>30 times that in a 1F-PCVs) can be measured.

(4) The thickness of the thin-film scintillator and the threshold for signal processing is optimized for the selective measurement of α-rays. Therefore, even in high γ-ray environments(>1 Sv/h), only α-aerosols can be selectively measured.

In order to confirm the response to highly concentrated α-aerosols, the IAAM (excluding the heating section) was installed at the glovebox dismantling site of a MOX fuel facility, and a long-term field test (> 6mo.) was conducted. The IAAM exhibited a significantly faster response time (τ= 1s) compared to conventional dust monitors (τ= 100s). In addition, the counting rate of the IAAM did not saturate throughout the entire period, proving its long-term continuous operability.

J04

Application Example of Research Findings: Investigation inside Unit 2 Well

Akihiro TAGAWA[1]

[1]Japan Atomic Energy Agency

Abstract

Supported the investigation of air dose rate in the well of TEPCO's Fukushima Daiichi NPS Unit 2 by combining JAEA's original research results and Eichi project research results and applying them to the decommissioning site. Measurement accuracy was ensured by combining a newly developed sensor capable of measuring even high radiation doses with conventional measurement methods.

1. Introduction

TEPCO needed to conduct a survey of dose rates in the reactor wells of Fukushima Daiichi NPP Unit 2. JAEA proposed a method to confirm safety and measure dose rates and conducted the actual measurements.

2. Overview of Application Example of Research Findings

2-1. Preparation stage of work

JAEA predicted the Cs concentration in the reactor well section and the upper part of the inside of D/W by simulation. Since the dose rate inside the well was about 40 Sv/h, safety measures during the work were considered.

2-2. Investigation in the reactor well

Measurement accuracy was ensured by combining a newly developed sensor capable of measuring even high radiation doses with conventional measurement methods. Furthermore, LIBS analysis of the recovered materials collected during the work revealed a trace amount of salt from seawater injection.

2-3. Feedback to JAEA research

The simulation was reviewed based on the survey data.

Source distribution predictions were improved to reproduce the measured air dose rates in the wells.

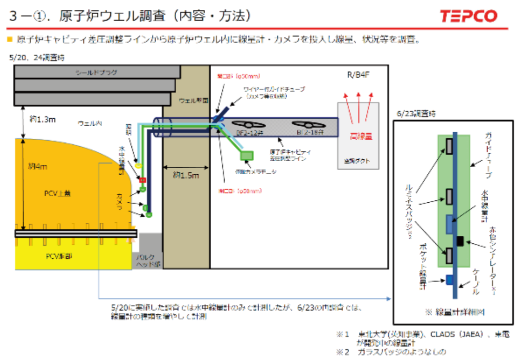


Figure1. Status of tests in the reactor well of Unit 2 reported to the Nuclear Regulation Authority by TEPCO[1]

3. Conclusion

Supported the investigation of air dose rate in the well of TEPCO's Fukushima Daiichi NPS Unit 2 by combining JAEA's original research results and Eichi project research results and applying them to the decommissioning site. TEPCO presented a letter of appreciation to JAEA.

References

[1] NRA web site [in Japanese], https://www.nra.go.jp/data/000358693.pdf

J05

Development of handheld alpha/beta imaging detector for contamination measurement

Yuki Morishita[1]*, Mikio Higuchi[2], Junichi H. Kaneko[1,2], Yuichi Kitagawa[2], Jun Akedo[3], Mitsugu Sohma[3], and Hiroaki Matsui[3]

[1]JAEA, [2]Hokkaido University, [3]AIST

Abstract

We have developed a handheld alpha/beta imaging detector that can be carried to the site of the Fukushima Daiichi NPS. The detector consisted of two scintillators (plastic scintillator and Ce:La-GPS ceramic scintillator), allowing discrimination between alpha and beta particles. The developed detector can be used to detect plutonium contamination at the site of the Fukushima Daiichi NPS.

1. Introduction

When alpha nuclides such as plutonium are measured, radon progenies are also measured because they also emit alpha particles. It is necessary to discriminate between them at the site of the Fukushima Daiichi NPS. In addition, since beta nuclides as well as alpha nuclides are present at the Fukushima Daiichi NPS, it is necessary to discriminate between alpha and beta particles. For discrimination between plutonium and radon progeny, visualization of the alpha-particle distribution by imaging is important [1]. In addition, a Phoswich-type detector is effective for the discrimination of alpha and beta radiation. Therefore, we have developed a handheld alpha/beta imaging detector that can be carried to the site.

2. Materials and Methods

The detector consisted of a 5 cm square plastic scintillator for alpha particle detection and a 5 cm square $(La, Gd)_2Si_2O_7$ (Ce:La-GPS) ceramic scintillator for beta particle detection. The Ce:La-GPS scintillator was optically coupled to a 5 cm square multi-anode photomultiplier tube (H12700A, Hamamatsu Photonics KK, Hamamatsu, Japan). The surface of the scintillator was shaded with a double layer of Aluminum Mylar. The output signals from the multi-anode photomultiplier tubes were amplified and summed by the center-of-gravity calculation circuit before being collected by the data acquisition system. The collected information was calculated by a field-programmable gate array (FPGA) in the data acquisition system to obtain information on the incident position and energy of alpha and beta particles. The detector was powered by a battery installed in the lower part of the device. A belt holder on the side of the case allowed the instrument to be carried on a shoulder belt. A laptop computer with a touch panel was installed in the upper part of the collection unit. In a previous test using the same detector, we measured alpha nuclides at the Fukushima Daiichi NPS and confirmed that it is capable of discriminating beta nuclides from alpha nuclides [2].

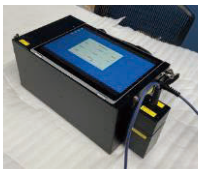


Figure 1. Handheld alpha/beta imaging detector

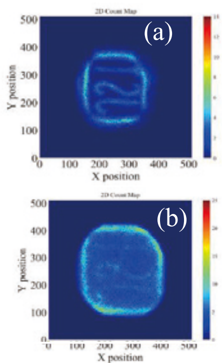


Figure 2. Imaging results: (a)alpha, and (b)beta

3. Conclusion

The developed detector can be carried to the site of the Fukushima Daiichi NPS and used, for example, to detect plutonium contamination of a worker's body surface.

References

[1] Morishita, Yuki, et al. IEEE Transactions on Nuclear Science 67.10 (2020): 2203-2208.

[2] Morishita, Yuki, et al. Radiation Measurements 160 (2023): 106896.

J06

Research and Development of Digital Technologies to Explore Radiation Source Distributions for Exposure Reduction ~ Current Research and Development Progress ~

Masahiro Suzuki, Yuto Aoki, Takashi Yamaguchi, Masahiko Machida,
Hiroko Miyamura, Koji Okamoto
Japan Atomic Energy Agency

Abstract

In order to facilitate the full-scale implementation of fuel debris removal at the Tokyo Electric Power Fukushima Daiichi Nuclear Power Plant (hereinafter referred to as 1F), it is necessary to establish a safe access route within the highly radiation-intensive reactor building (hereinafter referred to as "R/B"). For the purpose of the establishment, it requires environmental improvements such as decontamination of highly-intensive radiation sources and shielding measures. Based on the radiation dose measurement data at the site, Japan Atomic Energy Agency (JAEA) has examined inverse estimation scheme of highly-intensive radiation sources and developed a system that incorporates not only virtual reality (VR) but also mixed reality (MR, AR) to evaluate the effectiveness of decontamination and shielding measures. This report presents an overview of the research and development achievements to date and introduces necessary efforts for enhancing functionality to ensure practical applications at 1F.

1. Introduction

To improve the environment at 1F site, it is crucial to identify highly-intensive radiation sources based on environmental information including radiation dose measurements. Among lots of such activities, evaluating the reduction in working person’s exposure through decontamination and shielding measures is one of the most essential tasks. Using structural data and radiation-dose spatial distribution data within R/B, JAEA has developed a prototype system that identifies the intensity and location of radiation sources and simulates radiation dose variations resulting from decontamination and shielding measures in the virtual space. To verify the effectiveness of the aforementioned development system, validation analysis was conducted using structural data and spatial radiation dose data from Japan Materials Testing Reactor (JMTR), because the positions and intensities of radiation sources had already been identified.

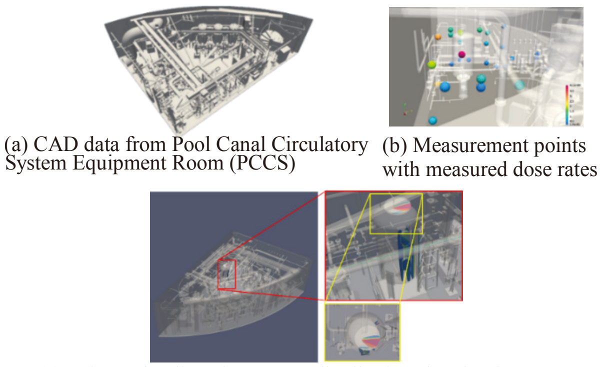
2. Main developments to date

This section presents the confirmation results of the system's effectiveness using data from JMTR (Figure 1).

(1) Through inverse estimation analysis based on machine learning techniques such as Least Absolute Selection and Shrinkage Operator (LASSO), positions and intensities of radiation sources can be determined using structural data and measured radiation dose values (points meeting the criteria for successful inverse estimation are specified using the measurement point indication tool).

(2) The radiation dose distribution is calculated using the estimated sources by using Particle and Heavy Ion Transport Code System (PHITS), and validity of the proposed method is confirmed by comparing the calculated values with the measured values.

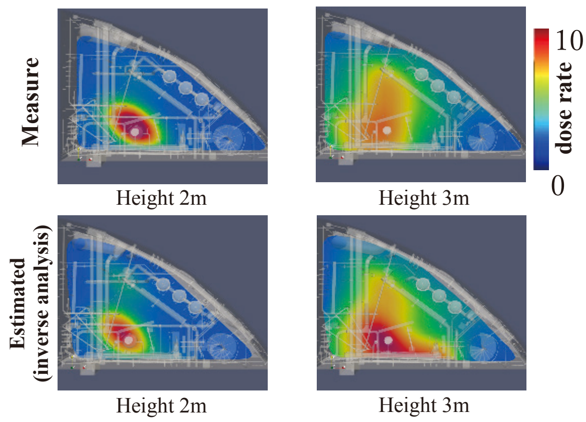
(3) Applying the system to JMTR, it is confirmed to be possible to identify positions and intensities of contamination sources.



(a) CAD data from Pool Canal Circulatory System Equipment Room (PCCS)

(b) Measurement points with measured dose rates

(c)Estimated radioactive source distribution showing hotspot areas on the surface of structures colored blue to red in the PCCS



(d) Comparison of dose rate distributions between measured and estimated ones

Fig.1 Validation experimental tests using JMTR

3. Research and Development to Enhance Applicability at 1F Site

Starting from the present fiscal year, research and development efforts has been conducted to enhance the applicability of the prototype system at 1F site. The following items have been pursued and will be intensively investigated:

a. System development for on-site applicability enhancement

Based on the environmental information (structures, images, radiations, etc.) measured at 1F site, a system will be developed to create radiation source and dose rate estimation maps, allowing for rapid visualization of highly-intensive radiation source locations and high-dose areas at the site.

b. Research and development to further enhance applicability at the site

To improve accuracy of the estimated radiation sources and dose rate estimation maps, innovative schemes to obtain more environmental information will be explored, and additional functionalities to support the developed system will be investigated.

Acknowledgements

This work was carried out in a subsidy program of “Project of Decommissioning and Contaminated Water Management”, entitled “Development of Technologies for Work Environmental Improvement in R/B”.

J07 Simulation systems of individual external exposure dose using monitoring data of environmental radiation.

Kazuya Yoshimura[1]

[1]Japan Atomic Energy Agency

Abstract

Assessment of exposure doses and risk communication with residents are required to lift evacuation orders around Fukushima Dai-ichi Nuclear Power plant. We built three simulation systems to evaluate external exposure doses based on behavior of residents and integrating ambient dose rate at each location, depending on the objects of the evaluation.

1. Introduction

In Fukushima, evacuation orders have been lifted with the progress in decontamination. Assessment of exposure doses and risk communication with residents are important requirements for the lifting of evacuation orders. Therefore, the tools to easily evaluate the exposure doses by residents and to facilitate the risk communication are desired.

2. External exposure dose simulation systems

2-1. Calculation of external exposure

Exposure doses are calculated by multiplying the integrated ambient dose rate for each location by the conversion factor from ambient dose rate to effective dose [1].

2-2. Simulation system for assumed behavior pattern (Predictive evaluation)

This system is a web-based simulator that can estimate exposure doses by selecting places on a map and inputting behavior data. Routes between places are automatically searched. Since the simulator is used on a browser, it can be used on a variety of devices, such as PCs and smartphones (Fig. 1).

2-3. Simulation system for behavior pattern taken by resident (Retrospective evaluation)

A smartphone application had been developed for the retrospective evaluation. This system records the place and time using GPS in the smartphone and estimate exposure doses along the behavior pattern of user. The application can automatically and continuously record the location information in background operation.

2-4. Simulation system for areal management of the external exposure (Evaluation for population)

This system facilitates dose management for populations entering certain areas, such as evacuation zones. This system consists of a smartphone application to record the place and time by GPS and a PC application that calculates exposure doses according to the behavior pattern, which can be transferred to the PC via a two-dimensional barcode.

3. Conclusion

We developed three systems to evaluate external exposure dose for individual behavior. Some systems have been applied to local government for risk communication with residents and will relieve residents and help their return.

Reference

[1] Sato, R. et .al, 2022. Journal of Radiation Protection and Research. 47(2): 77-85.

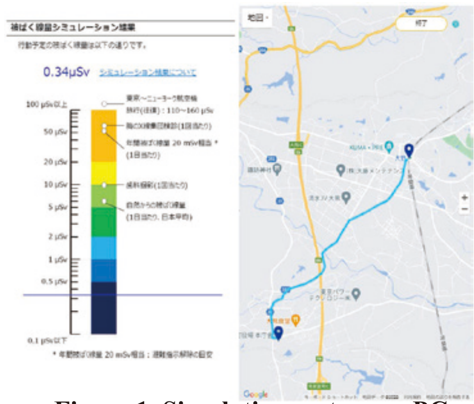


Figure 1. Simulation system on PC

J08 Chemical Researches Assisting in Decommissioning Fukushima Daiichi Nuclear Power Station Including Material Corrosion, Radioactive Nuclides Behavior and Waste Solidification

Takahito Aoyama[1], Youko Takahatake[1], Azusa Ito[1], Fumiyoshi Ueno[1], Takashi Okada, Ryuji Nagaishi[1] and Yoshikazu Koma[1]

[1]Japan Atomic Energy Agency (JAEA)

Abstract

Decommissioning Fukushima Daiichi Nuclear Power Station requires wide spectrum of R&D. Japan Atomic Energy Agency is contributing basic research besides supporting technology application, and recent results of basic research for the field of material and waste management is briefly reviewed.

1. Introduction

To help sustainable decommissioning work and implementing waste conditioning/disposal technology, material corrosion, radioactive contamination and waste solidification have been investigated.

2. Material Corrosion

The PCV and RPV of damaged reactors are in a severe environment for corrosion of steel materials due to heavy contamination. To enhance knowledge on the effects of β-ray, corrosion experiments using ^{90}Sr was conducted. It was found that the amount of H_2O_2 generated by radiolysis was greater than that of γ-rays using ^{137}Cs with the corresponding dose rate, which increased the corrosion potential [1].

3. Radioactive Nuclides Behavior

Fission products is the major concern for contamination, although ^{60}Co of activation products is often detected and is a key-nuclide for determining activity in solid waste generated at general nuclear power plant. Correlations between ^{60}Co and some nuclides found in various contaminated materials were revealed to suggest future utilization of ^{60}Co as a key for the waste originated from the fuel damage in operation [2].

4. Waste Solidification

The water decontamination system generates secondary waste such as sludge and slurry. Alkali-activated materials (AAM) is a candidate solidification method of the waste. AAM is initially amorphous and then crystallized in a solidified AAM[3]. To identify the key factors of crystallization, silica structures in sodium silicate as an AAM reagents have been investigated by Raman spectroscopy.

5. Conclusion

It is expected that the findings will be utilized for lasting decommissioning and waste management. And JAEA will continue basic investigations for Fukushima Daiichi NPS regardless of the above topics.

References

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[2] Y. Takahatake et al., Correlation with Cobalt-60 for radioactive nuclides in solid waste generated at Fukushima Daiichi NPS, International Topical Workshop on Fukushima Decommissioning Research 2022 (FDR2022), October 14-16, 2022, Fukushima, Japan (2022).

[3] O. Abdelrahman et al., Impact of Na/Al Ratio on the Extent of Alkali-Activation Reaction: Non-linearity and Diminishing Returns. Frontiers in Chemistry, 9:806532 (2022).

J09

R&D on Waste Management for Decommissioning Fukushima Daiichi Nuclear Power Station

Yoshikazu KOMA[1], Naoya KAJI[1]

[1] Japan Atomic Energy Agency

Abstract

Methodologies for safe management of radioactive waste from Fukushima Daiichi Nuclear Power Station is now under development including characterization, conditioning and disposal aiming at a knowledge base.

1. Introduction

For decommissioning Fukushima Daiichi Nuclear Power Station, R&Ds on waste management technologies has been conducted. Status on characterization, conditioning and disposal is briefly reviewed.

2. Characterization

Radiochemical analysis of samples collected inside the site has been continued for these years at the facilities, which are located in Ibaraki, besides that, the Radioactive Material Analysis and Research Facility of Laboratory–1 started its operation and it will mainly contribute [1]. The data obtained is utilized to clarify the behavior of radioactive nuclides on contamination, and to develop a method estimating the radioactivity inventory. The data and the suite of methodologies will form a knowledge base in the future [2].

3. Processing/Conditioning

To mitigate the risk of secondary waste from the water decontamination, a method to stabilize them for long-term storage is required. To realize such conditions for solidification, it is needed to consider variety of parameters, and then, an efficient method of estimating resulted physical properties and to screen possible conditions for the best. And, basic data concerning long-term stability is also experimentally obtained.

4. Disposal

On supposing large uncertainties in amount and properties of waste, a method to investigate disposal safety is under development. For the wide options of disposal concept, story board, scenario, FEP (Features, Events and Processes) and model will be discussed. Feedback will be provided to waste characterization and R&D on conditioning, and a rationalized concept will be derived. Basic data of nuclide behavior in an environment is also collected through field sampling to support quantitative evaluation.

5. Conclusion

To establish the optimized and economic management stream for each waste, R&Ds will be continued with stakeholder cooperation.

Acknowledgements

This work includes part of the results from works under the subsidy program “Project of Decommissioning and Contaminated Water Management” conducted by the Ministry of Economy, Trade and Industry.。

References

[1] Do, V. K. et al., this forum. [2] Koma, Y., Workshop on the Innovative Techniques and Technologies to Support Characterisation and Decommissioning of Complex and Legacy Sites, OECD/NEA (2022). [3] Michael Ochs et al., Applied Geochemistry, Volume 136, 105161 (2022).

J10

Interference-free determination for long-lived radionuclides based on solid-phase extraction combined with ICP-MS/MS

V.K. Do, Y. Ohta, M.B. Irvin, R. Aita, E. Murakami, S. Homma, H. Iwahashi, T. Furuse

Japan Atomic Energy Agency

Abstract

New analytical methods for determining long-lived radionuclides (^{93}Zr, ^{93}Mo, ^{107}Pd, and ^{126}Sn) using solid-phase extraction method and inductively coupled plasma triple quadrupole mass spectrometry (ICP-MS/MS) have been developed for the analysis of the rubble waste collected at Fukushima Daiichi Nuclear Power Plant (FDNPP).

1. Introduction

The radioactive material analysis and research facility 1 (Laboratory-1, at the Okuma Analysis and Research Center) has been constructed to analyze the decommissioning waste collected from 1F. Long-lived radionuclides can be measured by mass spectrometry, which is more rapid and more highly sensitive than radiometry. The paper summarizes our recent results in the development of analytical methods for ^{93}Zr, ^{93}Mo, ^{107}Pd, and ^{126}Sn by ICP-MS/MS, aiming at applications to the measurement of samples collected in the vicinity of FDNPP.

Table 1: Summary of the developed analytical methods

Nuclides	Recovery (%)	IF	BEC (Bg/g)	Method detection limit (Bg/g)	Regulatory trench disposal (Bg/g)
^{93}Zr	> 90	~10^5	4 ×10^{-3}	1.7 ×10^{-3}	1200
^{93}Mo	> 90	~10^5	0.4	0.2	11
^{107}Pd	~80	10^4-10^5	0.16	5 ×10^{-2}	1.2 ×10^3
^{126}Sn	> 95	~10^5	0.03	6.1 ×10^{-3}	1.3

IF: interference removal factor; BEC: Blank-sample equivalent concentration

2. Experimental design

Commercially available standards and home-made concrete-dissolved samples were used as simulated samples to characterize the purification and measurement capability. Accordingly, the initial samples were purified by multiple steps based on solid phase extraction columns. The obtained samples were then measured by an 8900 ICP-MS/MS instrument (Agilent, Japan) to evaluate the detection capabilities for the selected radionuclides.

3. Results and discussion

Table 1 summarizes the obtained results about the developed analytical methods. High chemical recoveries were achieved for all the nuclides. The IF factors and BEC denoted that the interference from sample matrixes was effectively removed by chemical separation and the measurement by ICP-MS/MS [1,2]. The obtained method detection limits are a few orders of magnitude lower than the regulatory limit for trench disposal of the radioactive rubble wastes.

4. Conclusions

We have developed new analytical methods for determining ^{93}Zr, ^{93}Mo, ^{107}Pd, and ^{126}Sn in solid wastes. The determination of the long-lived radionuclides by ICP-MS/MS results in shortening the measurement and simplifying the chemical sample treatment.

References [1] *Journal of Radioanalytical and Nuclear Chemistry* **327**, pages 543–553 (2021); [2] *Journal of Radioanalytical and Nuclear Chemistry* **331**, pages 5631–5640 (2022)

J11 Measurement of Radioactive Materials Contained in the ALPS Treated Water by JAEA as a Third Party

Yoshihiro Tsuchida[1], Naoya Kaji[1] and Ritsuro Tokumori[1]
[1]Japan Atomic Energy Agency

Abstract

Measurement of radioactive materials contained in the ALPS treated water by JAEA as a third party has been being conducted in Radioactive Material Analyses and Research Facility Laboratory-1 (hereafter 'Lab-1') for the purpose of highly objective and transparent measurement.

1. Purpose and background of the measurement

Based on the policy of the government, JAEA has been conducting the measurement of radioactive materials contained in the ALPS treated water as a third party for the purpose of highly objective and transparent measurement. The measurement has been being conducted in Laboratory-1, that was completed in June 2022.

2. Contents of the measurement

The following items are carried out by measuring the ALPS treated water before it is released into the ocean.

- Measurement of 29 nuclides for which it is stipulated in the implementation plan (which is approved document for license) to confirm that they are within the release standards (below regulatory standards)
- Measurement of 39 nuclide that TEPCO HD voluntarily confirms that it does not exist significantly from the viewpoint of suppressing rumors.
- Measurement of tritium concentration in ALPS treated water

3. Methods of the measurement

The tritium concentration in ALPS treated water is measured using a liquid scintillation counter, which is capable of efficiently measuring very weak energy β-rays emitted by tritium, after pretreatment to remove impurities that interfere with measurement. For nuclides other than tritium, impurities that interfere with the measurement are removed through pretreatment, purification of the nuclides to be analyzed is performed, and then measurements are performed using appropriate analyzers.

4. Results of the Measurement[1]

The first measurement was started at the end of March 2023 and the results was published on June 22nd. The sum of concentration ratios required by law for 29 nuclides other than tritium was 0.28 (<1), confirming compliance with regulatory standards. It was confirmed that none of the other 39 nuclides were significantly present. The tritium concentration was confirmed to be 140,000 Bq/L.

References

[1] JAEA Website https://fukushima.jaea.go.jp/okuma/alps/

J12 Summary of JAEA's efforts for fuel debris retrieval

Tomohiro Tomitsuka[1], Shin-ichi Koyama[1], Naoya Kaji[1], Satomi Kakutani[1]
[1]Japan Atomic Energy Agency

Abstract

JAEA is steadily proceeding preparations for full-scale mock-up testing of retrieval equipment, visualization of radiation applied by DX, development of analysis method and the characterization of fuel debris for the upcoming trial retrieval of fuel debris from Fukushima Dai-ichi Nuclear Power Station (1F). Prior to the operation of Laboratry-2 of Okuma Research and Analysis Center, JAEA plan to start fuel debris analysis at hot laboratories in the Ibaraki area. The related main activities are shown below.

a) Continuous support for full-scale mock-up testing at the Naraha Center for Remote Control Technology Development

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

b) Visualization of environment, dose and radiation source distribution for reducing exposure

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

c) Development of fuel debris analysis and research Facility at the Okuma Analysis and Research Center

d) Preparation for fuel debris analysis

Various kinds of analysis of samples (wiped smear or deposits) obtained from 1F internal investigation were performed to clarify the components and micro-structure of fine particles therein. Those analysis data were utilized for estimating the characteristics of fuel debris.

e) Analysis system and improving the quality of analysis data

The simulated fuel debris was prepared and supplied to the hot laboratories in Ibaraki area in order obtain a common understanding for the technology for evaluating the characteristics of fuel debris. As a first important step, elemental composition was analyzed using the own technology possessed by each laboratories to establish the evaluation method as an approach for sample with unknown composition.

f) Development of remote, quick and simple analysis technology for fuel debris

Development of an image clarification method using deep learning for improving the operator's spatial awareness.

g) Evaluation of fuel debris measurement technology and radiation characteristics

h) Estimation of internal reactor situation by analysis of measurement data and accident progress scenario

J13 JAEA's Efforts for Human Resource Development and Regional Commitment in Fukushima

Tomohiro Tomitsuka[1], Ryoichiro Kuroki[1], Satomi Kakutani[1]
[1]Japan Atomic Energy Agency

Abstract

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

1. Introduction

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

2. Efforts for Human Resource Development

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

3. Technology development with local industries in Fukushima

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

O01 Remote monitoring system used in a severe radiation environment

Utsuki Sekioka[1], Minoru Watanabe[1] and Nobuya Watanabe[1]
[1]Okayama University

Abstract

When testing a certain device in a severe radiation environment such as nuclear power plants, a remote monitoring system is necessary to confirm whether an operation is correct or not. Therefore, we have developed a remote monitoring system used for radiation-hardened optically reconfigurable gate arrays by using a DE1-SOC field programmable gate array (FPGA) board which includes an FPGA block and an ARM processor. The remote monitoring system can generate control signals and a test vector and can monitor the output signals of an operation on an optically reconfigurable gate array.

1. Remote monitoring system

We have developed a remote monitoring system used for radiation-hardened optically reconfigurable gate arrays [1]. The remote monitoring system entirely consists of remote host personal computer (PC), a DE1-SOC field programmable gate array (FPGA) board and a target device or an optically reconfigurable gate array. The remote part of the remote monitoring system has been implemented onto a DE1-SOC FPGA board which includes an FPGA block and an ARM processor. Linux (Ubuntu) works on the ARM processor on the DE1-SOC FPGA board. A remote host personal computer (PC) and the ARM processor on a DE1-SOC FPGA board are connected with ethernet. We can login to the ARM Linux system remotely through the PC. The ARM processor can provide some commands onto a circuit on the FPGA block.

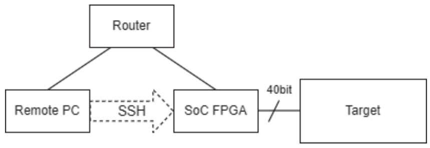

Figure 1. Experiment System.

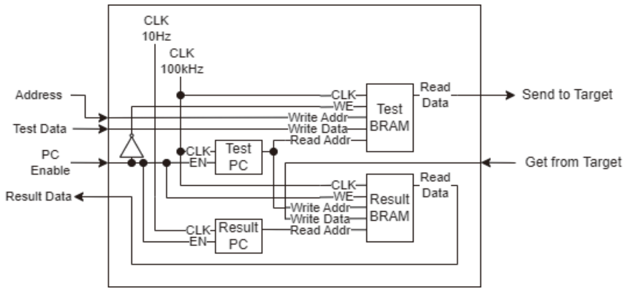

Figure 2. Block diagram of a circuit on an FPGA.

```
RUN:ADDR          :REF:CLK:CCLK:RST:CRST:MODE:SWING:TB  :WAIT
0  :00000000000000:0  :  0:   0:  1:   1:   1:     1:0000:0000000000000000000000
0  :00000000000001:0  :  1:   1:  1:   1:   1:     1:0001:0000000000000000000000
0  :00000000000010:0  :  0:   0:  1:   1:   1:     1:0010:0000000000000000000000
0  :00000000000011:0  :  1:   1:  1:   1:   1:     1:0011:0000000000000000000000
0  :00000000000100:0  :  0:   0:  1:   1:   1:     1:0100:0000000000000000000000
   .
   .
   .
1  :00000000000000:0  :  0:   0:  1:   1:   1:     1:0000:0000000000000000000000
```

Figure 3. Data example on BRAM.

2. FPGA block operation

Control signal patterns and test vectors are transferred from the PC to the ARM processor and finally stored on block random access memory (BRAM) on the FPGA block. The data on BRAM is constantly provided onto a target devices or an optically reconfigurable gate array at a certain clock frequency. The output signals from the target devices are also monitored and stored onto BRAM. Then, the storing BRAM data is also transferred to the PC through ethernet. The data is compared with correct data on the PC.

References

[1] H. Shinba, M. Watanabe, "Radiation-hardened configuration-context realization for field programmable gate arrays," Applied Optics, Vol. 59, Issue 19, pp. 5680-5686, 2020.

O02

Design example of a triple modular redundancy ALU, a register file, and a program counter for a processor

Masato Isobe, Minoru Watanabe, and Nobuya Watanabe
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Abstract

Under a strong radiation environment such as decommissioning situation of nuclear power plants, processors must have high-radiation tolerance. Currently available processors are always weak for radiation and are easily broken by radiation for a short period. Therefore, we have been developing a triple-modular-redundant processor to increase the radiation tolerance. This paper presents design examples for an Arithmetic Logic Units (ALU), a register file, and a program counter.

1. Research Background

Currently, space-grade processors are available. However, the life-time of the space-grade processors is limited to a extremely short period when the processors are applied to decommissioning work in nuclear power plants. So, we have introduced triple modular redundancy (TMR) for processor to increase the total-ionizing-dose tolerance.

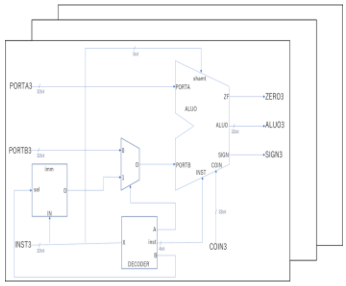

Figure 1. ALU.

2. Each module

2-1. Triple modular redundant Arithmetic Logic Units (Figure 1)

A triple modular redundant ALU includes three ALU units was designed.

2-2. Triple modular redundant register file (Figure 2)

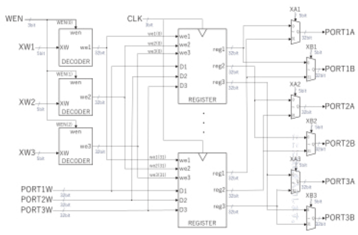

Figure 2. Register file.

Firstly, a triple modular redundant register was designed. The register consists of majority voting circuits, flip-flops, and selectors. Since the majority voting operation is automatically executed on the circuits at every clock cycle, the circuit has high soft-error tolerance in addition to high total-ionizing-dose tolerance. The entire triple modular redundancy register file was constructed by using the triple modular redundant registers.

2-3. Triple modular redundant Program counter (Figure 3)

A triple modular redundant counter consists of majority voting circuits, flip-flops, adders, and selectors. At every clock cycle, the majority voting operation is automatically executed.

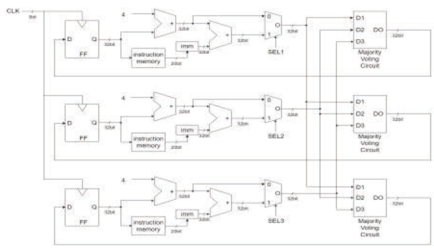

Figure 3. Program counter.

3. Conclusion

We have designed a triple modular redundant ALU, register file, and program counter. In the future, a perfect triple modular redundant processor will be designed and implemented onto a field programmable gate array (FPGA).

References

[1] H. Shinba, M. Watanabe, "Radiation-hardened configuration-context realization for field programmable gate arrays," Applied Optics, Vol. 59, Issue 19, pp. 5680-5686, June, 2020.

O03

Photodiode current range measurement result of an optically reconfigurable gate array VLSI

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Abstract

Since very large-scale integrations (VLSIs) are vulnerable to radiation, VLSIs used in strong radiation environments such as space and nuclear power plants must use a certain radiation tolerance increase method. We've been developing a radiation-hardened optically reconfigurable gate array that use a perfect parallel configuration as one of the methods. This paper presents the allowable photodiode current range measurement result of an optically reconfigurable gate array VLSI.

1. Optically reconfigurable gate array

Optically reconfigurable gate array consists of three components: a programmable gate array VLSI, a holographic memory and a laser array [1]. Although the optically reconfigurable gate array is a type of field programmable gate arrays (FPGAs), the configuration procedure is different from the FPGAs. The optically reconfigurable gate arrays can be configured optically in parallel. Such optical parallel configuration can support our proposed reparable VLSI concept. In the reparable concept, even if a part of the programmable gate arrays and/or a part of the configuration circuits are broken by radiation, the defective VLSI can continuously be used by avoiding these defective regions and by using non-damaged regions. As a result, the radiation tolerance of the programmable gate array VLSIs can be increased.

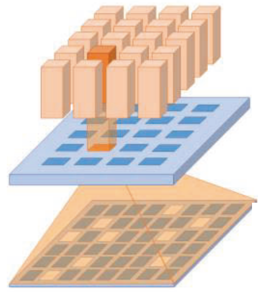

Figure 1. Structure of an optically reconfigurable gate array

2. New optically reconfigurable gate array VLSI

A new optically reconfigurable gate array VLSI that never use flip-flops for its configuration circuit. A photodiode is directly connected to each programming point of the programmable gate array. In this case, the photodiode current must be adjusted to a suitable value. The current is controlled by the gate voltage of transistors. Here, we have confirmed the allowable gate voltage range of the transistors is 1.87 V - 2.61 V for a certain holographic memory system on an optically reconfigurable gate array.

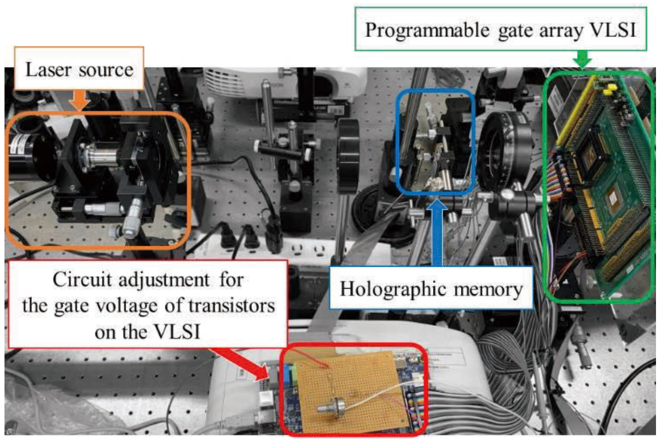

Figure 2. Experiment system

Reference

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

O04

Evaluation of low-voltage operations of an optically reconfigurable gate array VLSI

Yuki Shimamura, Minoru Watanabe and Nobuya Watanabe
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Abstract

Since currently available very large-scale integrations (VLSIs) are vulnerable to radiation, soft errors and permanent failures frequently happen on VLSIs in high radiation environments such as the decommissioning situation of nuclear power plants. Therefore, we have been developing a radiation-hardened optical reconfigurable gate array VLSI. This paper presents an evaluation result of low-voltage operations of the radiation-hardened optically reconfigurable gate array VLSI.

1. Optically reconfigurable gate array

An optically reconfigurable gate array consists of three components: a programmable gate array VLSI, a holographic memory, and a laser array. Since VLSIs are valuable to radiation, we propose a repairable VLSI concept applied for the optically reconfigurable gate array. In this concept, even if a part of the programmable gate array and/or a part of the configuration circuit are broken by radiation, operations on the programmable gate array can be executed continuously by reconfiguring the programmable gate array. To realize it, we must use parallel configuration architecture. The optically reconfigurable gate arrays can supoprt the parallel configuration architecuture. As a result, 1 Grad total-ionizing-dose tolerance could be achieved using the optically reconfigurable gate array. However, when using the optically reconfigurable gate array, a stabilized power supply unit with the same radiation tolerance as the optically reconfigurable gate arrays is necessary. Nevertheless, the total-ionizing-dose tolerance of currently available stabilized power supply units is always limited to up to 100 Mrad which is 1/10 times lower radiation tolerance than the optically reconfigurable gate arrays. Therefore, since the radiation tolerance of stabilized power supply units is not enough, we are considering that the optically reconfigurable gate array must be driven by battery cells directly without any stabilization. In this case, the voltage of battery cells is constantly decreased while the optically reconfigurable gate array works.

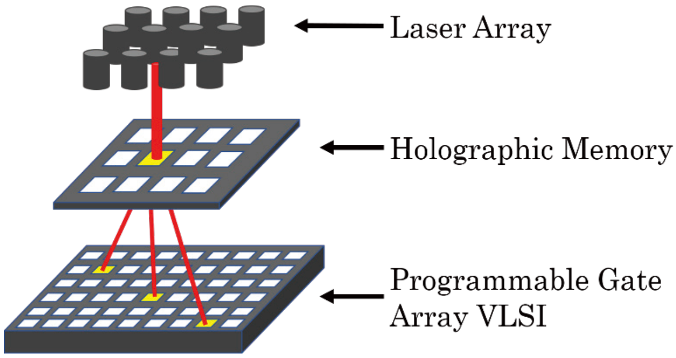

Figure 1. Structure of optically reconfigurable gate array

2. Evaluation result of low-voltage operations of the optically reconfigurable gate array VLSI

We have evaluated the allowable operating voltage range of an optically reconfigurable gate array VLSI. Normally, the VLSI requires 1.8 V for the core and 3.3 V for I/O. A 2-bit adder circuit has been implemented onto the optically reconfigurable gate array VLSI to clarify the allowable operation voltage range. As a result, the allowable operating voltage range was 0.918V – 1.8V when the I/O voltage is 3.3 V. When using a low I/O voltage of 1.65 V, the allowable operating voltage range was 0.796V -1.65V.

Reference

[1] T. Fujimori, M. Watanabe, “A 1.15 Grad total-ionizing-dose tolerance parallel operation oriented optically reconfigurable gate array VLSI,” 2019 IEEE 5th International Workshop on Metrology for AeroSpace (MetroAeroSpace), June, 2019.

O05 Total-ionizing-dose tolerance of an optically reconfigurable gate array VLSI

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Abstract

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

1. Introduction

At the Fukushima Daiichi Nuclear Power Station, decommissioning work is currently being executed. However, when using a robot in such environment, there is a problem that integrated circuits inside the robot are easily broken by radiation. Therefore, we have been developing a radiation-hardened Optically Reconfigurable Gate Array (ORGA)[1][2].

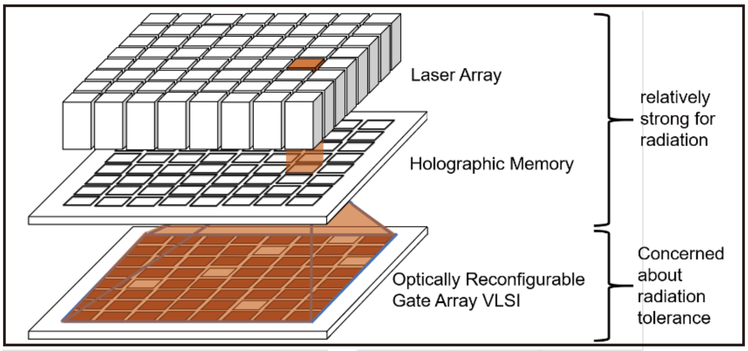

Figure 1. Radiation tolerance of an ORGA system.

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

An ORGA consists of a laser array, a holographic memory, and an ORGA-VLSI (Fig.1). If we can use a programmable architecture for VLSIs, even if a part of a programmable gate array is broken by radiation, the non-damaged area of the VLSI can be used continuously (Fig.2) so that the total-ionizing-dose tolerance of the VLSI can be increased. To realize it, we have introduced parallel configuration architecture onto the ORGA instead of serial configuration architecture. Since laser arrays and holographic memories are relatively strong for radiation (Fig.1), ORGAs can achieve high-radiation tolerance. In this experiment, an ORGA-VLSI was exposed to 290 Mrad gamma radiation. Under the radiation environment of the Fukushima Daiichi Nuclear Power Station, although the existing integrated circuits will not operate normally within 10 hours, the ORGA can operate for 120 days.

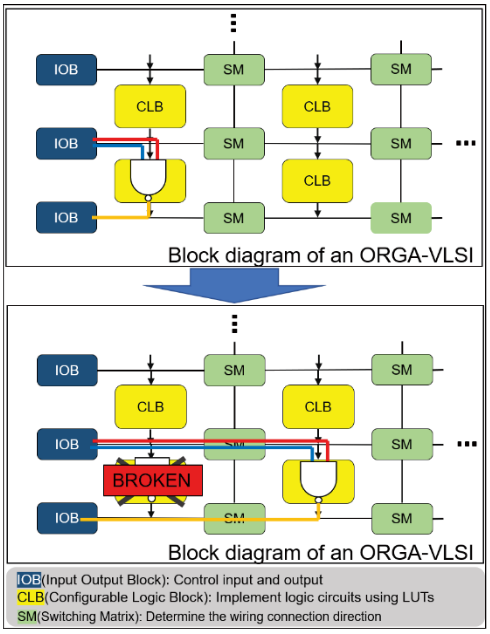

Figure 2. Repair operation of when a part of a programmable gate array is broken by radiation.

3. Experimental result

In this experiment, only a 6.9 % performance degradation was confirmed at an absorbed dose of 290 Mrad. The small degradation can be out of consideration.

References

[1] T. Fujimori, M. Watanabe, "Optically reconfigurable gate array using a colored configuration," Applied Optics, Vol. 57, Issue 29, pp. 8625-8631, Oct., 2018.

[2] T. Fujimori, M. Watanabe, "Parallel light configuration that increases the radiation tolerance of integrated circuits," Optics Express, Vol. 25, Issue 23, pp. 28136-28145, Nov., 2017.

O06 An optically reconfigurable gate array driven by an unstabilized power supply unit

Masashi Tsujino , Minoru Watanabe and Nobuya Watanabe

Okayama University

Abstract

Radiation-hardened integrated circuits are required because current integrated circuits are easily be broken by radiation and cannot operate properly under a high-radiation environment. We have developed an optical reconfigurable gate array VLSI which has high total-ionizing-dose tolerance. However, since stabilized power supply units are vulnerable to radiation, the life-time of optically reconfigurable gate array VLSIs is limited to that of the stabilized power supply units. This paper presents an optical reconfigurable gate array VLSI driven by an unstabilized power supply unit.

1. Optically reconfigurable gate array

An optically reconfigurable gate array is shown in Fig. 1 [1]. The optical reconfigurable gate array consists of a laser array, holographic memory, and programmable gate array VLSI. The holographic memory stores circuit information. The circuit information stored in the holographic memory can be selected by the laser array. The selected circuit information is programmed onto the gate array VLSI via photodiodes. The optically reconfigurable gate array could achieve 1 Grad total-ionizing-dose tolerance.

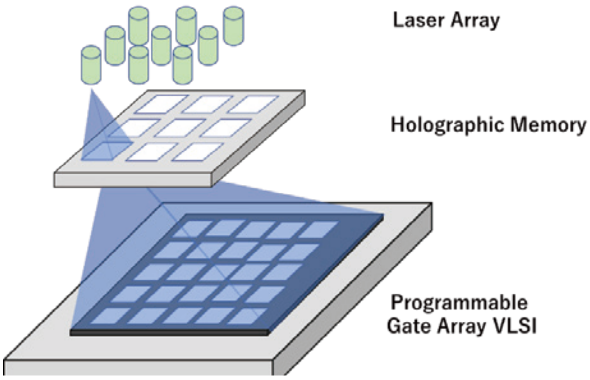

Figure 1 optically reconfigurable gate array.

2. Unstabilized power supply unit

Figure 2 shows the circuit diagram of unstabilized power supply unit. Since power transistors used for stabilized power supply units are weak for radiation, the unstabilized power supply unit never uses power transistors. Instead, the unstabilized power supply unit consists of a transformer, capacitors, a bridge diode, and some diodes. Figure 3 shows the waveform of the output voltage obtained by the power supply circuit. The output voltage varies from 0.94 V to 1.41 V at a frequency of 120 Hz. We have tested the optically reconfigured gate array's operation by using the power supply unit. As a result, it has been confirmed that the optically reconfigured gate array driven by the power supply unit can work correctly.

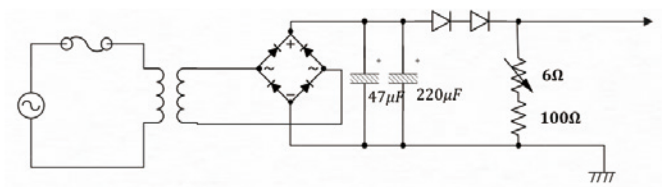

Figure 3 Circuit diagram of power supply

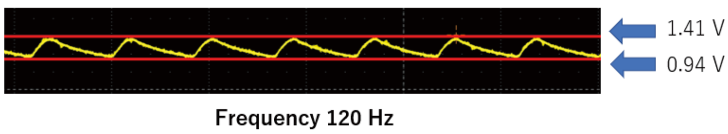

Figure 2 Output voltage from the power supply unit.

Reference

[1] H. Shinba, M. Watanabe, "Radiation-hardened configuration-context realization for field programmable gate arrays," Applied Optics, Vol. 59, Issue 19, pp. 5680-5686, June, 2020.

O07

A mono instruction set computer architecture on an optically reconfigurable gate array VLSI

Soma Imai, Minoru Watanabe and Nobuya Watanabe

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Abstract

We have been developing an optically reconfigurable gate array (ORGA) VLSI which can be reconfigured within a few nanoseconds. Exploiting such high-speed dynamic reconfiguration, the programmable gate array's performance can be increased by introducing the simplest architecture. This paper presents one of them or the evaluation results of a Mono instruction set computer (MISC) architecture.

1. Optically reconfigurable gate array (ORGA)

An optically reconfigurable gate array consists of three components: a programable gate array VLSI, holographic memory, and a laser array. A lot of configuration contexts can be stored in the holographic memory and addressed by a laser array. The configuration contexts stored in the holographic memory are dynamically programmed onto the programmable gate array VLSI within a few nanoseconds. The architecture can provide large virtual gates. For example, even if the physical gate count of a programmable gate array VLSI is 1,000,000, if a holographic memory can store 1000,000 configuration contexts, virtually, 1000,000,000,000 gate count can be realized. This architecture can realize a huge virtual gate count VLSI.

2. Mono instruction set computer (MISC)

Many computers today use reduced instruction set computer (RISC) architecture. However, since processor's performance depends on the complexity of architecture, the simplest processor is the best performance processor and RISC is not the best performance processor. We propose mono instruction set computer (MISC) architecture as the best performance processor. The MISC has only one instruction. Instruction change from one MISC to another MISC can be realized by reconfiguring hardware itself. Using optically reconfigurable gate array, which can support a high-speed reconfiguration, for MISC architecture, the reconfiguration overhead can be out of consideration. So, the best performance processor can be realized.

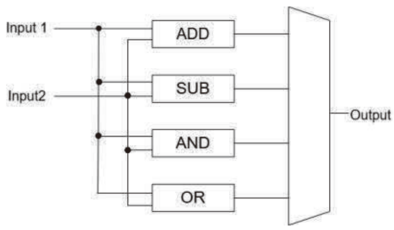

Figure 1: RISC ALU.

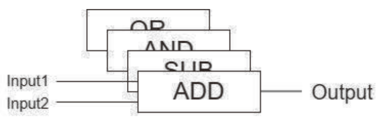

Figure 2: MISC ALU.

3. Evaluation results of MISCs

A 4-bit adder, a 4-bit subtracter, a 4-bit logical AND, and a 4-bit logical OR operations were respectively implemented as a MISC processor onto an optically reconfigurable gate array. By using Model-SIM simulation, the processing time of the 4-bit adder, 4-bit subtracter, 4-bit logical AND, and 4-bit logical OR operations were measured as 3.07 ns, 3.26 ns, 1.01 ns and 0.99 ns, respectively. On the other hand, in the case of a RISC-ALU including the same functions, the processing time of the 4-bit adder, 4-bit subtracter, 4-bit logical AND, and 4-bit logical OR operations were 7.25 ns, 7.25 ns, 6.26 ns and 6.26 ns. As a result, it has been confirmed that MISC processors can realize higher performances than RISC processors.

Reference

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

O08 Realization of a wafer-scale VLSI by using optically reconfigurable gate array architecture

Atsushi Takata, Minoru Watanabe and Nobuya Watanabe
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Abstract

Current very large-scale integrations (VLSIs) are always fabricated on a small die since a wafer has a lot of defects. However, if we can exploit a fully parallel configuration on optically reconfigurable gate arrays, a wafer including defects can be used as a programmable gate array. This paper discusses a realization method of a wafer-scale VLSI based on an optically reconfigurable gate array architecture.

1. Optically reconfigurable gate array

An optically reconfigured gate array consists of a programmable gate array VLSI, a holographic memory, and a laser array. In current field programmable gate arrays (FPGAs), the gate array is configured electrically in a serial connection. On the other hand, optical reconfigurable gate arrays can be reconfigured optically in parallel using a holographic memory and a laser array. Parallel configuration is robust for any defect mode while serial configuration is weak for defects.

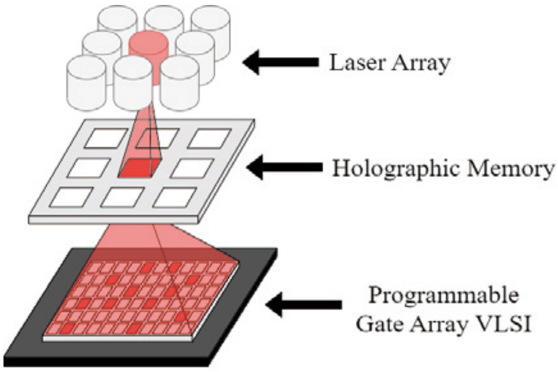

Figure 1. Overview of an optically reconfigurable gate array.

2. Realization of a wafer-scale VLSI using an optically reconfigurable gate array architecture

A wafer-scale VLSI based on an optically reconfigurable gate array architecture allows some gate array regions to be defected. Even if the wafer-scale VLSI has many defects, its configuration is possible. Figure 2 shows four examples of a 16-bit adder circuit using different hardware resource. These four programmable gate arrays are assumed to include about 15 defects for 56 configurable logic blocks (CLBs), or about 27% of the total number of CLBs. However, the 16-bit adder can be implemented by avoiding the defects. This indicates that a wafer-scale VLSI can be realized by using an optically reconfigurable gate array architecture.

Figure 2. 4 implementation examples of a 16-bit adder circuit avoiding defects.

Reference

[1] H. Shinba, M. Watanabe, "Radiation-hardened configuration-context realization for field programmable gate arrays," Applied Optics, Vol. 59, Issue 19, pp. 5680-5686, June, 2020.

O09

Sequential circuit implementation onto optically reconfigurable gate array VLSI using a ring oscillator

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Abstract

Currently popular very large-scale Integrations (VLSIs) are vulnerable to radiation. Devices used in high radiation environments are always susceptible to soft-errors and permanent failures. Sequential circuit must use a clock signal generated by a crystal oscillator. However, the crystal oscillator might be broken under high radiation environments. Therefore, this paper presents a ring oscillator circuit used for sequential circuits instead of crystal oscillators. The ring oscillator circuit has been implemented onto an optically reconfigurable gate array VLSI and the correct operation could be confirmed experimentally.

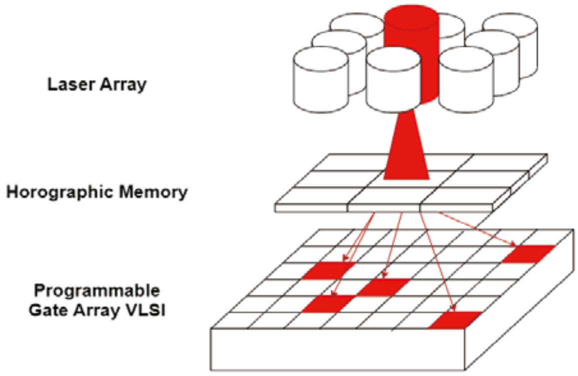

Figure 1. Structure of optically reconfigurable gate array.

1. Optically reconfigurable gate array

Optically reconfigurable gate array consists of three components: a programmable gate array VLSI, a holographic memory and a laser array.

The holographic memory on an optically reconfigurable gate array is the equivalent of serial ROM in existing FPGAs. The holographic memory is a highly radiation tolerant device and can store large amounts of circuit information. The circuit information is addressed by the laser array. Finally, a configuration context pattern can be read out from the holographic memory and finally programmed onto the gate array VLSI in parallel.

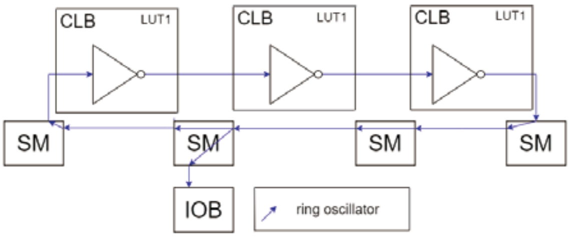

Figure 2. Structure of ring oscillator.

2. Ring Oscillator in operation

We have realized a ring oscillator using three look-up tables as shown in Fig. 2. The ring oscillator circuit has been implemented onto an optically reconfigurable gate array VLSI and the correct operation could be confirmed experimentally. Although redundant design is difficult when using crystal oscillators, using a ring oscillator for sequential circuits, if one ring oscillator is broken by radiation, another ring oscillator can help the situation. As a result, robust system can be realized.

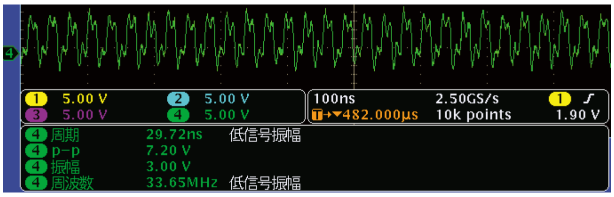

Figure 3. Ring oscillator operation.

Reference

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

MEMO

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