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Basic study on the separation of radioactive cesium from the incinerator ash by using Electrokinetic (EK) method

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ABSTRACT

In March 2011, a large amount of radioactive cesium (hereinafter referred to as Cs) was dispersed over a wide area as a result of Fukushima Daiichi Nuclear Power Plant accident after the devastating earthquake that occurred in Japan. The incinerated ash containing Cs (>8,000 Bq/kg) generated from the environmental decontamination was temporarily stored as designated waste in interim storage facilities, but the shortage of storage space has become a problem. In order to reduce the amount of requiring management, the cesium concentration in incinerated ash should be less than 8,000 Bq/kg. The purpose of this study is to reduce the volume of designated waste that needs to be preserved by removing Cs from incinerator ash using Electrokinetic (EK) method. EK treatment is a combination of water electrolysis, electrophoresis and electroosmosis. Two electrodes are placed in the polluted materials (i.e. incinerator ash for this study) with electrolyte solution and a DC voltage is applied in between the electrodes. The metallic pollutants (i.e. Cs, heavy metals etc.) are supposed to move according to the direction of electric field. It can be understood that the Cs ions will move to the cathode side and thus it will be removed from the incinerator ash. There are many reports on heavy metals using EK treatment but a few reports with radioactive materials. At first, a preliminary experiment was conducted with commercially available ash polluted with stable Cs with the concentration ratio 50 mg/kg. We found that more than 80% of stable Cs was separated from the ash with the EK treatment. Then we performed the EK treatment with two types of radioactive contaminated ashes (i.e. fly ash and bottom ash). The radioactive concentration of Cs was about 3,000 Bq/kg to 5,000 Bq/kg for the ashes. We succeeded in the separation of Cs from the ashes. The details will be discussed in the conference. Further works will be carried out with various test conditions, such as, the type of electrolyte and the structure of the test apparatus etc.

Keywords: fly ash, bottom ash, electrokinetic (EK) method, Cs removal

Development of Tools for the Examination of Post-Burst Rods Exposed to LOCA-type Events in the SATS system

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ABTRACT

During a loss of coolant accident (LOCA) in a nuclear reactor, nuclear fuel and cladding is rapidly heated, causing an eventual balloon and burst of the associated cladding. For high burnup fuel, there is concern that some of this fuel could disperse from the burst opening. In this work, new tools and their corresponding preliminary results are presented to improve understanding of the burst characteristics and to evaluate potential fuel dispersal during LOCAs involving high burnup fuel. Post-burst, the standard method of burst analysis has been to use calipers to measure burst width, length, and maximum cladding strain. While useful, this measurement method can be imprecise and is limited to a few points along the rod. To improve on this method, a scanning system is used in this work to measure the rod diameter more completely by using a light curtain profilometer and an optical camera. The system consists of 3 axes of motion: length-wise along the rod, rotation of the rod, positioning of a camera. On each axis, scans can be conducted with step sizes of up to 0.1 mm and 0.1 degrees for axial and angular displacements, respectively. Post-examination data reconstruction can then be used to identify the precise burst geometry and to provide 3D burst data to modelers for replication. In this work, data on 4 tubes (3 having been burst) was collected at various axial and angular respective steps sizes. Large step sizes (5 mm and 45 degrees) were used to collect images on the camera for full rod image stitching. Rod edge and diameter data was then collected along the entire rod at steps of 1 mm and 5 degrees, respectively. This was followed by a higher resolution scan of the burst region at 0.2 mm and 1 degree step sizes to reveal burst geometry. For very small bursts (~ 3 mm maximum width) a higher resolution of 0.1 mm and 0.1 degrees was also used, for a total of around 900,000 data points. Data was then post-processed to extract burst edge information and compute circumference, tube circularity, and caliper-equivalent diameter at each point, as well as the maximum width and length of the burst region. These data, as relevant, were compared to caliper-measured burst parameters and found to be consistent with them.

A second system for fuel dispersal analysis was also developed and consists of a mounting plate, two high precision displacement motors, and particle collection tray. After a test, the fuel sample is taken directly from the SATS system and loaded onto the mounting plate. The rod is then subjected to sinusoidal oscillations with amplitudes from 1 to 28 mm and frequencies from 1 to 10 Hz. Dispersed material is then collected in the tray and weighed/sieved as necessary. This system is demonstrated in this work using fuel surrogate ceramic sands and pellets in as-burst rods. Future work will demonstrate the use of these systems on actual fuel samples.

Keywords

High burnup nuclear fuel, burst, LOCA, SATS, Fuel Dispersal

Performance assessment of FBTR driver fuel after 105 GWd/t burn-up operated at a uniform high LHR of 400 W/cm

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ABSTRACT

The burnup limit of mixed carbide fuel (70%PuC-30%UC) of Fast Breeder Test Reactor (FBTR), India was set at 155 GWd/t based on post-irradiation examinations (PIE) of irradaited subassemblies at various burnups of 25, 50, 100 and 155 GWd/t. These fuel subassemblies were mostly irradiated at linear heat rating (LHR) of ~320W/cm, with sodium inlet temperature ~350°C. For carbide fuel subassembly (FSA) irradiated at LHR of 400 W/cm and higher sodium inlet temperature (380°C) from the beginning of life, the burnup limit estimated based on thermo-mechanical modeling is ~100 GWd/t. To validate the model and to explore the feasibility of extending the burnup beyond 100 GWd/t, PIE of an FSA irradiated to 105 GWd/t was carried out in hot cells of Radio Metallurgy Laboratory, of Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam, Tamil Nadu, India. This paper will discuss the irradiation performance of this Mark-I carbide fuel based on the examinations such as gamma scanning, fission gas release measurements, microstructural examinations and quantitative analyis on fuel porosities from microstructures.

PIE was carried out on a set of representative fuel pins extracted from the FSA. Axial scanning of the fuel pins carried out using a HpGe based gamma spectrometry system provided information on the axial swelling of the fuel stack and the fission product distribution. Maximum axial swelling of fuel among 8 pins examined was ~ 4%, which is on par with that of 155 GWd/t burnup (LHR: 320W/cm) fuel examined earlier.

Fuel pins of 105 GWd/t burnup (LHR: 400 W/cm) were punctured in hot cells and fission gas was analyzed with gas chromatograph using hydrogen as the carrier gas. The isotopic composition of the fission gas was also analysed using a Quadruple Mass Spectrometer. The maximum internal pressure observed in the fuel pins was ~2 MPa and the fission gas release was estimated to be 10-17.5%, which were similar to that of 155GWd/t burnup lower LHR fuel.

Metallographic examinations were carried out on fuel pins showing the higher axial fuel swelling. Transverse and longitudinal fuel pin sections were metallographically prepared by sequential grinding, polishing and etching. The micrograph of fuel pin transverse sections indicated closure of the fuel-clad gap. Circumferential cracks were present in the peak power location of the fuel pin. Porosities quantified using image analysis technique, indicated that fuel has reached restrained swelling with a porosity exhausted zone at the periphery. No clad carburization was observed at the fuel clad interface.

PIE of 105 GWd/t burnup fuel operated at LHR of 400W/cm, indicated higher fuel swelling similar to that observed for 155 GWd/t BU fuel operated at lower LHR. The higher LHR (400 W/cm) and higher operating temperatures of 105 GWd/t FSA led to higher center line temperatures as well as clad temperatures resulting in higher fuel swelling.The salient results of the fuel performance assessed from the gamma spectrometry, fission gas release and microstructural analysis will be presented and the factors that are considered for burnup extension beyond 105GWd/t would be discussed.

Keywords: Mixed carbide fuel, Post irradiation examination (PIE), Linear Heat Rating (LHR), Fuel swelling, fission gas release etc.

Studying the respirable airborne contamination from spent nuclear fuel fractures

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Key words: **Spent Nuclear Fuel (SNF), SNF Aerosols, Consequence Analysis of SNF Failure**

Spent Nuclear Fuel (SNF) assemblies, retrieved after its useful lifetime in a reactor, are stored in cooling ponds and then transferred into stainless steel canisters for further storage and disposal. An accidental cladding fracture during this timeline can result in the release of radioactive SNF dust, which is now available for contamination during potential canister failure events. This work focuses on understanding the source term of respirable aerosol fractions from fractured SNF rods under postulated accident scenarios.

In this regard, an unpressurized 6-inch ZIRLO clad SNF rod segment with a local burnup of 63 GWd/MTU was fractured under 4-point bending at the Irradiated Fuels Examination Laboratory at ORNL, and the radioactive aerosols released were collected and characterized. The results show a total dust mass of ~4.6 mg was deposited on the collection apparatus (Figure 1), and ~0.5 mg of respirable aerosols collected in the cascade. The collected dust was analyzed separately using ICP-MS to obtain an isotopic distribution of the source term.

The test results will inform a number of studies including the amount of source term and particle size distribution available to a canister breach in storage. This work will also help in site boundary considerations and predictive models looking at respirable aerosol release from accident scenarios.

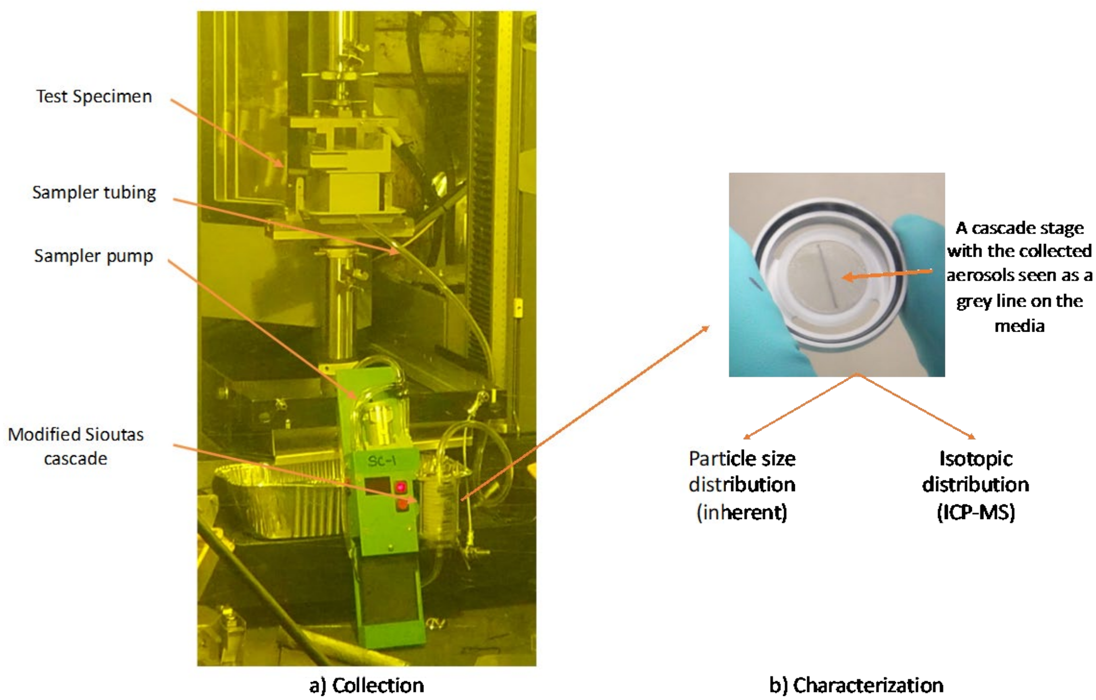

Figure 1. The two steps involved in this study: a) The 4-point bending test rig with the aerosol collection setup b) A cascade stage with the collected aerosol particles before characterization.

Hot Cell Design and Fabrication Corporate Experience

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ABSTRACT

Robatel's activities have been exclusively focused on the nuclear industry for civil, research, medical, and defense for the past 60 years. During this time, Robatel has developed a comprehensive range of skills and experience including design and manufacturing of hot cells, glove boxes, transportation casks, and waste processing equipment. Since 1998, Robatel has designed more than 45 unique hot cells in various countries for research institutions, commercial nuclear power plants, pharmaceutical, and fuel cycle companies. Each design developed by Robatel is unique and specifically tailored to meet the functional specifications supplied by the customer while complying with local regulatory requirements. This paper discusses several hot cells designed and fabricated by Robatel. The most recent project for TRIUMF included the design, manufacturing, inspection, and onsite commissioning of the ARIEL hot cell in Canada. The ARIEL hot cell is composed of 316L containment liners, 230 mm of lead gamma shielding, shielded and leak tight docking systems, access hatches, feedthroughs, penetrations, manipulators, shielded windows, cranes, and turntables. Additionally, the ARIEL hot cell included a bracing structure for seismic protection. The hot cell was assembled at Robatel's factory in France to test for leak tightness and streaming paths and to test the functionality of telemanipulators, HVAC, electrical, and mechanical systems. The lead slabs were cast and machined in-house for this project following Robatel's standard procedures for lead hazard controls. To comply with the Canadian codes of construction, the design and engineering analyses were reviewed and approved by a licensed professional engineer in Canada. In 2017, Robatel fabricated, tested, and commissioned a mechanical hot cell for the Institute of Materials Engineering to test irradiated materials at the Australian Nuclear Science and Technology Organization (ANSTO). This project included three cells with 190mm of lead shielding, leak tight containment enclosure, eight master slave telemanipulators, four shielded windows, and mechanical testing equipment, such as Charpy-V and tensile testing tools. Robatel was also responsible for the manufacturing, inspection, testing, and commissioning of a pharmaceutical hot cell for INVAP in Argentina. The hot cell was designed for handling Mo-99 and I-131 and featured lead and steel shielding thicknesses of 200mm and 330mm, respectively. The INVAP hot cells also included six telemanipulators and three lead shielded glass windows. These past design and fabrication projects uniquely position Robatel in the nuclear market as an experienced supplier of hot cells, shielded enclosures, and glove boxes for a wide range of applications, such material testing, pharmaceutical, waste processing, and defense programs. This paper will describe three recent hot cell projects with emphasis on the application, design, fabrication and testing, integrated equipment, and automated handling processes.

Keywords: hot cell, radiation protection, lead shielding, containment enclosure, nuclear waste processing

Hot Cell Material Transfers
Managing Vertical Loading and Offloading Horizontally

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ABSTRACT

This presentation is an overview of material handling at Pacific Northwest National Laboratory's Radiochemical Processing Facility (RPL), specifically around the loading and unloading of items originally designed for vertical handling including the modification or fabrication of support structures to accommodate a horizontal configuration.

Vertical clearances within existing facility structures both inside and outside of the hot cells provided significant restraints on meeting changing mission needs and project requirements. New methods of loading and offloading materials within the hot cells in a horizontal fashion were established and successfully implemented.

RPL teamwork developed a sled and track system to support receipt of a NAC cask. The cask is picked from the shipping container, placed onto the sled, and lowered into a horizontal position. The sled is then pulled into a containment where the lid is remotely removed. The cask is mated with the hot cell port, its contents retrieved, the cask is backed away, and then lid replaced. Finally, it is removed from the containment and placed back into the original shipping container.

Another cask often used is the Dry Transfer Cubicle Cask Insert (DTCCI) housed inside of a Model 10-160B Transport Cask. This insert cask is removed from the transport cask and placed within the facility. Using a three-point pick method and dual hoist overhead bridge crane the cask is lifted and rotated to a horizontal configuration. Once horizontal the cask is placed onto a hydraulic lift table and is mated with the hot cell where its contents are removed.

Internal transfers of high dose materials from one hot cell complex to another are done by use of specially designed horizontal casks. RPL has several different casks for distinct items and various levels of dose. These casks are equipped with a transfer scoop for receipt and offloading of materials and are used to support the transportation of high-dose waste from other hot cells to the High-Level Radiochemistry Facility for downsizing and packaging.

Shielded Waste Drums (SWD) may also be packaged horizontally and allow for the removal of low-level waste with a dose consequence of up to 1Rem. SWDs may be direct loaded in-cell or prepositioned for receipt of waste ex-cell.

Shielded Waste Cask Assemblies (SWCA) allow for the packaging of waste exceeding 1Rem. SWCAs are specifically designed and used as waste containers for transuranic and low-level waste streams. A specially designed lifting device hoists and rotates the cask 90 degrees into a horizontal configuration and then places it onto a specially designed cart. The cart is equipped with an auger that runs a transfer scoop out of the cask where a steel pipe is housed. The pipe can be loaded in-cell separate from the cask where it can be packaged with waste. The packaged pipe is then loaded back into the SWCA.

Where limitations on vertical clearances exist, alternative methods are identified and implemented. It is because of courageous ideas that the team at the RPL is successful in meeting and exceeding project needs and requirements.

Keywords: 10-160B transport cask, hot cell, material handling, material transfers, NAC, radiochemical processing laboratory, shielded waste cask assemblies, shipping casks, waste packaging.

Transient Fission Gas Release System for Irradiated Fuel under Loss-of-Coolant Accident Conditions

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Abstract

The transient fission gas release (tFGR) during the temperature increase associated with a loss-of-coolant accident (LOCA) in light-water reactors (LWRs) might be a significant contribution to the total pressure in a fuel rod and may cause an unexpected rod burst. The lack of data related to tFGR continues to be a key gap in understanding LWR cladding burst behavior under LOCA conditions. To fully characterize this behavior, tFGR data must be collected from several different systems. The available data suggest that tFGR testing must be performed under fuel-pin-relevant pressure (~10 MPa) to be characteristic of in-pile conditions. The US Department of Energy's Oak Ridge National Laboratory has developed a system to measure the integral tFGR from irradiated fuel segments. This system was designed to integrate with the existing Severe Accident Test Station (SATS) and to build upon decades of experience capturing fission gas to characterize fuel behavior. The tFGR system consists of a sweep-gas system to transport gases from the in-cell SATS apparatus to an out-of-cell fission gas detection system composed of a series of cold traps to capture the off-gas from the heating tests. It also includes a gamma spectrometer to detect and measure ^{85}Kr . A Mass Spectrometer is also available for the detection and quantification of stable gases. Initial system testing operations have been completed, and ^{85}Kr collection and measurement were verified along with the ability to detect stable inert gases. A tFGR test with a high-burnup fuel specimen is expected to be performed using the hot-cell system in fall 2023 .

Optical Microscopy Advanced Post-Irradiation Examination Visual Upgrades in the Hot Fuels Examination Facility Argon and Metallograph Loading Cells

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ABSTRACT

Visual examinations conducted during the non-destructive phase of post-irradiation examinations (PIEs) is a critical step in PIE and experiment performance assessment. Microscopy and additional destructive techniques such as microhardness are also fundamental to determine properties and status of nuclear materials after irradiation. Refurbishment and replacement of visual examination equipment in the metallograph loading cell at the Hot Fuels Examination Facility at the Idaho National Laboratory was completed for improved optical microscopy of nuclear fuel specimen. A modular inverted microscope (Leica DMi8) was replaced, and a new capability to support optical examination of fuels with a digital microscope (Leica DVM6) was developed and tested. A modular, removable camera was also developed and tested in-cell to improve visual examination efforts. In addition to the microscopy upgrades, micro-indentation hardness measurements were improved via development and installation of a vibration platform. The microscopes and micro-indentation hardness tester visual and measured data transfers were also upgraded to utilize fiber optic USB 3.0. A shielded fuel specimen container was constructed for in-cell use to reduce degradation of experimental infrastructure and extend the life of equipment used in-cell. In addition to visual upgrades to the metallograph loading cell, an automated visual examination machine miniature stage was developed, tested, and installed in the Hot Fuels Examination Facility argon hot cell for optical imaging of fuel specimen. The mini stage includes vertical and horizontal linear stages and a rotary stage that enables 360° imaging of fuel specimen up to 14-inch in length, 12-inch width, and 14-inch depth. These upgrades largely improved the quality of the data gathered with pre-existing equipment.

Keywords: Optical microscopy, post-irradiation examination, hot cell, visual examination,

Modifying analytical equipment without original equipment manufacturer support for hot cell use

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ABSTRACT

Cyclife Aquila Nuclear (CAN) designs and supplies special purpose equipment to all nuclear sectors globally. CAN boasts an established engineering team comprising of industry leading experts in equipment integration; both complete and partial.

Over the years this has manifested in many forms:

1. Sub-contracted design work to the Original Equipment Manufacturer (OEM)
2. Modification to Commercial Off The Shelf (COTS) equipment
3. Design of bespoke containment environments for interfacing with COTS equipment
4. Ruggedising and modularisation of COTS equipment
5. Production of factory specials to specific requirement specifications

With technical support and endorsement from the OEMs, where required.

For options 1 & 2 (above) the OEM is often required to share significant Intellectual Property (IP) (3D model data etc.), subject to a Non-Disclosure Agreement (NDA). Although this has been successfully navigated on numerous occasions it is expected and understood not all OEMs will feel comfortable sharing such IP.

What happens when an OEM cannot support a modification project?

1. This is not necessarily a surprise; it is recommended that key questions like this are answered in early stages such as deselection and optioneering
2. What is the risk?
   a. Is specialist knowledge required? OEM may still be able to provide this on a consultancy basis
   b. Detail information and explicit design intent is not known, such as geometries, assemblies, sensitive parts and complexity of equipment
   c. Process risk if modification is not successful
   d. Reputational risk if modification is not successful

CAN are currently working on a modification where this is the case.

The project is the modification/integration of a Rheometer, a laboratory instrument that measures a given fluid's response to an applied torque and/or shear force. The instrument shall ultimately be used in a Hot Cell environment.

Key points for the modification:

1. Modularisation to enable posting into and out from containment will constrain the maximum permissible size of assemblies
2. All Operation & Maintenance (O&M) of the equipment shall be performed exclusively via Master Slave Manipulators (MSMs)

3. Key instrument modules shall need to be assembled together, and disassembled, in containment using MSMs
4. Integration of environmental shielding to aid radiation resistance

What did we do to mitigate risk?

1. Engagement with the OEM on a consultancy type basis to ask questions to their technical team
2. Procurement of a COTS instrument; permitting disassembly in-house to gain a detailed understanding of internal parts of the instrument, including but not limited to
    a. Assembly envelope sizes
    b. Key component geometries
    c. Analytical process functionality
    d. electronic parts
3. Sharing of modification concepts with the OEM technical team to get guidance and input expertise where possible
4. Completed preliminary design & are now engaging with the OEM to produce a factory special
5. Rather than engaging the OEM for ancillary O&M tooling, CAN have taken on the design as Hot Cell experts

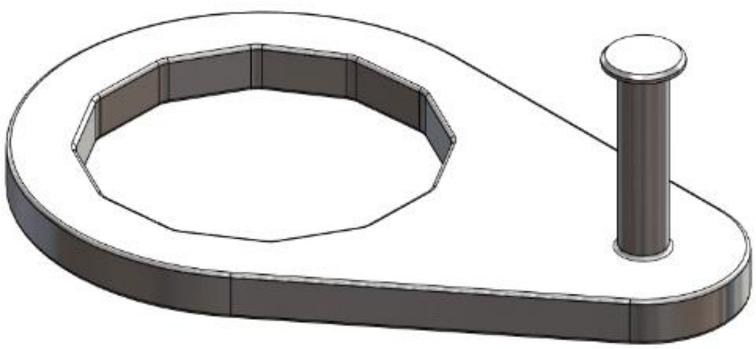

Figure 1 - Sample pot lock nut tool

Figure 2 - Sample pot cap

“AD System” a

Safe, fast and ergonomic Glove changing system

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ABSTRACT

During maintenance of glove boxes one of the most dangerous operation in term of safety and radioprotection remain the glove replacement. Many glove boxes still use “manual gloves” which mean that the glove is only maintained on the glove ring by a groove and a ribbon of adhesive tape. During this cumbersome glove change process there is also a high risk of mismanipulation leading to a contamination incident.

In glove boxes where plutonium is handled, it is even worse as plutonium powder is very volatile. Customers using this equipment since years (around 5000 gloves changed by year) only install this equipment now as it reduces a lot of incidents during a glove change.

In order to solve this issue LaCalhène has developed a glove system where the tightness is not based on the groove and tape but on a specific special seal maintained in an ejecting glove ring.

In addition to that the glove can be change very easily using an ejecting device, which avoid any human error during the process. The glove is changed using a push-push system that by design allows that at least of the two lip seals is in contact with the glove ring avoiding any containment issue.

Gloves can be to installed on the flange, but also sleeves to remove wastes or PE plugs the glove ports needs to be closed for the operators. As the PE plug has also a longer lifetime than a glove cost can be lowered using these kind occessories.

Characteristics:

The design concept gives containment continuity during glove change-over

Ejectable ring for fast glove changes without breaking containment

Glove port designed for the human arm

Ejection tool with integral passage for easy / rapid change-over

Several glove material are available according to the use (chemical/mechanic constraints etc)

Keywords: Glove system, Glove ports, glove box, safety, gloves, radioprotection

FUEL TRANSFER IN THE USED FUEL PACKAGING PLANT FOR CANADA'S CANDU REACTOR USED FUEL

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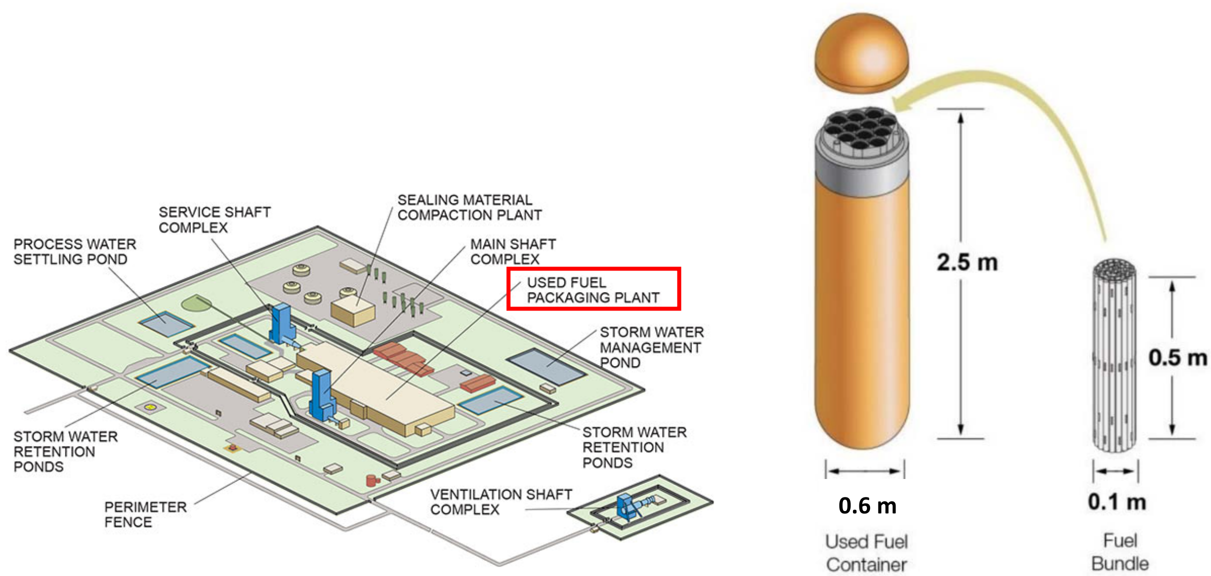
ABSTRACT

The **Nuclear Waste Management Organization (NWMO)** is a not-for-profit established in 2002 in accordance with the *Nuclear Fuel Waste Act (NFWA)* in Canada. The NWMO is responsible for designing and implementing Canada's plan for the safe, long-term management of used nuclear fuel. The plan, known as **Adaptive Phased Management (APM)**, requires used fuel to be contained and isolated in a **Deep Geological Repository (DGR)** [1] [2]. This DGR facility will be a novel and crucial infrastructure for Canada's sustainable energy future. The selected site will host **surface facilities** for the DGR as illustrated in Figure 1(a). The **Used Fuel Packaging Plant (UFPP)** being a vital part of the facility will include all necessary structures, systems, and components for: (1) Receiving used fuel transportation packages; (2) Loading used fuel into long-term storage containers, Used Fuel Containers (UFCs) (shown in Figure 1(b)); (3) Sealing, inspecting, and dispatching filled UFCs for underground placement.



(a) (b)

Figure 1. (a) NWMO's concept for the DGR surface facilities. The **Used Fuel Packaging Plant** (UFPP) is highlighted by the red box. (b) The long-term storage container, the **Used Fuel Container** (UFC) concept is shown adjacent to a used CANDU fuel bundle.

Most of the used fuel assemblies in Canada are CANDU fuel bundle configurations. This creates unique opportunities and challenges in the design process for used fuel handling and packaging systems. CANDU fuel bundles are natural uranium and the fuel bundles are approximately 0.5 m in length. These characteristics are essentially unchanged over 50 years of nuclear power use in Canada.

The natural uranium and small size of the CANDU fuel bundles presents no criticality risk outside of the reactor, making remote handling of the bundles easier at the UFPP.

However, the total used fuel inventory is substantial, and unique to the development of Canada's UFPP and DGR. Canada has an inventory of 3.2 million used CANDU fuel bundles with a mass of 60,000 metric tons. A total of 5.5 million bundles with a mass of 106,000 metric tons are expected when the UFPP and the DGR are operational in 2045 [2]. In comparison, the United States of America, the largest user of nuclear power in the world, has an inventory of 241,500 fuel assemblies as of 2015 with a mass of almost 80,000 metric tons [3].

The UFPP systems for fuel handling and packaging must be highly reliable and transfer the entire Canadian inventory of used fuel into UFCs for placement in the DGR. The concept for the used fuel identification and loading system in the UFPP that has been developed by NWMO and its partners will be discussed. The overall conceptual design of the UFPP was discussed at HOTLAB in 2022 [4].

Keywords: CANDU, used fuel transfer, nuclear waste, remote handling

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Modification of large hot cell for complex nuclear fuel experiment assembly and disassembly

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ABSTRACT

The Hot Fuel Examination Facility (HFEF) at the Idaho National Laboratory (INL) is a large, multipurpose hot cell used mostly for Post Irradiation Examination of nuclear experiments. As part of fuel testing programs, it has recently been modified to support the assembly, disassembly, and verification of water- and sodium-filled, fueled experiment vehicles exceeding two meters in length. Modifications include renovations to a ten meter deep pit in the floor of the air cell; the installation of a large work table with features to facilitate securing experiment capsules; the installation of an approximately 1.2 meter long feedthrough with instrumentation and power for an in-cell welding system; the modification of another 1.2 meter long feedthrough to provide power, signal, and gas access for various experiments; development of in-cell leak checking methodologies to verify that experiments have been sealed following remote assembly; development and installation of fixtures and tooling; and development of out-of-cell interfaces and control systems. All in-cell equipment is installed, operated, and maintained remotely. Additional modifications of even greater scope are currently underway on the argon cell to support sodium metal-filled experiments.

Keywords: Hot-cell, TREAT, Capsule, HFEF

Development of Machining Technique for Reactor Internals in Hot Cell

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ABSTRACT

Evaluation of irradiation effect and materials degradation on the reactor internal components from harvested materials is important for safe dismantling and improvement of operational safety for nuclear power plants. In this research project, evaluation of reactor internal component removed from Kori unit 1 are being carried out. As in similar overseas cases, crack signals were found in the baffle former bolts of Kori unit 1, and the root cause analysis of the defect were performed. To analyze the root cause of the defect, it is important to collect the harvesting specimens for post-irradiation examination and it should be carried out with the manipulator in hot cell. Therefore, machining technique in hot cell must be developed and performed necessarily.

In this technique, specimens are machined using wire electrical discharge machine and computer numerical control mill. And developed machining fixtures and machining programs are designed and manufactured for the hot cell and the baffle former bolt.

Tensile specimens were machined with the accuracy of 100 ㎛ through the remote processing. This technique is expected to be used for the purpose of harvesting specimens for post-irradiation examination of various types of dismantling reactor internal structures and the basis of irradiated material machining of Irradiated Material Examination Facility at KAERI.

Keywords: Baffle former bolt, Machining, Pressurized water reactor, Irradiation, Hot cell

Automatic inert gas fire suppression system in fuel PIE hot cells: upgrades and full-scale tests

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ABSTRACT

In 1997 an automatic inert gas fire suppression system was installed in the concrete hot cells used for fuel post irradiation testing. The system replaced the old Halon-based system and uses an inert gas mixture of nitrogen and argon called Inergen. Heat detectors in each cell are wired to a main control circuit which automatically trigger the gas release into the hot cell if the cell temperature exceeds 60 degrees Celsius. At each cell there is also a button where manual triggering is possible. A fire is suppressed by reducing the oxygen levels in the cell. The amount of gas needed to suppress the oxygen levels was calculated by volume of the cells. When triggered, the exhaust fans for the active ventilation are turned off, a main ventilation damper downstream the cells is closed and the inlet ventilation system is turned off. After installation the system was tested in one of the small 2x2x4 m^3 cells.

In 2017 the fire suppression system was erratically triggered, due to electric malfunction in central control, in the biggest cell 8x2.5x4 m^3. The activation of the suppression system created an overpressure in the big cell. The overpressure forced contamination, through various cell inlets, to adjacent rooms including the operating area and the transport area. Luckily, the event took place after operating hours and nobody inhaled the contamination. The event was reported to the Swedish nuclear authority who later rated the incident as a level 2 on the INES scale.

The decontamination took more than 1000 man-hours and compensatory measures to the failing fire suppression system had to be put in place before normal operations could continue in the hot cell facility.

After lengthy investigations, discussions with nuclear insurance companies, information gathering from other hot-cell facilities and fire suppression expertise from leading industry an upgraded automatic fire suppression system was installed and successfully full-scale tested in 2021. The major changes to the old system includes installations of automatically closing inlet ventilation dampers in each cell, keeping main ventilation damper open and keeping the exhaust fans running during an event. The system was tested in one small cell and the big cell with maintained under-pressure and fire extinguishing oxygen levels.

In the presentation more information on the causes tot the event, compensatory measures, clean up after the event, system upgrade details and lessons learned will be given.

Keywords: inert gas, fire suppression, hot cells, automatic

Image Enhancement on Short PIN PWR Fuel Contains Natural UO_2 Pellets Post Digital X-Ray Radiography Test Using ImageJ Program

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ABSTRACT

Non-destructive test using digital x-ray radiography is one of the important series in performance evaluation of nuclear materials and fuels. Image enhancement or image quality improvement is needed for further analysis of digital x-ray radiography image which tends to have lower resolution. The image enhancement process on digital x-ray radiography images is carried out to provide images that are easier to interpret so that resulting data can be used as a basis for evaluating performance after irradiation process. Tests using digital x-ray radiography on short pin PWR fuel contains UO_2 pellets was performed at distance 60 cm against the detector with voltage and current parameters of 120 kV and 1000 μA. Furthermore, image enhancement is executed on digital images using ImageJ which involves the process of aligning the image contrast, edge enhancement, processing histogram value, and adjusting image colors. Data results are adjusted from information that must be obtained from digital x-ray radiography test of short pin PWR fuel contains UO_2 pellets. Information obtained such as dimensional data, boundaries between pellet, and histogram that illustrate material or thickness differences. Further this information can be used as data input in mapping the concept of image enhancement short pin PWR fuel contains UO_2 pellets. Image output that produced fromimage enhancement process has a better resolution and quality to facilitate the process of analysis and interpretation also can be used as a standard method to perform image enhancement of digital x-ray radiography image of short pin PWR fuel contains UO_2 pellets after irradiation.

Keywords: image enhancement, digital x-ray radiography, short pin PWR, imageJ, interpretation.

PIE capabilities for the Second Target Station

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ABSTRACT

The Second Target Station (STS) project will upgrade the existing Spallation Neutron Source (SNS) that currently operates at Oak Ridge National Laboratory by delivering a new target station optimized for high brightness neutron production. At the heart of that system are several critical perishable components, including the target segment assembly. This assembly will be a cutting-edge development of spallation material bonded together using various manufacturing processes aiming to significantly improve operational lifetime and hence reduce need for maintenance periods. This target assembly is similar to other solid tungsten targets that are currently in operation, but this design also introduces several new developments and factors that will influence operations and lifetime prediction models. As this is a new development, it is considered to be one of the highest design risk elements for the project, therefore having the capabilities to perform root cause analysis tasks associated with component failures was deemed a project essential requirement. Post Irradiation Examination (PIE) activities have proven to be an invaluable resource for troubleshooting failures and improving component performance at the existing SNS target station. Therefore, as part of the design and development of the STS, a dedicated area to perform limited PIE activities is being planned to support the fundamental task of failure mode analysis on critical components.

This paper will provide an overview of facility requirements development, PIE-supported activities, and equipment required to support those activities for the STS project.

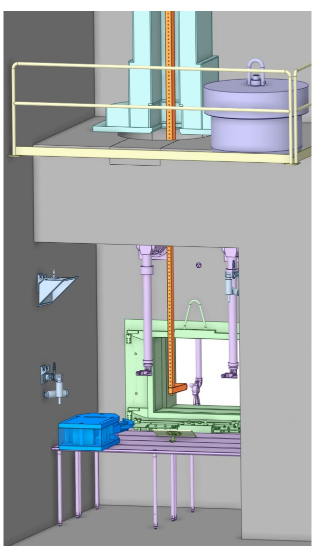

Figure 1: PIE Cell Layout Overview

Keywords: hot cell, post irradiation examination, spallation neutron source

Enhancing Conveyor Hot Cell Reliability: Replacement of Damaged Drive Wire Rope

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ABSTRACT

The Hot Cell Conveyor is a crucial apparatus used to securely transport irradiated shipments between separate cells. A recent visual inspection has revealed a defect in the drive steel wire rope, responsible for the movement of the conveying tray. To reduce the potential hazard of a complete wire rope failure, a detailed plan for its replacement has been developed.

The Hot Cell Conveyor is situated within a shaft positioned at the rear of the facility. The shaft includes loading apertures in each cell, an expedition window outside the Hot Cell, and a service hole at the entrance to Hot Cell No. 5. The replacement procedure will be executed through this service hole. Additional access points involve using the expedition window to reach a drive pulley and accessing an idler pulley through Hot Cell No. 1.

This replacement operation is initiated by detaching the existing damaged wire rope from the conveying system. A precise assessment of the pulleys, rollers, and guide mechanisms will be conducted to ensure they remain in optimal condition. Following that, the new wire rope will be installed, aligned, and tensioned to ensure flawless operation. The entire process will be carried out by a qualified team experienced in maintaining the conveyor system and adhering to safety protocols.

The suggested wire rope replacement procedure presents challenges due to significant contamination in the adjacent hot cells and limited shaft access. Dosimetry characterization needs to be performed before the replacement. Successful replacement ensures the continued functionality of the Hot Cell Conveyor and upholds the dedication to maintaining the highest safety standards.

In conclusion, replacing the damaged wire rope within the Hot Cell Conveyor supports the facility to continue to fulfill its crucial role in expediting the irradiated materials from research reactor LVR-15, thereby safeguarding personnel, the environment, and the seamless operations of the entire complex.

The presented results were obtained using the CICRR infrastructure, which is financially supported by the Ministry of Education, Youth and Sports - project LM2023041.

Keywords: Hot cells, Research Reactor, Conveyor, Replacement

Challenges for hot cells in the utilization of highly irradiated structural materials from the decommissioned Jaslovske Bohunice NPP reactor internals

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ABSTRACT

As nuclear power plants age and eventually reach the end of their projected operational life, it becomes essential to predict how the structural materials will behave over extended periods. Mechanical testing, corrosion-mechanical testing and microstructural analyses provide valuable data for modeling and predicting material degradation, embrittlement, and other aging-related effects.

One of the favorable ways how to expand the volume of available experimental results is the use of structural materials from decommissioned nuclear power plant units. However, the use of archive materials, especially highly irradiated materials from reactor internals, brings new challenges to the transport, handling and specimens machining procedures in hot cells infrastructures.

Paper is focused on the presentation of the current project realized from 2021 at the UJV Rez, a. s., Integrity and Technical Engineering Division, dedicated to the utilization of the structural materials from the reactor internals of decommissioned nuclear power plant Jaslovske Bohunice, Slovakia (Unit 1 and Unit 2, WWER-440 type). Highly irradiated structural materials extracted from the reactor pressure vessel internals will be used for an extensive experimental program of mechanical testing, corrosion-mechanical testing and structural analysis to provide results for improving the existing normative documents to ensure the long-term operation of WWERs. The paper describes the key challenges that have been addressed in the field of irradiated material blocks transport to the hot cells facility and the issues connected to the fabrication of test specimens for the experimental program.

Keywords: decommissioning, internals, handling, hot cells

Lirob - Advancing Automation in Hot Cells with LIROB

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ABSTRACT

Title:

In recent years, advancements in robotics and automation have revolutionized various industries, including applications in high radiation environments such as hot cells. Introducing Lirob, a cutting-edge robotic telemanipulator specifically designed for seamless integration into existing and newly designed hot cell facilities. This abstract presents an overview of Lirob's capabilities, highlighting its unique features and its potential contributions to enhancing safety and efficiency within hot cell operations.

Lirob offers a novel approach to automating hot cell processes by replacing traditional mechanical manipulators in standard wall openings between 160mm to 254mm. Its robust design enables it to handle payloads of up to 10 kg (20kg peak load) with ease and interact with hazardous materials and processes remotely, mitigating the risks associated with direct human intervention. This ability proves indispensable in scenarios where precision, safety, and efficiency are paramount concerns.

The key advantage of Lirob lies in its capability to automate processes, significantly reducing operator fatigue and physical strain. The robot's prowess in handling heavy objects eliminates the need for manual intervention, sparing operators from repetitive tasks that may lead to musculoskeletal issues. This aspect ensures a safer and healthier working environment, ultimately enhancing productivity and operational outcomes.

Lirob is engineered with user-friendliness in mind. Its intuitive programming interface empowers operators to effortlessly define and customize automated tasks while also having the option to intervene remotely and directly control Lirob's actions within the hot cell environment, granting them greater control and flexibility. Additionally, remote access enables operators to oversee operations from multiple locations, thereby promoting convenience and adaptability in real-world applications.This abstract seeks to underscore the transformative potential of Lirob in hot cell facilities. By augmenting human capabilities, this telemanipulator with robotic functionalities promises to streamline workflows, optimize resource utilization, and expedite critical processes. Attendees of the HotLab conference will gain insights into how Lirob is revolutionizing practices in hot cells and how its adoption could revolutionize other domains reliant on meticulous automation.

In conclusion, Lirob's integration into hot cells paves the way for a more efficient, safer, and advanced approach to remote handling tasks within the hot cell. As the nexus of robotics and high radioactive environments converges, Lirob emerges as a trailblazing solution, poised to shape the future of the nuclear discipline.

Fig. 1 to 4: 3-D Model of LIROB

Fig. 5: picture of LIROB

Keywords: Robotic, Hot Cell, Innovation, Automation, Operator, Efficiency, Comfort, Process.

Hot Cell Testing of Baffle Former Bolts Removed from Pressurized Water Reactor

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ABSTRACT

It is required to develop technologies that can comprehensively analyze various characteristics of the materials, which were exposed to high neutron irradiation environment for longer operating time.

Hot cell testing of the baffle former bolts removed from Kori unit 1 was performed to identify the root cause of the cracking as part of development of failure and degradation analysis technologies of reactor internal baffle former bolts from decommissioning NPP project.

Baffle former bolts removed form Kori unit 1 were examined and tested. The scope of work included visual inspection, dimensioning, radiation measurement and X-ray, PAUT examination as a non-destructive testing. Pull testing was conducted and the fracture surface was examined using optical microscope and SEM with chemical composition analysis. Tensile, IIT testing and metallography specimens were machined through a remote processing in a hot cell.

The defective indications of the UT response in the Kori unit 1 reactor were identified the crack of the baffle former bolts. The initiation and propagation of cracks shows similar patterns in the whole defective bolts, and the results of the fractographic examinations show the entirely intergranular cracking mode in the fracture surface. The length of bolts increased and the diameters decreased. Tensile specimens were machined with the accuracy of 100 ㎛ through the remote processing.

Keywords: Baffle former bolt, Pressurized water reactor, Irradiation, Hot cell testing

Abstract for HOTLAB 2023 Conference at Oak Ridge National Laboratory, Knoxville, TN

Chemical and Radiation Effects from Hanford Tank Waste on Tank-Deployable Reference Electrodes

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ABSTRACT

In some Hanford tanks, including 241-AY-101 (AY-101), 241-AP-102 (AP-102), 241-AW-105 (AW-105), and 241-SY-101, reference electrodes are deployed to monitor corrosion potential as part of the in-tank corrosion monitoring strategies to assess pitting corrosion and stress corrosion cracking risk. The objective of the work was to determine the chemical and radiation effects from actual tank waste of Tanks AY-101, AP-102, 241-AW-101 (AW-101), AW-105, 241-AN-106 (AN-106), and 241-AZ-101 on the tank-deployable reference electrode performance. Commercially available Ag/AgCl reference electrodes manufactured by Van London Company, eDAQ Inc., and Borin Manufacturing Inc. were exposed to waste collected from the tanks, and the test cells were set up in the hot cell at the 222-S Laboratory in 2021. Studies included weekly measurement of potentials of (i) tank-deployable reference electrodes and carbon steel electrodes made from alloys similar to the tank liners against saturated calomel electrode and (ii) the carbon steel electrodes against tank-deployable reference electrodes; bi-weekly measurement of electrochemical impedance spectroscopy of tank-deployable reference electrodes; and forensic analysis of posttest reference electrodes.

For all the electrodes, the potential fluctuated with time upon exposure to tank waste indicating performance degradation. Chemicals and radiation from tank waste had the most detrimental effects on the eDAQ electrode because of the nearly non-detectable thin layer of AgCl coating on the Ag wire and smaller electrolyte volume in the electrode to buffer the impact. Most of the eDAQ electrodes failed after exposure for several months. The potential of the Van London electrode fluctuated in the range of 40 to 200 mV with the most degradation in AW-101, which is the tank with the highest hydroxide concentration. The Borin electrode showed the best performance even under a highly radioactive environment, indicated by stable potential and impedance response. Posttest analysis showed that performance degradation of reference electrodes was mainly caused by intrusion and mixing of tank waste with the filling solution inside the electrode that degraded the frit material and changed the electrochemical behavior of the potential sensing element.

Corrosion potential of A537 or A515 Grade 60 carbon steel generally drifted in a positive direction over the test duration. It was the lowest in AW-101 and highest in AY-101, with AN-106, AP-102, and AW-105 falling between those two. Corrosion potential correlated well with the pH of tank waste, the higher the pH, the lower the corrosion potential. Because of the performance degradation upon exposure to tank waste, corrosion potential measured through Van London and eDAQ reference electrodes over-estimated the corrosion risk of the tank liner, with the eDAQ electrode being worse.

The reference electrode testing is on-going in the hot cell. In the meantime, we are developing the slow strain rate testing facility in the hot cell to assess the stress corrosion cracking risk in actual tank waste.

Keywords: reference electrodes, pitting corrosion, stress corrosion cracking, Hanford tank waste

Optimized Mechanical Testing Specimens from Nuclear Cladding Tubes for Hot Cell Testing

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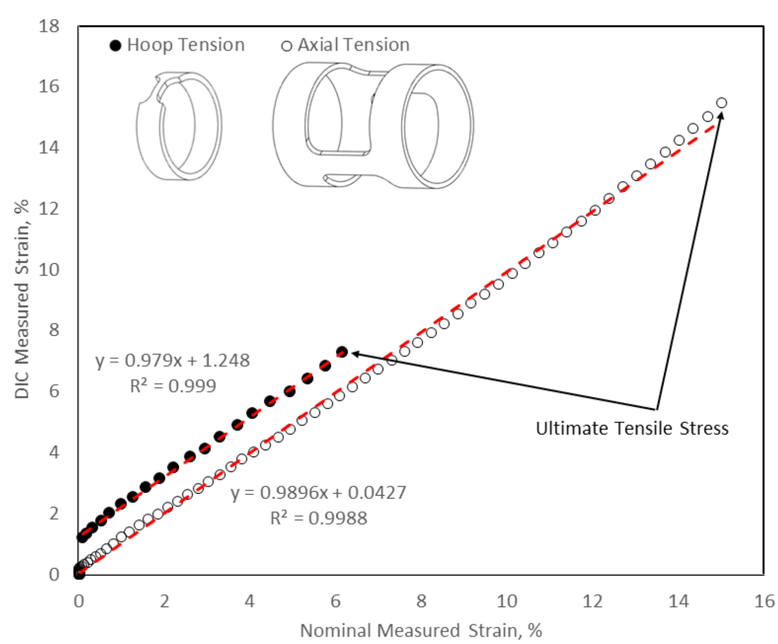
ABSTRACT

Maintaining the structural integrity of nuclear cladding tubes is the primary defense against fuel release. As a result, evaluating their mechanical performance is crucial to ensure the safety of the fuel rod both in and out of the reactor. However, because of the cladding's thin-walled tube geometry, several challenges exist for measurement of mechanical properties. Specifically, testing in the hoop direction induces bending that makes strain measurement difficult, frictional forces that make stress measurement difficult, or internal overpressures which does not inherently output strain. All these phenomena impact the nominal stress or strain measurements that would be obtained by mechanical test systems. Furthermore, with out-of-reactor materials there is a need to reduce specimen sizes to save the scarce material, but mechanical specimen preparation and testing must be done in a hot cell with manipulators with limited dexterity. In this work flat tensile, axial, and hoop direction mechanical specimens were machined out of bulk zircaloy among other materials and tested to correlate geometry-related effects. Hoop tensile and axial tensile geometries shown in figure 1 were machined out of nuclear-relevant alloys and results were compared to those from the unmachined component-level results to investigate scale-effects. Digital image correlation was utilized to evaluate geometry-related deformation effects to optimize understanding of nominal test output that is representative of a test in the hot cell. It was found that strain measurements from the nominal machine output correlated near 1:1 with those measured via DIC as shown in figure 1. Furthermore, a fully remote CNC machine was modified for hot cell insertion so the investigated mechanical specimen geometries can be machined out of irradiated cladding tubes in the hot cell. Comparisons of the mechanical test specimens and full-tube components and details on CNC machining are discussed.



Keywords: Nuclear cladding, Mechanical Properties, Mechanical Testing, Deformation, Machining

Post-Irradiation Examination of TRISO Fuel

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ABSTRACT

Tristructural-isotropic (TRISO) particle fuel for high-temperature gas-cooled reactor (HTGR) applications is an established technology; however, further development and performance testing are needed to commission the use of TRISO fuel in commercial applications. The U.S. Department of Energy Advanced Gas Reactor Fuel Development and Qualification (AGR) Program has undertaken an effort to support TRISO fuel qualification through a series of fuel development and irradiation activities. The AGR Program team are engaged in post-irradiation examination (PIE) and safety testing of compacts irradiated in the Advanced Test Reactor at Idaho National Laboratory. Herein, we present a PIE approach utilizing facilities housed within the Irradiated Fuels Examination Laboratory at Oak Ridge National Laboratory. Approaches such as high-temperature safety testing, deconsolidation and acid leaching of compacts, and analysis of individual particles with gamma spectroscopy, x-ray computed tomography, and microscopy are explored. A few results on PIE of AGR TRISO materials are discussed. Results and approaches presented further our understanding of how TRISO fuel performs under both normal and off-normal conditions, including evaluation of fission product migration and the manner in which failure mechanisms proceed.

Keywords: Tristructural isotropic, TRISO, post-irradiation examination

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Current Status of the Sample Preparation Laboratory at Idaho National Laboratory (2023)

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ABSTRACT

Idaho National Laboratory is currently building a new hot-cell facility called the Sample Preparation Laboratory (SPL). This user facility is designated as a beta-gamma facility and will support the characterization of structural materials for nuclear reactors. The status of construction of the facility will be presented. The design of the hotcell will be exhibited, providing attributes with its design for the activities associated with SPL. Planned activities and characterization equipment will be discussed including the use of shielded enclosures for advanced characterization and the use of a mechanical properties test cell. SPL will utilize robotics for loading activities of the instruments and the current status of development will be discussed.

Keywords: Hot cell, Design, Construction, Characterization, Robotics